Canadian Nuclear Safety Commission | Commission canadienne de sûreté nucléaire

CMD 26-H7.39-Ref1

Date: 2026-08-12

Reference from the Canadian Environmental Law Association

In the matter of

Ontario Power Generation

Application to refurbish Pickering Nuclear Generating Station and to renew licences for the Pickering Nuclear Generating Station and Waste Management Facility

Commission Public Hearing
Part 2

October 2026

Référence de l'Association canadienne du droit de l'environnement

À l'égard de

Ontario Power Generation

Demande concernant la réfection de la centrale nucléaire de Pickering et le renouvellement de ses permis pour la centrale et l'installation de gestion des déchets

Audience publique de la Commission
Partie 2

Octobre 2026

Document ID: QQQVZZNDK725-166150894-14787

Canada

VOLUME 1

DURHAM NUCLEAR AWARENESS, SLOVENIAN HOME ASSOCIATION & THE CANADIAN ENVIRONMENTAL LAW ASSOCIATION

APPENDIX B: SUPPORTING DOCUMENTS FOR SUBMISSION TITLED:
"***Comments on the Ontario Power Generation's Application to Refurbish Pickering Nuclear Generating Station Units 5–8 and to Renew the Pickering Nuclear Generating Station and Pickering Waste Management Facility Operating Licences"***

(Ref. 2026-H-7)

August 12, 2026

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Canadian Nuclear Safety Commission

Commission canadienne de sûreté nucléaire

Public hearing

Audience publique

April 4th, 2018

Le 4 avril 2018

Public Hearing Room
14th floor
280 Slater Street
Ottawa, Ontario

Salle des audiences publiques
14e étage
280, rue Slater
Ottawa (Ontario)

Commission Members present

Dr. Michael Binder
Ms Rumina Velshi
Dr. Sandor Demeter
Ms Kathy Penney
Dr. Marcel Lacroix
Mr. Timothy Berube

Commissaires présents

M. Michael Binder
Mme Rumina Velshi
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M. Marcel Lacroix
M. Timothy Berube

Secretary:

Mr. Marc Leblanc

Secrétaire:

M. Marc Leblanc

General Counsel:

Ms Lisa Thiele

Avocate générale :

Me Lisa Thiele

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Ottawa, Ontario / Ottawa (Ontario)

--- Upon commencing on Wednesday, April 4, 2018
at 11:18 a.m. / L'audience publique débute
le mercredi 4 avril 2018 à 11 h 18

Opening Remarks

M. LEBLANC : Rebonjour, Mesdames et Messieurs. Bienvenue à cette audience publique de la Commission canadienne de sûreté nucléaire.

The Canadian Nuclear Safety Commission is about to start the public hearing on the application by Ontario Power Generation, or OPG, for the renewal of the Nuclear Power Reactor Operating Licence for the Pickering Nuclear Generating Station.

During today's business, we have simultaneous interpretation.

Des appareils d'interprétation sont disponibles à la réception. La version française est au poste 2 and the English version is on channel 1.

Please keep the pace of your speech relatively slow so that the interpreters have a chance to keep up.

Les transcriptions seront disponibles sur le site Web de la Commission dans environ une semaine à

deux semaines.

To make the transcripts as meaningful as possible, we would ask everyone to identify themselves before speaking.

I would also like to note that this proceeding is being video webcast live and that the proceeding is also archived on our website for a three-month period after the closure of the hearing.

As a courtesy to others in the room, please silence your cell phones and other electronic devices.

Monsieur Binder, président et premier dirigeant de la CCSN, présidera l'audience publique d'aujourd'hui.

Mr. President...?

LE PRÉSIDENT : Merci, Marc.

Good morning and welcome to the public hearing of the Canadian Nuclear Safety Commission.

Mon nom est Michael Binder, je suis le président de la Commission canadienne de sûreté nucléaire.

I would like to begin by recognizing that we are holding this public hearing in Algonquin Traditional Territory.

Je souhaite la bienvenue aux gens ici présents, and welcome to those joining us via webcast.

I would like to start by introducing Members of the Commission.

To my right are Dr. Sandor Demeter and Ms Kathy Penney.

To my left are Dr. Marcel Lacroix, Mr. Timothy Berube and Ms Rumina Velshi.

We already heard from our Secretary, Marc Leblanc, and we also have with us here today Ms Lisa Thiele, Senior General Counsel to the Commission.

CMD 18-H5

Adoption of Agenda

THE PRESIDENT: I would like to start by calling for the adoption of the agenda as outlined in Commission Member Document 18-H5.

For the record, the agenda is adopted.

MR. LEBLANC: This is Part 1 of the public hearing. The initial Notice of Public Hearing and Participant Funding was published on September 29th, 2017.

A revised notice was posted on March 16th, 2018, to announce the location for Part 2 of the hearing.

Submissions from OPG and CNSC staff were filed on March 5th. They were made available on the CNSC website shortly after that date.

To support the CNSC's efforts in being an environmentally responsible organization, paper copies of the submissions will no longer be distributed on the day of proceedings as they are now available electronically on our website.

March 28th was the deadline for filing of supplementary information. We note that presentations have been filed by OPG and CNSC staff.

Part 2 of the public hearing is scheduled for June 26th, 27th and 28th, 2018, and will be held at the Hope Fellowship Church in Courtice, Ontario.

The public is invited to participate either by oral presentation or written submission at the Part 2 hearing. The deadline for the public to file a request to participate and a written submission is May 7th, 2018.

In terms of location in Courtice, just rest assured that we looked for a hotel or facility closer to the Pickering site and all over Toronto for that time period and nothing was available despite our using the assistance of the Tourism Bureau and other like organizations. We were well served in Courtice in the past, so we are pleased to go back there.

To assist the new Commission Members, I will ask the presenters and those answering questions to

please avoid using acronyms.

Mr. President...?

THE PRESIDENT: That's going to be tough.

So before we start I would like to share with you that we have representatives from Health Canada, Environment and Climate Change Canada, and the Office of the Fire Marshal and Emergency Management here with us.

Also joining us by teleconference are representatives from Fisheries and Oceans Canada.

Fisheries and Oceans, can you hear us?

MS THOMAS: Yes, we can. Thank you.

THE PRESIDENT: Thank you.

So I would like to start the hearing by calling on the presentation from Ontario Power Generation as outlined in Commission Member Document 18-H6.1 and 18-H6.1A.

I understand that Mr. Jager, you will make the presentation. Over to you. I guess not.

CMD 18-H6.1/18-H6.1A

Oral presentation by

Ontario Power Generation Inc.

MR. LOCKWOOD: For the record, my name is Randy Lockwood. Good morning, President Binder and Members

of the Commission.

I am the Senior Vice President, Pickering Nuclear at Ontario Power Generation and I am responsible for the safe and reliable operations of the Pickering Nuclear Generating Station. My career in the nuclear industry began in Douglas Point 40 years ago. Since then I worked at five CANDU stations across three continents in various roles of increasing responsibility.

Alongside me today, on my right, is Steve Gregoris, Deputy Site Vice President, Pickering; as well, on my left, Stephanie Smith, the Director of Operations and Maintenance, Pickering.

We are also pleased to be joined by our Nuclear President and Chief Nuclear Officer, Glenn Jager; and last but not least, other members of the OPG team seated behind me.

On behalf of the entire OPG team, it is our pleasure to be in attendance this morning.

We are here in support of our request for the renewal of the Pickering Power Reactor Operating Licence which expires August 31st, 2018. We have requested a renewed licence term of 10 years.

In a moment I will ask Glenn Jager to provide opening remarks, including a brief overview of Ontario Power Generation and how the Pickering licence

renewal plays a key role in our company's social licence.

I will then speak briefly about our specific request of the Commission and the fundamental commitments included in our licence application. I will also spend a little time to tell you how proud I am of Pickering's performance. The OPG team has achieved some of the best performance in the station's long history.

Steve, Stephanie and I will then expand on our commitments in the context of Pickering extended commercial operation.

I will touch briefly on the transition from the end of commercial operation to the start of the safe storage.

We also have a few short videos that will form part of today's presentation.

In addition, we plan to do a little show-and-tell with the emergency awareness kits. These emergency awareness kits are the very same packages that were supplied to people living near the Pickering site.

We hope you find our presentation to be informative and useful.

I will now pass the presentation over to Glenn Jager.

MR. JAGER: Good morning, President Binder and Members of the Commission. For the record, my name is

Glenn Jager and I am the President of Nuclear and Chief Nuclear Officer for Ontario Power Generation. I am accountable for all the nuclear operations within OPG.

OPG operates a fleet of nuclear units which includes Pickering, Darlington and three waste sites. We also own the Bruce site which is operated by Bruce Power. And we operate two combined cycle gas plants, two stations fuelled by biomass, 66 hydroelectric stations, and 264 control dams that control the water level throughout the Province of Ontario, including the Great Lakes. Three of our hydroelectric stations were developed in direct partnership with Indigenous communities located near those facilities. Together our facilities provide more than half of Ontario's energy needs and communities as far north as James Bay right down to Pickering.

We understand what it means and takes to operate in partnership with local and Indigenous communities, remote communities and we understand the obligations to operate safely and responsibly at all of our sites throughout Ontario. We understand the leadership and the example we set and the need to earn the trust of local communities each and every day. This is our social licence.

Pickering Nuclear is supported by OPG's full technical capability, financial resources and more

than 9,000 people throughout Ontario. OPG utilizes fleet standards and capability to ensure that Darlington, Pickering and Western Waste operate environmentally and to the highest standards of safety and reliability.

OPG is also part of a larger industry fleet. Through membership in the World Association of Nuclear Operators (or WANO), INPO, which is the Institute of Nuclear Power Operators, EPRI, which is the Electric Power Research Institute, CANDU Owners Group, which we refer to as COG, and others, we obtain standards of excellence, share operating experience and technical innovation. We receive training and assistance. We benchmark good practices and are subjected to critical peer evaluations of our performance. We support Canada's membership in the IAEA as part of various consultations and groups and are subject to oversight by both the IAEA and the CNSC. This network of support and oversight has resulted in both Pickering and Darlington achieving high levels of safety and reliability.

For example, last year Pickering operated three quarters of the year without an employee injury, so without a lost time accident certainly, but without injuring a single employee. This is quite remarkable performance. It is excellent performance that is really unheard of in the industry for a site that has six units

and almost 3,000 people on site. The proof of the effectiveness of this industry and corporate support is in our excellent performance, performance that Mr. Lockwood and his team will speak to today.

Pickering's greenhouse gas-free energy has contributed to the health and welfare of all Ontarians by providing clean energy, employment and supplying isotopes worldwide. As a result of OPG's 99 percent greenhouse gas-free energy from our nuclear and hydroelectric facilities, Ontario is the lowest greenhouse gas-emitting jurisdiction in the world. The safe work environment, clean energy, medical isotopes produced by Pickering are saving the lives of Canadians every day.

You know, I walked into Pickering on my first day at work 37 years ago and I was met with a very large plaque. A lot of the buildings around have changed since then, but the plaque is still there and I want to read to you in part what it says:

> "Dedicated to serve the people of Ontario, this nuclear power plant is a testimony to the spirit of adventure and ingenuity of Ontario Hydro, Atomic Energy of Canada Limited and the Manufacturers and to the skills of thousands of workers."

(As read)

Names of the companies have changed, but I am proud to say that as Chief Nuclear Officer that statement is as true today as it was 46 years ago when that plaque was dedicated.

I will turn the presentation back over to Mr. Lockwood.

MR. LOCKWOOD: Thank you, Glenn.

Randy Lockwood, for the record.

Commissioners, looking back at the start of my career and thinking about the early stages of Pickering, I recall the great sense of pride that existed at the plant and throughout the CANDU industry. Pickering was a major milestone in the Canadian development and implementation of the CANDU technology. I joined the Pickering team only a few years ago and I can tell you that this sense of pride has not wavered. In fact, it is growing.

As a team we are very proud that Pickering is heading into the next licence period with some of the best performance ever. This is a testament to the dedication and diligence of the employees and personnel who have been supporting the station's operation each and every day. I have had the privilege of working with many exceptional people throughout the nuclear industry in

several places around the world. The Pickering team is by far the best team I have ever worked with. They are the best team in the industry.

And while people have come and gone over the years, the plant too has changed. It's not the same as before, when it first began, it is better. It's far better. We have been investing in the plant all along and these investments have made the Pickering plant safer than ever before and more reliable. Our programs are solid. Our staff are highly trained and competent. For these reasons we are very confident that the plant can be safely operated and maintained, fit for service through to the end of commercial operations in 2024.

OPG will continue to invest in the station to make Pickering even safer tomorrow, to secure Ontario's clean power future and to make Pickering's last years the best years.

I would like to highlight some of Pickering's strong performance over the current licence term.

Pickering had an industry best all injury rate in 2017: .06 injuries for 200,000 hours worked. Said another way, this is the best Pickering industrial safety performance in the history of the site and I am very proud of our team for achieving that. Our collective radiation

exposure for workers has also been below target.

Pickering's equipment reliability has steadily improved, recording our highest Equipment Reliability Index, ERI, score in station history and it is continuing to increase. As a result, Pickering achieved record operating runs for two units in 2017, with Unit 5 at 632 consecutive days and Unit 1, 622 days. Recently, Unit 4 enjoyed a record run prior to shutting down for a planned maintenance outage on March 7th of this year.

The forced loss rate performance over the last three years has also been the best ever. And our maintenance backlog for safety critical equipment across the entire station, all six units, it is zero.

There have been no significant incidents over the current licence period. The plant has and continues to operate safely. We are pleased that Pickering received the CNSC integrated plant rating of fully satisfactory in each of the last two evaluations.

And as another major achievement, Pickering was issued a fish authorization by Fisheries and Oceans Canada in recognition of OPG's environmental stewardship for aquatic life.

These accomplishments are just a few examples to highlight Pickering's passion for excellence to protecting the public, the worker and the environment.

Investment for Improvement. In this slide I would like to impress upon the Commission some of the many significant investments we have made to further improve Pickering's safe and reliable operation.

During the current licence term, 2013 to 2017, substantial investments were made to implement numerous physical safety improvements at Pickering such as those listed on the slide. These safety enhancements significantly bolster the plant's defence-in-depth to guard against the very highly unlikely beyond design basis events.

Most notably, emergency mitigating equipment, or EME. Both Phase 1 EME and Phase 2 EME are in place and implemented at Pickering, ready for deployment in such highly unlikely events. With these safety improvements the plant is safer today than it has ever been. We will describe EME provisions in more detail later in the presentation, but for now I would just like to show you with this slide what this impressive fleet of equipment looks like.

As mentioned, we have both Phase 1 EME and Phase 2 EME. They are in place at Pickering, available, tested and ready for deployment.

Phase 1 EME consists of portable pumps and generators. The upper right, shown in red, is one of the

portable pumps. The main purpose: supply water to cool the fuel. The upper left, in green, portable generator. Main purpose: supply power to essential equipment such as instrumentation.

Phase 2 EME represents a substantial investment that focuses on protecting containment integrity. This includes larger electrical generators, such as shown here on the lower left photo. They can restore power to various containment systems, such as filtered air discharge, air conditioning units, hydrogen ignitors. There are five such generators with a total capacity of approximately 7 megawatts.

As outlined and supported in our licence application, we are requesting a 10-year licence renewal for Pickering's power reactor operating licence. The licence renewal will encompass continued commercial operations to the end of 2024, as well as transitioning the units to safe storage by 2028.

Associated with our submission is our request for the Commission to approve to operate the fuel channels up to 295,000 effective full-power hours, as this would correspond approximately to the end of 2024 for the lead unit. We consider that a 10-year licence term is necessary, acceptable, and appropriate for the following reasons:

It's consistent with a timeframe that is normally associated with periodic safety reviews in Canada;

It allows OPG to expedite the post-shutdown activities and ensure efficient and smooth transition to safe storage by using existing proven procedures. Please note that these type of activities are already licensed and were already successfully carried out to place Units 2 and 3 in safe storage;

It aligns with the current licence term of the Pickering Waste Management Facility. When that licence expires, OPG plans to seek a single decommissioning licence for the station -- for the site;

And lastly, a 10-year licence renewal provides regulatory certainty for our shareholder, the Province of Ontario, and our ratepayers for continued Pickering baseload generation during Darlington refurbishment and major component replacement at the Bruce.

Pickering's commitments: Throughout our licence application and CMD, we have put forth a set of six commitments which we view as fundamental to supporting our licensing case for extended operations of Pickering station.

These are general commitments which reflect the main areas of responsibility that OPG bears to the public and to the CNSC. They acknowledge public

interest and CNSC site expectations; uphold nuclear safety, namely in the areas of nuclear safety, fitness for service, staffing, low operational impacts, public transparency and engagement, and continued investment. During our presentation today, we will briefly describe how each of these areas are addressed.

I will now pause to pass the presentation over to Steve Gregoris.

MR. GREGORIS: Steve Gregoris, for the record.

I am the Deputy Site Vice President for the Pickering Nuclear Generating Station. I have worked at OPG for 27 years. I started my career as an engineer at the Darlington Nuclear Generating Station when it was under construction. Since then I have held a number of positions including certified shift manager and operations manager at Pickering, and Director, Operations and Maintenance at Darlington.

Recently, I completed a secondment with the World Association of Nuclear Operators in Atlanta. This work assignment provided me with a unique perspective of the nuclear power industry, which I brought back to my work at OPG.

On a more personal note, my wife also worked for OPG Nuclear. We have lived in the Durham region

for over 20 years and have raised our family in this community.

In the next several slides, I will elaborate on our efforts in support of Pickering's fundamental commitment to nuclear safety.

Pickering's first fundamental commitment is to nuclear safety. Specifically, nuclear safety will be assured, such that plant personnel, the public and the environment are protected. I can personally attest that not only for Pickering, but across OPG's nuclear fleet, safety is our core value. It underpins everything we do at our stations. Safe operation is at the heart of our social licence and of our operating licence.

In this part of the presentation, I will touch on some key aspects that support the assurance of nuclear safety at Pickering.

Defence-in-depth is a universally accepted cornerstone of nuclear safety. With a defence-in-depth approach, multiple and redundant barriers are in place to protect the public and the environment. As depicted in this simplified illustration, Pickering has five basic levels of defence-in-depth for the overall design and operation of the station, consistent with international practice.

Defence-in-depth is extremely robust. It

encompasses engineer structures and equipment with multiple backups designed to high standards and regularly inspected, tested and maintained. It also includes people-based barriers, such as administrative processes and procedures, as well as rigorous training and the fostering of a healthy safety culture. Through levels 1 to 3, these measures serve to prevent events from occurring or to effectively mitigate unlikely events that are postulated within the plant's design basis.

With the post-Fukushima implementation of emergency mitigation equipment, or EME, during the past licence term Pickering has provided additional barriers for the extremely unlikely event of a beyond design basis accident.

Basically, in level 4, EME is used in one of two ways. First, EME is used to prevent an event from progressing to severe accident and, second, together with severe accident management guidelines, the same EME can be used to mitigate a severe accident in the extremely unlikely event one were to occur. As will be discussed later, even more defence-in-depth measures are being pursued as part of our completed periodic safety review.

Lastly, further defence-in-depth for public protection is provided in the fifth level through offsite emergency preparedness.

These barriers serve to illustrate that there are many diverse ways to put water into the reactor cores to cool the fuel under any circumstance, and because of this a severe accident at Pickering is an extremely unlikely event.

This figure is for illustrative purposes to depict the many diverse and flexible deployment options for the use of EME in the extremely unlikely event of a beyond design basis accident.

At Pickering the EME is located and safely secured onsite near to, but separate from the station. With Phase 1 EME pumps, water can be drawn from the lake and injected into the heat transport systems, steam generators, or calandria vessels of any Pickering unit using a variety of deployment routes. Any of these options provide an effective means to cool the fuel.

In addition, the Phase 1 EME pumps can provide make-up water for the irradiated fuel bays to keep the used fuel covered with water and cooled. Phase 1 EME also includes portable generators and uninterruptable power supplies to power instrumentation for monitoring.

As part of Phase 2 EME, larger generators can be used to repower various equipment, including containment air coolers, hydrogen ignitors, and a filtered air discharge system which enables controlled and filtered

venting. It should be noted that EME Phase 2 can also be used to repower station equipment that will provide additional methods of cooling, in addition to Phase 1 EME.

Staff are trained and regularly practice the use of EME. Phase 1 and 2 EME can be quickly connected to provide the necessary supplies of power and water. We have demonstrated that EME can be rapidly deployed with ample time to mitigate a wide range of extremely unlikely events.

Simply put, if ever needed, EME can be deployed virtually anywhere onsite through many possible routes and by diverse means to provide emergency water and power to cool the fuel and protect containment.

In support of Pickering licence renewal, OPG has systematically conducted a periodic safety review, or PSR. The PSR focused on safety-significant items and has confirmed that current plant operation is safe.

Furthermore, the PSR considered all aspects important to continued safe operation and concluded that the plant can be safely operated through to the end of commercial operation in 2024.

The PSR was conducted in accordance with CNSC requirements and consistent with industry best practices. It assessed the extent to which Pickering meets modern regulatory requirements and standards. And from

this part of the process, OPG identified additional safety enhancements to further strengthen the already robust defence in-depth of the plant. The safety enhancements are documented in the integrated implementation plan, or IIP, which OPG is committed to. The IIP has been accepted by CNSC staff.

On top of all the EME safety benefits, with the PSR additional safety enhancements, Pickering will be even safer in the next licence term.

The PSR safety enhancements include various plant modifications. For instance, through a station interconnection, firewater from the Units 1 and 4, diesel-driven firewater pumps, shown in the slide, will provide an additional and diverse source of firewater to Units 5 to 8. This modification will serve to further enhance the Units 5 to 8 firewater system.

In addition, piping modifications will enable firewater addition to the Units 1 and 4 heat transport systems, steam generators, and calandria vessels. These modifications will significantly bolster Pickering's defence-in-depth by providing more diverse ways, separate from EME Phase 1, to cool the fuel in the extremely unlikely event of a beyond design basis accident.

In summary, Pickering will be even safer through the IIP actions, 63 in total, to be completed by

2020. This figure shows the safety benefit of the various Pickering improvements for Units 1 and 4. It shows the results of a probabilistic safety assessment, or PSA, for large release frequency, or LRF, on a per-unit, per-hazard basis. As shown by the middle group of results, compared to the left-most group, the post-Fukushima improvements, namely EME, have significantly lowered the estimated risk for the station as it is today.

Today, the units are well below the safety goal for each hazard and have improved by factors in the range of approximately two to ten from the pre-Fukushima values. As shown by the group of results on the far right, the estimated risk will be significantly further reduced by the PSR modifications, once implemented.

In driving towards continued safety improvement, OPG has established more stringent administrative safety goals as targets. The PSA administrative safety goals are aligned with new build requirements and are one order of magnitude more stringent than are standard PSA safety goals, as shown in the figure. With the PSR modifications, Units 1 and 4 are expected to be below the PSA administrative safety goals. These goals are already met for Units 5 to 8. In summary, the Pickering plant has evolved to nuclear safety levels that are comparable to modern new build requirements.

At this point, I'd like to summarize how EME and PSR modifications work together to protect containment and the public.

Pickering has a very robust containment design. This is evident by the very low values of LRF which are below OPG's PSA safety goal. The defence-in-depth at Pickering provided by our installed design basis equipment and the additional Phase 1 EME portable pumps and generators, ensures it is extremely unlikely that a beyond-design basis event progresses to the point where containment filtered venting is required.

The PSR modifications will provide even more defence-in-depth by providing additional ways to cool the fuel, making it even more unlikely that a beyond design basis event progresses to the point where containment filtered venting is required.

Nonetheless, Pickering has filtered venting capability today through the existing filtered air discharge system shown here with no power and for multi-unit events. Further, with the addition of the larger generators of Phase 2 EME, power can be restored to the containment air coolers and hydrogen ignitors to further protect containment structural integrity, and to the filtered air discharge system to provide additional diverse ways to enable where containment filtered venting.

In summary, we are confident that the potential for a large offsite release has been practically eliminated.

Pickering nuclear safety is supported and confirmed through comprehensive deterministic and probabilistic safety analyses. These complementary tools demonstrate the effectiveness of the plant safety provisions and the robustness of defence-in-depth. For instance, our PSA confirms that the safety goals are met for each reactor unit and hazard analyzed.

You may recall during the previous Pickering re-licensing hearings, concerns were raised around the overall risk of the site and whether the estimated risk for individual units should be simply multiplied by the number of units and added across all hazards. Well, it's not that simple and there is no international consensus on whole site PSA methodology. Regardless, OPG accepted the challenge and has since completed a whole site PSA for Pickering.

OPG updated the Commission on December 14th of last year when we presented a summary of the Pickering whole site PSA. This work has been conducted in collaboration with the CANDU owners group. It is a first-of-a-kind study that is at the forefront of PSA development internationally. The work is also summarized

in addendum "C", as in Charlie, of our CMD.

The whole site PSA supports that the overall risk of the whole Pickering site is low. In fact, total site-wide large release frequency representing all units and all hazards is lower than the safety goal that is applicable for an individual unit and a single hazard. This is a strong indication of low site risk and further supports the robustness of the overall plant defence-in-depth.

We have demonstrated that the risk of a nuclear accident is extremely low. Nonetheless, we remain prepared in the very unlikely event of a nuclear emergency.

We recognize that both onsite and offsite emergency preparedness are important measures in ensuring the safety of the public. For that reason, the plans that are in place are robust. Our readiness to respond to a beyond design basis event has been recognized as industry leading, and our emergency plans have been practiced and integrated with all levels of government.

Furthermore, EME is available at both Darlington and Pickering. Procedures are issued, staff are trained, and a response using EME has been practiced in a number of drills and exercises.

OPG continues to improve in the area of emergency preparedness. We are following up to assess any

necessary changes to our plans as a result of the updated provincial nuclear emergency response plan. Various diverse means of public emergency learning are in place, including sirens, radio, television, social media, and cellphones, and OPG has worked with various government organizations on presentation-distribution of potassium iodide pills, or KI pills, to all residents, businesses and institutions within 10 kilometres of the station.

Also, a website is maintained to provide information on the use of KI pills, including in the nine most common languages spoken within 10 kilometres of the plant. Furthermore, KI pills are made available for people within 50 kilometres of the plant for anyone who wants them.

Together with various external agencies, OPG worked hard to reach out and communicate with the public about emergency preparedness. Based on polling results, we are confident that the general public knows what to do in the extremely unlikely event of a nuclear emergency.

We have an ongoing series of public communications on the topic of emergency preparedness to help keep the public informed and prepared. Public awareness sessions are held on an annual basis.

We also distribute information brochures

containing KI and emergency preparedness information to households within 10 kilometres of the station. For instance, we held our flashlight campaign in 2014 and our KI and "Prepare to be Safe" campaigns in 2015 and 2016. More recently, in 2017, we distributed emergency awareness kits to residents in the vicinity of the Pickering plants.

Commissioners, we provided you with your very own emergency preparedness kits. These are the same kits that were distributed to residents within 10 kilometres of the station for them to keep handy, just in case.

A couple items that I would like to point out with the kit: The first thing is that there is space in front of the kit. That was intentional. The "Prepare to be Safe" KI pills that were previously distributed, are meant to fit within the case so that you have everything in one package.

Also included in the case is a pamphlet that describes to residents how to be prepared and what to expect in the unlikely event of a nuclear emergency. There is also a pen with a flashlight so that instructions and information can be read even in the case of loss of power and there is a magnetic checklist to make sure that you are prepared that can be placed anywhere convenient so it can be easily referenced.

The last thing I would like to point out is that this kit provides information to the public so they can be prepared for any emergency such as a loss of power and not just a nuclear emergency.

Our biggest test of emergency preparedness at Pickering took place in 2017. On December 6 and 7, we led the execution of Exercise Unified Control, where more than 800 participants, from over 30 agencies, federal, provincial, municipal, and non-government agencies, participated over the course of two days.

The following video will provide an overview of Exercise Unified Control.

--- Video presentation / presentation vidéo

MR. GREGORIS: Commissioners, we hope you found the video both enjoyable and informative.

I would like to note that the exercise was based on a severe accident scenario. EME was deployed, filtered venting was required, and protective action limits were reached, requiring evacuation.

There were also several new innovations employed during the exercise. For example, the new Unified RASCAL Interface, or URI, dose projection software was used by OPG, the province and the CNSC. The real-time automatic data transfer plant information emergency summary page was accessed by the CNSC and the province, and the new P25

radio system, common between OPG and Pickering Fire Services, was used, enhancing communications between these groups.

Exercise results demonstrated that there are robust emergency plans in place at all levels to deal with a nuclear emergency. Further enhancements to these plans are being implemented as part of the lessons learned.

Of particular value, this exercise strengthens the relationships with the various participating agencies. This will further ensure effective and well-coordinated response activities in the unlikely event of a nuclear emergency.

I will now pass the presentation over to Stephanie Smith.

MS SMITH: Good morning. I'm Stephanie Smith, for the record. I am the Director of Operations and Maintenance at Pickering, which is otherwise known as the plant manager.

I have worked for OPG Nuclear for 28 years. I spent nine years as a certified shift manager. I have experience in engineering operations, maintenance and fuel handling.

I've also had senior emergency roles in the OPG nuclear emergency response organization, including emergency shift manager, operations support manager, and

emergency response manager.

I too have lived in the Durham Region for the last 26 years, and I have raised a family here as well.

I'm now going to talk about fitness for service.

With respect to fitness for service, our stated commitment is as follows.

Systems, structures and components at the plant are fit to continue commercial operations to the end of 2024, and the inspection programs will ensure fitness for service during the next licence period. In the next couple of slides, I will discuss how this is supported by our aging management program. I will also explain how we ensure our fuel channels are safe and robust.

The comprehensive programs that are in place today are how we always know that the plant will be safe and reliable to operate to the very last day. We know that the condition of equipment in a plant can change gradually over time. That is why we have in place an effective and comprehensive aging management framework to ensure that critical safety equipment remains fit for service at all times.

We know that the plant is in good condition. We know because we inspect. Periodic inspections are rigorously performed on all major

components and structures to ensure fitness for service on an ongoing basis. Component and equipment health is evaluated regularly and trended by means of technical evaluations, inspections, maintenance, and testing in accordance with the requirement of applicable standards, OPG's integrated aging management program, and the life-cycle management plans which are in place for major components, such as fuel channels, feeders, and steam generators. These plans are submitted to the CNSC staff regularly.

Fuel channels. We appreciate that fuel channel fitness for service is a topic of specific regulatory interest, especially in the context of extended operation. OPG can assure the Commission, the CNSC staff, and the public that the fuel channel fitness for service will be maintained, with margin, to the end of 2024 for all Pickering units. This corresponds approximately to 295,000 effective full-power hours for the lead unit. At this service life, there's still ample margins to the fuel channel criteria. This is true for all of our units. The projected hydrogen equivalent levels, less than 100 ppm, are well within the validation basis of the fracture toughness models that are used to support fuel channel fitness for service.

Our confidence in ensuring fuel channel

fitness for service to 2024 is based on technical assessments, industry research, a well-established aging management program, and the mitigation measures to ensure ongoing fitness for service. We have committed to the IIP actions to support safe operation of the fuel channels to 2024. In short, we understand the fuel channel aging process, and have proven our ability to manage it well.

A more detailed and scientific explanation in support of the assurance of the fuel channel fitness for service is available in Addendum B of our CMD.

MR. LOCKWOOD: Thank you, Stephanie. For the record, Randy Lockwood.

Engage workforce. Staff are qualified and competent to operate the plant, and this will be maintained throughout the next licence period, including sufficient staffing numbers.

Staff engagement is a very important underlying part of this. I will briefly talk on some of the related topics of staff qualification, safety culture, and staffing management.

An engaged workforce is about having the right number of people, with the right skills and the right attitudes, doing the right work at the right time. Our nuclear management system provides a framework that establishes the process and programs required to ensure

Pickering achieves its safety objectives, continually monitors its performance against these objectives, and fosters a healthy nuclear safety culture. This is not possible without an engaged workforce that is aligned around common goals. It starts with awareness, understanding, and commitment by all levels of the organization. It involves promoting individual and department ownership and accountability, and it drives multifaceted initiatives all the way from the front-line worker to the leadership level.

Our extensive training programs are used to develop and maintain competent personnel, with the necessary skills and knowledge to safely carry out their work.

We periodically evaluate our safety culture to recognize positive attributes, and to identify areas for further improvement. A panel, consisting of the senior plant leadership team, meets each quarter to discuss the status of the nuclear safety culture at Pickering. A station-wide assessment was conducted at Pickering in 2015 and concluded that there is a healthy nuclear safety culture, a healthy respect for nuclear safety, and that nuclear safety is not compromised by production priorities. Another station-wide assessment is scheduled later in the year.

Our safety culture is also subjected to periodic external evaluation by industry peers, which confirms our safety culture.

We have initiatives under way to address OPG's demographics, especially through investments in knowledge management for ongoing operations.

Long-term hiring strategies are also in place to address staffing needs through to and beyond the shutdown of the station.

To ensure a supply of leaders, and a continuity of critical roles, a company-wide succession-planning process is in place. Staff engagement and motivation will remain a key focus at Pickering to ensure continued safe and reliable operation.

With respect to the impacts of normal operation, our stated commitment is that impacts of plant operation to the public, workers and the environment will continue to be of low risk, and adequately mitigated, while continuing to provide the various societal and environmental benefits of plant operation. This encompasses a wide range of topics. I will touch on a few of these topics in the upcoming slides, and our CMD does have more information.

Our performance is strong in all areas of environmental protection. For instance, our targets for

radiation dose to the public are much more stringent than regulatory limits, and we strive to keep them as low as reasonably achievable. The public radiation dose from Pickering remains a fraction of the 1 percent of the regulatory limit, and is negligible in comparison to natural background radiation levels.

Our environmental monitoring programs in the vicinity of Pickering assess the impact of operations on the environment and human health. Results are made available to our community and to the public on our external website.

Pickering is continually working to reduce the amount of low and intermediate level nuclear waste it produces, thereby reducing our environmental footprint so that future generations will not be unduly burdened.

Pickering's irradiated fuel bays provide a storage of used fuel until it can be transferred to the dry storage containers for interim storage at the separately-licenced Pickering waste management facility.

Also, in January of this year, Fisheries and Oceans Canada issued a fish authorization for the Pickering station in recognition of the mitigating measures in place and planned for aquatic species. This is, in part, thanks to our successful fish-diversion system, which features an extensive net that can be seen in the photo.

We are very, very proud of our safety record, and we work hard to maintain a safe work environment. Our conventional safety performance remains strong.

With respect to radiation safety over the licencing period, there were no radiation exposures at Pickering that exceeded regulatory dose limits or our much more stringent administrative dose limits.

Collective radiation exposure is an industry standard measure of radiation safety performance. Our performance in this area has been better than target in each year during the current licence term. We attribute this success in part to our As Low As Reasonably Achievable, or ALARA, safety culture, and to the programs and industry-leading technology that our staff have developed through innovation to minimize radiation exposures.

The following are just a couple of examples.

Major reductions in reactor-phase dose rate have been achieved by installing a combination of an overhead shielding canopy and end-fitting shielding tiles.

Also, we are using robots to avoid worker dose, and to improve monitoring of the plant. These robots, as you can see in the upper corner there, are used

in high gamma, neutron, tritium environments. They are very versatile and can help with equipment monitoring and inspections, and to support dose-mapping strategies for personnel entering medium or high hazard areas. These robots can go virtually anywhere in the plant, and, as such, they significantly help to reduce worker dose.

I will now pass the presentation back to Randy Lockwood.

MR. LOCKWOOD: Thank you, Stephanie.

For the record, Randy Lockwood.

OPG recognizes that members of the public, stakeholder groups, and local communities have a legitimate interest in Pickering operations. For that reason, we remain open and transparent, and frequently engage with these groups.

Our stated commitment, transparency and appropriate public and indigenous engagements, will continue. In the area of the Pickering station, OPG is aware of the aboriginal and treaty rights, interests and assertions of several indigenous communities. These communities include Williams Treaties First Nations, on whose traditional territory the station is sited, as well as the Mohawks of the Bay of Quinte, and the Métis Nation of Ontario, Region 8. OPG meets regularly with these representatives from these communities to provide details

of our nuclear operations, and to address any issues and concerns. We take these discussions very seriously and follow up on all feedback that we receive.

From a company's perspective, the OPG indigenous relations policy provides guidance and direction to ensure that meaningful discussions and information transfer occurs consistently across the entire province. In June 2017, OPG made changes to its supply chain policies and processes to encourage and support those indigenous businesses, who have a desire to become active vendors in the OPG supply chain. A minimum 5 percent indigenous content goal is now assigned to bids, and OPG encourages our existing vendors to formulate joint ventures and subcontracting opportunities.

As a company, OPG has successfully partnered with indigenous communities on hydro electric projects. In each case, the first nation is an equity partner and will earn a long-term stable source of revenue. These first nations also benefitted from the construction of these facilities through employment and procurement.

As well, OPG and Nanticoke Solar are working with Six Nations Development Corporation on a solar generating station on the site of what was once Ontario's largest coal-fired generating station.

Part of our indigenous engagement takes

place through a corporate citizens program. In 2017, OPG funded 84 indigenous community projects focused on youth, science, culture and sports.

At OPG, we're committed to gaining a greater understanding of indigenous peoples' cultures and traditional land use.

Public trust is a core key driver of our social licence, the ongoing approval of the community, and other stakeholders for our operations. It requires daily commitment. It means being part of the community, sharing its values, and supporting what matters to the community and to ourselves. In Pickering, that means meeting regularly with elected representatives and community members, it means partnering with dozens of environmental, educational, and community-building groups. It's a commitment we hold dear and that we demonstrate every day through our support of the community's safety and environmental causes, because our employees don't just work in the region, they live there too.

In keeping with the spirit of openness and transparency, a substantial amount of information is made publicly available on our external website, including numerous documents associated with our present licence renewal request. Such documents include key PSR documents, as well as PSA summary reports.

Continuing to invest. A recurring theme you heard us say today is how OPG is continuing to invest in the Pickering plant.

Our commitment. OPG will continue to invest in Pickering to support our objectives, including to improve equipment reliability to ensure fitness for service until the end of commercial operations, and to further enhance nuclear safety.

In addition to some of the physical improvements already described in the context of enhanced nuclear safety, our continued investments include additional inspections and maintenance of critical components to ensure safe and reliable operations.

We're also continuing to invest to protect the environment. Whether it's through implementation of dikes at the station, the establishment of biodiversity and wildlife habitat programs, or our ongoing salmon stockade program, we will continue to invest.

At OPG, we will continue to invest in our people, our plant and in support of our community. At Pickering we're also investing and leading the industry in the development and the application of innovative technologies.

This short video highlights our use of innovation and technology to improve safety, reliability

and station culture.

--- Video presentation / Présentation vidéo

MR. LOCKWOOD: Over the next licence term we plan to continue investing over $300-million of committed actions in support of extended operations.

This slide summarizes that a high level were just some of the -- are key areas our investments are aimed at. With these investments the plant will be even safer and more reliable in the future.

End of commercial operation. At OPG we have extensive experience not only in operating reactors for power production, but also managing the successful transition to their eventual permanent shutdown. With the end of commercial operation partway through our requested licence term we understand the need to carefully plan the various activities that will ensure not only a smooth and safe post-shutdown phase, but also a safe transition period in the time before shutdown.

On the next slide I will briefly explain how we plan to address these considerations into our sustainable operations plan and our post-shutdown stabilization activities plan.

In looking ahead into the transition period from the operating station to the end of commercial operation, we understand the special considerations

involved. This is part of our sustainable operations plan or SOP.

The SOP will be submitted to the CNSC staff five years before permanent shutdown of the first unit at Pickering.

Upon the end of commercial operation, stabilization activities include the defueling and dewatering of the shutdown reactors. The stabilization activity plan or SAP documents OPG's planning efforts for stabilization of the Pickering station and the safe storage state.

Both the SOP and the SAP comprehensively take into account all of the 14 CNSC safety and control areas in the context of these transition periods. We're highly confident in our ability to safely execute this work as evidenced by the successful safe storage of Units 2 and 3 which have been in place for several years now and which will inform our plans for the rest of the units.

Furthermore, we have conducted a predictive effects assessment and concluded the associated activities are of low risk to human health and the environment.

At this time I'd like to share with you a video that reflects our enthusiasm and optimism around Pickering extended commercial operations to 2024.

--- Video presentation / Présentation vidéo

MR. LOCKWOOD: As a team, we're very proud of that video's message. It reflects who we are, that's our plant, and we're proud to work, live and be a part of the local community.

In summary, I would like to reiterate the importance of Pickering in providing clean, safe, reliable, low-cost electricity to the people of Ontario.

We continue to protect the public, the workers and the environment and we continue to foster open and respectful relationships with the communities that have understandably interest in our operations.

At OPG, nuclear plants have had a long history of safe operation. We are one team, we are one OPG and our staff have remained and will continue to be highly skilled, engaged and motivated in keeping Pickering operating well now and well into the future through to 2024.

The Pickering plant has continued to evolve and improve over time, meeting or exceeding regulatory requirements and applicable standards. To this end, we have seen increasing high levels of plant performance, some of the best ever in the history of the plant.

Through continued investments, the

Pickering design has advanced to be safer than ever before. Through the integrated implementation plan, Pickering will be even safer going forward.

In conclusion, we're confident that OPG is qualified and the Pickering plant is able to continue its safe and reliable commercial operations through to the end of 2024 and to transition the station to safe storage by 2028. In doing so, we will continue to make adequate provisions to protect the public, the workers and the environment while respecting our national security and international obligations.

To close, I would like to speak to the image on the slide. For the context, very large flags hang on each end of both turbine halls at Pickering.

On one end an Ontario flag. It reminds us of our role, power with purpose, to provide clean, safe, reliable, low-cost power to the people of Ontario and to provide this power in a consistent and transparent manner that maintains public trust and confidence. We will always operate our station with minimal impact to human health and the environment.

On the other end of each turbine hall hangs an equally large Canada flag. Pictured here some of the team members that helped hang those.

We're proud of our heritage and the

important role Pickering plays in the CANDU industry. An image of the Pickering station is synonymous with Canada deuterium/uranium technology throughout the nuclear industry worldwide. I know this to be true having worked offshore in the industry for more than a decade.

Commissioners, the Pickering team are engaged and they're determined to continue safe and reliable operation of our station to 2024. We are -- we're the best team in the industry and we're committed and we're aligned around one simple mission, safely operate Pickering with improved performance year over year such that our last bit is our very best.

It's been a pleasure and an honour to appear before you today and speak on behalf of the Pickering team. Thank you for your attention and we'd be pleased to take any questions you may have.

THE PRESIDENT: Thank you. You gave us a lot to chew on, which means it's lunchtime.

--- Laughter / Rires

THE PRESIDENT: So, we will resume the hearing at 1:30. Thank you.

--- Upon recessing at 12:35 p.m. /

Suspension à 12 h 35

--- Upon resuming at 1:33 p.m. /
Reprise à 13 h 33

THE PRESIDENT: Good afternoon.

I would like to move now to the presentation from CNSC staff as outlined in CMD 18-H6 and 18-H6.A. I understand that, Mr. Frappier, you will start the presentation. Over to you.

CMD 18-H6/18-H6.A

Oral presentation by CNSC staff

MR. FRAPPIER: Thank you.

Thank you and good afternoon, Mr. President, Members of the Commission.

For the record, my name is Gerry Frappier and I'm the Director General of the Directorate of Power Reactor Regulations here at the CNSC.

With me today is Dr. Alex Viktorov, Director of Pickering Regulatory Program Division and Heather Overton, Senior Regulatory Program Officer of the same division.

Also, regulatory and technical staff from CNSC are present and will be available to answer questions, should the Commission have any.

The presentation complements CNSC staff's written submission found in CMD 18-H6. I'll start first with the purpose of the presentation and to explain its structure.

The CNSC staff presentation provides information regarding OPG's application to renew the power reactor operating licence for the Pickering Nuclear Generating Station.

In this presentation we will provide a summary as well as some highlights from CNSC staff's evaluation of the licence application and assessment of Pickering NGS' safety performance during the current licensing period. Full details of the CNSC staff assessment can be found in CMD 18-H6.

The current power reactor operating licence for Pickering expires on August 31st, 2018. OPG has applied to renew the Pickering operating licence from September 1st, 2018 to August 31st, 2028. Concurrently, OPG is seeking Commission's approval to operate fuel channels at Pickering's Unit 5 and 8 up to 295,000 effective full power hours. Note that the current licence authorizes OPG to operate Pickering fuel channels up to 247,000 effective full power hours.

As you can see from the outline, after a short introduction, this presentation will touch on a

number of subjects important from the point of view of giving a complete picture of Pickering's safety performance.

We will briefly describe CNSC staff's evaluation of the licence application, results of the recently completed periodic safety review, specific aspects related to the end of commercial operation, outcomes of regulatory assessment of OPG's safety performance, several areas of particular regulatory focus of interest, extended operation of the fuel channels and the proposed licence and Licence Condition Handbook.

At the end we will summarize CNSC staff's conclusions and recommendations regarding OPG's application to renew its operating licence.

I will now provide some background information on Pickering Nuclear Generating Station.

The Pickering Nuclear Generating Station consists of eight CANDU pressurized heavy water reactors and their associated equipment, structures and facilities which were designed, constructed and are operated to produce electrical power.

Units 1 to 4 were retubed in the late '80s/early '90s, Units 2 and 3 are in safe storage state and will remain in safe storage until the eventual decommissioning of the station. Units 1 and 4 and Units 5

to 8 are operational. Each operating reactor has a nominal electrical output of around 515 megawatts electric.

In this presentation, or in staff's responses, we may sometimes refer to Pickering A and to Pickering B which were the previous station designations. So, for the record, Pickering A refers to Pickering Units 1 and 4 and Pickering B to Units 5 to 8.

The Pickering Nuclear Generating Station is located on the north shore of Lake Ontario in the City of Pickering in the Regional Municipality of Durham, Ontario. The facility lies 32 kilometres northeast of downtown Toronto and 21 kilometres southwest of Oshawa. The facility is owned and operated by Ontario Power Generation Incorporated and generates about 14 per cent of Ontario's electricity.

The Pickering nuclear site also contains the Pickering Waste Management Facility which is licensed separately under a Class 1B Waste Facility Operating Licence.

Pickering also produces cobalt-60, a radioactive isotope used for sterilization of medical equipment and food products. Cobalt-60 is harvested from the irradiated reactor components that are removed from the reactor during planned outages.

As all nuclear power reactors in Canada, Pickering reactors are a CANDU design. These reactors were designed by Atomic Energy of Canada Limited, they use heavy water as a coolant and as a moderator and are fueled on-power with natural uranium.

A special feature of Canadian multi-unit nuclear power plants is that reactors are placed within an interconnected containment with a large vacuum building.

On the photograph, Pickering Unit 1 is the one with a white circle on the top, Unit 4 is on the very left, Units 5 to 8 are on the right and the large structure in front of it is the vacuum building. It's also possible to see the water intake channel with the fish diversion system barrier net.

The current power reactor operating licence for Pickering expires on August 31st, 2018. OPG has requested the licence to be renewed for a period of 10 years. As OPG intends to cease commercial operations at Pickering on December 31st, 2024, the requested licence period will cover three distinct phases: operations of the reactor on-power until December 31st, 2024; a stabilization phase which will include post-shutdown cooling, defueling and dewatering that will last approximately three to four years; and then the beginning of safe storage for the units that are not in safe storage now.

The proposed licence and accompanying Licence Condition Handbook sets the requirements and expectations for continued operation, as well as for the transition from operation to safe storage state. Later on in the presentation this transition will be further discussed.

I will now pass the presentation over to Dr. Viktorov who will discuss CNSC staff's evaluation of the OPG application for licence renewal.

DR. VIKTOROV: Thank you, Mr. Frappier. It's Alex Viktorov and I am the Director of Pickering Regulatory Program Division.

Commission Member Document 18-H6 provides details of CNSC staff's evaluation of the licence renewal application, as well as important highlights from the ongoing CNSC staff compliance oversight.

This presentation will summarize from CNSC staff's perspective the most significant aspects of the licence application review.

In August, 2017, OPG submitted an application for the renewal of its Pickering power reactor operating licence. This was followed by submission of supplemental information regarding implementation of new or revised regulatory documents and standards.

To arrive at the conclusions and

recommendations presented to the Commission in CMD 18-H6, CNSC staff assessed OPG's licence application and supporting information, examined the licensee's past performance and compliance with regulatory documents and requirements, and considered other matters of regulatory interest. Examples of such matters include Indigenous engagement, public outreach, response to previous actions from the Commission and follow-up on the Fukushima action plan.

In anticipation of OPG application for the licence renewal, CNSC staff set out prerequisites for the information to be included in the application. These prerequisites were based on the past best practices and aligned with the CNSC regulatory document, REGDOC-1.1.3, Licence Application Guide, Licence to Operate a Nuclear Power Plant.

Once the application was received, CNSC staff evaluated its completeness and proceeded to detail the assessments to confirm that the operation of Pickering has been and would be meeting regulatory requirements.

CNSC staff also confirmed that the appropriate regulatory documents have been identified for inclusion in the licence. The Licence Condition Handbook has compliance verification criteria.

Furthermore, CNSC staff has assessed

whether OPG will have a sound operating strategy considering that the plant will cease power production during the next licence period.

Staff recommendation regarding the licence renewal application are not based only on examining the application. During the current licence period CNSC staff have been performing oversight compliance activities and collected information characterizing safety performance of the licensee. The main items of compliance oversight activities include inspections and technical assessments.

The CNSC Pickering site office employs a number of highly qualified inspectors who conduct various types of inspections and, where necessary, results of inspection are supported by technical experts from Ottawa. On this slide we give numbers of inspections conducted during the current licence period.

Technical assessments represent in-depth critical reviews by CNSC technical specialists on the information provided by OPG. Such information may be in forms of scheduled reports, safety analysis, events reports, provisional OPG governance or research and development reports.

As can be seen from this slide, over the year CNSC staff maintains steady regulatory oversight of Pickering safety performance.

To sum up, the CNSC staff submits evaluation of OPG's application for licence renewal, conducts themselves in a robust regulatory framework, relied on factual evidence collected through CNSC oversight activities, as well as submitted by OPG, used modern practices and tools and built on practices that have been proved successful in the past.

In the next few slides I will go over the main points of the periodic safety review conducted by OPG in support of its licence application.

A periodic safety review, or PSR, is a comprehensive evaluation of the design, condition and operation of a nuclear power plant. A PSR is proven to be an effective way to obtain a comprehensive view of the plant's safety and to identify safety improvements. The PSR is complementary to, but doesn't replace regulatory oversight activities.

CNSC required that OPG perform a PSR if Pickering operation were to be extended beyond 2020. Regulatory document REGDOC-2.3.3 establishes CNSC requirements for the conduct of PSR and it identifies the four phases in the conduct of such reviews.

The first phase is the development of a basis document which sets out principles, rules and scope and discuss the methodology for the conduct of such review.

The PSR document is produced by the licensee and reviewed and accepted by CNSC staff, if deemed sufficient.

The second phase of PSR results in preparation of safety factor reports. These reports contain detailed technical assessment that compares the plant design, actual condition of systems and operational processes to the modern standards.

In the third phase, a global assessment report integrates findings of the safety factor reports into an overall assessment of the plant condition and helps determine practical actions to enhance the safety.

Finally, in the fourth phase, identified safety improvements are brought together in the integrated implementation plan. The integrated implementation plan is accepted by CNSC staff and will form part of the licensing basis of the facility.

The conduct of Pickering PSR included the review of 74 modern standards. The outputs of these reviews resulted in 63 actions included in the integrated implementation plan. They need to align the station design and processes with modern requirements.

Among the highest ranked safety enhancements included in the scope of the IIP are those related to: first, updating the life cycle management plan

for the major components such as steam generators, feeders, calandria internals and fuel channels; two, revising safety analysis impacted by effects of aging; three, implementing upgrades to strengthen protection of containment integration; and, four, performing condition assessments of piping system and components such as valves, pumps and heat exchangers for the extended period of operation.

Implementation of the integrated implementation plan would be required by the proposed licence condition 15.1 and all IIP actions are targeted for completion by December 31st, 2020.

This slide presents the process to derive the actions which were ultimately included in the PSR Integrated Implementation Plan.

In the conduct of the Periodic Safety Review, OPG identified 120 gaps from the reviews against the 74 modern codes and standards, and the review of the Continued Operations Plan and re-examination of the post-Fukushima action plan.

CNSC staff reviewed all OPG submissions and further identified 23 gaps.

Using a global assessment method, OPG consolidated the 143 gaps into 51 global issues representing common themes or related subjects. The global assessment also resulted in 117 resolution plans.

These resolution plans were then screened to determine whether they need to be included in the Integrated Implementation Plan, as follows.

Twenty-two (22) resolution plans were judged to require no further action since the issues either have already been resolved or plans existed for their resolution outside of the Periodic Safety Review process.

Thirty-five resolution plans were determined to be acceptable deviations since the issues were of very low safety significance or other options were established to meet the intent of safety requirements.

Another 35 resolution plans were identified as requiring actions in the Integrated Implementation Plan and developed into the so-called resolution statements or more specific projects with concrete deliverables.

Also, 25 resolution plans were labelled as cross-reference, meaning that their scopes are covered under the other 35 plans.

The resolution statements were ultimately expressed as 63 distinct actions to be executed by OPG in the Integrated Implementation Plan.

Results of the Pickering Periodic Safety Review are the product of a detailed and systematic assessment by OPG that required more than two years of

efforts to complete.

CNSC staff remained engaged in close regulatory oversight throughout the project and undertook a meticulous independent examination of the PSR method, scope and outcomes. Based on the CNSC staff conclusions, CNSC staff accepted the PSR Basis Document and, subsequently, the Integrated Implementation Plan.

CNSC staff conclude that:

- OPG has performed a PSR review in accordance with CNSC requirements;

- Pickering design, operation, processes and management system will ensure continued safe operation of Pickering to the end of 2024;

- the actions identified in the IIP will further enhance safety of the station; and

- Pickering Nuclear Generation Station will only be operated if fitness for service of the structures, systems and components important to safety is continuously assured.

CNSC staff will be evaluating OPG's execution of the IIP and updating the Commission on the status of the IIP actions as part of the annual NPP Regulatory Oversight Report.

In the following slides we will describe how the end of commercial operation fits within the

requested licence period.

On the 28th of June 2017, OPG informed CNSC that all Pickering units would cease commercial operation on December the 31st, 2024. After the units are shut down, the reactors will undergo stabilization activities when fuel is removed from the reactor cores and placed in the irradiated fuel bays. Most systems will be drained and de-energized. Those systems required to assure cooling of irradiated fuel, radiation control and security provisions will remain operational. The beginning of the safe storage stage in 2028 marks also the beginning of the station decommissioning.

OPG's strategy for operation and eventual decommissioning of Pickering consists of five stages as shown on the diagram. It includes:

- continued operation until December 31st, 2024;

- stabilization activities from 2025 to 2028;

- safe storage with surveillance from 2028 to 2050;

- station dismantling and demolition from 2051 to 2061; and finally

- site restoration from 2061 to 2065.

The dates at this time are tentative.

OPG is requesting a 10-year licence. Therefore, the requested licence period would cover stages 1 and 2, as well as the beginning of stage 3.

CNSC staff recognize that the approaching end of power generation and transition to safe storage will pose unique challenges and require special licensing provisions. Such challenges will necessitate dedicated regulatory oversight that is not fully covered by baseline programs.

In anticipation of the permanent shutdown of the Pickering reactors, the proposed licence contains site-specific Licence Condition 15.4. In particular, the condition will require OPG to implement and maintain plans for the end of commercial operation, such as a Sustainable Operations Plan and a Stabilization Activity Plan.

Should OPG decide or be required to operate any unit beyond December 31st, 2024, the compliance verification criteria under this Licence Condition will require that OPG notifies CNSC no later than December 31st, 2022, of its intent to operate any unit beyond December 31st, 2024; and provide a timeline and scope for the assessment of the safety impact of operations beyond 2024.

The Sustainable Operations Plan, or SOP, covers the period leading to the end of commercial operation. It is to be developed and implemented at least

five years prior to the permanent shutdown of any unit and updated annually.

The Stabilization Activity Plan, or SAP, covers the period after permanent shutdown leading to safe storage. The SAP is to be developed and implemented at least three years prior to the permanent shutdown of any unit and implemented immediately after the unit is shut down.

The two plans, SOP and SAP, must demonstrate how requirements within all 14 Safety and Control Areas will continue to be met during this period.

It is expected that the two plans will include the following elements:

- the long-term workforce planning;

- human performance and staffing strategy while approaching the unit shutdown, and from the unit shutdown through the transition period to safe storage; and

- demonstration that systems, structures and components will remain fit for service while they are required to assure safety.

OPG is expected to continuously analyze emerging issues during this period leading to the end of commercial operation.

This slide shows the relationship between the regular CNSC programs and activities and the additional

plans for overseeing the end of commercial operation.

CNSC staff will conduct regular compliance oversight activities to ensure the licensee implements and maintains programs required by the licence and in accordance with the compliance verification criteria.

The PSR and resulting IIP identify safety enhancements for the continued safe operation of Pickering during this period. CNSC staff will also assess implementation of the IIP by executing focused compliance verification activities.

The Sustainable Operations Plan and Stabilization Activity Plan identify unique incremental activities within each Safety and Control Area that may not be covered by existing programs. CNSC staff will assess the deliverables under these two plans and follow up with OPG as required.

Staff will include updates on the progress of these activities in the annual NPP Regulatory Oversight Report.

Commission Member Document 18-H6 presents CNSC staff's assessment of Pickering's safety performance in each of the 14 Safety and Control Areas in detail. Outcomes in all Safety and Control Areas, as well as their constituted specific areas, were summarized to reflect Pickering performance during the current licence period.

Here I would like to point out that the systematic presentation of the regulatory oversight for all regulated activities in well-formulated Safety and Control Areas is one of the strong features of the Canadian regulatory regime which compares favorably with the international counterparts. This approach allows a comprehensive and balanced consideration of all aspects important to safety.

This slide tabulates Pickering safety performance ratings for the 14 Safety and Control Areas during the current licence period.

"SA" indicates a satisfactory rating and "FS" represents a fully satisfactory rating. As you can see, Pickering performance has been satisfactory or fully satisfactory across all areas.

Results for 2017 have not been fully processed yet, but CNSC staff expect that there will be no change from the performance in 2016. Results of 2017 will be presented to the Commission with the annual NPP Regulatory Oversight Report.

In one Safety and Control Area, namely, the Radiation Protection, results went down from fully satisfactory to satisfactory in 2016. The specific reason for this will be explained later on in this presentation.

Overall, CNSC staff compliance oversight

confirms that Pickering Nuclear Generating Station has a good safety record. There are currently no major safety issues.

CNSC staff concluded that:

- OPG meets regulatory requirements in all Safety and Control Areas and received a satisfactory rating or higher in each area;

- licensee programs are implemented and maintained effectively and in accordance with the regulatory requirements in all Safety and Control Areas;

- no workers or members of the public received radiation doses in excess of the regulatory dose limits;

- releases of nuclear and hazardous substances are controlled, monitored and assessed, and were below regulatory limits;

- safety enhancements and improvements continued to be made during the current licence period.

In other words, OPG is qualified to carry on activities authorized by the licence.

CMD 18-H6 provides detailed information on Pickering's performance in all Safety and Control Areas.

The next few slides will focus on selected topics judged to be of particular interest to the stakeholders. I will pass the presentation over to

Ms Overton who will discuss these topics.

MS. OVERTON: Thank you.

My name is Heather Overton, for the record.

I will be discussing five areas of regulatory focus, these are:

- Probabilistic Safety Assessment;
- fitness for service of fuel channels;
- radiation protection;
- environmental protection; and
- emergency preparedness.

Probabilistic Safety Assessment, or PSA, provides numerical estimates of the likelihood of various scenarios and their consequences. It also provides risk insights which can be used for design or operational improvements to further enhance plant safety.

OPG completed S-294 compliant PSAs for Pickering A and Pickering B by 2014. In accordance with regulatory requirements for periodic updates, OPG submitted an updated Pickering B PSA in December of 2017 and will submit the updated Pickering A PSA by December of 2018.

In addition, OPG will submit the additional analyses required by REGDOC-2.4.2, which has superseded S-294, by the end of 2020. These will include the consideration of other sources such as irradiated fuel

bays and other plant operating states.

During the licence period, OPG also developed and implemented a Risk Improvement Plan, as requested by the Commission following the Pickering licence renewal hearing in 2013. The Risk Improvement Plan included physical, procedural and analytical changes to further reduce plant risk, focusing on the dominant risk contributors. As will be shown on the next slide, the dominant risk contributors for Pickering A is the Fire Level 1 and Level 2 PSA and Pickering A Internal events Level 2 PSA. The implementation of the Risk Improvement Plan resulted in significant reductions in the PSA results.

Lastly, during the licence period OPG also completed a whole-site PSA for Pickering, as requested by the Commission. I will discuss the whole-site PSA a bit later in the presentation.

This slide shows the current Pickering A and Pickering B PSA results. The Pickering B results are those submitted as part of the 2017 update and the Pickering A results are those from the 2014 submission, except for fire hazard which was reassessed in 2017 as part of the Risk Improvement Plan.

The red line on these graphs indicates the safety goal (or safety goal limit) and the purple line represents the safety goal target, which is sometimes

called the administrative safety goal.

The safety goal limit represents the limit of risk tolerability above which actions must be taken to reduce the risk. The OPG safety goal limit for core damage frequency is 1x10 to the -4 occurrences per year and for large release frequency it is 1x10 to the -5 occurrences per year per unit per hazard. They are consistent with the international practice for existing nuclear power plants which are documented in IAEA Publication INSAG-12.

The safety goal target, set one order of magnitude lower, represents the desired objective towards which the facility should strive, provided that measures to reduce risk are practicable. The safety goal target for core damage frequency is 1x10 to the -5 occurrences per year and for large release frequency is 1x10 to the -6 occurrences per year. Theses are consistent with the CNSC design requirements for new nuclear power plants.

As can been seen from the charts, the unit aggregated core damage frequency, which is the chart on the left, and the large release frequency, the chart on the right, for all hazards meet the safety goal limits for existing plants.

Core damage frequency and large release frequency for all hazard groups meet administrative safety targets except for Pickering A internal fires for core

damage frequency and large release frequency, and Pickering A internal events for large release frequency.

As OPG is currently updating the Pickering A PSA to credit the Risk Improvement Plan projects, it is expected that the overall Pickering A core damage frequency and large release frequency will be further reduced.

Whole-site PSA characterizes the overall risk of the nuclear facilities at the site, taking into account the benefits from the emergency mitigating equipment added as part of the Fukushima action plan. The scope of whole-site PSA includes the impacts from both internal and external hazards on all reactor units and on all other major radiological sources at the site such as the irradiated fuel bays.

OPG submitted a concept level whole-site PSA methodology in February 2014, which was reviewed and accepted by CNSC staff.

OPG submitted the Pickering whole-site PSA report in December 2017.

The Pickering whole-site PSA results were presented at the Commission Meeting on December 14th, 2017, in CMD 17-M64.1.

To summarize some of the information provided during that update, this slide shows the results

for the site-level aggregated core damage frequency and large release frequency.

The charts show that:

- aggregated site core damage frequency is below the unit-based core damage frequency safety goal of 1 in 10,000 years; and the

- aggregated site large release frequency is below the unit-based large release frequency safety goal of 1 in 100,000 years.

After OPG completes the Pickering A PSA update planned for the end of 2018, it is expected that the whole-site aggregated core damage frequency and large release frequency can be further reduced.

As a reminder, a whole-site PSA is not a requirement but is regarded as a complementary safety assessment for quantifying the overall nuclear site risk, with the acknowledgement that this is a new and not yet mature methodology. Both the CNSC and industry are following the international developments for improving the methodology.

CNSC staff are currently reviewing OPG's whole-site PSA submission and will provide the Commission with the assessment results and conclusions in the NPP Regulatory Oversight Report.

I will now discuss fitness for service of

fuel channels.

Managing the aging of major components is integral in ensuring safe long-term operations. As part of the Integrated Aging Management Program at Pickering, OPG has lifecycle management plans for all major components.

Life cycle management plans identify the activities that are undertaken to continuously assess component condition and their fitness for service. This includes activities such as evaluation of the component's aging degradation mechanisms, inspection methods and repairs or replacement strategies.

Of particular regulatory focus is the aging management of fuel channels, in light of OPG's request to operate Units 5 to 8 up to 295,000 effective full power hours. This request will be discussed in more detail later in this presentation.

In January of this year, CNSC staff provided a technical update to the Commission on fuel channel evaluations for Canadian nuclear power plants. CMD 18-M4, which is available on the CNSC website, specifically addressed the pressure tube fitness-for-service requirements and evaluation methodologies that are established in CSA standards.

CNSC staff continue to review the results from fuel channel inspections that occur routinely during

planned inspection outages. Inspection results to date confirm that the safety margins are being maintained. OPG's fuel channel lifecycle management plan identifies mitigating strategies should fitness-for-service assessments not be able to meet acceptance criteria up to the end of the evaluation period.

During the licence period, OPG continued to implement an effective radiation protection program to protect the health and safety of persons inside the facility and ensure that occupational and public exposures are below regulatory dose limits and maintained as low as reasonably achievable, or ALARA.

Pickering's rating in the Safety and Control Area declined from fully satisfactory to satisfactory in 2016. This was due to non-compliances related to the calibration and use of fixed and semi-portable gamma monitors. Despite those deficiencies, CNSC staff confirmed that OPG continued to ensure the protection of their workers through other protective measures such as routine surveys, work plans and dosimetry requirements. There were no exceedances of action levels or dose limits. As a result of CNSC enforcement actions, OPG took immediate measures to return to regulatory compliance. Staff will continue to follow up with OPG on remaining corrective actions through regular compliance

verification activities.

This slide presents the distribution of annual effective doses to all monitored persons at Pickering. The vertical red line on the far right of the slide indicates the annual regulatory dose limit of 50 mSv for nuclear energy workers. As you can see, in the current licence period, there were no radiation exposures at Pickering that exceeded the annual regulatory dose limit. The majority of doses received by workers at the facility are at or below 1 mSv, which is the regulatory limit for a member of the public.

CNSC staff are satisfied that the radiation protection program at Pickering is effective in maintaining doses below regulatory limits and ALARA, and protecting the health and safety of persons.

The CNSC conducts an Environmental Assessment under the *Nuclear Safety and Control Act* for all projects to ensure the protection of the environment and the health of persons. The environmental assessment report considers current operations and future activities.

The report was produced using a number of different sources, as shown in the blue boxes on this slide, including OPG Pickering environmental protection measures, the updated Environmental Risk Assessment and the Predictive Effects Assessment, results from CNSC's

Independent Environmental Monitoring Program and other regional monitoring, as well as the results of annual compliance reports and the preliminary decommissioning plan for Pickering. The CNSC Environmental Assessment Report is provided in Appendix F of CMD 18-H6.

CNSC staff's environmental assessment concluded that OPG has and will continue to make adequate provision for the protection of the environment and the health of persons.

In April 2017, OPG submitted an updated Environmental Risk Assessment for the Pickering site based on effluent and environmental monitoring data for the five-year period between 2011 and 2015. CNSC staff completed a detailed technical review of the 2017 ERA and found it to be consistent with the methodology of CSA Standard N288.6-12. CNSC staff concluded that OPG's ERA for Pickering provides a complete evaluation of all potential risks to human health and the environment associated with the facility operations, and that meaningful adverse ecological and human health effects due to releases to air and water from Pickering are unlikely.

CNSC staff's assessment of the ERA was also provided to the Commission in CMD 17-H5.B during the licence renewal hearing for the Pickering Waste Management Facility in 2017. Intervenors were also given an

opportunity to review the ERA during the Waste Management Facility licence renewal hearing process.

Since the submission of this current CMD, OPG has addressed CNSC's recommendations to validate several ERA conclusions and improve the quality of the next ERA update. This also includes the two specific items in the environmental assessment report related to thermal discharge matters.

The last area of regulatory focus that I will discuss is emergency preparedness.

OPG has effective emergency management and fire protection programs at Pickering that meet regulatory requirements. In addition, as shown on this slide, OPG undertook numerous activities during the current licence period to enhance emergency preparedness and response capabilities. OPG will also update its emergency response plans to reflect the changes in the Provincial Nuclear Emergency Response Plan.

CNSC staff conclude that OPG has sufficient provisions at Pickering for emergency preparedness and response.

On December 6th and 7th, 2017, OPG conducted a full-scale exercise at Pickering. Exercise Unified Control was designed to test the capacity of onsite groups and offsite agencies to respond to a nuclear

emergency at the Pickering site. The exercise simulated a severe accident and tested OPG's ability to respond to extreme events, including the use of the portable emergency mitigation equipment and connection points added to the plant post-Fukushima. Multiple external agencies, including the CNSC, participated in the exercise.

In addition to simulating CNSC response in case of a nuclear accident, CNSC staff conducted an inspection of the OPG performance. Inspection results confirmed OPG readiness to respond effectively to a serious accident.

The Commission heard earlier today from the Office of the Fire Marshal and Emergency Management about the new Provincial Nuclear Emergency Response Plan, or PNERP.

During the public consultation period, CNSC staff provided comments to the province, focusing on the PNERP's alignment with national and international standards. CNSC staff also modeled accident scenarios at Ontario nuclear facilities to support validation of the planning assumptions.

CNSC staff have confirmed that the updated PNERP conforms to both CSA N1600 and IAEA standards on emergency management.

When discussing the emergency response to

a nuclear accident, it's important to keep in mind that all practical steps are taken to prevent such accidents from ever happening, in other words, to make them extremely unlikely. The recent evaluations of the risk of severe core damage and large release frequencies were presented earlier when discussing the PSA results. These results demonstrate that the likelihood of a serious accident remains very low.

The reduction of the likelihood of severe accidents comes from several improvements, such as:

- completion of the Phase 1 and Phase 2 emergency mitigating equipment to supply emergency make-up water to the steam generators, heat transport systems and calandria vessels, thus ensuring continuous fuel cooling and monitoring;

- strengthening containment robustness by assuring provision of cooling water and power for containment air coolers;

- installed hydrogen passive autocatalytic recombiners in all Pickering units; and

- implementation of Severe Accident Management Guidelines to guide plant staff in preventing and mitigating accident progression to a severe accident.

The Integrated Implementation Plan commits OPG to implement additional safety enhancements that will

ensure capabilities exist to prevent an accident from progressing to a severe accident and to mitigate the consequences should a severe accident occur.

In this next series of slides, I will provide a summary of the following subjects:

- CNSC outreach, public consultation activities and Participant Funding Program information;

- First Nations and Métis engagement;

- a summary of the IAEA OSART Mission to Pickering;

- information about the Pickering *Fisheries Act* authorization; and

- an update on the Fukushima Action Plan.

The CNSC communicated to the public, stakeholders and Indigenous groups information regarding the regulatory process for the renewal of the Pickering licence. A variety of communication methods were used such as traditional news release, feature articles and social media. In addition, a CNSC Open House was held at the Pickering Community Centre on March 8th, 2018.

The Notice of Public Hearing and information about the Participant Funding Program was issued in September of 2017.

Participant funding allows for the provision of new, distinctive and valuable information to

the CNSC through informed and topic-specific interventions. Up to $100,000 in participant funding was made available.

A Funding Review Committee, independent from the CNSC, reviewed the funding applications and made recommendations.

The CMD reported that nine recipients had been awarded participant funding. Since then, an additional group has also been awarded funding. The ten recipients, including two First Nation communities, will be reviewing OPG's licence renewal application and other documents and preparing written and oral interventions for Part 2 of the Commission hearing.

As an agent of the Crown, the CNSC understands the importance of building relationships and engaging with Indigenous peoples in Canada.

The CNSC ensures that all its licensing decisions under the *Nuclear Safety and Control Act* uphold the honour of the Crown and consider any adverse impacts to potential or established Aboriginal and treaty rights.

CNSC staff have provided information to and engaged with interested First Nations and Métis, including meetings, to strengthen relationships and better understand issues and concerns related to the Pickering Nuclear Generating Station.

Issues and concerns raised to date include

impacts on localized fish populations; perceptions and impacts on harvesting; transportation and storage of waste; participation in the CNSC's Independent Environmental Monitoring Program; and understanding CNSC processes and practices.

All of the identified First Nations, First Nations regional representative bodies and Métis Nation of Ontario Region 8 have been encouraged to participate in the review process and in the public hearing to advise the Commission directly of any concerns they may have in relation to this licence application.

The CNSC has and will continue to meet with First Nations and Métis Nation of Ontario Region 8 to discuss any issues or concerns and maintain productive and respectful relationships.

At the request of the CNSC, in 2016 an IAEA Operational Safety Review Team, or OSART, conducted a 19-day mission to evaluate Pickering's operational safety performance against IAEA safety standards. OSART missions provide member states with the opportunity to share best practices and to support continuous improvements to the facility operations.

The 2016 OSART team concluded that management at Pickering Nuclear Generating Station is committed to improving the operational safety and

reliability of the plant. The team identified eight good practices, 11 suggestions and 10 recommendations.

To date, the OPG corrective action plan is approximately 85 percent complete. OPG is targeting to complete the remaining improvement actions by Quarter 2 of 2018, with the exception of the recommendation related to managing alcohol and drug testing of key staff in safety important roles. Independent of this recommendation, the Commission has published REGDOC-2.2.4, which is the Fitness for Duty Volume II on Managing Alcohol and Drug Use. OPG has submitted its implementation plan for this REGDOC as of the end of March.

With respect to the *Fisheries Act* authorization, prior to 2015 OPG had been working towards applying for a *Fisheries Act* authorization, after being informed by CNSC staff that Pickering operations would require one.

Following an episodic fish impingement event at Pickering in 2015, Fisheries and Oceans Canada, or DFO, issued a warning and determined that OPG must apply for an authorization under section 35(2) of the *Fisheries Act*.

OPG submitted their application to DFO in July of 2017 and the authorization was granted in January of 2018.

The authorization requires that OPG install the Fish Diversion System barrier net by May 1st and that it remain in place and functioning until November 1st of each year. As well, OPG is required to offset the residual harm caused to fish and fish habitats with offsetting measures.

As per the Memorandum of Understanding between the CNSC and DFO, CNSC is responsible to inform DFO of any non-compliance with the conditions of the authorization. The DFO authorization is valid until December of 2028.

Lastly, an update on the Fukushima Action Plan. All Fukushima action items have been closed for Pickering and one station-specific follow-up action remains open, as indicated in the CMD. OPG has recently requested closure of that station-specific action item and CNSC staff are reviewing the request.

In a complementary evaluation performed during the Periodic Safety Review, OPG reassessed Pickering Fukushima action items to determine if there were any impacts associated with plant operation to 2024 and whether the closure basis of Fukushima action items remains valid in the context of extended operation.

CNSC staff reviewed the results of OPG's reassessment and identified an additional gap that was

incorporated into the PSR Integrated Implementation Plan. The additional gap is related to the Fukushima action item for protection of containment integrity during severe accidents. The corresponding Integrated Implementation Plan actions will strengthen Pickering's provisions for controlled filtered venting.

As committed in the Integrated Implementation Plan, OPG will implement safety-enhancement modifications and design changes to protect containment integrity and ensure additional barriers exist to prevent the initiating event or accident from progressing to a severe accident and to mitigate the consequences if a severe accident occurs.

Specifically, OPG will:

- complete interconnections to make fire protection system water available to the steam generators, heat transport system and calandria in Units 1 and 4; and

- complete the necessary power and support service connections required to restore the functionality of the main volume vacuum pump.

Actions to provide emergency power and cooling water to the air cooling units in all reactor units as well as emergency power to the hydrogen igniters and Filtered Air Discharge System are already complete. CNSC staff will continue to confirm that procedures and

instructions are in place to operate existing venting provisions in case of a severe accident, such that the containment integrity is protected while minimizing the releases.

OPG has implemented and will continue to implement physical and programmatic changes that will further enhance safety by ensuring the existence of provisions to prevent an initiating event from progressing to a severe accident and ensuring availability of options for controlled filtered post-accident venting.

OPG will implement the majority of IIP modifications and provisions related to protecting containment integrity to ensure controlled filtered venting by December of 2018. The remaining two actions will be completed by December of 2020.

CNSC staff will update the Commission on the status of actions related to the protection of containment integrity as part of the annual Regulatory Oversight Report for Nuclear Power Plants.

I will now pass the presentation back to Dr. Viktorov to discuss OPG's request to operate Pickering Units 5 to 8 up to 295,000 effective full power hours.

MR. VIKTOROV: Thank you.

Once again it's Alex Viktorov, for the record.

Currently, Pickering station is approved to operate to 247,000 effective full power hours. OPG is seeking the Commission's approval to operate fuel channels of Pickering Units 5 to 8 up to 295,000 effective full power hours. This is the maximum value expected for any unit before the end of commercial operation. Units 1 and 4 had their pressure tubes replaced in the mid to late 1980s, therefore their accumulated effective full power hour values are significantly lower.

OPG has implemented measures to maintain fuel channel fitness for service as required by the licence and the Licence Conditions Handbook, including:

- execution of Periodic Inspections Program;

- consideration of internal and external operating experience; and

- implementation of outcomes from research and development projects.

The safe operation of fuel channels is demonstrated through assessments of the current and expected conditions based on an understanding of relevant degradation mechanisms. Research activities as well as inspection and maintenance programs provide data to periodically validate the input parameters for these assessments.

In January 2018, CNSC staff provided a technical update to the Commission on fuel channel evaluations. As discussed in CMD 18-M4, of all the stressors, the dominant contributor to risk of a fuel channel failure is deuterium ingress, which has a direct impact on the material fracture toughness. Deuterium ingress is expressed in terms of hydrogen equivalent concentration. Adequate fracture toughness is required to prevent a fuel channel from abrupt rupturing if a small crack develops during operation.

OPG, along with the other Canadian utilities, uses two fracture toughness models which are covering hydrogen equivalent concentration levels up to 120 parts per million. By the end of 2024, the lead unit at Pickering in terms of Deuterium ingress is estimated to reach hydrogen equivalent concentration levels below 100 parts per million.

In the next licence period OPG and the industry will continue research activities to enhance the current understanding of fracture toughness and other fuel channel aging-related degradation mechanisms. OPG will document these activities in a Pickering Fuel Channel Readiness Plan. The Readiness Plan will include research activities needed to enhance the fracture toughness models used to evaluate safety margins.

Information from the Fuel Channel Readiness Plan will be submitted annually for CNSC staff review.

CNSC staff expect that the ongoing and planned activities will continue demonstrating that the fuel channels are fit for service up to 295,000 effective full power hours.

Regardless of the maximum approved value of the effective full power hours, OPG must always demonstrate to CNSC staff that fuel channels are robust enough to continue safe operation of Pickering reactors.

Based on current hydrogen equivalent concentration predictions, Pickering fuel channels are not expected to reach 120 parts per million before the end of their target service life.

However, in the case that future predictions for any fuel channel exceed, or even approach, 120 parts per million before the end of commercial operations, CNSC staff recommends a specific licence condition 15.3 to be added to the licence. This licence condition will ensure OPG will maintain fracture toughness sufficient for safe operation and will assure the presence of adequate compliance verification criteria.

CNSC staff recommend that the Commission approve the operation of fuel channels of Pickering Units 5

to 8, up to a maximum of 295,000 effective full power hours.

I will now briefly discuss the proposed licence and Licence Conditions Handbook. Both these documents are included in CMD 18-H6. Please note that at this time these documents are considered drafts and may change prior to Part 2 of the Commission hearing.

The Power Reactor Operating Licence is a fundamentally important document as it establishes the licensing basis, or the set of conditions under which the applicant would be allowed to operate a facility. It also describes the specific activities which the licence holder is authorized to conduct, as well as the period during which such activities may be conducted.

The Licence Condition Handbook spells out specific criteria used by CNSC staff to verify that the conditions of the licence are being met. The existing Licence Condition Handbook for Pickering, as well as recently approved Licence Conditions Handbooks for other Canadian NPPs, were used to develop the proposed Licence Condition Handbook for Pickering.

In addition to requirements, the LCH provides guidance. Up-to-date CNSC regulatory requirements and Canadian standards are used for setting the regulatory requirements and expectations.

CNSC staff strives to use standardized wording of the licence conditions for all power reactor operating licences. This approach promotes consistency in licensing and compliance verification and helps achieve common interpretation of the licence requirements for all nuclear power plants.

The standardized licence conditions require the licensee to implement and maintain programs which govern safety-related activities in a systematic and comprehensive manner. In addition to the standard licence conditions, six site-specific conditions are proposed, which will be listed later on.

It is important to note that the licence does not prevent the licensee from modifying its operating conditions or limits. The changes with a neutral or positive impact on safety are considered to be within the licensing basis. The changes with negative safety implications shall be justified and approved by the Commission.

OPG has continuously demonstrated satisfactory performance and a commitment to seek safety improvements in all areas. Safety performance has been stable or improving during the current licence period.

Results of the periodic safety review demonstrate that the facility can be operated safely, and

adequate safety margins have been and would be maintained. The Integrated Implementation Plan identified further safety enhancements to be carried out in the coming years.

On this basis CNSC staff are recommending a 10-year licence period, as requested by the licence applicant. However, within this 10-year period, OPG is only authorized to operate the power reactor on-power until December 31, 2024. Should circumstances arise requiring OPG to operate any reactor unit past this date, OPG is required to notify the CNSC and obtain Commission approval, as this will be a change to the licensing basis.

Safety performance of Pickering will continue to be reported annually to the Commission. This practice provides the public with frequent updates regarding the outcomes of CNSC regulatory oversight activities.

At this point, I would like to provide evidence that the proposed draft Licence and Licence Conditions Handbook allow for adequate regulatory oversight. There will be no reduction in regulatory requirements compared to the current licence.

There are currently a total of 25 CNSC regulatory documents referenced in the draft Licence Conditions Handbook as compliance verification criteria. Twenty-two documents are referenced in the current licence

as well, but four of those in the new Licence Conditions Handbook will be recently revised versions. Three new CNSC regulatory documents have been added as updated requirements in the proposed Licence Conditions Handbook.

Earlier versions of regulatory documents are being replaced by newer versions in the spirit of continuous improvement. The full list of the regulatory documents can be found in the appendix section of the Licence Conditions Handbook.

In addition to the CNSC regulatory documents, there are currently a total of 25 CSA standards referenced in the Licence Conditions Handbook. Fourteen of those are listed in the current licence, and five of them were updated. Eleven new standards have been added in the proposed Licence Conditions Handbook.

In addition to regulatory documents and CSA standards setting requirements, a further number of documents are identified as guidance, often with the intent of transitioning such documents to requirements following an appropriate period for implementation.

I will now pass this presentation back to Mr. Frappier for CNSC staff’s conclusions and recommendations.

MR. FRAPPIER: Thank you. For the record, it's Gerry Frappier.

Commission member document, CMD 18-H6, describes in detail CNSC staff's conclusions and recommendations for the Commission's decision regarding the licence renewal application. In the next couple of slides I will summarize the CNSC staff position regarding OPG's application for renewal of the Pickering operating licence.

Based on the assessment of OPG's application and past safety performance, CNSC staff conclude that, as per section 24(4) of the *Nuclear Safety and Control Act,* OPG is qualified to carry out the activities authorized by the licence, and in carrying out the licensed activities OPG has made and will continue to make adequate provision for the protection of the environment, the health and safety of persons, the maintenance of national security, and measures required to implement international obligations to which Canada has agreed.

CNSC staff recommend that the Commission accept CNSC staff's conclusions presented in CMD 18-H6 and exercise its authority under the *Nuclear Safety and Control Act* to renew the licence to authorize OPG to carry out the activities listed in Part IV of the licence from September 1, 2018 to August 31st, 2028.

It is also recommended that the Commission authorize OPG to operate fuel channels of Pickering Units 5

to 8, up to a maximum of 295,000 effective full power hours.

With respect to Licence Condition 3.2, CNSC staff recommend the Commission delegate the authority for consent to restart a reactor after a serious process failure to the following CNSC staff: The Director of Pickering Regulatory Program Division; the Director General, Directorate of Power Reactor Regulations; the Executive Vice-President and Chief Regulatory Operations Officer of the Regulatory Operations Branch. This delegation of authority would be unchanged from the current licence.

Furthermore, CNSC staff recommend that the Commission accept the standardized and station-specific conditions included in the proposed licence. The six licence conditions specific to Pickering Nuclear Generating Station require OPG to:

One, implement the Integrated Implementation Plan;

Two, maintain Units 2 and 3 in a safe storage state;

Three, maintain pressure tube fracture toughness sufficient for safe operation;

Four, implement and maintain plans for the end of commercial operation;

Five, implement and maintain a Cobalt-60 program and;

Six, limit the activities of import and export of nuclear substances to those occurring as contaminants in laundry, packaging, shielding or equipment.

Thank you, Mr. President, and members of the Commission. We are prepared to any questions you may have regarding this presentation.

THE PRESIDENT: Thank you.

So colleagues, what I propose we do is we go the normal one question, per commissionaire for as many rounds as we need.

Let me start with Ms Velshi.

MEMBER VELSHI: Thank you, Mr. President.

If we can go to Slide 43 of OPG's presentation, I would like to start off, first of all, by commending Pickering nuclear station for its performance. Your comments that the plant is safer today than it has ever been, and more importantly that your last day will be your very best were extremely, extremely reassuring, and I commend you for your performance.

On that last slide, you painted a picture of the turbine hull and the Ontario flag at one end, and the Canadian flag at the other end. If you look at the slide, you've got a picture of your team. I just wanted to

comment it would be kind of nice to have some women in that group as well because you went to extreme lengths to present an image and that's missing in a big way.

So to my first question then, and it is around future and all fitness for service, when is the first unit expected to hit the current limit of 247,000 effective full power hours?

MR. LOCKWOOD: Randy Lockwood, for the record.

I am going to ask Kathy Charette to answer your question here directly. You asked specifically when we would reach 247,000 effective full power hours. While Kathy positions, your comment is well accepted about the photo at the end of the slide.

I accept that, but I also would point out, and I'm very proud that Stephanie Smith, plant manager first, right. I'm very proud of that.

MEMBER VELSHI: As you should be.

--- Laughter / Rires

MEMBER VELSHI: As you should be, yes.

MS CHARETTE: Thank you, Randy.

I'm Kathy Charette. My name is Kathy Charette. I am the Senior Manager of the Fuel Channels, Life Confirmation Project, for the record.

The first unit, well as you know -- well,

as you have been told, and as I explained to you, we do frequent inspections of our fuel channels. Based on the information that we have, we are able to predict the rates of our aging mechanisms.

For our Unit 6, the Unit 6 reactor is the first that is expected to reach to 247,000 EFPH. That will occur around mid-2019, according to our current projections.

MEMBER VELSHI: Thank you. And when your lead unit reaches the 295,000 effective full power hours, what does your model say your expected hydrogen equivalent level is expected to be at that time?

MS CHARETTE: Kathy Charette, for the record.

Our models predict that we will below 100,000 effective full power hours. I mean --

MEMBER VELSHI: Parts per million.

MS CHARETTE: -- parts per million, thank you.

MEMBER VELSHI: Right.

MS CHARETTE: Of hydrogen concentration at that time.

MEMBER VELSHI: Right. So I have seen that. I wanted to know --

MS CHARETTE: The number?

MEMBER VELSHI: -- you say there is an adequate safety margin. Is there a range of numbers? Is it like 100, is it like 50? What is the expected parts per million?

MS CHARETTE: The deterministic method, which is a bounding assessment that we have, currently it predicts 95 PPM on the highest unit. That's, you know a 25 PPM margin against the --

MEMBER VELSHI: Excellent.

MS CHARETTE: -- limit.

MEMBER VELSHI: And I recall very clearly about five years ago when you folks were here for your licence renewal and asking the Commission to extend, allowing larger effective full power hours.

And you may recall there was a question asked: So are you likely to come back in front of us and ask us for additional allowance on the EFPH? You were fairly adamant "no". So here you are today.

And so my question now is because, you know, 10 years is a long period from now and lots can change. What -- if your hydrogen level was 120 parts per million, what would be the associated equivalent, effective full power hours? What does your model say it would be?

MS CHARETTE: It's Kathy Charette, for the record.

I don't have that information with me.

MEMBER VELSHI: Is that number available? I mean, clearly the regular -- the CNSC staff are prepared in anticipation of an extension beyond 10 years to operate the units. I just wondered if you have that information. Is that -- I know you don't have it now, but is that information available?

MS CHARETTE: Kathy Charette, for the record.

I will find out if it's available and get back to you.

MEMBER VELSHI: Thank you. That will be helpful. Thank you.

MR. LOCKWOOD: Randy Lockwood, for the record.

While Kathy is here, it might be important to set the context around 295,000 equivalent full power hours and the assumptions that had gone into that are conservative from the perspective that it would be the best operating set of conditions we could get, right. The reality is it will probably fall somewhere significantly less than that between now and the end of 2024.

MEMBER VELSHI: Thank you.

Staff, do you have any comments on that?

MR. FRAPPIER: Gerry Frappier, for the

record.

So first, I want to be clear that we do not anticipate that they are going to operate beyond December 2024.

We wanted to be clear for the Commission that there is provisions within the licence that will require them to come back here if they were ever to think of that, but we have no indication and we have no desire, one way or the other, for them to go beyond that term.

As discussed, we have been looking at what we are confident with from a safety margins perspective. The 295,000 full power hours and the fact that even at that time, they will be below the 100 parts per million gives staff a lot of confidence that that's not going to be an issue.

THE PRESIDENT: Thank you.

Dr. Demeter...?

MEMBER DEMETER: Thank you. Just first a comment, and then a question for OPG.

Just to remind you again, in your documentation you're still referring to dose as "REM" and, like it said in the last with the Bruce hearing, it's confusing enough to try to relate these risks with -- but I was really happy in your supplement -- Figure 8, in your supplement of your CMD, you had a title that looked at

person Sieverts, but the legend on the actual graph was person REMs. So you're getting there halfway on that one, just a note.

So the question I have -- first, though, I'll get staff to clarify to make sure I am interpreting the table right. It is Table 8 on your CMD, page 67 -- yeah, staff CMD. Yeah.

So I am looking at the last row of that table which says "deferrals of preventive maintenance" and the trend is average work orders -- 2015 is 120, 2016 is 110, and for the first two quarters of 2017 it's 92.

So if I do simple linear analyzation that would equal 184, so the trend would not be down. I just want to make sure I'm interpreting the table correctly and then I'll ask my question again.

MR. FRAPPIER: Gerry Frappier, for the record.

I would ask Eric Lemoine to please come and explain the table.

--- Pause

MR. LEMOINE: For the record, my name is Eric Lemoine. I am the Director of the Systems Engineering Division.

So on Table 8 you are referring to the deferrals of preventive maintenance rows, and the average

work orders: In 2015, 120; 2016, 110, and the average work orders for Q1 and Q2 as 92.

MEMBER DEMETER: Correct.

MR. LEMOINE: So this is a per unit, per quarter average. So this is taking into account the full year.

MEMBER DEMETER: So in 2015, the 120 is a per unit, per quarter average, not the whole year?

MR. LEMOINE: Sorry. No, that is the average for the year per quarter. I have the actual graph here in front of me which we graph per quarter and then we apply a trend line, which it does demonstrate.

But I will ask Yong Chang Liu to perhaps explain it a little bit more clearly than myself.

MR. YONG CHANG LIU: Yong Chang Liu, Technical Specialist, for the record.

The date is per -- you know because we get a quarterly date, so Q1 to Q4 average per unit. So we get an annual -- basically this is get an annual per unit, quarterly based data, because we have to -- for example, Q1 plus Q2, plus Q3, plus Q4, plus Q5, divided by four because this is a quarterly base per unit.

MEMBER DEMETER: Okay. So that someone reading the table, they would think this would be the absolute number of work orders for 2015.

So I don't need to ask my question anymore because the way I was reading it is that the number of work orders if it was annualized, was going up considerably and the trend -- but it's really confusing, I have to say, because there is no indication this is a per unit, per quarter average. It's an absolute number.

THE PRESIDENT: So can I piggyback on that question? I have two questions about that.

First of all, why is that good when the industry average is way below? That's number one.

Number two, if you go down that particular page there is a review that I assume was done by CNSC and actually come to a conclusion that lack of adequate resources were found for maintenance.

Now, for some of you, who have been with me for a long time, know that's my favourite indicator "maintenance". So how is that -- how is that even possible for Pickering that there will be a lack of resources for maintenance?

So maybe staff and then maybe OPG.

MR. LEMOINE: For the record, my name is Eric Lemoine.

So we are not saying that it's good in our document here. We are saying it has a low safety significance. So one thing that is important to note here

is the trend has actually been trending down from year to year over the licence period. However, it is as you said, above the industry average. It is also important to note that the industry average has also been trending down over the licensing period for Pickering as well.

And this was one of the reasons why we initiated a desktop review and came to that conclusion that you have already referenced, later on in the document.

THE PRESIDENT: I thought we heard that OPG, in fact, were bragging about Pickering now being at the top of their game. So maybe somebody can explain this.

MR. LOCKWOOD: All right. Randy Lockwood, for the record.

Not speaking directly to the table there, but to put a couple of things in context and then I am going to ask Jason Wight to also talk a little bit about where we are with our value-based maintenance program and how that's leading to working on the right equipment at the right time and, in turn, driving up equipment reliability which is in turn meaning that the units are running better and resulting in less challenges to the operators, and then ultimately that's a safer plan. So to put this back in perspective here, it's about prioritization and working on the highest priority first.

So to put this in perspective then, again

for the end of 2016, our target for corrective backlog, CCs and CNs -- in the business that's what they are called -- was 100 and we ended the year. We had a target of 55, and we ended the year at 116 per unit.

Where we sit at the end of 2017, we set a target for ourself of 28 per unit, at the end of 2017 we were around 26 work orders per unit. We exceeded our target. That's still not industry best, but of that category the most important, the critical equipment, the CCs, we strove to drive that down to zero, and we have done that. Through the first quarter of 2018, where that sits, not per unit, across the entire plant, is zero. Okay?

Where are we now? The next part is to work down in the next priority, and that's the deficient category, DCs. We set a target for ourself of 15 work orders per unit by the end of the year. We will get there. I'm quite certain of that. We are somewhere around 19 right now, okay, so we will continue to drive that down toward industry best.

The way that we see this is we'll probably be at industry best somewhere in 2019 in those categories that I mentioned. That is what we mean by, "that is good" and "it's a favourable trend".

There's many things that go into the Equipment Reliability Index, including forced loss rate,

including this backlog that I'm discussing, and also the PMs that we were also discussing here.

Our PM backlog, we are driving that down as well, and we're looking at a different maintenance strategy, value-based maintenance, which has been adopted in certain parts of the industry, to leads that we're working on, the right equipment at the right time, making the plant more reliable.

At that point, I think it's a good time to turn over to Jason to talk about PBM.

MR. WIGHT: Jason Wight, Director of Engineering, for the record.

The value-based maintenance program that Randy was talking about is exactly what he just discussed. It's the right work at the right time, with the understanding that we can't fix everything in the power plant at once, we have to focus on the critical work.

It's also employing advanced technology. We have a monitoring and diagnostic centre that were just employed to use advanced pattern recognition and machine learning to help us diagnose and determine when equipment should be maintained, all part of the value-based maintenance program.

With regard to deferrals, I saw the table, and there is some context that needs to be set there. If

you look at our critical one and critical two pieces of equipment in deferrals, it's gone way down in the past six months to a year, so a very positive trend overall in critical work deferrals. if you look at the bulk, it's a bit different, because you start to focus on other non-critical work as well, so the value-based maintenance project is all about that, ensuring we understand what the critical work is, we get the right maintenance strategy, we put advanced diagnostic tools, monitoring, using some of the equipment and technology that the industry is using, such as advanced pattern recognition, APR, and machine learning, determining the right work at the right time, and keep reliable.

If you look at our Equipment Reliability Index, that's a good indication of how reliable we are, and also the effectiveness of value-based maintenance. Our Equipment Reliability Index at Pickering has been the best it's ever been, and it's expected to improve throughout the next quarter of this year, and at the end of 2018, so some great progress by the team in general using that technique.

MEMBER DEMETER: Thank you. That helps me understand the table and some of the background.

I have to say from other industries, mine, health care, whenever there is a funding crunch it's usually the preventive stuff that falls off, and if you are

continually behind industry standard in that it appears, on the surface of it, that you're getting to the critical, the must-do stuff, and it's a resource issue to not get to the preventative stuff, which is not necessarily mission critical at the time but better for the long term. There is a perception that being behind in that gives an impression of potentially a resource issue, because you are getting to the critical stuff. The preventative non-critical stuff takes funding.

THE PRESIDENT: Let me jump in further on this.

The way I interpreted this table is you guys were to close down in 2020, so my fear is always as you go closer and closer to closing down, less and less money gets invested in backlog maintenance, so if you want to be the best on the last day, somebody gave a quote, "To be as good at the last minute as now", you can't allow the backlog to go below industry.

Am I misreading this or you got there because you're about to close down?

MR. LOCKWOOD: Randy Lockwood, for the record.

Many different messages there.

First of all, speaking to the Commissioner's point, this is not a case where we're

cutting funding and trying to reduce staff or something like that. This is not a case of that. Right?

Speaking to President Binder, point accepted. We have a backlog. That backlog developed probably for a number of reasons, including the maintenance strategy that was in place, and, hence, why we're looking at and revising the maintenance strategy, to ensure that we're working on the most critical equipment at the right time. It's not a case of we're short resources, and we decided we'd cut the resources, and we decided not to work in certain areas, no, it's going at the backlog in the sense of prioritizing what is important first, working it down and working onto the next priority, making sure staff understand that's what we're doing, and engaging them and aligning them around that. That's where we've had success. We've been very clear to staff it's to drive the backlog down on critical equipment, and that will in turn make the plant run and operate better, and like we said earlier reduce the challenges for the operators, which results in a safer Pickering. That's our focus.

THE PRESIDENT: Thank you.

Mr. Berube.

MEMBER BERUBE: To change direction just a little bit here away from maintenance provisions and into something I think is a little interesting to me anyhow, it

might be interesting to the public as well, I'm looking at the PSA, the whole-site PSA calculations. You're saying at this point that there's no requirement for that, but how confident are you that the whole-site PSA model is adequate and accurate, and that the results are indicative of what we actually expect?

Then, staff, if you could basically follow up with that, your ideas on that as well?

MR. MANLEY: Robin Manley, for the record. I'm the Vice President Nuclear Regulatory Affairs.

I'm going to ask Jack Vecchiarelli to provide a sort of technical commentary on the whole-site PSA with respect to its accuracy.

I would say that this is a first-of-a-kind work and industry-leading, so there's always opportunity to learn and grow down the road, but we're very proud of the work that we've accomplished, and we think it's a very good indication of the safety of the plant.

Jack, over to you.

MR. VECCHIARELLI: Jack Vecchiarelli, for the record. I'm the manager of Pickering relicensing, and I'm also the chairman of the steering committee for the industry joint project on whole-site PSA.

The question was around our confidence in the whole-site PSA model. I would say that we're very

confident in its adequacy, its ability to provide a good indication of the whole-site risk. That's based on the fact that the approach used for the Pickering whole-site PSA is to leverage existing PSA work, which is per unit based, but carefully analyzing the results to account for single unit and multi-unit scenarios, properly accounting for that, so those underlying PSAs which feed into the whole-site PSA are based on well-established approaches, guidelines used by OPG, methodologies accepted by CNSC staff, consistent with industry best practice, and IAEA and ASME standards, so these are very well-established, mature PSAs that are underlying the whole-site PSA modelling approach.

To be careful also, we have to remember that a PSA is not a predictor, so the adequacy of it, I would say, is in terms of its thoroughness, comprehensiveness, and usefulness to be able to get an indication of site risk, and it does do that.

MR. FRAPPIER: Gerry Frappier, for the record.

From staff's perspective, we view this as very much a leading internationally new sort of area, if you like, of use of PSA. It was very much driven by the Commission over the past few years, trying to look at a reasonable answer to a simple question, which is: if this

is the risk of one unit, and here I am with six units, plus a fuel bay, plus a waste facility, now what's the risk? I think the whole-site PSA is starting to really pull that together so that we can say all of those components are in it. I think, as Mr. Vecchiarelli just mentioned, they're certainly based on rock solid PSAs that are done at the unit, so we know the underlying data is strong.

How things have been combined seem to be very reasonable, but to be honest, as a regulator, we're also working with our international colleagues, who are also quite interested in this. There are a lot of sites around the world where there's multiple units, and they are also doing reviews, both of the methodology here and looking at other approaches, so I think we're going to learn a bit from that over the next year or two and see whether there's improvements to the methodology we'd like to see, but I would say as of right now it's certainly an example, perhaps the best example that there is in the world, of how to do this.

THE PRESIDENT: Just so we understand, slide 17, from OPG, was really a very instructive slide. It shows before Fukushima, after Fukushima and now, with all the EMEs.

I understand that staff is still reviewing the whole site, and I thought I understood that your review

will not be available for part two, because OPG, as far as I understand, now claim they've done the work and staff have not accepted it as of yet.

MR. FRAPPIER: Gerry Frappier, for the record.

On your slide 17 -- sorry, not your slide, but OPG's slide 17 -- I think what it's really indicating is a progression of a reduction in risk that has occurred over the past few years because of the implementation of the Fukushima improvements. We would certainly concur with that, and that the site has got a lower-risk profile now than it did in the past.

With respect to the actual PSAs, the requirement for a rock-solid PSA, if you like, were met by OPG back with their submission in 2014. However, obviously they've done changes to the plant, and as we would expect, they're updating those PSAs as they make changes to the plant, and you're seeing some of the improvements that those changes are making with respect to the risk profile.

From our perspective, we are reviewing -- there was a submission -- and perhaps I'd ask Ms Yolande Akl to come up and give the details -- there was a submission since 2014, both for Pickering B and for Pickering A. Those are in the process of being reviewed, and I'd ask Yolande Akl for the detailed scheduling of

that.

MS AKL: Yolande Akl, for the record.

I just want to start by giving a little bit of context about the modelling and how strong is the modelling.

When we accepted in 2014 the basis PSA, the PSA is required, by our regulatory document S-294, to be as built and as operated, so the modelling is already for a single unit as built and as operated.

OPG has submitted to us a methodology in 2014 that we reviewed, and staff accepted the methodology. Since then, we were following that progress into doing the whole-site PSA based on this concept methodology, and we have been reviewing the submission since 2014 -- sorry, since 2017, end of '17, when they gave the submission, so in general we agreed with the way they approached the methodology in removing the double counting. That was one of the hurdles of the one unit and the six units, and the four units, and so on.

In general, we agree with the approach that OPG has taken in providing this PSA, and working on this PSA. We will be completing very soon our review, probably before Day Two, Mr. President, so we will have these results ready and communicated to OPG.

THE PRESIDENT: Thank you.

I should have mentioned that we should not lose the opportunity -- we have representatives from Environment Canada, Health Canada, and Fisheries and Oceans available for colleagues to ask questions. I'm told that the Office of the Fire Marshal is to leave by four o'clock, so if anybody wants to engage on some emergency management questions, now is the time, before they disappear on us.

Next is Ms Penney.

MEMBER PENNEY: Thank you.

I just want to say I'm going back to fuel channels and aging.

With respect to fuel channels, two tools that were provided that were very useful and instructive in me trying to understand the challenges around this, one was OPG's Addendum B, that was quite good, and the other was CNSC staff's presentation of January that I watched online. While I'm not an engineer, I did find them helpful.

I had a question. I'm looking at OPG's slide 27 about fuel channels. It says, on the bottom in blue, "fuel channel aging is understood and well managed." Unless I've missed something, I do think there's still some research involved in the models that are used. In fact, on that slide it says, "supported by research and development." My question really to OPG, and then to CNSC staff, is how confident we are about fuel channel aging

being understood and well managed, and how much of a safety margin are we talking about?

MR. LOCKWOOD: Randy Lockwood, for the record.

I'm going to ask Kathy Charette to speak to the details about our confidence, why we are confident, and what goes into the fuel channel aging management program.

While she is getting into position, a couple of factors to consider here is we have a lot of experience with fuel channels in the CANDU industry, throughout the industry and within OPG as well. It's a well-managed and seasoned program. There's R&D that does into that to make sure that we understand the aging mechanisms.

As well, our aging management program and our life-cycle management programs, specifically for major components like fuel channels, heaters and boilers, complies with regulatory standards, regulatory documents, and associated CSA standards.

I would also like to point out to the Commission that we just recently, as part of the IIP action, submitted our fuel channel readiness plan that, like I said, was an action item coming out of the IIP.

Before I turn it over to Kathy, there was

an earlier question asking when we would reach 120 ppm hydrogen would it affect full-power hours. Although that's not a fitness for service limit, by our assessment it would be beyond 350,000 equivalent full-power hours. I hope that answers your question.

Kathy.

MS CHARETTE: Thanks, Randy.

Kathy Charette, for the record.

I'll just describe OPG's life-cycle management process for fuel channels, just to give you the information that you're seeking.

OPG's program for managing the aging of our fuel channels is mature and robust, and it is supported by a long history of OPG-funded research and development activities to improve our understanding of active fuel channel degradation mechanisms. Our fuel channel aging management program complies with regulatory requirements, as Randy said, which are consistent with international standards, and it is implemented through the fuel channel's life-cycle management plan in order to demonstrate ongoing fitness for service to the target end of life of our operating units.

The fuel channel aging management program uses a "Plan-Do-Check-Act" model, which is depicted in Figure B.4, that's Bravo point four, on page 102 of OPG's

written submission. The program consists of five major elements, which include operating guidelines to ensure that our fuel channels are operated in a manner that mitigates the effects of aging, rigorous in-service inspections, and periodically removing pressure tubes from our reactors for examination and destructive laboratory tests to monitor condition, maintenance activities to manage degradation mechanisms where required, engineering assessments to confirm that fitness for service requirements and CSA standards are satisfied, and an extensive research and development program to improve our understanding of active degradation mechanisms, extend the validity of existing models, and either develop or improve models and methodologies for the ongoing assessment of this fitness for service.

The life-cycle management plan describes the inspections and other activities that are required to maintain fitness for service, and the plan is adjusted as needed, based on learnings acquired from inspections, internal or external operating experience, and research and development outcomes.

OPG and our partners Bruce Power and Canadian Nuclear Laboratories initiated the joint fuel channel life management project in 2009 to address end-of-life aging concerns in order to extend the knowledge

of fuel channel degradation mechanisms beyond our current operating period, and to develop improved methodologies and analytical tools to support the assessment of fitness for service late in life.

Outcomes from the fuel channel life management project support fitness for service assessment of the Pickering fuel channels in key areas, such as fracture toughness, and annulus spacer integrity.

OPG and our project partners have shared key outcomes from the fuel channel life management project with industry through the Canadian Standards Association's standards. A major outcome from the fuel channel life management project is the issuance of two pressure tube models -- fracture toughness models, which were adopted into CSA Standard N285.8, to replace a model that could not be used for assessing fitness for service during late life operation. Work on the fuel channel life management project to further develop these fracture toughness models will continue into the next licence period.

OPG would like to point out that our end-of-life predictions for hydrogen concentration in the Pickering fuel channels are bound by the fracture toughness models and the CSA standard, and considerable margin, as we just discussed, to the licence condition handbook limits that CNSC staff has proposed in their CMD exists for all

the Pickering units.

Based on the inspections, assessments, R&D work completed to date, confirmatory actions in the life management plan for a strengthened service, and the availability of mitigating actions where required, OPG is confident that Pickering fuel channels will continue to operate safely to 2024.

Does that answer your question? It's a long answer.

MEMBER PENNEY: It was good information. I heard you say you were certain and confident. I guess is there anything -- how certain, how confident? What safety margins are there?

MS CHARETTE: Kathy Charette, for the record.

MEMBER PENNEY: Maybe I'm asking the wrong question.

MS CHARETTE: I'll try to answer it.

Kathy Charette, for the record.

We have actually gone through unit by unit, degradation mechanism by degradation mechanism, evaluated the degradation rates, and projected out to end of life, where we stand, and we have confirmed that we do have adequate margin to end of life on all degradation mechanisms based on those assessments, so we are confident

in our assessment because we have a deep understanding of the degradation rates of the mechanisms in our reactors right now.

THE PRESIDENT: And you will report annually on any progress in the deterioration, and we can get actual data, together with some of the research that's still going to be, if I understand correctly, on burst possibilities and the whole leak before break research is still going on; right? So, we'll get annual update on that.

MS CHARETTE: Kathy Charette, for the record.

Yeah, we're continuing to do our annual updates. We're also required to submit semi-annual updates on the fracture toughness model developments as well. So, those activities will continue.

MEMBER PENNEY: Are we able to hear from staff on this, on their confidence and certainty?

MR. FRAPPIER: Gerry Frappier, for the record.

And I'll ask Glen McDougall to come up and help me with the answer here.

But just to your question, of course, fuel channels are the heart of the reactor, have been of great interest since day one and there's been lots of different

research going on with respect to how they evolve, how they degrade, how they age, if you like, and we require that research to continue.

So, the fact is there's research going on. It's not a surprise, it's actually something we want because we want to always be looking over the horizon because new years are coming, pressure tubes are aging.

I think what we're saying here is, there is a good understanding of what the degradation mechanisms are. We do understand how pressure tubes age.

And while that's interesting, from a regulator's perspective really the key thing is, do we have margins to safety from the perspective of leaks, leak before break as the President mentioned, and other aspects which Mr. McDougall will mention in a minute.

I think what we can say with confidence is, we understand where the pressure tubes are, where the fuel channels are right now. We want to have research ongoing since there's --not just at Pickering, but other stations as well that are operating longer and longer, and so we're going to want to have continual research. And as the fuel channels age, the degradation mechanism that's the most important has evolved as well.

And so, as OPG mentioned, now there's a lot of emphasis on fracture toughness where that now seems

to be the aging process that's turning into being the key parameter. And so, industry is looking to develop better modelling capabilities in those areas.

But from the regulator's perspective, perhaps I'll ask Mr. McDougall to explain how we get our confidence.

MR. McDOUGALL: Glen McDougall, for the record.

So, the first question had to do with how certain staff are about the available information and the prognosis for the future.

As Mr. Lockwood and Ms Charette said, OPG relies heavily on a life cycle management plan for their fuel channels and that integrates a number of different things that we pay special attention to as staff. It integrates R&D, not just R&D that OPG is doing itself, but across the industry. It pays special attention to operating experience from other plants because Pickering units don't always suffer the same problems as some of their competitors do.

And they also pay a lot of attention to periodic inspection results of their own fuel channels. So, they're keeping track of what they know and some emerging issues that might be coming out that they might need some R&D or they might wish to do additional

inspections to try and keep track of that.

So, the life cycle management plan pulls all of that information together in one place and becomes the basis for OPG to plan its future inspections of its fuel channels.

The plan does that for all the different degradation mechanisms and for many of those mechanisms, the presentation that staff gave to the Commission in January showed that in terms of the level of understanding and the amount of uncertainty that remains with those mechanisms, staff's assessment is that industry is in a very good position to be making predictions about the future of those.

As Mr. Frappier said, one of staff's key focal points right now are issues which only really come into effect towards the end of life of fuel channels, when the hydrogen levels get high enough. And the key one, as noted in staff's CMD, was fracture toughness. And, in terms of that, industry has an extensive R&D program planned to look into uncertainties in the prediction of fracture toughness.

But one of the ways that staff and industry keep track of those uncertainties, or try to get a handle on them, is through the CSA standards which require routine removal of pressure tubes for destructive

examination because for fracture toughness that's the only way that you can actually make a measurement and sort of take a temperature, if you like, to find out the status of toughness in the tubes.

Taking up your second question on margins. Canadian licensees do two types of assessments. They do one type of assessment for pressure tubes that are inspected, those are the so-called fitness-for-service assessments.

And OPG consistently meets CSA acceptance criteria for all of those assessments that they perform and staff has no concerns about the margins that are applied, they are industry standard margins.

In terms of pressure tubes that have yet to be inspected, licensees perform risk assessments. And the two most common kinds President Binder mentioned are leak before break and fracture protection assessments. Both of them rely on predictions of fracture toughness.

At the current time, Pickering has done assessments of those that carry on into the near future and both of them meet CSA acceptance criteria, again, which take into account what the industry considers to be acceptable margins.

THE PRESIDENT: Okay, thank you. Dr. Lacroix?

MEMBER LACROIX: Well, thank you.

I would like to push a little further on the question of Kathy. I think you've already answered half of my question, but this is the second half.

Thus far I've heard rate of aging and aging are degradation and I would like you guys to be more accurate on your terms. And my question is addressed first to OPG, next to the staff, and of -- watch my hands -- unfortunately, there's no blackboard over here, so I would have plotted on the blackboard.

But if you plot -- this is the origin, the origin is the microphone, okay. On the vertical axis -- if you put the degradation and on the vertical axis you put time, my question -- there are many aspects to my question -- is the degradation resulting from hydrogen ingress a linear function of time?

Is the degradation of pressure tubes caused by the high temperature, high pressure and high neutron flux a linear function of time?

If it is, will I feel confident that you guys know perfectly well what you're doing? What if it is not, then I'm not so sure.

Also, these degradation mechanisms occur at the same time. Are they coupled, are they independent? If they are coupled, then is it possible that one phenomena

exacerbates the other one and vice versa? In other words, are we facing a phenomena that we may call a positive feedback effect?

Now, maybe this effect has escaped you guys because you have been operating the reactors for less than 250 equivalent full-time power hours, how you call it, full power hours, but above this limit could it be possible that we enter a zone where the relation as degradation as a function of time becomes a more complicated function of time? And could it be a step function?

These are the questions that I'm asking myself. So, I would like to hear from OPG and next from the staff.

MR. LOCKWOOD: All right. Randy Lockwood, for the record.

I appreciate your question. Many aspects to your question to answer, all right. So, I'll turn it back to Kathy to start and then we'll probably call others to the mic to answer different parts and try and tie it together, all right.

Okay. Kathy?

MS CHARETTE: Okay. Thanks, Randy.

I'm actually just going to turn it straight over to Sara on this one. We've been giving each other the eye here, so she's ready for it, I think.

MS IRVINE: Hi, I'm Sara Irvine, I'm Manager of Strategic Planning for the Fuel Channel Life Confirmation Project with OPG.

So, as I understand it, the question is, is the rate of degradation associated with hydrogen concentration linear or could it accelerate; is that correct?

MEMBER LACROIX: Sorry. The rate of degradation is the slope.

MS IRVINE: M'hmm.

MEMBER LACROIX: So, if you have a linear function it's a constant rate and it's easy to extrapolate above 250 hours. So, the rate is the slope, the degradation is the function, all right.

MS IRVINE: So, Sara Irvine, for the record.

So, right now the Pickering reactors are operating in a regime that is below a condition that we call terminal solid solubility dissolution of hydrogen in the tubes.

The rate right now of hydrogen ingress is steady and it's understood. Now, there is a condition called, again, TSSD above which we have hydrides solid in solution.

Since we're not there yet, we have

research going on right now to understand what this degradation, if it might accelerate in the future. Early results show that is not a big effect. So, that is our understanding.

Now, having said that, at December, 2024 we duly expect a very small population of tubes to be beyond that condition. So, while it's acknowledged that it could happen, it is a very small impact on the lead, the uptake unit.

MEMBER LACROIX: Thank you.

MS CHARETTE: Kathy Charette, for the record.

Just to add to that, I just wanted to also explain that all this research that we're doing it actually is -- we're doing it for the purpose of actually looking ahead, it's not just to see where we are now. We are actually doing research at conditions to reflect end of life conditions.

So, for example, when we do our burst tests we will take material out of the reactor, we will artificially introduce additional hydrides in it, so we're actually creating an end of life condition of a hydrogen equivalent concentration that we would see in 2024 as opposed to today, and we are also projecting out the stresses that would take place or that you'd find in a tube

at end of life as well, and then we would test under those conditions to ensure that at end of life we understand what the conditions will look like.

So, we are able to actually look ahead through our research and development programs.

THE PRESIDENT: Okay. Staff, you want to add to this?

MR. FRAPPIER: I'd ask Mr. Glen McDougall to add to that.

MR. McDOUGALL: Glen McDougall, for the record.

Mr. Commissioner, I heard four different questions. I'm just going to repeat the questions back first to make sure that I'm actually staying on point.

The first one you asked about the degradations that are related to hydrogen levels, are they known to be linear.

You asked about the degradations that are due to temperature and fast flux and whether they are likewise linear with time.

You asked if they were not known to be linear, what impact that would have on our understanding of uncertainty and you questioned whether some of the degradations might, in fact, be relied -- or rely on combinations of some of those parameters.

And your fourth question was, could the degradation be more complex with time. You suggested one example would be a step function or something like that.

So, I'll take those in order.

For most of the presentation that you've heard today the main degradation mechanism that we've been concerned about that's related to hydrogen levels is pressure tube fracture toughness.

The answer to your question is that the industry uses two models because the dependence of fracture toughness on different parameters varies with temperature. So, above about 250 degrees C.

It turns out that the current R&D is showing that the toughness is not related that strongly to hydrogen levels. So, it's not a question of whether it's linear or not, the hydrogen is not having that much of an influence.

However, when you get to the conditions that are more typical of reactor start-up and cool-down, so you're below 250 degrees C, there is a strong dependence on hydrogen level, but the current R&D is not sufficient to let us tell whether it is linear or not.

Industry has done a reasonable number of tests in the last five years to try and explore what the trends could be in terms of temperature and hydrogen level,

but I think calling it linear would be an oversimplification.

On your second point you asked if some degradations were due to temperature and fast flux, whether those are linear with time.

Unfortunately, the answer is going to be, it depends on which mechanism you look at. There are some types of pressure tube degradation that are pretty close to being linear with time, there are others that are not. Examples of ones that are not would be the creep of the pressure tube, so that's the deformation of the pressure tube, it gets larger in diameter, the wall gets thinner, the pressure tube gets longer, it expands inside the reactor. Those relations are not linear with time.

But precisely for that reason, industry has done an awful lot of measurements of those so that they have a good handle on exactly what type of relationship there is with time and that forms the basis for much of their predictive capability.

In terms of the effect on uncertainty, I think the important thing to remember is that industry applies, and staff concurs with this approach, they apply a boot strapping approach basically. You collect inspection data from your reactors, you use R&D to some extent to be able to guide what you should be inspecting and when.

Once you have the inspection data, then you turn that data over to modellers who try and come up with predictions that carry you a short time into the future.

And then during that period when you're using your predictive model, you continue to collect data and you're always checking yourself. You're saying, where does my model say I'm going to be and where are my actual pressure tubes telling me that I am and wherever there's a deviation the CSA standards prescribe a mechanism that the licensee has to go through to reconcile those differences.

And our job as staff is not to do the modelling for industry or to do the predictive curves, what we do is, we monitor that discrepancy process to make sure that industry is following the standard.

Your final question is, could degradation become more complex with time. And the answer is that experience shows us that there are some degradation mechanisms that remain fairly simple with time. They were next to linear in the very beginning and they remain next to linear right now.

However, there are a small number of degradation mechanisms that show it's turning out with time that there are -- Mother Nature has new things to show us, a few new wrinkles to show us.

The good news story for fracture toughness is that industry has recognized some of these wrinkles and they have an R&D program in place to try and examine some of these subtleties and I believe their intention is to try and take the results of that R&D and roll that back into their models as an improvement.

But it's all with the developing picture, there was never a point where you can be complacent and say, well, the model's good enough, I think we can stop now.

And my -- CNSC staff are confident that that's not the approach that OPG is taking.

THE PRESIDENT: Just to be clear, that's all very nice, but I wouldn't rely on the modelling as a substitute to the actual measuring and more frequent measurement if your model doesn't give the exact, precise parameters.

So, let's put it this way, you want to be a good forecaster you've got to forecast often and you forecast more and more based on actual data; right?

And the other thing is, since we can't forecast everything, I'm always relying on your assurance that no matter what will happen it will be leak before break, so we will catch whatever we miss through the modelling.

I just want to make sure that we're not putting all our eggs in this modelling basket.

If somebody wants to react to that, go ahead.

MR. McDOUGALL: Glen McDougall. Just to follow up on that. The whole point of a leak before break -- a leak before break assessment, Mr. President, is to cover the possibility that there are uncertainties that you haven't thought of.

In current pressure tubes -- or in the Canadian fleet we have not had an incident of break before leak, that's the bad case, the Doomsday. We have not had anything like that since 1983. Since then we've changed the pressure tube material and we've dramatically improved the way we make pressure tubes and the way we operate reactors.

But the way a leak before break assessment works is, it starts from the assumption that you already have a crack in a pressure tube and it's already leaking, then you apply your various models for hydrogen levels, for fracture toughness and you check to see that, even in the face of all of those bad things, the safety systems and the operator performance is still sufficient to be able to safely shut down the reactor.

And I said a few minutes ago that there

were two types of risk assessments that Pickering has done for its fuel channels. One of those is a leak before break assessment and CNSC staff have accepted the outcome of that assessment.

THE PRESIDENT: Thank you. We need to take a break, but before we break I cannot let the Office of the Fire Marshal leave without -- I thank you for this preparedness little package, I always carry it when I go to Pickering.

--- Laughter / Rires

THE PRESIDENT: But my question is really, I'd like to know -- you didn't give us any statistics, how many of those kits were actually distributed and what was the reaction from -- you know, from the clients or from the household upon receiving this? Did they know what it's for?

I mean, I'm just curious to know about the level of understanding of what's going on with these kits.

MR. BURNS: Dr. Binder, it's Scott Burns, I'm the Vice-President of Security and Emergency Services for Ontario Power Generation, for the record, I can address your question.

So, to target the residents and businesses within the detailed planning zone, formerly the primary zone, there's approximately 200,000 residents and

businesses that we send that package out to.

We have done polling in terms of our distribution and we've done different polls, so in some we've seen 80 per cent remember getting the package in 2015-'16 and we had very positive results in the polling, upwards of 90 per cent in terms of the uptake on these packages and what they're for.

Continually over -- we heard earlier in our presentation the 'prepare to be safe' was targeted for 2015 and '16. We also did a reach in another package, a flashlight package in '14 and this, again, was an upgrade in '17.

So, we continually want to reach out to our community to remind them because they can put this in a drawer and forget about it and we don't want them to do that. So, we've had -- our polling has suggested we've had good uptake in the community.

And another aspect of this, the distribution of KI and information around KI is that we're partners with the province and with the City of Toronto and the Region of Durham. All of our websites kind of reference to go to each other's websites and talk about KI, so there's lots of information there and people can obtain those KI if they wish and they haven't at our 'prepare to be safe' website.

And, in addition to that, we have sort of a newcomer program for people that are coming in that have not received the package in a previous distribution. So, three times a year we have a partnership with Canada Post that identifies residents, businesses that have moved in and have not received a package and those get sent to those folks.

THE PRESIDENT: So, the Fire Marshal...

MR. NODWELL: Thank you. Dave Nodwell, for the record.

There's not much I can add to what Scott Burns has already has already indicated. I would like to say, however, that it is an excellent model. We have throughout the province, in the vicinity of Bruce Power, Pickering, Darlington, and so forth, we have public education committees that work on products like this and they're comprised of the facilities, local municipalities, in this case Durham Region, the City of Toronto, as well as the province is engaged on those committees as well.

So, again, we've had very good feedback on this and, as I said, it's an excellent model that I know the other pub ed committees are having a look at and consider this a best practice.

THE PRESIDENT: So, who actually does the sending? Is it your office, the Office of the Fire Marshal

or the OPG? And also, what do you do with schools and school boards, do they get specialized treatment?

MR. NODWELL: Dave Nodwell, for the record.

On schools, the primary linkage with the school boards is through the municipality, Toronto, Durham Region in particular. So, in very general speak, they are very familiar with the nuclear technology. They've been receiving KI pills for decades literally and have very extensive plans.

And I'm sorry, there was another part to your question, President Binder, I didn't quite catch.

THE PRESIDENT: Just who actually mails this or send them or deliver them?

MR. NODWELL: Dave Nodwell, for the record.

I understand that OPG actually sends them out to the individual homes and so forth.

THE PRESIDENT: Thank you. Anybody else to the Office of the Fire Marshal?

Go ahead.

MEMBER VELSHI: It really is a question for OPG, but with the Office of the Fire Marshal rep here.

With your polling, have you had any feedback about PNERP and any comments around the

implementing plans for the municipalities around you and the level of support that you are giving them with that? Your opportunity to give them that feedback here. It's to OPG.

MR. BURNS: Okay. So, Scott Burns, for the record.

As the PNERP has been rolled out, only approved in December of last year and the province has been going out, as I mentioned earlier, to the communities. We've also been doing community presentations for a number of years in terms of our nuclear emergency preparedness plans. So, we do get feedback on the control -- or the contingency planning zone recently.

It is -- as mentioned earlier, so it basically advises the municipalities that they have to consider certain things. I think the feedback is they'd like a little bit more clarity around that, but there is a number of aspects of emergency planning in the implementing plan that directs municipalities to consider a number of specific items in terms of keeping the community safe.

But if I was to sum it up in one question it would be, they would like to see it be more direct like some of the other zones, but I'd defer to the province in terms of the recent feedback they may have had.

MR. NODWELL: Thank you. Dave Nodwell,

for the record.

Mr. Burns is correct. There was a lot of discussion in our consultations with responders around the contingency planning zone, specifically what it entails and what it does not entail.

As a result, we went from the fairly high level master plan drilling into the implementing plans where there was an annex created specifically to address what is entailed in the contingency planning zone. That was developed in consultation with the City of Toronto and Durham Region, amongst other designated communities, and I feel that at this point we have pretty much resolved those concerns around the specific requirements of the contingency planning zone.

THE PRESIDENT: Anybody else, question on this?

Okay. We're going to take a 15-minute break, come out at five after four.

Thank you.

--- Upon recessing at 3:52 p.m. /

Suspension à 15 h 52

--- Upon resuming at 4:09 p.m. /

Reprise à 16 h 09

THE PRESIDENT: Okay, we are ready to proceed.

Ms Velshi, you have the floor.

Okay. Dr. Demeter...?

MEMBER DEMETER: Thank you.

I had a question for Health Canada if they are still online.

THE PRESIDENT: Health Canada is here or online?

--- Laughter / Rires

MEMBER DEMETER: It's kind of like the Elvis story, they have left the building, right.

So referenced in the emergency planning talk was your 2016 Canadian guidelines for protective action during a nuclear emergency and that was a 2016 draft. So one clarification. Is it still a draft or if it's not, what's the date for the final?

The second part of that is you have a complementary document -- and I must have seen it somewhere, I can't exactly reference it -- for dosimetric criteria for nuclear emergency planning and response, a 2017 draft. Is there any information in that document that

might further influence PNERP or is it in conjunction with PNERP?

So the first one is, is the first one still a draft or what's the deadline for finalization? The second one is the second document, because it deals with dosimetric criteria, is it going to influence PNERP?

THE PRESIDENT: And the third one is what can it tell us about recovery?

MR. AHIER: Brian Ahier, Health Canada, for the record. Thank you for the question. I'm glad you asked it when you did as I was just on my way out.

So you have referenced two documents. In fact, they are one and the same document. So the 2016 draft was using a previous title for the document, which was basically the same as the previous 2004 version. The title of the document was revised to "Generic Criteria and Operation Intervention Levels for Nuclear Emergency Planning and Response" because in fact that is exactly the information that's in the document. So the title now aligns with the content.

This is a document that has been in development for several years. It has been finalized. It was finalized in 2017. It was improved internally within Health Canada in 2017 and now it is just going through the final formatting process so that it can be published and

then posted on the Canada.ca website. So that document is finalized.

You have asked what the link is with the Provincial Nuclear Emergency Response Plan. The document was developed in very close collaboration and consultation with our federal-provincial-territorial stakeholders, so it went out for a couple of rounds of public consultation. It also had a couple of rounds of consultation, both multilateral and bilateral conversations with the provincial organizations. All the stakeholders, whether they were federal or provincial, provided quite a bit of feedback, which is why it took a bit of time to get the document into the format that it is, but it was very good feedback. It recognizes the fact that they are guidelines, which means that they are voluntary, we can't force them on the provinces, so we wanted to take the time to get the agreement of our stakeholders such that they would be incorporated. And in fact, we are actually quite pleased that we have produced a very good document and they have been incorporated into the Ontario Provincial Nuclear Emergency Plan and in fact New Brunswick is on the point of adopting them as well once it's formally published. So there is a close link between the PNERP and the Health Canada document on generic criteria and operational intervention levels.

And to segue into the third question, the document does not address long-term recovery. There is generic criteria and an operational intervention level for relocation, but we made the conscious decision to leave the recovery part for another document, and that is a document that is being led by the CNSC with the involvement of Health Canada and that's the framework for recovery. And we would expect to see some sort of alignment between at least the relocation criteria that we have in the generic criteria document and then what eventually gets captured into the recovery framework document.

THE PRESIDENT: So CNSC, give us a sense of the timing on this.

MS PURVIS: Good afternoon. Caroline Purvis, Director of Radiation Protection Division, for the record.

So the REGDOC, we are tentatively planning to publish for public comment -- this is the recovery REGDOC -- in the summer. This has followed a discussion paper which was published in 2017 where we went out to seek stakeholder input on the development of a REGDOC that would cover the aspects of recovery.

THE PRESIDENT: Dr. Demeter...?

MEMBER DEMETER: That's good. I'm glad you clarified that it has been put into one document.

THE PRESIDENT: Anybody else?

So I can allow Health Canada to leave, but before, maybe one final thing. You think now that Canada is in a good position for the 2019 EPREV -- is it EPREV -- EPREV kind of a study, or peer review I guess.

MR. AHIER: Brian Ahier, for the record.

We are in the process of preparing for the EPREV. As has been mentioned earlier during Ontario's presentation this morning, the mission is planned for June of 2019. About two weeks ago the IAEA has formally accepted our invitation to do that. So we had sent -- as you may be aware, we had sent the IAEA the request to hold an EPREV in February of 2017. There were many steps that needed to be put in place before they formally accepted the invitation, including the completion of a national self-assessment. In order to achieve that, a planning committee across all jurisdictions was established, a steering committee across all jurisdictions was established, and we have been using those mechanisms to put together not only the self-assessment but also the overall planning. So I think from the governance point of view and our workplan in order to prepare ourselves for the EPREV, and as well based upon the outcomes of the self-assessment, that we are well placed for that mission that will happen in June 2019. The next step will be a preparatory meeting

with the IAEA and the technical team lead that's scheduled for May of 2018, so next month.

THE PRESIDENT: Okay, thank you. Thank you very much.

Back to Ms Velshi.

MEMBER VELSHI: Thank you, Mr. President.

Staff, I just wanted to confirm with you the 2017 performance assessment for Pickering. Will that be available before day 2?

MR. FRAPPIER: Gerry Frappier, for the record. So the 2017 Regulatory Oversight Report will not be ready for Part 2. We intend to present it a little bit further on in the fall. However, I expect the results with respect to the ratings, if not 100 percent complete, would be close enough for us to at least give an indication and perhaps with a bit of an indication which ones are still sort of in transit. To be honest, it's just with all the hearings going on this spring we are a bit behind on that work.

MEMBER VELSHI: Yes. Because even if it's your preliminary results, I think it will be helpful to show those.

If we can go to staff's Slide 23, I had a couple of questions on that, please. This is the slide that shows end of commercial operation. So the way this

has been laid out is if Pickering is granted a 10-year licence, then in 2028 they would be moving into a decommissioning licence, is that...

MR. FRAPPIER: Gerry Frappier, for the record.

That's the nominal plan right now, so that's what we would see happening, yes.

MEMBER VELSHI: And if they were to come a couple of years before 2024 to say, sorry, shutdown is going to be a bit later and if everything moved, then they would need an operating licence for whatever that extended period of time is before they are moving to decommissioning?

MR. FRAPPIER: Gerry Frappier, for the record.

So the licence that we are talking about today would be valid until 2028. The licensee can, before that date if they wanted to, come and ask for a different licence or a licence to have different activities. If they are looking to -- if you are talking about what might happen after 2028, if this schedule, if you like, shifted a couple of years, there are actually options there. They could be looking to do a decommissioning licence anyways a little bit early. There is nothing that requires that that transition to safe storage state be done under an operating

licence. So for instance, Hydro-Québec with G-2 is doing their current stuff under a decommissioning licence.

MEMBER VELSHI: Thank you.

And for the safe -- the stabilization activity plan and the sustainable operations plan, I noticed that the REGDOCs are draft. I think I saw that in the LCH. So why are these not finalized or what's the timeline to get them final?

MR. FRAPPIER: Gerry Frappier, for the record. I will pass it back to Dr. Al Omar in a second, but I'm a bit confused by the question because I don't believe our REGDOCs cover this.

MEMBER VELSHI: I thought it said they had to develop the plan in accordance with -- and it was draft requirements. I'm sorry, it didn't say a draft REGDOC, I think it said whatever the requirements were, but it had draft next to it.

MR. FRAPPIER: So now we got to what you are asking, so I would ask -- is it Heather? Al. Al Omar will answer that one.

DR. OMAR: Al Omar, for the record. I am a Senior Regulatory Program Officer.

We don't have a REGDOC for the SOP and SAP. However, we have experience with that since 2011 when Pickering decided not to be refurbished. Currently, we

updated these requirements and expectations in a formal letter to OPG and that's what we are referring to, and this update was done in 2017.

MEMBER VELSHI: So there is no REGDOC, but the requirements are very clear and it's not -- I don't know where I had seen a draft then.

MR. VIKTOROV: If I may, it's Alex Viktorov.

Indeed, there are no REGDOCs and we don't plan to develop a REGDOC for this particular aspect of operation. Requirements are already sent to OPG in a formal letter and they are captured in the Licence Conditions Handbook.

MEMBER VELSHI: Thank you.

And if you can -- you mentioned about G-2. Are there any other kind of big learnings from that that this slide -- that would be reflected in this slide from a regulator's perspective? Is anything being contemplated different for Pickering?

MR. VIKTOROV: Alex Viktorov.

We do have experience, as mentioned, with Gentilly-2, also Pickering Units 2 and 3. Indeed, there are some specific nuances that are experienced during this particular transition from operation to gradual transition to safe storage. However, overall they are making the

plant safer. It is de-energizing the fuel and removing the radioactive components. This work needs to be done with proper attention to risk and hazards, but the past experience shows that it can be done, has been done safely and we will learn from this experience. Right now the requirements which were already mentioned for SAP and SOP do focus on certain areas, in particular human performance.

THE PRESIDENT: Okay. So now let me try to understand. Before Pickering decided to ask for an extension to 2024, they were to come to us in 2018 giving a transition plan for shutdown in 2020, if I understand correctly. Because you are going from full operation to shutdown, there is staff implication, that's a lesson we have learned from Gentilly-2, the whole operation changes, so they should have had already a plan ready almost now for a possible shutdown in 2020. So I would like to know what that would have been and what would it look like for the shutdown in 2024, particularly if you lay -- staff, let me start with staff. Throughout those phases, presumably staff will gradually go down. How do you maintain? I'm going back to how do you make sure that you are as good the last day as now when your most competent people will bail out. Let me put this as an assumption. So how do you retain everybody with the same competency until the end and what is the plan for downsizing over time?

MR. FRAPPIER: Gerry Frappier, for the record. Perhaps I will start and then I'm sure OPG will want to add.

So you are correct, how to do this transitioning and the thinking around the sustainable operation plan has progressed -- has had a lot of thought put into it already because of the expectation in 2020. It was very different with respect to G-2 because G-2 was not -- did not have this long plan in front of it about closing down. It all of a sudden made a political decision and decided to close down, so there was no need for that plan. But they did have to scramble to put together the equivalent of the plan for transitioning to safe shutdown and, in hindsight, it would have been better if that plan was more developed earlier, but of course they didn't think they had the need for it.

With respect to Pickering, this is something that has been seen coming for a while and so there have been discussions between staff and OPG on what are some of the key things, but perhaps I will let OPG talk to where they are at with their planning for the SOP.

MR. LOCKWOOD: Thanks, Gerry. Randy Lockwood, for the record.

We will probably go about this with several questions in amongst that discussion. We will

start like this, have Robin talk a little bit about the background previous, because we have submitted a sustainable operations plan in the past, 2016 as I recall, have Robin talk a little bit about that, come back to me and I will tell you a little bit about where we are now and how we are planning to 2024. Speaking to the staffing, really it comes down to three simple areas, in my opinion, the staffing, training and how are we going to maintain them motivated or engaged; correct? And then post-commercial, end of commercial operations, I will have Steve talk about -- because we do, we are working on this right now of course, as you would expect -- have Steve talk a little bit about where we are post-2025 by the end of commercial operations. That way it should give you a feel about where this status is now, where we are going, right?

Robin.

MR. MANLEY: Okay. Robin Manley, for the record.

As Mr. Lockwood just said, I'm going to speak a little bit about the historical background here and then turn it back over to him.

So as you said, President Binder, the Commission and staff required us to make preparations for the previously planned end of life in 2020 and we submitted to staff, starting back around 2011 or thereabouts, the

first of the sustainable operations plans, or the SOP as we called it, and that has been an ongoing process. We submitted on the order of four or five of them annually as late as 2016 when it became apparent that we were asking to continue operations to 2024. And the CNSC staff had been providing us with feedback along the way on those plans.

And it's talked about a little bit in our Commission Member Document in section 1.3.4 on end of commercial operations and, you know, it's clear that the work that we have done to date has set us up for the future and that we understand that there are special considerations that we need to take account of, that it's not just normal operations as you approach end of life. So there are things that we need to think about that could be special around organizational change, staffing, managing safety culture, managing human performance, and of course the equipment and reliability strategy.

Other aspects. When we looked at it in the initial sustainable operations plans, we looked at it and we went there's nothing obvious in this safety and control area that's likely to change, transport and packaging, for example, what particularly. But we're not keeping, you know, blinkers on here. We are recognizing the fact that we need to continue to look every time and if something changes then we would take some sort of action

and develop additional plans and strategies to deal with it as we got closer. So that's the historical practice. We have met the regulatory requirement to submit the SOPs and they are in place and available as the planning basis today for whatever might come in the immediate future.

With that, I will pass it back over to Randy.

MR. LOCKWOOD: All right. Randy Lockwood, for the record. Thanks, Robin.

So I'm going to start with putting this in context. The shutting down and the end of commercial operations for Pickering has significant impact on the company, OPG as a whole, right? It's very important that we start off from that perspective. It is a significant source of generation, hence revenue, and it is far bigger than just who is working at site, okay.

So with that in mind, even our strategic plan for the entire company has been changed in terms of timeline out from five years to seven years to consider the significance of Pickering shutting down. So that will give you some context of where this -- the magnitude of this and the way the company is assessing it and the amount of resources and attention that they are paying to it.

With that in mind, we have to start with the end in mind, what does OPG look like in 2025, what is

it going to look like in 2030, 2035, because there's a whole number of things that enter into this in terms of workforce demographics and we are a company, we still want to grow the company and move forward. That's why what I will have Steve talk to in a moment is what does 2025 look like. We are going to start with the end in mind and say what is the work and what is the organization going to look like, how many staff do we have, et cetera, et cetera.

It comes back to up until the end of commercial operations though, I mean we are a large company, we have staffing resource planning that looks out 10 years, we are very comfortable to say this is what it takes to safely operate and maintain Pickering. Little doubt about that. And also, it's important to be thinking post-2025 what is the company going to look like and how are we going to transition to that. And that's some of the parts that Steve will talk to in a moment.

So where we are now is we know what the staffing looks like to run Pickering to the end of 2024 and there is a lot of work ongoing in the company around this transition. As well, we do have talent and recruitment. That has not changed. The company is not going to end with the shutting down of Pickering. I mean it's an asset. At some point the end of life comes with every asset, so we know that's coming. So we still are out there recruiting

the talent.

And as well, succession planning is key in all this, that we make sure we have the right leadership and that we are planning that we have the right knowledge to continue to operate the company. For argument's sake, that might be just Darlington, the Darlington Station. Are we making sure we are developing the right leadership, we have the right knowledge and skills, and from a knowledge perspective, a knowledge maintenance perspective, to still run the industry safely going forward and efficiently. So that's also ongoing, but largely those are just things that we do every day.

The training, I don't see that there is much change there. We will still have to -- between now and the end of commercial operations, we still have to maintain staff, skills, knowledge, competency. We have a very well-established mature training program at OPG that I can tell you that's recognized around the world. Important to all that is on boarding, you know, making sure that staff quickly come up to speed when they come into OPG, and from an efficiency point of view and as well an engagement piece, if you understand the behaviours and the culture at OPG that safety is priority, okay.

Why is that important? Because not just the shutting down of Pickering but the age of our industry,

there is a lot of attrition, there is a lot of people retiring. So regardless of whether we were shutting down Pickering, there is still a lot of new staff going to be coming in the door and that's why I make mention of that. So we have a very well-established on boarding program as well as continuing training. That is going to continue and be maintained because we have to maintain people's skills and qualifications.

The last piece before I turn it over to Steve is the engagement piece and the culture. I spoke to that this morning in the presentation. It's important that we -- and we have done this and we are going to continue to do this because that's why we are having success, is making it very clear to people where we are going. There are no false hopes. I can assure the Commission that every time I tell people we are running to the end of 2024, that's what we are doing, right, and that as well what I was going to say -- and it's very clear that they understand our priorities.

For example, at Pickering right now it's drive down the backlog, it's human performance and it's teamwork, okay, and everyone is aligned around that at site. And what comes from that is success. As I mentioned earlier, we are driving down the backlog, we are having the best generation that we have ever had, our safety record

is -- you know, this is industry leading, like Glenn talked about during his talk this morning, for a six-unit site and we are all very proud of that. We have to continue doing that. And it's having a positive effect, plus what the video showed this morning in terms of innovation. We are out there. You might not think of that, but we are leading the industry, Pickering, in terms of looking for new ways of solving old problems and a lot of the industry is starting to look at us that way and that is helping the culture and it is attracting new people that want to stay here, right?

As well, we have to transition people with experience from Pickering over to Darlington because ultimately that will be the station we have.

And I will leave you with a final thought before turning it over to Steve, is several external groups and evaluators have come into Pickering expecting they are going to find a morale problem and quite surprised that they do not, and quite the opposite. We received in our recent WANO audit a strength in teamwork and every evaluator agreed with that. So we are going to continue what we are doing because it's having the right positive effect and people are aware that the end of life is coming, right? That's just the way it is, okay. As well, we have to work with the PWU and the society representatives to

make sure that collectively the communication is consistent on all venues. And post-2025 -- I will turn it over to Steve -- because there is a lot of work ongoing for what the organization will look like post-2025.

Steve.

MR. GREGORIS: Steve Gregoris, for the record.

Just before I go to 2025 and what we are looking at for the organization, I am part of that project, I wanted to give some perspective to the Commission here because I have been away from Pickering for over three years. I worked at Darlington for a couple of years and then I was down in Atlanta, as I mentioned. Prior to that though, I had 21 years at Pickering. I know the Pickering plant, I know the Pickering people. Coming back to the plant after three years I see change and I see good change in that plant. And it's not just that the plant is performing better than it ever has, but there are new people in the plant. We hired new people last year, we hired new people the year before. So that's new people with fresh ideas, a new generation that approaches work differently, with a lot of new ideas, and that has allowed other people to move up into supervisory positions. So then what we have there are new leaders that are very energized and committed to the organization. The

leadership and the approach of leadership at the plant has changed. It's a facilitative type of leadership and it really empowers people. It lets people lead, make decisions where appropriate and really drive the company forward, and that has really again energized not only the leadership but the entire organization and that's why you are seeing some of the results you are seeing.

The last thing I will mention is around innovation and we talked about it. We are using innovation to solve problems we have never been able to solve before, I am seeing them personally, and we are using innovation to be more efficient. And more importantly, we are allowing people to think beyond boundaries that we used to have in place and because of that we are developing solutions, like I said, to problems and having people come up with new ways to be more efficient and drive this business forward. So the energy, the motivation, the engagement and the performance at Pickering has significantly changed and what I would offer you is people want to be part of it, it feels good. And we are going to hire again this year and we won't have a problem hiring people. People want to work for a good company and a good station and Pickering is just that.

To move on to down the road when Pickering shuts down, so we have a companywide project that is going

to look at the entire company's organization. So clearly, the Pickering plant is going to shut down. Once it shuts down, the amount of resources won't be the same as when it's operating. So clearly, there is an impact on the nuclear organization, the corporate functions and companywide, because some of those functions in corporate are shared with a renewable generation part of the business as well as nuclear.

So the project right now is developing an organization for the entire company based on what the company will look like as we know it today down -- we are looking towards 2030-2035. It has some initial assumptions, so there is some growth in the company we see, but clearly for nuclear it will be Darlington is the only operating plant, with the ability to scale up should anything change.

That organization -- we are moving through that design, it is involving a number of parts of the company -- will be set towards the end of June. Once that's in place we will understand the people demand and what the organization will look like going forward. The next step then is to move towards the supply side, how are we going to manage our workforce and the changes to the organization to make sure there's a clear transition. So we will do that with implementation plans and change

management plans and that will happen each year as we move through business planning. So, as Randy said, our business planning has moved to a longer duration and all those changes will be part of business planning each year to ensure changes are made in a gradual sense to get to the end result.

THE PRESIDENT: So right now, I mean just a quick answer. Just now, are there any numbers that can take you from 3,000 operating now to 2025 into the SAP process? Is there planning numbers? I'm just trying to get a feel whether those numbers will be in the plan that you have to submit. I assume in 2019 you will have to come up with a SOP and SAP, if I understand correctly.

MR. GREGORIS: Steve Gregoris, for the record.

So there certainly is planning that takes into account the changing numbers. I am going to ask Art Rob to speak specifically to those numbers because they are an input into this overall plan I described. Again, once we shut down there will be reduced numbers and I will ask Art to speak to that.

MR. ROB: My name is Art Rob, for the record. I am the Vice President of Decommissioning at Ontario Power Generation.

So just to build on what Steve has

indicated, there is an overall strategy in OPG and of course it's looking at the supply and demand sides of the resource profiles required for the safe shutdown phase at Pickering. Right now of course the Pickering project is looking at some of the work involved to get the plant into the safe shutdown stabilized phase which is covered in this licence period. As part of that work, we are looking at individual systems and the classification of systems, and stating reports and different planning to figure out the actual work involved. And we are using of course the basis of the earlier work done on Pickering Units 2 and 3, which are already in a safe storage phase. So we are working to the timelines, to support the timelines of Steve's overall project for staffing and working aggressively to complete some of our detailed planning on the work to get a resource profile for the shutdown phase that you are asking.

THE PRESIDENT: Okay. We need to move on. Mr. Berube...?

Go ahead, please.

MEMBER PENNEY: I had a question for DFO if they are still on the line. Maybe we could just check and see if they are still in the room.

MS THOMAS: Yes, DFO is still on the line. It's Jennifer Thomas, for the record.

MEMBER PENNEY: Okay, great. Thanks. So

my question really had to do with the authorization. In the documentation that we got it said it was coming and what we heard today is that it has been issued and I think what I heard was that all offsets were accepted. So my question for you, DFO, is, you know, what follow-up monitoring or otherwise requirement is there on OPG and is it you who does that or expects that or is it CNSC? Thank you.

MS THOMAS: It's Jennifer Thomas with Fisheries and Oceans, Fisheries Protection Program, and I am the Manager of Regulatory Review, for the record.

So regarding the authorization, yes, it has been issued and we could provide a copy if the Commission would like that. It was issued on -- I have the date here. No, I don't have the date on the last page of it. It was issued in January I believe of 2018. The exact date I can't quite recall. There is offsetting outlined in the authorization as well as monitoring and reporting requirements. It's a fairly detailed monitoring and reporting requirement and Pickering would be working with both CNSC and DFO on those monitoring and reporting requirements based on the MOU that we have with CNSC.

And did you want to know specifically what sort of monitoring was being asked for, and reporting?

MEMBER PENNEY: Sure, in a brief way.

That would be very interesting, thank you.

MS THOMAS: Sure, okay. So I'm just skimming through the authorization here that I have in front of me. So we have the offsetting at Big Island, the wetland complex, and we are going to be looking at production and there is a detailed monitoring plan that was put in their application with Simcoe Point. Again, we did some modelling at that particular wetland and again we are going to be doing some monitoring again to report back on the biomass that was created at that wetland. And then we also have the stocking program. And specifically for the stocking program, just that it happened. I don't think there is any specific monitoring related to that piece.

MEMBER PENNEY: One other question. So when you say there is monitoring, follow-up monitoring required for the stocking program in the two wetlands, is that monitoring that DFO undertakes in addition to what you are requiring OPG to do?

MS THOMAS: So the way the monitoring would work, like I said, we would be working with the Canadian Nuclear Safety Commission on this and OPG is required to submit a monitoring report in -- it's outlined in the authorization the dates that these reports are required, and then we would be reviewing them with CNSC and providing comments back to OPG. So that's one aspect of

the monitoring program and we will likely go out and do some of our own monitoring by doing our own site visits and such.

MEMBER PENNEY: Thank you for that. A question for CNSC. Does this in any way collaborate with your Independent Environmental Monitoring Program that you undertake?

MR. FRAPPIER: Gerry Frappier, for the record. Perhaps I will ask Dr. Ducros to give us a sense on how what we are going to be doing is part of the monitoring and how it might fit into independent environmental monitoring.

DR. DUCROS: Dr. Caroline Ducros. I am the Director of the Environmental Assessment Division, for the record.

As Ms Thomas mentioned, the reporting comes from OPG. From the CNSC perspective we would do the reviews alongside in collaboration with DFO. However, we do have inspectors onsite that would look at any kind of sampling techniques in terms of losses.

With respect to your question about the Independent Environmental Monitoring Program, I will pass that to my colleague, Ms Sauvé.

MS SAUVÉ: Kiza Sauvé, I am the Director of the Health Science and Environmental Compliance

Division.

The Independent Environmental Monitoring Program doesn't really interact with the *Fisheries Act* authorization. We are adding fish to the IEMP at Pickering, but it is not in relation to the *Fisheries Act*.

Something I will add is that you won't see this *Fisheries Act* authorization in the LCH, in the Licence Conditions Handbook. As OPG is responsible for following all authorizations that they have, it's under the MOU that CNSC gets involved in reviewing. And we are actually right now going through the *Fisheries Act* authorization line by line to ensure that we have a robust compliance program to ensure that OPG is meeting all their requirements on behalf of DFO.

THE PRESIDENT: So right now, are both CNSC and DFO satisfied with the way the authorization is being executed? DFO, why don't you start?

MS THOMAS: Thank you. It's Jennifer Thomas, for the record.

To date, yes, we are satisfied with the way the authorization is being implemented. I don't know if CNSC wants to add anything.

DR. DUCROS: Caroline Ducros, for the record. I will concur.

THE PRESIDENT: Thank you.

MEMBER PENNEY: Thank you.

THE PRESIDENT: Back to Mr. Berube.

MEMBER BERUBE: I'm just looking at the timeline for this proposed licensing phase here and of course we are seeing a lot of transition, a lot of activity. We have already spoken to that. I am more curious about the actual SAP phase of this thing because at that point we are going to start tearing things apart and moving a lot of materials, certainly activities that have been done before but not at this magnitude and probably not at this speed. I'm curious, the question specifically I have is, looking at the four-year timeline that you have to actually turn this into a safe storage state, if you are going to make the end of licence as indicated on Slide 23 here as presented by the staff, do you feel that you actually have the resources to do that? How firm are you with this?

MR. LOCKWOOD: Randy Lockwood, for the record. I appreciate your question. Again, I will direct Art Rob to speak to the details.

You make mention, yes, there's Unit 2 and 3 that we have experience from, but we also are informed from recent activities at Darlington refurbishment in terms of how to defuel the reactor in a timely manner, as well how to drain the heat transport, how to drain the

moderator. So there's a lot of learnings and experience coming from that as well, not just Units 2 and 3. And in terms of timelines, we can probably look at how long it is taking to do those various activities at Darlington and extrapolate them over to Pickering, right, which will be undoubtedly quite a bit different from 2 and 3, off the top of my head, just because of the way that things have evolved. It is probably a good point for Art to talk a little bit about the details of the planning of the various activities in the SAP phase.

Art.

MR. ROB: Art Rob, for the record. Thank you for your question, Commissioner.

The question you had was about the magnitude of products and how fast we have to deal with them as we go through the decommissioning phase. I guess just to be clear on the record, the preliminary work we are going to do as part of the licence period is to actually safe-state the reactors on the six units at Pickering. Safe-stating for us involves of course removing all the fuel and all the water in the reactors and then end-stating some of the systems to reduce the actual maintenance burden and simplify the plant footprint. As part of that work there is of course some known work as we have learned from the previous safe-stating at Pickering 2 and 3. We also

benchmark of course in industry with our industry partners and Hydro-Québec and learn what they have learned through the process of doing that, but it's not the actual -- that phase of the work is not the demolition of the plant and there isn't large quantities of waste or something like that to be generated or transferred during this period of time.

MEMBER BERUBE: Okay. Thank you for that.

Staff, just out of curiosity, based on our monitoring activities of what has happened at Pickering so far and our understanding of their performance, do you think that these timelines are reasonable?

MR. FRAPPIER: Gerry Frappier, for the record.

Yes. As mentioned, the big steps that we would be looking at having there is defuelling the reactor and then moving to dewater it. Both of those steps are pretty well understood. Of course fuelling, they are doing on a constant basis, so the timelines we would see as being appropriate timelines. And they will, at the point of 2028, come back for a new licence and at that point in time they would be looking at some of the perhaps more difficult steps of the operation.

MEMBER BERUBE: So just to follow on that, we would be talking about human resources, but this can be

done comfortably in this timeline? There is not going to be pressure on staff at these facilities to actually move at a pace such that it could increase the potential risk to their radiation protection, security, safety, that kind of thing?

MR. ROB: Art Rob, for the record.

I guess one of the things I want to assure the Commission of course is that, much like most work we do, we want to protect of course the strong plant performance this plant has enjoyed as it moves towards the end of its life. Keeping that in mind, we do the detailed planning for doing things like decommissioning in this first phase of safe-stating the units. Those are primary areas of focus, the safety and of course the security, the rad dose that people will take doing this work. These are all important considerations in the detailed planning that we are doing for that work.

Currently, I would say that there is the pace of decommissioning or the pace of safe-stating these units is something I think that is fairly well understood and of course it will count on us with some of the technical limitations of the plant, but for the most part I think the staff part is not the biggest risk to that work being completed safely.

MEMBER BERUBE: Thank you.

MR. FRAPPIER: Gerry Frappier, for the record. Perhaps I would ask Dr. Al Omar to add to that from a staff perspective.

MR. OMAR: Al Omar, for the record.

I think this is why we required OPG to develop and submit the SAP three years before they get into the stabilization agreements. I think the concern here will still continue to be how you transition even the workforce from a workforce to shut down to a workforce to stabilization. So you may need some new skills and knowledge. You may change the workforce and base it on outsourcing, for instance, but there will be change in the staff. In addition, we will be looking into perhaps developing new tools maybe for that based on the experience of Unit 2 and 3. How you are going to deal with some units shut down but stayed in guaranteed shutdown state while you are defuelling, dewatering at least two units at a time. How are you going to deal with the fuel transfer from the reactor to the wet storage, the irradiated fuel bays, and so on and so forth. I think what we have to do is during the submission, development of the SAP from 2022 to 2024, I think we will go, as we used to do for the SOP, just to see whether the expectation and what OPG will manage, even transition in the organization itself will be sufficient for the staff to be satisfied that the activities

undertaken between 2024 and 2028 will continue the units to be safely operated or surveillanced, as well as how they manage again the human element itself. How will you manage human performance that will continue into the SAP.

I would like to invite my colleague Ed Leader just to add the site experience during the shutdown of, and isolation of, 2 and 3.

MR. LEADER: For the record, my name is Ed Leader, I am the Power Reactor Site Office Supervisor at the Pickering site.

Just to add to Al, our current compliance verification plan goes out five years. Our compliance verification criteria will not change during that period and, as Al said, once we get the initiation of the SAP, then we will adjust our compliance verification criteria accordingly. But we do have experience monitoring Units 2 and 3 safe shutdown. We did a separate project for that and there was no additional risk to the workers, public or the environment during that project.

THE PRESIDENT: Thank you.

Dr. Lacroix...?

MR. GREGORIS: Steve Gregoris, for the record.

I just want to add, for the activities that we are looking at in the SAP specifically, so those

are activities that we perform on a daily basis or in outages, so we do for the most part have procedures to defuel and to dewater. We drain the moderator systems in outages, so we know how to do that. We have proven procedures and we know how to manage those processes and that draining safely. We also partially drain the heat transport systems, so again we know how to move that water and do it safely.

I also want to stress that most recently with Darlington Unit 2 we have proven technologies to drain and dry these systems. So that again, as Mr. Lockwood mentioned, is valuable experience for us to bring forward so that we can do this work safely, we can do it correctly and in a timely fashion.

THE PRESIDENT: Okay. Thank you.

Dr. Lacroix...?

MEMBER LACROIX: Out of curiosity, what do you do with the water? Can you reuse it in other facilities or do you simply dispose of it?

MR. GREGORIS: I'll ask Art Rob to speak specifically because that will be a larger quantity of water normally, but we certainly have facilities to store the water, we certainly want to keep the water, it's D2O, so it's worth a lot to the industry and so our intention is to have the ability to store that water and to hold onto

it, but I'll ask Art to speak to the details.

MR. ROB: Art Rob, for the record.

So the D2O product, as Steve mentioned, there is a quantity coming available as Pickering shuts down. There is about 3,500 megagrams of heavy water on the Pickering site. The plan detail -- detailed planning is looking at places to actually store the water. And actually between the two facilities, Darlington and Pickering, there is enough purpose-built tankage to actually store all water safely during the transition time.

Overall in the industry, OPG is looking at a long term strategy for heavy water and, of course, the other players in the Canadian market of course are also looking with interest at the overall long term plan for heavy water management.

MEMBER LACROIX: Okay, this is my real question now.

--- Laughter / Rires

MEMBER LACROIX: We changed from a safety and controlled area of security.

On page 112 of the licence application, there is one page devoted to cyber security, and I must say that although I sort of understand that you are in the process of complying to the regulation, this is my perception.

First, I will give you my perception of cyber security and next I will ask you, I would say, a multifaceted question.

To me, cyber security is the protection against malware, that is, software that may corrupt, steal, or destroy data, but it can also destroy physical objects, such as to pumps and generators. So the first part of my question is that, is the philosophy of cyber security implemented at OPG similar to the defence-in-depth that has been adopted for the physical facilities?

And the second part of my question is, are you aware of any attempt to infiltrate your system at Pickering, how many times, and what sort of a threat are you facing? Is it from outside? Is it from inside?

And before you answer my question, don't spare me the technical details. Please, talk to me about firewalls, encrypted files, blocked USB flash drive boards, system robustness, redundancy, quarantine, anti-virus, isolation, cloud computing, air gaps between internet and your computer system.

Thank you.

MR. BURNS: First of all, thank you for your questions. Scott Burns, for the record.

I'm not going to be able to answer your technician question, but I'll give an overarching answer,

and then Mike Benjamin is here with us and can provide some more details, based on your question.

So OPG does take cyber security very seriously. We do have an overall defence-in-depth philosophy. We manage our assets in a secure, vigilant, and resilient manner.

We have been working hard at cyber security improvements on the site since 2010 and our goal to be compliant with the CNSC standard is to have all of those aspects of our system completed in 2019.

We have done this on a priority basis and we initially looked at our significant cyber assets first, so in terms of the nuclear operations, security systems, emergency preparedness systems and fire systems. Those are some of the significant assets, and then we are looking at some of the more cyber essential assets in terms of the next level of priorities.

We do have awareness campaigns in terms of the phishing. So we test our employees right across the country. That's been a very successful program to educate and provide awareness for all of our employees around the importance of being aware of potential attacks. So we actually have phishing exercises and validate and measure how we're doing there. That has been effective in educating our staff and creating a greater level of

awareness.

In terms of answering some of the detailed aspects of your question, I would turn it over to Mike Benjamin, our manager for cyber security.

MR. BENJAMIN: Thank you, Scott.

My name is Mike Benjamin. I am a Senior Manager, Cyber Security for OPG.

As Scott said, since 2010, we've had robust controls in place to protect those -- especially those assets, significant cyber assets that support nuclear safety functions.

And OPG does display an overall defence in-depth strategy. This is right across the company. We have a matrix organization integrating our security for the business, our renewable generation stations, and as well as our nuclear plants. This defence in-depth strategy takes the place of technical administration and operational controls.

Most importantly, our industrial control systems are isolated and segregated, separated from the business IT side. But the defence in-depth is broader than that. We isolate. We have intrusion detection systems. Our perimeters are controlled by firewalls. Technical controls are things like these firewalls.

THE PRESIDENT: Be careful with security,

technical security, things that describe the information. So keep it general, please.

MR. BENJAMIN: Okay, I will.

We do have operational controls as well and things like bringing USBs. There are specified procedures that a USB must be scanned before it's put into a piece of control equipment.

We have a systematic approach to training in OPG nuclear that also applies to cyber security. So our cyber security specialists -- our subject matter experts have specialized training, that is that they have to qualify for.

All of our nuclear employees have general employee training every year where they have refresher training on awareness. And as Scott said, we have monthly campaigns to train our employees to recognize and report emails that are being sent into the company to try and fool them into losing their credentials or introduce malware.

Without going into any details, you're asking, how do we have incoming threats? One of the major threats faced by the company is those types of phishing emails which we recognize and eliminate once we find them.

We also recognize that things do happen, so we spend a lot of time in emergency preparedness and incident response. Specifically, last year we took place

in the GridEx exercise right across North America. There were over 450 organizations that took place. This was right across North America including utilities. That was combined cyberattacks along with physical attacks on our nuclear plants. We included our nuclear plants in that exercise as well as our renewable generation plants.

We also are very much working with other industries. We have joined the new Canadian Cyber Threat Exchange where we trade intelligence with industries across Canada.

Within Ontario, we have monthly cyber meetings to collaborate with Hydro One and the ISO to exchange operating exercise and intelligence.

And of course, we're moving to the new CSA N290.7 standard. That's being done in collaboration with the other CANDU owners, through the CANDU Owners Group. So we're working very closely with CNL, with Bruce Power, as well as Point Lepreau, so as we broaden our existing controls and make sure that we all move together towards the CSA N290.7 compliance.

THE PRESIDENT: Okay. Thank you.

MR. BENJAMIN: Thank you.

MR. LEMOINE: This is Eric Lemoine, for the record, if I could just add CNSC's perspective on that.

I'm over here. Sorry.

--- Laughter / Rires

MR. LEMOINE: So we actually did a Type 2 cyber security inspection in February of this year against the current requirements in the Licence Condition Handbook, and a preliminary summary of findings was shared with OPG. The inspection report is not complete yet.

I can't get into the details of the findings unless we were to go *in camera*, for example, but what I can say is that the Pickering cyber security program does meet the current requirements and there were no safety significant findings.

THE PRESIDENT: Okay, thank you.

I would like to move -- Ms Velshi...?

MEMBER VELSHI: Thank you. A question on periodic safety review, so if we can turn to Slide 17 of staff's?

So this slide shows that the CNSC staff identified 23 gaps. It's a very simplistic question. Is this something that OPG should have picked up? Is there judgment involved in identifying gaps?

MR. FRAPPIER: Gerry Frappier, for the record.

I would ask Dr. Al Omar to respond to that.

DR. OMAR: Al Omar, for the record.

These involve coding standards and safety factors -- we call it task reviews -- really requires the assurance of existence of things or adequacy of things or effectiveness of things, things meaning programs, procedures.

So when OPG submits these documents as well as the reassessment of the original continued operation plant, continued operation plant is equivalent to an IIP, integrated implementation plan. That was produced back in 2010 for the Pickering B original refurbishment.

So when you reassess the continued validity of the continued operation plant findings to close to 2020, by operating to 2024, there is different perspectives from the way the licensee see the reassessment from the way the CNSC staff see it and make the determination on the necessity to continue a particular issue or elaborate on a particular issue. Most of our findings came, in fact, from the reassessment of the continued operation plant and the impact of operating to 2024 on the original findings.

Another major assessment came from OPG's reassessment of the Fukushima action items and we found that we have some issues that have to be included in the current IIP to ensure the integrity of containment under severe accident.

MEMBER VELSHI: Thank you.

And then I also read -- I think it was one of the IAEA/OSART recommendations was that the periodic safety review safety should have looked at waste management facilities. And I know you said Pickering has got an implementation plan to address those recommendations.

Before I come to Pickering, did I get that right that waste management facilities were not assessed as part of the PSR and should they have been, or will they be?

MR. VIKTOROV: Alex Viktorov.

We briefly know whether the waste management facility or waste management practices outside of a power station were not part of some scope. Therefore, relatively well defined standard scope of their emissions, and we went with that.

Again, the waste management facility at Pickering site has a different licence and governed by different regulatory requirements, and different practices. However, we are considering right now expanding the PSR practice to other facilities.

MEMBER VELSHI: So as far as the specific recommendation of the IAEA or OSART is there -- what would the action item say to that recommendation?

Maybe I should ask Pickering that because I guess their recommendation was to you.

MR. GREGORIS: Steve Gregoris, for the record.

I am going to ask Mike Ruffolo to speak to the details of your question.

MR. RUFFOLO: Mike Ruffolo, for the record. I'm the manager of the periodic safety review for Pickering.

Just simply, there is -- there is currently no plan to do a periodic safety review for the Pickering waste management facility. I'm not aware of a plan to do such an exercise going forward.

MEMBER VELSHI: And correct me if I'm wrong, that was a specific recommendation or it wasn't a recommendation?

MR. FRAPPIER: Gerry Frappier, for the record.

So I think we're not quite following your line of questioning. So in the OSART area there was not a recommendation.

--- Pause

MR. FRAPPIER: Okay. So maybe we're just a bit confused as to what area you're talking about.

While you're looking for that, I think Mr. Jammal wants to add something.

MR. JAMMAL: It's Ramzi Jammal, for the

record.

With respect to the use of the periodic safety review for waste management or low-risk facilities that is wishful thinking from the IAEA perspective. So the implementation for the PSR for low-risk facilities or even waste facilities is a work in progress. So it's a potential future recommendation based on the adoption of each member's state with respect to a future to come of periodic safety review.

To date, there is no standard to be used for waste facilities. I want to make that clear from the Commission's perspective.

Usually, in the peer review submissions, they will make a reference to good practice of another country that is using a PSR or other methodology with respect to the waste management. In Canada, we do not use enriched fuel and we do not do repossessing, so the PSR really to such a low-risk facility does not add any safety value.

MEMBER VELSHI: Thank you. That's very helpful.

And then just for my benefit, Pickering, what's the level of effort involved in doing a PSR?

And then I'll ask the same of staff. I mean, is this years of work? Okay.

MR. RUFFOLO: Mike Ruffolo, for the record.

The Pickering PSR2 took about two and a half years to prepare to get to this point.

MEMBER VELSHI: How many FTEs?

MR. RUFFOLO: Well over 100.

MEMBER VELSHI: Thank you.

And staff?

MR. FRAPPIER: Gerry Frappier, for the record.

So, undertaking that periodic safety review is a major undertaking. It's a very extensive review of both what is your current state of your facility, and also how does that compare to modern standards, which of course your facility would not have been built to.

In the particular case here, just so that we're clear, Pickering had done a periodic safety review a few years back, thinking they were closing in 2020. With the decision to continue operation to 2024, we require them to do a delta PSR, if you like, so a more fulsome PSR than what they had looked at before. But even in that case, it's still a major undertaking with certainly a couple years of effort and, as mentioned, hundreds of people on their side.

Similarly on our side to review it is a

major review that took -- takes us certainly a year or so of undertaking. I don't have the FTEs right off the top of my head, but it would be substantial.

MEMBER VELSHI: Yeah, it's just good to get a sense of the level of effort. Thank you.

THE PRESIDENT: Dr. Demeter...?

MEMBER DEMETER: Thank you. I'll change gears a bit here.

The base for my question is on page 29 of staff's CMD -- CNSC CMD -- but I will direct the question to Pickering. And it's to do with the management safety control area and organization.

So in 2012 OPG did this business transformation initiative and it centralized a number of services across nuclear, hydro and thermal, and that included training, environmental management, information, management items, service management, et cetera. This is not unique to this industry. It has happened in a lot of other industries. They centralized a sort of shared service model.

One of the things it does is it pulls out some of that on the ground expertise. Rather than someone going down the hall and talking, you've got to call a number and get a ticket and have someone else call you.

I'm not sure it works in your shared

services, but I was wondering with this transition to a more centralized service across three sectors, nuclear, hydro and thermal, are there any risks to losing some of the content-specific issues from an HR point of view, relative to fatigue and the drug and alcohol coming down the road. Because they might be applied differently to the employees in the three different services and you've got sort of one HR that's trying to deal with a broader side of employees, so risks and benefits of this sort of centralization of your HR services?

MR. MANLEY: Robin Manley, for the record.

So if I understand your question, you are looking for an answer to the specific issue around centralization of the human resources function and its --

MEMBER DEMETER: It's HR and training and environment management.

MR. MANLEY: Okay, all of them.

MEMBER DEMETER: That sort of takes it out of your shop and puts it into a bigger shop that now has to deal with a broader audience.

MR. MANLEY: Okay.

MEMBER DEMETER: So how do you make sure that you still get your needs met?

MR. MANLEY: Right, okay. So first off, when the business transformation of sort of centralization

of certain functions happened, we did go through a systematic and rigorous change management plan that, you know, was aware of that risk, right.

So if I could pick on environment, for example, as one of the functions that was centralized, we have maintained a very strong link into nuclear from the environment function.

So we have a Director of a Nuclear Operations Environment. He is here with us today, Raphael McCalla, and he is embedded within the nuclear organization. He is not off at head office somewhere where you can't get him at the end of a phone. In fact, he's in the same building with me. He's in the same building with Glenn Jager, Nuclear Engineering and the rest of our team, and he has resources that are available.

Training -- you know, the Vice President of Training has been off and at our nuclear executive committee meeting and we have training staff, particularly in nuclear certification training, who are 100 percent focused and dedicated around training of nuclear. Training in radiation protection, these people are embedded in our plants and in our nuclear training facilities. So they aren't at arms-length and, you know, losing contact.

In terms of human resources, if I could pick another one, we have human resources business partners

and a Vice President of Human Resources Nuclear who is part of the nuclear team and nuclear executive committee.

All of the centralized management functions or programs that support the nuclear organization whether it's environment, supply chain, training, human resources, they all report on their performance to the nuclear executive committee and the Chief Nuclear Officer Glenn Jager, through our fleet view reporting system. So they are required to come out and report, at least on an annual basis and more frequently if there are any kind of issues as to the performance of their programs, any kind of gaps, deficiencies or what have you.

And when CNSC staff perform inspections of those functions they're reporting their results of any findings that they have to our site vice presidents, like Mr. Lockwood. So they can't -- they are not away, and they can't be away. They are part of us. They are integrated into all of the work that we do.

MEMBER DEMETER: From staff's point of view, as is tradition, made it better, worse, the same?

MR. FRAPPIER: Gerry Frappier, for the record.

I would ask Pierre Lahaie to come and give some additional information.

But as Mr. Manley said at the very end,

from our perspective how they organize themselves is certainly of interest but it still remains that they have a licence. The licence is for the Pickering site. The licence requires them to have security. It requires them to have environmental, different things like that, and we will do our reviews and our audits and our performance based on that.

The fact that, just because Scott is here, you know, the security function is central led does not change the fact that we expect the site to meet the licence requirements. How they choose to organize to do that is a different thing.

But perhaps Mr. Lahaie can talk about our evaluation of the organizational transition.

MR. LAHAIE: Pierre Lahaie, for the record. I am the Director of the Management System Division.

I just want to point out that at the get-go when OPG announced they were going to go to this BTI, they involved staff in our human organizational performance division and in our division in management systems, and we basically walked through it with them, asked all the -- you know, stupid, inane questions to get the good answers and we were -- it was kind of a surveillance for us to see how they were approaching this.

At the end of the day, our biggest concern was whether the accountability for all of those cross-functional programs resided within nuclear. We had lots of discussions with OPG on this, and at the end of the day we were satisfied that they had the points of contact within nuclear that were accountable for those programs, as well as accountability of the corporate program owners to the CNO, as Mr. Manley just highlighted a minute ago, when they do the fleet view review process.

We also followed up recently with a Type II inspection on organizational roles and responsibilities, and we came out with the understanding that as far as organizational change goes, and structure of the organization, it's in line with our regulatory requirements and our compliance verification criteria, and it's aligned with all of the discussions we had over the last few years.

MEMBER DEMETER: Thank you very much.

THE PRESIDENT: Thank you.

Mr. Berube.

MEMBER BERUBE: I have something a little easier probably for you to answer. I'm looking at post-Fukushima activity, especially the upgrades that you've done since lessons learned have come out on that, specifically the EME equipment that's now in the field.

The questions are basically: how often do you spin that equipment up; how often do you exercise the people that are actually using that equipment; what are the chief lessons learned in that; do you actually score that activity; and, how do you use all that knowledge to basically improve your performance overall in this area?

Staff, if you could also comment on the performance evaluations from Exercise Unified Control over what you've seen in the field, and how you feel about that?

MR. GREGORIS: Steve Gregoris, for the record.

As far as practising the use of emergency-mitigating equipment, we did have a multi-unit exercise that was performed in 2015, which had staff deploy that equipment and exercise the use of it. Again, we practised that in 2017 with Exercise Unified Control.

We also have training. The training is done on a fixed cycle, and that's set. It's a five-year training cycle, and in that cycle there is training, practical training, on using the emergency-mitigating equipment. That was done between 2015 and 2017 as well. Initial qualifications to use that equipment does use a practical evaluation, which is scored. You need to meet the minimum score to get that qualification.

MR. BURNS: Scott Burns, for the record.

You wanted some information about Exercise Unified Control as well, and the results of that.

Overall, we've been very pleased with the results of the exercise. It was a year of planning with all of the organizations involved.

In terms of the evaluation, there's really three levels of evaluation, our own internal evaluation, the evaluation by the CNSC, and we're just waiting on the finalization of a report from International Safety Research, who did some coordination for us in terms of the off-site partners. We're just waiting for some results on that.

The purpose of the exercise, as you know, was to validate our program effectiveness. We have received feedback from the CNSC that we did successfully achieve that.

The other aspect of this is obviously learning and strengthening our program, and we've identified some key areas for us to look at and improve in. We saw, not unexpectedly, communication with all of those organizations, and all of the people involved inside our organization. Communication is a key learning. Interesting for us, we had a focus group specifically looking at social media. We had a simulated social media cell that IIR put together. We've learned a lot of lessons

from that in terms of corporate communication and communicating with the public, and utilizing social media, and the sort of higher-level expectations around that. That was really some interesting learning for us that we'll be certainly taking forward.

In terms of some of the success highlights, we spoke earlier about the Unified RASCAL Interface Dose Projection Software, so we implemented that in 2017, and trained our people. That was used successfully in the exercise, as well as the plant information emergency management summary page real-time event status of what was happening, and that information going to the CNSC in real time.

We heard earlier about our new radio system, our P25 radio system for security and fire. That was successfully tested in the interoperability between our ERT, emergency response team, and Pickering Fire Services, who responded and went to a common interoperable channel, and that was successful.

I also want to mention that more recently we had an exercise in security at Pickering utilizing the same system. Again, the success of that system was very positive for us.

So we did have a good exercise, and we have some good lessons learned to apply to our program and

strengthen our program going forward.

THE PRESIDENT: Something for both. Why are some of what I thought was post-Fukushima, make-up water, fire water, some of the ventilation, was bumped to the IIP. Why wasn't it already done? Why has it not been done by now? What am I missing? I thought that there was a lot of work that still got pushed to the IIP, where I expected it to had already been done by now.

Maybe we'll start with the CNSC. Was that not kind of a requirement?

MR. FRAPPIER: Gerry Frappier, for the record.

I think that certainly all the Fukushima action items, as you're suggesting, were done, and in fact were done. There are connections for the water. All that are done.

I'll ask Dr. Al Omar to give us more things.

There are additional things that we're now looking -- given that we have an IIP, given that they're going to be operating a little bit longer, we are looking at some additional potential for improvements. Some of those are yet to be implemented, and they will be implemented over the next couple of years, under the action of Dr. Viktorov.

DR. VIKTOROV: Alex Viktorov.

Just to maybe repeat the same message, it's done. The Fukushima action plan is completed for Pickering. That's categorically done.

THE PRESIDENT: I can't remember, but in both submissions, I thought make-up water to the core was the number one priority to deal with a doomsday scenario, and I'm reading here that's it not done as yet, so somebody correct me.

DR. VIKTOROV: The original plan provided make-up connections according to the Fukushima plan. However, OPG, and we challenged them to do so, continue working to either add additional make-up connections or strengthen and simplify the earlier-provided connections, so it's building up on already accomplished actions.

MR. GREGORIS: Steve Gregoris, for the record.

Just to summarize, President Binder, all the Fukushima action items are completed. All of the required water connections that were required as part of the Fukushima action items are complete and were completed some time ago.

The additional actions for water connection are enhancements to further improve safety. They were developed as part of the PSR process, and they

will improve the total site risk. That's the additional actions that are outstanding.

MR. FRAPPIER: Gerry Frappier, for the record.

Perhaps I could ask Al Omar to add a couple of points on that.

THE PRESIDENT: Let me read you just a couple of sentences. Okay?

> "However, emergency fire water cooling piping connection to Pickering Unit 1 will be completed during a planned 2020 outage, and the restoration of emergency power to one main volume vacuum pump will be completed by June 2019."

And I just picked up something on this:

> "Completion of necessary power and support service connections required to restore the functionality of a main volume vacuum pump."

I mean there's a whole to-do list. Why are these not Fukushima kind of -- I take the point that you can always find some improvements, but I thought all of these were Fukushima requirements.

MR. FRAPPIER: Gerry Frappier, for the

record.

I'll ask Al Omar to give a little bit more detail on this, but all the things that were required for Fukushima, to provide the water and all that stuff, have been done.

During the PSR process, of course we're looking at additional things that can be done, so that's part of what you're reading there, that were not required as part of the Fukushima actions but are increasing the safety of the plant.

Perhaps Dr. Omar can give us some details.

DR. OMAR: Al Omar, for the record.

As per my previous answer to the reassessment of the Fukushima action items, it was added to the PSR, because the Fukushima action items were okay to 2020, now they are operating to 2024, so we opened again the reassessment of Fukushima action items. That was done in, I think, March 2017, so it has been an issue continuing for discussion during that reassessment.

By default, the Fukushima action items Phase I and Phase II have been in progress by OPG, and as OPG indicated, all actions related to Phase I and Phase II have been completed by end of March 2018, so during the PSR we discovered that some issues remained to be completed, only two actions, because we can't provide an additional

source of water from the fire water. Therefore, this new source of water has to be connected to Unit 1 and Unit 4, and we need an outage to do that. That's the reason one of the actions will be completed in 2020.

Regarding the other one, OPG did a study in 2014 for the controlled venting system, and they have determined that the value added, the safety value added for an external venting system like that for Lepreau, is not huge because they have existing systems they can use for the same purpose, so during that study and the reassessment we said, "Okay, what options do you have?" In discussing the options with OPG, we found that using the FADS -- the filter air discharge system, can be utilized. We went to a site visit and we inspected that option and we found, while discussing the options with OPG, that another system can be added, another option can be added to ensure that when we release, the release will be controlled and filtered venting. That other option requires the provision of power to one of the main volume vacuum pumps, which required another upgraded component that would be added to Phase II of the Fukushima action items. That component, I believe, OPG has already purchased and it's on-site. The only thing left for that is to have the design, the implementation, and the execution to ensure the power is connected to the pump, and then the provision to make it functional doesn't

impact any other component that may be influenced by the provision of that source, so it's only two actions left.

THE PRESIDENT: I assume you will report the progress in the annual report on some of those things as you go along.

MR. FRAPPIER: Gerry Frappier, for the record.

Yes, we'll be reporting annually on the implementation of the IIP items.

THE PRESIDENT: Thank you.

MR. FRAPPIER: Go ahead, Steve.

MR. GREGORIS: Steve Gregoris, for the record.

President Binder, it's important that we go through this systematically so we understand what's been completed and what hasn't, and where that lies. So EME Phase I, that's providing water to the heat transport systems, the calandria vessels, and the steam generators. That is available on every unit at Pickering.

It also provides power for monitoring purposes, and that's available on every unit at Pickering, and that's been in place for several years now.

Emergency mitigating equipment Phase II is a larger power, and that is now available at Pickering as well.

A filtered air discharge system is available at Pickering right now. It's available in manual mode with no power, and it's available to support filtered and controlled venting for multi-unit events.

The EME Phase II provides power to that system, so there's additional diverse ways to use the filtered air discharge system with power beyond the manual capabilities so that there's multiple ways to use FADS for controlled filtered venting in a beyond design basis accident.

The additional modifications we're now talking about for Pickering 1 and 4 is to add fire water again, and similar to EME Phase I, to Units 1 and 4 heat transport systems, calandrias, and steam generators. It's an additional way, another diverse way, to add water. What it does is it proactively looks at preventing an accident from moving to a severe accident, and if it does move to a severe accident, it mitigates the consequences because it's timely, it's a timely modification to add water. That's why we're making that modification, and it's part of our IIP enhancements to safety.

The improvements to the main volume pumps, which were discussed, the power is available to those pumps now. We need the support systems, mainly water for cooling of those pumps. We'll have that in place this year.

That's to provide an additional diverse flow path on top of the current three different ways to do filtered venting via FADS so that there's multiple ways to use filtered air discharge systems in the unlikely event of a beyond design basis accident.

THE PRESIDENT: Just for clarity, if you look at staff's document, page 45, you explained everything to me as a delta on an existing set of improvements you already implemented, but you'll see at the top of the page, page 45, there's one, two, three, they don't look like a delta, they look like a major improvement here, so there may be a clarification kind of thing that you may want to do for part two, that's what I would say, because when I read that, that looked to me like a lot of work that should have been done that has not been done. I take your point, we don't have to debate it now, but I think you'll want to clarify what staff are saying here.

MR. FRAPPIER: Gerry Frappier, for the record.

Just to be clear, from staff's perspective, we would agree with what was just mentioned, as these are add-ons. We view this as additional work that's going to be done under the IIP. Having said that, it probably could have been clearer in the CMD that these are on top of the Fukushima action items that had occurred

a couple of years ago.

I'd also, if you'll provide me with one moment, come back to Mrs. Velshi's discussion around the PSR for waste facilities. I do now have the point here that we tend to look at the OSART as being here's the actions we've agreed to do, here are our recommendations, but there's actually another layer of items that the IAEA provide, and those are called suggestions. There was a suggestion that the plant should consider initiating PSR for the Pickering waste management facility and western waste management facility. As Mr. Jammal indicated, we are not looking at making that a requirement at this time, so we do not expect to do any actions on that.

THE PRESIDENT: Okay.

Ms Penney.

MEMBER PENNEY: I have a groundwater question, I think for OPG first, and then a follow up from CNSC staff.

When I look at page 84 of the OPG document, and then I look at page 38 of the CNSC EA, I'll give you what I think is my summary, and then I have two questions. It's that below the turbine auxiliary bay there are two sumps or sumps, there are sumps. They actually are creating a hydraulic sink and they're affecting the groundwater in the area under that facility. I think when

I looked at the ERA, I may have seen that there as well.

My question is also about there's tritium -- I don't think there's tritium in this location, but there's tritium in some groundwater samples on-site. I do understand that it's not -- the samples that you've taken at the fence line indicate that you're not exceeding anything beyond the fence line, I think I'll say that, so I had a question about how we could have a leak below this tab. Is there no liner? Was there an expectation about a liner or it didn't need a liner? This sump that's being operated and affecting the groundwater flow on-site, is it going to be turned off when we get to end of operations, and does that mean anything with respect to the tritium on-site after the end of operations?

Perhaps OPG first, and then CNSC.

MR. GREGORIS: Steve Gregoris, for the record.

I will ask Raphael McCalla to give some additional details, but I'll start with the groundwater monitoring program at Pickering.

We have a well-established program. There's over 140 wells on the site, including a set of perimeter wells, as you mentioned. The program is there to do a couple of things, to look at groundwater flow, and the preference of flow. It looks at quality, and specifically

any changes in quality that may indicate a pathway from the station. It also looks at is there any off-site effects.

As you mentioned, the system is very sensitive. We've had cases where we've seen quality change and we've initiated investigations, and we've determined pathways and corrected those.

The flow, based on our monitoring, is preferentially north, away from the lake, and to the foundation drains that you're talking about. Those foundation drains then go to a sump that's a monitored pathway out of the plant, so we know exactly what goes out and we can control that. Okay?

As far as if we go beyond commercial operation and to shut down, those sumps and those monitored pathways will remain in service. That's part of our environmental monitoring program, and they'll remain in service as we go forward into the stabilization activities, and beyond.

MR. FRAPPIER: Gerry Frappier, for the record.

I'd ask Andrew McAllister to discuss about the EA associated with the groundwater.

MR. McALLISTER: Andrew McAllister, Director of the Environmental Risk Assessment Division.

OPG touched on a number of the points that

there is environmental monitoring in place, including groundwater. OPG is also transitioning. There's a CSA standard now on groundwater monitoring, 288.7, so they'll be transitioning to that. The expectation is that any changes to operations, if there are changes that could have a bearing on groundwater, then the monitoring program will be adjusted accordingly.

But, as you pointed out, the tritium readings in the groundwater at the perimeter are below, for example, the drinking water guideline of 7,000 becquerels per litre, acknowledging that groundwater on site is not used for drinking water purposes.

MEMBER PENNEY: A question for CNSC staff. So, would you expect that there would be a continuation of groundwater monitoring after end of operations?

MR. McALLISTER: Andrew McAllister, for the record. Yes.

MS PENNEY: Thank you.

THE PRESIDENT: Thank you.

Dr. Lacroix...?

MEMBER LACROIX: No.

THE PRESIDENT: Ms Velshi...?

MEMBER VELSHI: Thank you.

A question for OPG. So, after the fuel channels, what are the key challenges or life limiting

components or structures, and we've heard very little about your fuelling machines where we have talked about them in the past and issues about your limit of liability associated with them, but from a safety perspective primarily, what do you see as the big challenges for the next I guess six years?

MR. LOCKWOOD: Randy Lockwood, for the record.

I'm going to ask Jason Wight to talk about our aging management program, I think that's a good place to start and then see where that goes.

Specifically, we can go back to fuel handling if that's where it takes us, but let's look at the overall aging management -- MAAP program for a start and go from there and that will probably lead into life cycle management plans which I talked about earlier for major components, boilers, feeders, fuel channels.

Jason...?

MR. WIGHT: Jason Wight, Director of Engineering, for the record.

I guess before I start I'd like to just, you know, kind of give you some context. Pickering's been around for a while and we're considered old by years, but we've got a lot of refurbished components in our systems, we're like a brand new Vet in a lot of ways, so we're not

quite as old as we make out to be.

So, with regards to aging management, we have what we call an integrated aging management program and we talked a bit about life cycle management plans and LCMPs on major components, boilers, fuel channels, calandria, vessels, et cetera.

We also have CCAs, compliant condition assessments, we have an obsolescence management process, a very multi-disciplinary encompassing program that we use to monitor for aging, as well as a periodic inspection program as per our operating licence.

Just for example, we talked a bit about the work required for the PSR and the PSR -- even just going through the PSR, it's an aging management -- a way that we manage our components. And we looked at over a half a million components for fitness-for-service, relief valve programs, pump programs, pressure boundary.

We took a look very specifically at our component condition assessments and prepared over a thousand component condition assessments. And that's all part of an all-encompassing aging management process.

With obsolescence, we have an obsolescence management program, always looking at the age of our equipment and what we're required to replace, monitor, refurbish, et cetera.

And I talked a bit about the actual -- I talked a bit about innovation advanced technologies. I'm pretty passionate about, probably could take the rest of the evening.

But very specifically, we're using advanced technologies and as what we can do to monitor and manage our equipment as we go and as we operate and to ensure that we have reliable equipment and safe equipment.

We talked also a little bit about the IAEA and the OSART Mission. They took a look at our aging management process and they were quite pleased with what they saw. In fact, a lot of the refurbishment and a lot of the investment in the power plant has been paying dividends and very clearly they wanted to recognize us for that.

With regards to, you talked a bit about, you know, what would worry us with regards to aging management. Obviously, we've got a program for everything. And when it comes to actually the asset itself, it's really the major components that's the issue.

Fueling machines is more of a reliability and we've got plans with reliability and operating with our fueling machines.

But boilers, for example, would be something that we're inspecting and monitoring, we have life cycle management plans for our boilers. Those are

probably the major issues with regards to aging, it would be the big components, the big pumps, the big turbines, for example, an item that would cost a lot of money to repair and replace.

From a safety standpoint, there are no issues. From a safety standpoint, we are on top of, we manage, we have programs in place, we're using advanced technology as much as we can, using from industry OPEX, et cetera. So, really it's the investment compared to the larger components, but we are monitoring.

MEMBER VELSHI: Thank you. Again, very reassuring.

Staff, anything else you want to add?

MR. FRAPPIER: Gerry Frappier, for the record.

We're very happy with the requirements that the CNSC have put in place with respect to aging management. I think they're considered world standard.

We believe that -- or we are sure that Pickering is fully compliant with those aging management requirements, in particular life cycle management plans for all the major components. So, those are reported to us annually in quite detail.

We're concerned, obviously, just about nuclear safety in the context of the major components. We

are confident that they will have inspection programs and surveillance programs that will ensure that there's nothing that's going to be safety significant that gets through. Long before it becomes that, it will become an operational cost problem and they'll have to deal with them.

But, so at this stage, we don't see anything that we would consider a major safety concern.

THE PRESIDENT: Thank you.

Dr. Demeter...?

MEMBER DEMETER: I don't have a final question, but I just have a final comment.

Just to say that I really appreciate the quality of the reports that have come out in the presentations and a special mention for the OPG report, I really like the little explanatory boxes that help describe some of the terms and definitions of what some of the acronyms meant. So, that was very helpful, I think, to help someone who's reading this navigate what all this language and in and outs. I like those explanatory boxes. That's one thing I know that was quite well taken.

So, I don't have any other questions though.

THE PRESIDENT: Thank you.

Mr. Berube...?

MEMBER BERUBE: Just one other question.

It's not really related to the licence itself, but since you've offered the timeline on page 42 of your document, I'm just curious as to what the management decision is to maintain a safe storage state of what, 22 years, according to this timeline? Of course, that could be done sooner. So, is there a reason for the prolonged safe storage timeline?

MR. LOCKWOOD: Randy Lockwood, for the record.

Yes. Specifically, you're asking about the overall until we start dismantling and demolition; correct?

I'll ask Art to speak to that in a few minutes and perhaps Robin might want to add a few comments here.

But at this time, based on benchmarking and looking at around the industry, we've determined that the deferred decommissioning approach is the best way to go.

Overall, we've come to that conclusion for several reasons, but most notably for this audience would be, we think that's the lesser -- sorry, I should say, it would result in lesser impact to the public and to the worker in terms of overall dose that we allow the plant to decay before we start to disassemble it.

Art, is there something else you would like to add?

MR. ROB: Art Rob, for the record.

Yes. Just building on what Randy said, there is certainly a correlation to worker dose with, of course, the deferred strategy. So, employing this particular strategy does allow us to minimize the potential for -- or the opportunity maybe for dose reduction to workers and ease in dismantling.

Other considerations, of course, is that we still actually have fuel base to this plant for a number of years post-shutdown until about 2037 as the fuel has to have time to age in the fuel base before it can be taken to dry storage. So, there is going to be some requirements to maintain certain systems in the building anyway. So, there's less opportunities to actually start demolishing large parts of this plant until some of those other issues -- secondary issues are dealt with.

THE PRESIDENT: Thank you.

Ms Penney...?

MEMBER PENNEY: This question is around the EA, again, and the OPG ERA. So, on page 43 of the CNSC EA it talks about Frenchman's Bay and sediment sampling and contamination as elevated COPCs in sediment samples.

And in both here and in an OPG document it

makes the -- states that:

> "It should be noted that many of the COPCs identified in sediment samples from Frenchman's Bay were not related to the Pickering site, but due to urban stormwater runoff." (As read)

Now, I know that there's an appendix in the ERA that explains this technically, but I'd like to have someone from OPG to explain it to me.

MR. GREGORIS: Steve Gregoris, for the record.

I'm going to ask Jane Borromeo to explain details to yourself.

MS BORROMEO: Jane Borromeo, Section Manager, Environmental Projects.

Yes. So, Frenchman's Bay is the bay that's located to the west of the Pickering site. It has a wetland, it's an enclosed bay with a barrier beach and an opening to the lake, so there is an exchanger plate.

So, within the ERA we do indicate sort of the per cent contribution from the Pickering site. We calculated that value and that value is based on depending on the contaminant.

So, with or without Pickering's contribution, the risk in that bay does not change because

that bay is highly influenced by urban runoff.

MEMBER PENNEY: And I may have not have asked it right, but how did you make that determination? What's the science behind saying that most of the COPCs that are being exceeded there in sediment are not due to Pickering?

MS BORROMEO: Jane Borromeo, for the record.

So for -- so, we sampled both in the outfall of Pickering and we sampled within the bay. We used -- we had a hydrological model which determined the concentration and the contribution to the bay itself.

MEMBER PENNEY: And did you also look at discharges from Pickering?

MS BORROMEO: Jane Borromeo, for the record.

Yes. So, we looked at both effluent discharges in terms of the concentrations and measured values within the environment.

MEMBER PENNEY: Okay. Thank you.

THE PRESIDENT: Thank you.

Dr. Lacroix...?

MEMBER LACROIX: That's all right.

THE PRESIDENT: I have a couple of quickies here that I'm trying to understand.

First of all, somebody said there's no enriched uranium, yet in a couple of places I see enriched uranium, solid painted layer on tubes of fission chambers.

Where is this enriched uranium coming from and how is it controlled, who owns it?

MR. GREGORIS: Steve Gregoris, for the record.

So, fission chambers, the way they operate, they use enriched uranium. That is part of our licence to have those components onsite and those fission chambers are there to measure neutron power on the Pickering 1 and 4 reactors, they measure it on the outside of the reactor cores.

THE PRESIDENT: Well, where did it come from? I mean, we are -- if I understand correctly, we do not enrich uranium in this country, so where did this stuff come from? Maybe I'm misunderstanding what this is.

MR. GREGORIS: So, maybe I'll explain. So, these are components. They're actually -- the fission chambers are measuring flux on the outside of the core, so they use enriched uranium as part of the method to measure the flux and they're supplied from our supplier. I can get you that information, but I don't have the information on who supplies that equipment to us.

THE PRESIDENT: So, it's coming from

outside Canada, is that -- somebody can...? I saw someone walking from CNSC trying to help us, or not?

--- Laughter / Rires

MR. FRAPPIER: Gerry Frappier, for the record.

So, I've mentioned this is part of instrumentation that has been purchased and put on the two reactors, but I'd ask Dave Moroz to provide us with some detail.

THE PRESIDENT: It's part of the possession licence -- in your possession licence it's explicitly mentioned. So, what is it?

MR. JAMMAL: It's Ramzi Jammal, for the record.

Since I started I might as well finish it. This is not part of the fuel, so let me -- it's Ramzi Jammal for the record -- make it very, very clear that there is no enriched uranium that's being burnt in Canada in CANDU, so it's not uranium.

On the waste management side, there are no processing, again, there is no enriched fuel.

This component you are talking about is, as was mentioned, it's integral to a detector and it's been declared with respect to under the safeguard. So, Dr. Moroz is right next to me, he can give you the gory details

with respect to the inspection and the tracking and safeguarding of such very small quantity. I'm not sure if it's prescribed information, but I'm pretty sure Dave has the amount. So, it is very small quantity of enriched uranium used for the purpose of detection and monitoring in the detector.

DR. MOROZ: I'm sorry, David Moroz, for the record.

I don't have much to add, but these are used -- these fission chambers are used. OPG has them, as was stated, on Units 1 and 4 and the IAEA also uses them in fission chambers in its own detection equipment.

The HEU is in very small quantities and it is part of the equipment itself.

THE PRESIDENT: Okay. Every time you see enriched uranium, trust me, it will raise a question. I think you better come up with a better answer than what -- what is it, how it's controlled, where it's come from.

My other quickie is, on page 147, I actually was happy to see the normalized weighting factors for SCA. Everybody on the same page, 147?

So, I assume it's all add up to one. Somebody tell me why it adds to 1.4. What am I missing here? This is staff -- no?

MR. FRAPPIER: Gerry Frappier, for the

record.

So, yes, those are the normalized weighting factors for the safety control areas that we talked about in the past, but I'm not sure who I'm going to look to to explain if it doesn't quite add up to one and it should, as you say, so...

THE PRESIDENT: Okay. Should it? Am I missing anything? In the past --

MR. FRAPPIER: Let me take that out and come back with a proper answer for you.

THE PRESIDENT: Okay. And my final kind of -- I think we need to hear from Environment Canada about their -- you know, there's so many ministries now overseeing this facility from Ministry of Environment of Ontario and the Fisheries and Oceans and we heard from many other.

I'd like to hear from Environment Canada, were you satisfied with, you know, the thermal impact on the lake, the normal kind of thing that we always ask the same question and with some of the emission and impact on the environment?

MS ALI: Okay. Nardia Ali, Environment and Climate Change Canada, for the record.

And we have a very broad mandate, so we're usually involved in the ecological risk assessments. The

areas that touch our mandate, mainly in the area of water quality, species at risk. Those are sort of the main areas we cover. And water quality and discharge.

So, we reviewed sections of Ontario Power Generation's revised Environmental Risk Assessment Report. We're generally satisfied with the conclusion presented by OPG that the risk is low. However, we did -- there were a few areas where we've asked for more information as to how they arrive at their conclusions.

So, one of these is with respect to the whitefish. So, we've asked for -- okay, we've asked them to comment on the whitefish aspect.

MR. KIM: So as far as -- Duck Kim, for the record.

As for the whitefish, as many of -- or maybe not many of the Commission Members know, our primary concern in Lake Ontario is the thermally sensitive species and we've identified round whitefish as a thermally sensitive species of interest.

And the assessment that was provided by OPG is -- we are satisfied with their general conclusions that impacts on round whitefish is likely low. However, the information request that Ms Ali has -- my colleague has mentioned is in reference to data that was provided for other species that potentially might be impacted by the

thermal plume and that is smallmouth bass and emerald shiner.

And the information that was provided was for a five-year period in terms of exceedances of those species' MWATTs, or maximum weekly average temperature thresholds, but it's not clear whether the -- you know, when they -- if there was a cluster of exceedances in one particular summer that might have some impacts on that particular species, because in the discharge channel, water being warmer, these warm water species are attracted and they congregate and they concentrate in the discharge channel.

And, therefore, if we have a large number of exceedances in one particular year versus spread out, you know, one or two over these five years that was reported. So, we're asking for additional detail, precision in that data.

And we're also asking for additional information on stormwater, stormwater management at the site, the point source of the stormwater.

But at this point we're seeking additional information and we can -- we expect that by Day 2 we'll have the additional information and we'll determine what next steps should be at that point.

MS ALI: One thing I wanted to add is that

the Ontario Ministry of Natural Resources and Forestry actually did a study that showed that the population of an area lake wide rather than local, which makes the level of risk from temperature effects even lower. I just wanted to mention that.

And the other area we've asked for more information is with regard to some of the species at risk that are in the site. Like we agree with OPG's conclusions about the level of risk being low, but we just wanted a little more detail on the analysis that were -- on the analyses that were undertaken to arrive at that conclusion.

Another thing we reviewed is, we had one of our meteorologists review the probabilistic risk assessment for high wind and he was generally happy with the assessment done by OPG.

THE PRESIDENT: So, that would be very -- all of this would be very useful for Day 2 to bring some more information to the table.

MS ALI: Okay.

THE PRESIDENT: Did you want to add something?

MR. McALLISTER: It's Andrew McAllister, Director of Environmental Risk Assessment, just to complement Environment and Climate Change Canada's answer.

Back to the study that Ms Ali was

referring to was round whitefish. It was one that was led by the Ontario Ministry of Natural Resources and Forestry, but I want to acknowledge OPG had a large role in the data collection associated with that because one of the things that we were looking at is, you know, are we looking at isolated populations around these nuclear power plants, because these bluff systems seem to be areas of congregation for this species.

And so, there was work done on both sites and some reference sites which, from a genetics perspective, indicated sort of a metapopulation and its building upon the monitoring notion is that, you know, we're comfortable with the thermal risk assessment done to date where there's no indications of unreasonable risk, is not saying now that we're putting our pens down and stopping.

One thing that we did recommend that OPG did accept was to do two additional years of thermal monitoring. So some of these exceedances that we have noted are things that will be rechecked through the monitoring and that monitoring will in turn feed the update of the environmental risk assessment to continue on that sort of cyclical basis.

THE PRESIDENT: Okay. Thank you.

Ms Velshi...?

MEMBER VELSHI: Thank you.

Also some very quick questions and comments.

So staff Slide 31 on the whole-site probabilistic safety assessment, I just wanted to let you know I really like this slide.

A couple of comments and more points of clarification. One is around terminology and I think we have talked about this before, where OPG uses goals and administrative goals and administrative and its limits and targets and whatever and I just hope that we can get consistency. The very fact that it says it's a goal seems it's something one needs to strive for as opposed to something one needs to be less than. So I like the limits and the administrative limits, but something for consideration because it is confusing.

What I wasn't clear is what to expect for day 2 around this. After staff has completed its review, is the expectation that there will be another bar that will show what the all units, all hazards aggregation will be post-IIP? Is that what we can expect for day 2? Which is kind of what OPG shows, but they show it just per unit I think.

MR. FRAPPIER: Gerry Frappier, for the record.

Right now what you can expect for Part 2 is that we will have had a chance to review a bit more in detail the expectations on Pickering A and how it's coming down, as you say, on the per unit basis, so that's for Slide 30, if you like, and we will be able to confirm that we are comfortable with those or whatever.

With respect to Slide 31, which is the whole-site PSA, we did not have at this time any anticipation of making modifications for Part 2. The methodology that has been used is what it is. They accomplished as required by December and we are still sort of evaluating where we think we should go with all that.

MEMBER VELSHI: And again, it's consistent with our expectations, they are above the administrative limits but they have some improvement plans to try to address that. Thank you. So that was on PSA.

And my last one is an opportunity for OPG to comment on the Licence Conditions Handbook. Any comments or your thoughts on that?

MR. MANLEY: Robin Manley, for the record.

As I understand it, the Licence Conditions Handbook is currently in draft. It's based in significant part on our existing Licence Conditions Handbook which we are satisfied with. And we have been providing to CNSC staff our implementation timelines for the variety of

REGDOCs and CSA standards and the like and the CNSC staff presentation spoke to that. So, you know, with respect to the proposed licence conditions that CNSC staff have recommended, we consider those to be acceptable.

I will just check with my colleague Paulina Herrera to see if she has anything to add.

MS HERRERA: Paulina Herrera, Manager of Pickering Regulatory Affairs, for the record.

So I concur with Robin's statement overall. The LCH as drafted, it is consistent with our current LCH and the discussions that we have had with the CNSC staff as well as our submissions. We are conducting a more detailed line by line review and we are anticipating that to be completed prior to Part 2 hearings.

MEMBER VELSHI: Thank you.

THE PRESIDENT: What I didn't hear from you and I expected to hear, it's clear what the requirements are.

MS HERRERA: Paulina Herrera, for the record. Yes, the requirements are clear.

--- Laughter / Rires

MS HERRERA: As always.

THE PRESIDENT: Okay, that's good. We like clarity.

Dr. Demeter...? Mr. Berube...?

Ms Penney...?

MEMBER PENNEY: A quick question for CNSC staff. So looking at the Environmental Assessment Report Regulatory Oversight, page 13 of 82, it talks about OPG submitted on July 13th, 2017, a set of proposed DRLs, derived limits, and environmental action levels to the CNSC that has to conform to CSA N288 and then OPG is going to submit an implementation plan for the new DRLs and requests the Licence Conditions Handbook be updated accordingly. The question to CNSC staff is what is the timeline for that? Is that really -- if they submitted it in July last year, is it really going to mean a revision after the Licence Handbook, if it's approved, is issued?

MR. FRAPPIER: Gerry Frappier, for the record. I would ask Mike Rinker or Andrew McAllister to respond to that.

MS SAUVÉ: So it's Kiza Sauvé, I am the Director of the Health Science and Environmental Compliance Division. So I will start and I will have my colleague Gaétan Latouche to add in anything.

CNSC has done a review of the submission and provided comments back and so as the review continues we expect it to be implemented at the time that we have put into the presentation. But I might be missing your question. Sorry about that.

MEMBER PENNEY: It says that there's going to be a requirement to revise the Licence Conditions Handbook later. Can it not be done -- will it not be incorporated into the Licence Conditions Handbook if it's issued, if the licence is approved?

MR. MANLEY: Robin Manley, for the record. Perhaps I could help a little bit. I'm not sure if this necessarily addresses your issue.

The licence itself once approved by the Commission is relatively unchanging under normal conditions. However, the Licence Conditions Handbook is deliberately a document which CNSC staff can and do update on a regular basis as things evolve. So there is a great deal of flexibility for staff to add new requirements or new DRLs or what have you as time goes on. So that has been an ongoing practice which we have found to be successful for maintaining current conditions and current compliance verification criteria.

THE PRESIDENT: Go ahead.

MS LOVE-TEDJOUTOMO: Lisa Love, for the record. Just a little bit of clarity on that.

In the LCH currently you see the current DRLs as well as the regulatory documents for those DRLs. We have been reviewing a revised set of DRLs according to REGDOC N288.1, the newer version, which I think I have in

guidance, and it's just a matter of time before we put that in. And we are expecting -- it makes a lot of sense to have it start in January of 2019, so it's just a matter of working out the final questions and dotting the i's and crossing the t's and then we will revise -- at that point we will have an LCH that's issued and we will make a revision to that to put the new DRLs into the LCH.

MS SAUVÉ: Kiza Sauvé, for the record.

I'm just going to add one quick -- what Lisa was saying was it makes a lot of sense. One of the reasons it makes a lot of sense to put at the beginning of the year is that when the reporting of effluent and emission against the DRL, when we give you those graphs, it will show the full year, we won't have a changed DRL halfway through the year, which often causes a lot of confusion. So it really helps to have it come in at the beginning of the year as well.

THE PRESIDENT: Dr. Lacroix...?

Ms Velshi...?

MEMBER VELSHI: No, thank you.

THE PRESIDENT: Dr. Demeter...?

Mr. Berube...?

Okay, so I'm going to do the last one and that's really an important one, Slide 59. And staff, you should know better, you have gone to great lengths to be

very clear about what has changed, what is not changing and you are listing 25 CSA standards, 11 new. I know it says examples, but it begs the question where are the missing three?

MR. VIKTOROV: Well, the answer is very simple, there was not enough space to put all the new standards --

THE PRESIDENT: Wrong answer.

--- Laughter / Rires

THE PRESIDENT: Wrong answer. That's why I'm raising it. Please, it always begs the question, so I suggest you don't do that.

Any real serious question?

Okay, I thank you very much, thank you for your patience and we will see you in Part 2, I assume.

--- Whereupon the hearing concluded at 6:22 p.m. /
L'audience se termine à 18 h 22

Canadian Nuclear Safety Commission

Commission canadienne de sûreté nucléaire

Public hearing

Audience publique

June 23, 2026

Le 23 juin 2026

Outaouais Room
140 Promenade du Portage, Phase IV
Gatineau, Quebec

Salle Outaouais
140, promenade du Portage, Phase IV
Gatineau (Québec)

Commission Members present

Mr. Pierre Tremblay
Dr. Timothy Berube
Dr. Marcel Lacroix
Dr. Victoria Remenda
Ms. Andrea Hardie
Mr. Jerry Hopwood
Dr. Alexander McEwan

Commissaires présents

M. Pierre Tremblay
M. Timothy Berube
M. Marcel Lacroix
M^{me} Victoria Remenda
M^{me} Andrea Hardie
M. Jerry Hopwood
D^{r} Alexander McEwan

Commission Registrar :

Mr. Jude Samson

Greffier de la Commission :

M. Jude Samson

Commission Counsel:

Ms. Joanne Gordon

Avocate de la Commission :

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Gatineau, Quebec / Gatineau (Québec)

--- Upon commencing on Tuesday, June 23, 2026
at 9:00 a.m. / L'audience débute le mardi
23 juin 2026 à 9 h 00

Opening Remarks

THE PRESIDENT: All right, good morning and welcome to Part 1 of this public hearing of the Canadian Nuclear Safety Commission to consider Ontario Power Generation's application for the authorization to refurbish the Pickering Nuclear Generating Station and renew its Pickering Nuclear Generating Station and Waste Management Facility licences.

Je vous souhaite la bienvenue and welcome to all those joining us in person for this hearing, as well as those virtually via Zoom or webcast. Je m'appelle Pierre Tremblay. Je suis le président de la Commission canadienne de sûreté nucléaire.

I would like to begin by reaffirming the CNSC's ongoing commitment to upholding our responsibilities towards Indigenous nations and communities. Advancing reconciliation and relationships with Indigenous peoples who have rights and interests in relation to the nuclear sector in Canada is a priority for us. I recognize that

the Commission is conducting this public hearing from Gatineau, Québec, on the unceded territory of the Algonquin Anishinaabeg peoples and that the hearing participants are located in different parts of this country.

I'd like to acknowledge that the Pickering Generating Station and the Pickering Waste Management Facility are located within the lands and waters of the Michi Saagiig Anishinaabeg, the Gunshot Treaty and Williams Treaties, and the Williams Treaties Settlement Agreement, 2018. Pickering site is within the traditional territory of the Michi Saagiig and Chippewa Nations, collectively known as the Williams Treaties First Nations.

Let us pause for a few moments so that each of us can acknowledge the treaty and/or traditional territory on which we're located.

I note that we will not be discussing issues of Indigenous knowledge, engagement, or consultation today. This is because Part 1 of the hearing is for the licensee to discuss its proposed application and for CNSC staff to explain its regulatory review. Part 2 will include Indigenous nations and communities, so we will reserve the conversations about their knowledge, engagement, and consultation until that portion of the hearing, which will be held in the Durham Convention Centre in Ajax, Ontario, during the week of October 5th, 2026.

I will preside over today's hearing and would like to introduce the Members of the Commission who are here with me to consider this application. Commission Member Dr. Alexander McEwan, Commission Member Jerry Hopwood, Commission Member Dr. Victoria Remenda, Commission Member Dr. Marcel Lacroix, Commission Member Andrea Hardie, and Commission Member Dr. Timothy Berube. Jude Samson, the deputy Registrar, and Joanne Gordon, Commission counsel, are also joining us today.

I also want to acknowledge the expertise provided by a number of outside organizations, many of which are represented today at this hearing. Thank you to the following organizations for being present today to take our questions: Fisheries and Oceans Canada, Ministry of Labour, Immigration, Training and Skills Development, and Emergency Management Ontario.

I will now turn the floor to the deputy Registrar for a few opening remarks and explanation of the process.

JUDE SAMSON: Bonjour, et bienvenue à cette audience publique de la Commission canadienne de sûreté nucléaire.

During the hearing, we have simultaneous interpretation. Please keep the pace of your speech relatively slow so that the interpreters are able to keep

up. I would like to remind people participating in person to speak directly into the microphone in front of you. People participating virtually should use a proper headset with a boom microphone or an approved tabletop microphone.

As a courtesy to others, please mute yourself if you're not presenting or answering a question. For those in the room, please also silence your cellphones and other electronic devices.

L'audience est enregistrée et transcrite textuellement; les transcriptions se font dans l'une ou l'autre des langues officielles compte tenu de la langue utilisée par le participant à l'audience publique. To make the transcript as complete and clear as possible, please identify yourself each time before you speak. The transcript will be available upon request to the Registry within one to two weeks.

I would also like to note that this proceeding is being video webcasted live and that archives will be available on our website for at least three months after the close of the proceeding.

Live transcription is also now available with the live webcast of the proceeding. These services are available in both English and French, depending on the feed you are watching. You can access the feature by clicking the small CC or closed captioning button on the webcast.

For those in the room, please note that emergency exits are located at the back of the meeting room and at the side, and washrooms are in the hallway towards the back of the room.

Please refer to the revised agenda CMD 26-H8 for the list of items to be presented today.

All documents presented publicly today are available electronically on the CNSC website or upon request to the Commission Registry. The Notice of Public Hearing and Participant Funding was published on July 21, 2025. A revised notice was posted on January 22, 2026, to announce the hearing locations and change in hearing dates.

Part 2 of the public hearing is scheduled during the week of October 5th, 2026, at the Durham Convention Centre in Ajax. Further details will be posted on the CNSC website at a later date. Public is invited to participate in Part 2 of the public hearing by oral presentation, written submission, or both. Intervenors have until August 12, 2026, to submit their interventions.

During Part 2 of the public hearing, the Commission will consider updated submissions from OPG and CNSC staff along with written and oral submissions from intervenors.

Participant funding has been made available to intervenors to prepare for and participate in Part 2 of

this public hearing. The funding decision made by the independent Funding Review Committee is available on the CNSC website.

I will also note that OPG filed one request for confidentiality as part of this proceeding, and a copy of the confidentiality decision is available on our website.

Today, we'll begin with presentations from OPG and CNSC staff. The President will then lead a question and answer period. During the question period, if you're on Zoom and wish to provide an answer or add a comment, please use the raise hand function.

Président Tremblay, à vous.

THE PRESIDENT: All right, thank you. I'll now turn the floor to Ontario Power Generation for their presentation as outlined in CMDs 26-H7.1 and 26-H7.1A. The floor is yours.

CMD 26-H7.1/26-H7.1A

Oral presentation by Ontario Power Generation

STEVE GREGORIS: Good morning, President Tremblay and Commission Members. For the record, my name is Steve Gregoris, and I'm the chief nuclear officer at Ontario Power Generation. Thank you for the opportunity to present to you today. Next slide, please.

Before we begin, I'd like to add to President Tremblay's territory acknowledgement and acknowledge that OPG's facilities across Ontario are located on the treaty and traditional territories of Indigenous people. The Pickering site is the shared traditional and treaty territory of the Chippewa and the Mississauga Anishinaabe. As a company, we remain committed to fostering positive and mutually beneficial relationships with Indigenous people and communities across Ontario. Next slide please.

OPG's Reconciliation Action Plan, or RAP, was created in 2021 and serves as our road map for how we will work in partnership with Indigenous communities, businesses, and organizations to advance reconciliation. OPG is currently working on a RAP refresh for 2027 that will report out on our progress between 2024 and 2026 and include new actions to reflect our evolving journey to advance reconciliation.

Specific to engagement at the Pickering Nuclear Generating Station and the Pickering Waste Management Facility, OPG makes continuous efforts to engage with Williams Treaties First Nations rights holders as well as with communities that express interest in ongoing and proposed activities at the site. Engagement is ongoing, and we look forward to sharing more about our efforts in Part 2 of this hearing. Next slide, please.

OPG is Ontario's leading electricity supplier, providing safe, reliable, and low-cost power to the grid through a diverse generation portfolio. This includes nuclear stations at Pickering and Darlington, hydroelectric facilities, a solar generating station, a biomass station, and natural gas stations. For decades, OPG has generated electricity for the people of Ontario and remains committed to doing so for many decades to come.

OPG's leadership in nuclear operations is demonstrated through its achievements and recognition in Ontario and internationally. OPG has a strong record of successfully delivering large-scale refurbishment projects to high standards of safety and performance. The Darlington refurbishment project was delivered safely, ahead of schedule, and under budget through disciplined project management, strong oversight and continuous improvement.

This reflects OPG's strong safety culture, technical expertise, and effective risk management. We have demonstrated diligence and a passion for excellence, which we see as our responsibility to the people of Ontario. I also want to recognize our staff for their dedication to safety and operational excellence. Next slide, please.

With me today are Dave Rogers, senior

vice-president at Pickering Nuclear; Luca Ceccato, senior vice-president Nuclear Refurbishment; Kapil Aggarwal, representing Nuclear Sustainability Services; and Sarah Irvine, vice-president, Nuclear Reg Affairs, as well as several other members of the OPG team. We are here in support of OPG's request to renew the Pickering Nuclear Generating Station power reactor operating licence consolidated with the Pickering Waste Management Facility operating licence for the next 10 years.

Nuclear power is critical to meeting the constant and growing electricity demands of Ontario safely, reliably, and cost-effectively without the environmental impacts of greenhouse gas and carbon emissions. Today, nuclear provides the backbone of Ontario's electricity needs, supplying more than 50 per cent of Ontario's power 365 days a year, making it one of the lowest carbon-intensive jurisdictions in the world.

With more than 50 years of experience in safe and reliable nuclear operations, OPG has established a strong record of performance, with Pickering serving as a key contributor to that success. Throughout the current licence term, both the Pickering Station and the Pickering Waste Management Facility have achieved strong performance outcomes, upheld a long-standing safety record, and delivered many significant accomplishments.

These results are a testament to the diligence, professionalism, and commitment to excellence demonstrated daily by personnel across the facilities. OPG's unwavering focus on safety and operational excellence will continue to be a defining strength as the organization maintains its leadership in the industry and supports the safe and reliable operation of Pickering for another 30-plus years.

OPG's licence renewal application demonstrates that OPG continues to be qualified to carry on licensed activities and will make adequate provisions to protect the health, safety, and security of persons and the environment as well as maintain national security and measures required to implement international obligations. Next slide, please.

I would now like to pass the floor to Mr. David Rogers to further discuss our licence renewal application.

DAVID ROGERS: Good morning, President Tremblay and Members of the Commission. For the record, my name is Dave Rogers. I'm the senior vice-president of Pickering Nuclear and the licence holder of the Pickering Nuclear Station. I'm accountable for the safe and reliable operation of the Pickering Nuclear Generating Station and for ensuring OPG continues to meet its regulatory requirements.

As Mr. Gregoris noted, we are here today

requesting the renewal of Pickering Power Reactor operating licence consolidated with the Waste Management Facility licence for the next 10 years, from January 1st, 2027, to December 31st, 2036.

Consolidation of the existing power reactor operating licence and the waste facility operating licence into a single power reactor operating licence is administrative in nature and will not change the licensing basis. OPG will uphold all safety standards and regulatory requirements of both facilities.

The Pickering station has been operating safely and reliably, and OPG continues to invest in station equipment and all employees. The CANDU reactor is a robust technology with multiple safety features and a strong track record of safety and reliability. The Pickering Waste Management Facility has operated safely for 30 years with a strong track record of processing and storing used fuel in dry storage containers.

The application includes Pickering refurbishment project, a multi-year, multi-phase project which includes replacements, upgrades, and safety improvements to Units 5 to 8 as well as the supporting systems, structures, and components.

In parallel with refurbishment and continued operation of Units 5 to 8, OPG will progress the

decommissioning of Units 1 to 4 during the requested licence period, which includes stabilization activities and progresses to storage with surveillance as well as limited non-nuclear dismantling as per our approved detailed decommissioning plan.

OPG's robust safety and operational performance, extensive experience, proven technology, along with the CNSC's robust regulation supports our request for a 10-year licence term for the Pickering station and the Pickering Waste Management Facility. Next slide, please.

Pickering has been a key part of Ontario's electricity system for more than 50 years. The station consists of eight units, four of which are currently in operation. Together, these four units provide approximately 2,100 megawatts of clean power, meeting about 10 per cent of Ontario's electricity demand. Pickering is recognized as a top performing nuclear station and also, in addition to generating safe, reliable electricity, Pickering plays an important role in producing 20 per cent of North America's supply and 10 per cent of the world's supply of cobalt-60, a critical medical isotope used to sterilize single-use medical equipment and supporting food safety.

Recent examples of Pickering's strong performance and recognition include: receiving Electricity

Canada's President's Award of Excellence for employee safety, recognizing OPG's top decile performance in both total recordable injury frequency and lost-time injury severity rates, underscoring OPG's long-standing commitment to workplace health and safety; number two, maintaining Wildlife Habitat Council Gold Certification, demonstrating OPG's commitment to environmental stewardship; and receiving international recognition in 2025 for Pickering's ability to operate at the highest level of performance, marking the second consecutive time the station has achieved this distinction. Next slide, please.

Pictured here is the northeast-facing view of Pickering site. On the left is Units 1 through 4. Units 2 and 3 are in safe storage and will remain in this state. Units 1 and 4 ceased operation in 2024 and safe storage activities are ongoing.

On the right comprise Units 5 through 8. These units are licensed to operate until the end of 2026. The plan is for Units 5 to 8 to shut down in September 2026.

The Pickering Waste Management Facility is composed of two sites, Phase 1 and Phase 2. Phase 1 is located within the protected area of Pickering Station, southeast of Unit 8, and consists of a dry storage container processing building and two dry storage container storage buildings. Phase 2 is located northeast of Phase 1

and is contained within its own security protected area within the boundary of the Pickering site. This site contains two dry storage buildings.

Next slide, please.

I'd like to pass the floor to Mr. Luca Ceccato to provide more details on the licences and refurbishment activities requested in OPG's Application.

LUCA CECCATO: Good morning. For the record, my name is Luca Ceccato. I am the Senior Vice-President, Nuclear Refurbishment at Ontario Power Generation. I am responsible for delivering this major project safely, on time and on budget, with a strong emphasis on quality and safety.

Next slide, please.

The proposed refurbishment at Pickering has been recognized as both Canada's largest clean energy project and Canada's largest infrastructure project. This $26.8 billion investment will significantly grow Ontario's GDP by an estimated 38.2 billion over the coming decades of station operation, with 17 billion of GDP growth linked to the refurbishment project itself. This will create thousands of highly skilled jobs for Ontarians, retaining and growing nuclear power generation and infrastructure building expertise in Ontario.

The station will continue powering about two

million homes and businesses safely and reliably. Refurbishment planning efforts include reviews of operating experience and lessons learned from OPG's Darlington refurbishment and hydro projects as well as other nuclear refurbishments.

To facilitate the sharing of information, skills, expertise and insights across projects and teams, staff placements from the Darlington refurbishment project and knowledge transfer sessions have been introduced. These sessions are designed to ensure essential knowledge is effectively communicated and preserved within the organization.

Next slide, please.

OPG has a proven track record of successfully conducting refurbishment activities at Darlington Station. Refurbishment enables the removal and replacement of key components of each reactor and associated equipment. OPG has engaged in procurement and prerequisite activities to ensure readiness to proceed pending regulatory approvals and completion of planning activities.

Refurbishment at Pickering will begin with the defuelings of Unit 5 and 7 near the end of 2026, where teams will prepare for refurbishment activities to begin in early 2027. Defueling of Unit 6 and 8 is expected to start in 2027, with refurbishment activities commencing in

mid-2027. Staggering resources between units will enable Unit 5 to return to service in 2031. Lessons learned will be used to return Units 6, 7 and 8 to service in 2032, '33 and '34, respectively.

Next slide, please.

The Pickering refurbishment project will include six bundles: retube feeder and boiler feeder replacement; boiler replacement; turbine generator; deep-water intake; balance of plant, and facilities and infrastructure. We'll go through each bundle in more detail in the next slides.

Next slide, please.

The retube and feeder replacement bundle involves the replacement of 760 feeder tubes and 380 fuel channels assemblies in each reactor. Other essential components supporting these two major assemblies will also be replaced, including end fittings, feeder cabinet insulation and instrumentation.

Lessons learned from past refurbishments will be used to ensure we minimize the likelihood of contamination, and comprehensive inspections will be performed before and after installation to ensure the integrity of all replaced components.

Next slide, please.

The boiler replacement bundle will support the

removal and replacement of all 48 boilers across Units 5 to 8, with 12 boilers being in each unit. Each boiler is to be lifted through an engineered access point at the top of each reactor building dome using an external crane. The installation of the new boilers will follow through the same access point. After the replacement of the boilers, engineered access points will be restored.

Next slide, please.

The turbine generator bundle involves replacing and upgrading critical components to ensure reliable, efficient and modernized operation. This includes the replacement of both high and low pressure turbine spindles as well as the moisture separator and reheat systems. Turbine control systems upgrade include modernization of turbine protection and control systems to increase system reliability and alignment with industry standards.

In addition, rotor and stator rewinds, stator core, generator cooling system and the excitation system will be replaced. This work will result in a higher output of approximately 30 megawatts per unit by increasing the efficiency of the turbine equipment.

Next slide, please.

The construction of the deep-water intake, similar in concept to the intake at the Darlington Station, is proposed at Pickering during the requested licence

period. Drawing water from deep below the surface of the lake ensures a stable supply of clean, cold water to the forebay. The installation is expected to improve thermal efficiency and support for increased electrical power output. Engineering design, construction, and infrastructure planning are ongoing.

While OPG's preferred option is to complete the intake project during the next licence period, the existing surface water intake remains a safe and viable option for supplying cooling water at Pickering. The new intake is not required to meet any regulatory requirements, and no part of the new intake is credited in deterministic or probabilistic safety analysis. It is an enhancement only and does not constitute a safety improvement. OPG will confirm the feasibility of completing the deep-water intake before Part 2 of this hearing.

Next slide, please.

The balance of plant bundle includes two main areas of work to enhance both component and overall system reliability and to support the extended operational life of the station. The first is the modifications to conventional electrical and nuclear systems, including detailed inspections of systems and components related to emergency coolant injection and emergency water supply. The second includes specialized projects such as upgrades

to the digital control computers, improvements to fire protection systems, and upgrades to the Pickering training simulator to ensure it is identical to the station's main control room.

Next slide, please.

The facilities and infrastructure bundle includes work currently in progress that will be completed by the end of 2026 in preparation for the refurbishment. Activities include construction of new buildings to accommodate growing needs and improve functionality, upgrades to existing infrastructure to meet current standards and requirements, optimizing space allocation for both personnel and material to improve efficiency and productivity, and enhancing site logistics to ensure smooth operation and minimal disruptions such as shuttle bus options and new parking lots.

Next slide, please.

I would like to pass the floor to Ms. Sara Irvine to provide an update on periodic safety.

SARA IRVINE: Good morning. For the record, my name is Sara Irvine, and I am the Vice-President of Nuclear Regulatory Affairs.

To support safe operation over the next operating period, OPG is conducting a comprehensive periodic safety review, or PSR, which evaluates the design,

condition and operation of the station against modern codes and standards. This is a safety assessment encompassing the cumulative effects of station aging and modifications, operating experience and technical developments to ensure a continued high level of safety throughout the station's operating lifetime.

PSR3 builds upon previous safety assessments: the Pickering Integrated Safety Review, or ISR, PSR2 and PSR2 Bravo, the Darlington ISR and the Darlington PSR, for programmatic components applicable to Pickering.

The most recent Pickering safety reassessment, PSR2 Bravo, ensured continued safe operation of the Pickering Station to the end of December 2026, while the objective of PSR3 is to assure safe operation of the refurbished Units 5 through 8 for an additional 30-plus years. The safety factor reports, completed in May 2025, reviewed OPG programs and procedures identified in the *Licence Conditions Handbook*, along with previous regulatory commitments and actions. They also considered previously identified gaps from earlier PSR and ISR assessments.

The Global Assessment Report, expected in Quarter 3 of 2026, will present the PSR results and evaluate the station's overall defense in depth. It will document a ranked list of global issues and will include OPG's overall assessment of the station's continued safe

operation.

The Integrated Implementation Plan, or IIP, is expected in Quarter 3 of 2027, and it will document the actions and schedule for the IIP. It will include a detailed list of safety enhancement initiatives and their implementation dates. The PSR3 IIP will be submitted to the CNSC for acceptance in accordance with REGDOC 2.3.3.

The periodic safety review will be completed before any of the units are returned to service post-refurbishment, and the timeline for completion of IIP actions are subject to acceptance by CNSC Staff.

Next slide, please.

The Global Assessment Report currently under development has concluded the following. The PSR3 confirms the Pickering Station has operated safely and reliably, with multiple areas exceeding modern codes and standards and effective barriers maintained across all levels of defence in depth.

The global assessment process identified seven strengths in a wide range of areas where the Pickering station exceeds modern requirements. The identification of these strengths is consistent with results from recent trends of key performance indicators which demonstrate the station has been operated in a safe and reliable manner.

The draft global assessment identified the

following scope opportunities. Check and replace key equipment and improve structural monitoring as needed. Update equipment qualification records and maintenance plans after refurbishment. Strengthen aging management plans for long-term operation and equipment obsolescence. For unreplaced nuclear Class 1 components, reaffirm the service life for extended operation. Reaffirmation that used fuel storage remains adequate for extended operation.

Reduce single-point failure risks by adding redundancy and reviewing other vulnerabilities. And finally, improvement to fire protection systems to address code gaps with practical upgrades.

Next slide, please.

OPG has also identified safety improvement opportunities to enhance nuclear safety as demonstrated by safety analysis. For core cooling, there are three opportunities.

Firewater makeup will be enhanced with an alternate water supply for Pickering Units 5 to 8 heat transport system and the moderator system. The emergency water supply system will be upgraded with additional flow paths and increased system reliability to increase core cooling capabilities. And for heat sink diversity, new sources of water to the core will be installed through dedicated pumps and firewater connections.

For enhancements in containment, there will be a new containment filtered venting system, a system dedicated to providing a filtered relief pathway, mitigating containment overpressure and reducing airborne contaminant releases in the event of a beyond design basis accident.

Relief capacity will be installed on the end shield and calandria vault to improve margin. And for external hazard protection, the Unit 7 control equipment room exterior wall will be strengthened to better resist debris carried by strong winds.

These safety improvement opportunities and the major component replacements will be key inputs to the development of the IIP scope.

Next slide, please.

I would now like to pass the floor to Mr. Kapil Agarwal.

KAPIL AGGARWAL: Good morning. For the record, my name is Kapil Agarwal, and I represent the Nuclear Sustainability Services Organization, which is responsible for OPG's nuclear sustainability activities, including oversight of waste management and decommissioning planning and operations within OPG's nuclear waste division.

Alongside the refurbishment and continued operation of Pickering Units 5 to 8, OPG will continue to

transfer used fuel into dry storage containers and progress decommissioning activities for Units 1 to 4.

Next slide, please.

For more than 50 years, OPG has been a world leader and innovator in nuclear waste management, including transporting, processing and interim storage. Our proven programs and resources will be leveraged to prevent, safely manage and harness the waste and byproducts associated with the station's operations over the proposed licence term.

The facility processes and stores used fuel generated at the Pickering Station in dry storage containers, or DSCs. To support on-site interim storage of future DSCs from the current and extended operation following refurbishment, a fifth used fuel dry storage building is planned to be in service in 2027. We are also constructing the Pickering component storage structure to support the on-site interim storage of low and intermediate level waste generated from refurbishment and decommissioning activities.

Next slide, please.

During the proposed licensing period, OPG will proceed with select non-nuclear dismantling included in the CNSC-accepted detailed decommissioning plan. The plan includes the removal of outbuildings and some non-nuclear components.

OPG plans to continue managing all decommissioned units safely and responsibly throughout and beyond the licensing period, ensuring the ongoing protection of workers, the public and the environment.

Next slide, please.

I would now like to pass the floor back to Mr. Dave Rogers.

DAVID ROGERS: Dave Rogers, for the record.

OPG maintains strong performance across all safety and control areas. For the purpose of this presentation, I will highlight a select few to illustrate how we continually ensure and strengthen safety.

Next slide, please.

Safety analysis. OPG's safety analysis program demonstrates the overall safety of the station and the fundamental safety functions to control reactor power, cool the fuel, and contain or limit any unlikely releases from the station. The program confirms the station meets safety requirements and demonstrates compliance with regulatory limits.

The safety analysis serves as a basis to establish the safe operating envelope for the station. OPG updates the Pickering probabilistic safety assessment every five years. The latest assessment concluded the safety goals were met with adequate margin and the overall risk to

the public from the station's operation is low.

During refurbishment, the station configuration will change, and OPG will perform reactor safety assessments and safety analysis throughout the project to ensure the continued safety of the station.

Operations at the Pickering Waste Management Facility is also supported by a comprehensive safety analysis, which is reviewed every five years and updated as required to reflect changes in site layout, operational experience and information supporting the assumptions made in the assessments.

The 2023 safety report demonstrates the facility continues to operate within regulatory limits.

Next slide, please.

Operating performance. Our operations program at both Pickering Station and Waste Management Facility focuses on health, safety, security, and radiation and environmental protection, ensuring licensed activities are carried out in accordance with regulatory requirements, ensuring safety is maintained. Through continued investment in equipment, training, personnel, station reliability continues to be among the best in the industry. Continuous improvement is central to OPG's approach, fostering a strong learning culture by applying operating experience, lessons learned and best practices to support

sustained safety and operational excellence.

In addition, operations oversight has been built into the refurbishment preparations to ensure safety considerations are addressed and activities are planned and executed with appropriate oversight.

Next slide, please.

Conventional health and safety. The safety of the public and workers is of utmost importance to OPG. We put our people first by taking proactive steps and continuously improving to prevent incidences before they occur.

OPG applies a comprehensive and safety effective management system to protect all personnel. The Employee Health and Safety Policy outlines OPG's approach and commitments to health and safety, including the requirements and accountabilities of all employees. This policy is implemented through OPG's Environment, Health and Safety Management Systems program, which manages workplace risks and supports a healthy, injury-free workplace.

Refurbishment and decommissioning activities will be carried out in accordance with this program, which applies to both OPG employees and contract partners. Contract staff will be trained and qualified to the same standards expected of OPG personnel. OPG recognizes that strong oversight is essential to successful project

execution. A dedicated health and safety team will provide daily support and oversight to ensure contractors meet OPG's safety expectations.

Next slide, please.

Radiation protection. Pickering's Radiation Protection program is designed to protect the health and safety of persons by keeping radiation doses well below regulatory limits and optimizing them to be as low as reasonably achievable, or ALARA.

Throughout the licence term, collective and individual doses were managed well below administrative and regulatory dose limits at the Pickering Station and at the Pickering Waste Management Facility. This is due to a number of factors, including strong equipment reliability, low unit forced loss rate, and implementation of dose reduction initiatives in work planning.

In 2024, OPG received the Canadian Nuclear Society Innovation Achievement Award for developing and implementing a groundbreaking shielding process using additive manufacturing, minimizing radiological exposure to workers. In the same year, OPG was also recognized by Electricity Canada with the Centre of Excellence Awards for its 3D laser scanning program.

Refurbishment and decommissioning activities will follow OPG's radiation protection program and related

procedures. We are incorporating lessons learned from the Darlington Refurbishment Project. This includes ensuring radiological safety is considered in design changes, including steam generator removal and zoning changes. At the Pickering Waste Management Facility, dose monitoring plans for future buildings will be established within this program.

OPG strives for excellence in radiation protection through a focus on continuous improvement and implementation of enhancements based on industry standards and internal assessments.

Next slide please.

Environmental Protection. OPG is committed to protecting the environment now and for generations to come. Our comprehensive environmental programs aim to minimize potential impacts from the station's operations on human health and environment through multiple barriers to control and minimize emissions along with ongoing monitoring. Our program regularly samples municipal drinking water, air, and soil and we post these results publicly on opg.com.

OPG updates the environmental risk assessment of the Pickering Nuclear site at least once every five years, consistent with REGDOC 2.9.1. The 2022 assessment confirmed the Pickering Station and Waste Management Facilities continue to operate in a manner that is

protective of the health and safety of the public and environment.

OPG prepared a Predictive Environmental Risk Assessment, or PERA, for the refurbishment and continued operations of units 5 to 8, the decommissioning of units 1 to 4, and the project activities associated with the Pickering Waste Management Facility.

The predictive PERA concluded that most activities are not likely to result in adverse effects to human or ecological receptor groups evaluated. Some aquatic habitat will be disturbed or removed, and while some individual exceedances of air quality or noise guidelines were identified, monitoring and mitigation efforts will be implemented to confirm no adverse effects.

OPG conducted climate change resilience assessments for the Pickering Station and the Pickering Waste Management Facility. These assessments confirmed that both facilities can safely operate under projected future climate conditions with no impacts to nuclear safety.

Next slide please.

Emergency management and fire protection. The Provincial Nuclear Emergency Response Plan establishes a framework of response for the unlikely event of a nuclear emergency in Ontario. The plan outlines roles and

responsibilities of all organizations and we take pride in the strength of our relationships with these partners.

Under this plan, OPG has a duty to safely manage our nuclear stations to protect the public, the environment, our staff, and to provide timely and accurate notifications to interfacing organizations.

The Nuclear Emergency Preparedness Program is documented in the Consolidated Nuclear Emergency Plan. This plan ensures OPG has adequate provisions for preparedness and response capabilities, provides a framework for interaction with external authorities and defines how OPG's commitments under the Provincial Nuclear Emergency Plan are implemented.

Emergency preparedness is regularly practiced through frequent drills and exercises. These drills are designed to validate response procedures, strengthen coordination with external partners and ensure staff readiness under realistic conditions.

Next slide please.

Community engagement. Beyond the safety and control areas I've outlined, at OPG we know our role in the community is about much more than safely generating electricity. It's about being a good neighbour. Our relationships within the community are important to us.

Engagement with key stakeholders and the public

is ongoing and we look forward to sharing more about those engagement efforts in part two of this hearing.

Next slide, please.

In summary, OPG continues to demonstrate strong performance in all safety and control areas and has had many significant achievements during the current licence term. At OPG, we are committed to operating our stations safely and to the highest levels of performance. Our workforce is highly qualified, engaged, and motivated, and we prioritize safety above all else.

We hold ourselves accountable to the highest standards and are committed to continuous improvement through comprehensive programs that are aligned with industry best practices to drive year-over-year improvement.

OPG values the relationships it has built with our stakeholders, including host communities and members of the public, and is committed to continued meaningful collaboration and engagement with Indigenous Nations and communities across OPG's operations.

Thank you for the opportunity to present to you today. We look forward to answering any questions you may have.

THE PRESIDENT: All right, thank you for that.

We will move directly to the presentation by

CNSC Staff as outlined in CMDs 26-H7 and 26-H7.A. So please proceed.

CMD 26-H7/26-H7.A

Oral presentation by CNSC staff

DR. ALEXANDRE VIKTOROV: Good morning, President Tremblay, members of the Commission, and everyone who participates in this meeting.

For the record, my name is Dr. Alex Viktorov. I am the Director General of the Directorate of Power Reactor Regulation.

I'm joined today by Ross Richardson and Daniel MacDonald from the Pickering Regulatory Program Division, and other CNSC staff will participate as well, either in person or virtually.

We are here today to provide the Commission with information regarding OPG's application to renew operating licence of the Pickering Nuclear Generating Station and Pickering Waste Management Facility.

Our presentation today will summarize the information provided in CNSC staff Commission Member Document CMD 26-H7.

As President Tremblay stated in the opening remarks, the focus of Part 1 hearing is on the technical

aspects of OPG's application and CNSC staff assessments.

Information pertaining to consultation and engagement with Indigenous nations and communities, as well as issues raised in public interventions, will be discussed during Part 2 of this hearing scheduled for October 2026.

Next slide.

In today's presentation, CNSC staff will provide an overview of the Pickering Nuclear site and the purpose of this meeting. An outline of the scope of OPG's application, key findings from CNSC staff assessment of the said application, including those concerning the proposed refurbishment of Units 5 to 8, decommissioning of Units one to 1 to 4, and operation of the Pickering Waste Management Facility.

We also provide a summary of OPG's performance during the current licensing period and share information regarding CNSC public engagement and wrap up with staff overall conclusions and recommendations.

And now I'll pass the presentation over to my colleague Ross Richardson.

ROSS RICHARDSON: Thank you, Alex. For the record, my name is Ross Richardson, and I'm the Director of the Pickering Regulatory Program Division.

So in this section, I will provide a brief introduction to the Pickering Nuclear site and the matter

before the Commission today.

So the Pickering Nuclear site consists of the Pickering Nuclear Generating Station, or Pickering NGS; and the Pickering Waste Management Facility, or Pickering WMF. The Pickering NGS is comprised of eight CANDU power reactor units, along with various supporting structures and equipment.

Units 1 to 4 came online between 1971 and 1973 and are permanently shut down. Units 2 and 3 have been in a safe storage state since 2010, and Units 1 and 4 were shut down in 2024 and have been defuelled and dewatered and are transitioning to safe storage. Units 5 to 8 first came online between 1983 and 1986 and are currently operating.

The Pickering Waste Management Facility is also located on the Pickering Nuclear site. It includes two locations, Phase 1 and Phase 2, that store nuclear waste produced at the Pickering NGS. Used nuclear fuel held in dry storage containers, or DSCs, is handled at a processing building and stored in various storage buildings across Phase 1 and 2.

There are currently four DSC storage buildings in operation, with a fifth currently under construction at Phase 2. Intermediate-level waste in the form of components from past Pickering A retube activities is also stored at the Pickering Waste Management Facility in dry

storage modules.

So the Pickering NGS and WMF are currently licensed under separate operating licences. The current power reactor operating licence for the Pickering NGS authorizes the operation of Units 5 to 8 until the end of 2026, followed by the transition of all units to a safe storage state. And the current waste facility operating licence authorizes the construction of up to six total dry storage container storage buildings and the construction of the Pickering Component Storage Structure, or PCSS. And both licences are valid until August 2028.

And so the purpose of today's hearing is to consider OPG's application to renew the operating licences of the Pickering NGS and WMF for a 10-year period and to consolidate these licences into a single operating licence. And OPG has applied for early licence renewal to align with the schedule of the proposed Pickering refurbishment project.

And so in this section I will provide some highlights from OPG's application.

OPG's application makes several requests to the Commission, including authorization to continue to operate the Pickering NGS Units 5 to 8 following refurbishment, early renewal of the operating licences for the Pickering NGS and WMF for a 10-year period from January 1, 2027 until

December 31, 2036, consolidation of the operating licences for the Pickering NGS and WMF into a single power reactor operating licence, inclusion of a licensed activity related to decommissioning of Units 1 to 4, and authorization to increase the maximum storage capacity of the currently under-construction Storage Building 5 at the Pickering WMF from 1,200 to 1,410 dry storage containers.

OPG has also provided additional documents to support its application. The Predictive Environmental Risk Assessment, or PERA, is a future-looking predictive evaluation of the environmental risks associated with the planned activities at the Pickering nuclear site. These include operating the Pickering NGS and WMF, decommissioning Units 1 to 4, and refurbishing Units 5 to 8. The climate change resilience assessments identify potential hazards to the Pickering NGS and WMF due to a changing climate, as well as future detailed assessments and mitigations that may be necessary. And finally, OPG's safety assessment for the Pickering Waste Management Facility provides a technical safety basis for storing additional dry storage containers in Storage Building number 5.

And so this section provides some of the high-level findings of CNSC staff's assessment of OPG's application and supporting documents.

CNSC staff assessed OPG's licence renewal application against the requirements of the applicable regulations, including the *General Nuclear Safety and Control Regulations* and the *Class I Nuclear Facilities Regulations.* And CNSC staff are satisfied that OPG's application and associated supporting materials are sufficient and meet CNSC regulatory requirements.

CNSC staff have also conducted detailed technical assessments of OPG's application and supporting documents and have found that OPG has demonstrated that it has suitable programs and processes in place to safely refurbish and return to service Pickering Units 5 to 8, continue decommissioning work on Units 1 to 4, and continue operating the Pickering NGS and WMF.

OPG has also demonstrated through the PERA and climate change resilience assessments that the proposed activities are not predicted to result in adverse effects to people or the environment. Any anticipated impacts will be mitigated through mitigation measures and monitoring, and OPG has adequately considered the possible impacts of climate change on the Pickering nuclear site.

I will now pass the presentation over to Daniel McDonald for further details on CNSC staff's assessment of the activities proposed in OPG's application.

DANIEL MacDONALD: Thank you, Ross.

For the record, my name is Daniel McDonald, Senior Regulatory Program Officer in the Pickering Regulatory Program Division.

In this section, we will cover OPG's proposed refurbishment of Units 5 to 8.

OPG's application and supporting documents provided an overview of the planned scope for the proposed refurbishment of Pickering NGS Units 5 to 8. This includes replacing major reactor components, such as the fuel channels, calandria tubes, feeder pipes, and steam generators. OPG intends to remove all steam generators through engineered openings constructed in the roof of each unit, which would then be sealed after completion of the work.

OPG is also planning to construct a deep-water intake to supply cooling to the station from deeper in the lake.

OPG's proposed refurbishment project also includes, but is not limited to, replacements, upgrades, maintenance, and improvements to many additional structures, systems, and components, including the turbine generator system, digital control computers, standby generators, and more, as shown on this slide.

OPG's proposed deep-water intake, or DWI, is a major new addition planned as part of the refurbishment

project. The current preliminary DWI design is a large, 1.5 kilometre long, 10-metre-wide tunnel out into Lake Ontario.

OPG's application and supporting documentation note several expected improvements of a DWI, including improved cooling performance for the Pickering NGS, particularly during hot or cold weather; reduced fish impingement; and enhanced resilience to climate change, such as resilience to algae blooms.

OPG's PERA considers the DWI construction and notes mitigation and monitoring measures that will reduce the anticipated impacts, such as erosion control and the replacement of disturbed aquatic habitat features.

Since it would be constructed within and beneath Lake Ontario, construction of the deep-water intake requires separate authorization from Fisheries and Oceans Canada. CNSC staff have and will continue to engage with Fisheries and Oceans Canada to share relevant information pertaining to the deep-water intake project.

OPG is currently conducting a periodic safety review, or PSR, to support the proposed refurbishment and continued operation of Units 5 to 8. This new PSR is called PSR 3 and is being conducted in accordance with a CNSC staff-accepted PSR basis document.

A PSR is a comprehensive evaluation of the

design, condition and operation of a nuclear power plant against modern codes and standards. OPG has submitted 15 15 PSR3 safety factor reports to CNSC staff, which CNSC staff found were conducted in accordance with the PSR basis document. OPG is currently working to submit the Global Assessment Report, or GAR, by August 2026, and the Final Integrated Implementation Plan, or IIP, is expected to be submitted to CNSC staff for acceptance by August 2027. CNSC staff are satisfied with these proposed timelines.

The GAR and IIP capture the results of the PSR. The GAR presents the findings, overall conclusions, and any identified safety improvements that will be considered. The IIP documents the proposed safety improvements committed to by the licensee, as well as the timeline for implementation. In accordance with REGDOC-2.3.3, OPG is required to submit the IIP to CNSC staff for review and acceptance. OPG has committed to post summaries of these documents on its website when available. During the proposed licence period, CNSC staff will verify that all completion criteria have been met prior to closing any IIP commitments.

Requirements related to the PSR are captured under the new proposed Licence Condition 3.4, which requires OPG to conduct a PSR at least every 10 years and implement the IIP. Additionally, during the proposed

return to service of units from refurbishment, the verification that related IIP actions are complete will be a prerequisite to the release of any associated regulatory hold points. OPG will be required to submit annual status reports on the IIP during the proposed licence period, as well as notify CNSC staff of any changes to the IIP. CNSC staff will keep the Commission updated on the status of OPG's PSR and IIP as part of the status report on power reactors that is presented at public meetings of the Commission.

CNSC staff are satisfied that all elements of the PSR submitted to date have met regulatory requirements and will verify that OPG's GAR and IIP are conducted in accordance with regulatory requirements and according to the committed timelines.

To conduct the planning, preparation, and execution of refurbishment work, OPG has established a Pickering Refurbishment Program under the Pickering Nuclear Management System. This program includes the use of engineer, procure, construct (or EPC) contractors, who will perform the majority of the refurbishment work. OPG's existing health and safety programs apply to all workers at the Pickering NGS and WMF, including these EPC contractors.

OPG's refurbishment program also incorporates operating experience and lessons learned from other

refurbishment and construction projects, such as the Darlington NGS refurbishment and construction of the Darlington New Nuclear Project. CNSC staff have reviewed the governance documentation underpinning the refurbishment program and are satisfied that OPG's refurbishment program meets regulatory requirements.

Licence Condition 2.4 of the current and proposed operating licence for the Pickering NGS requires OPG to implement and maintain a certification program in accordance with REGDOC-2.2.3. Following the shutdown and defueling of units 5 to 8, all certified operations staff at the Pickering NGS will be moved to inactive status. This is because the workers will be unable to meet minimum shift work requirements while the units are shut down. To reinstate inactive certified staff, REGDOC-2.2.3 defines several requirements, including simulator-based training and a period of work under supervision, where returning certified staff work with active certified staff to confirm that they can perform their duties competently and safely. Since all certified workers will be in inactive status when Unit 5 would return to service from the proposed refurbishment, OPG will be unable to meet work under supervision reinstatement requirements and has proposed an alternative approach.

OPG has proposed to verify that workers can

safely perform their duties during the guaranteed shutdown state of Unit 5 instead of during a period of work under supervision. In the guaranteed shutdown state, the reactors are not powered and no credible design basis accident scenarios exist. Proceeding beyond the guaranteed shutdown state would require the release of a CNSC regulatory hold point, which we will discuss later in this presentation.

CNSC staff are satisfied that OPG's proposal is adequate and does not introduce any additional risk to the Pickering NGS. Therefore, CNSC staff recommend that the Commission accept OPG's proposal for the return to service of Unit 5. Once Unit 5 has returned to service, OPG will again be able to meet the work under supervision requirements for subsequent reinstatements.

CNSC staff on-site inspectors will continue to inspect OPG's conduct during the proposed refurbishment project. This will include verifying that all refurbishment work is performed in accordance with regulatory requirements. In addition, OPG will submit documents and reports during the refurbishment project detailing project execution and milestones. CNSC technical specialists will review and assess these submissions and verify that all applicable regulatory requirements are met. CNSC staff will further verify that all prerequisites are

met and complete prior to recommending the release of any regulatory hold points.

CNSC staff are recommending that the Commission include four regulatory hold points during the return to service of each unit from refurbishment. These four recommended hold points are similar to those used for the Darlington NGS refurbishment project, and are, (1), prior to loading fuel in the reactor unit, (2), prior to removing the guaranteed shutdown state, (3), prior to exceeding 1 per cent full power, and (4), prior to exceeding 65 per cent full power. Licence Condition 15.2 of the proposed operating licence requires Commission authorization to remove each hold point, and the proposed *Licence Conditions Handbook* describes the necessary prerequisites.

CNSC staff recommend that the Commission delegate the authority to verify that the necessary prerequisites for the release of each hold point have been met to CNSC's Executive Vice-President and Chief Regulatory Operations Officer and to the Director General of the Directorate of Power Reactor Regulation. Should the Commission accept this delegation, either of the noted staff would be authorized to verify that the conditions have been met for release of the regulatory hold point once OPG has demonstrated completion of all required prerequisites.

On the next slides, I will provide some information on OPG's ongoing decommissioning work associated with Units 1 to 4.

OPG's application included a copy of the Detailed Decommissioning Plan, or DDP, for Pickering Units 1 to 4. OPG submitted the Detailed Decommissioning Plan and Storage with Surveillance Plan to CNSC staff for acceptance in December 2024. The Detailed Decommissioning Plan builds on the previous Preliminary Decommissioning Plan and describes activities limited to the removal of outbuildings and non-nuclear components that are no longer necessary to support the safe operation of the facility.

CNSC staff accepted OPG’s DDP package in June 2025, and OPG commenced some activities described in this Detailed Decommissioning Plan in September 2025. OPG continues to provide prior notice of any work which CNSC staff review to ensure such work is in accordance with the accepted DDP.

OPG is also currently transitioning Units 1 and 4, which shut down in 2024, to storage with surveillance under an existing stabilization activities plan.

OPG is required to update the DDP every five years, and each revision is focused on the next five years of planned work.

During the proposed licence period, CNSC staff

will continue to verify that OPG's ongoing decommissioning work is conducted in accordance with regulatory requirements. CNSC staff are recommending that the Commission include a licensed activity in the proposed operating licence pertaining to the decommissioning of Units 1 to 4 for the purpose of storage with surveillance up to and not beyond the removal of outbuildings and non-nuclear components. Decommissioning of nuclear structures, systems, or components which is beyond the scope of the current DDP would require the future authorization of the Commission.

In this section, we will cover CNSC staff's assessments of OPG's application pertaining to the Pickering Waste Management Facility.

OPG is currently constructing the Pickering Component Storage Structure, or PCSS, and a fifth DSC storage building at the Pickering WMF. Storage Building 5 is a double-capacity building that combines the originally planned fifth and sixth storage buildings. Therefore, it is considered to satisfy the existing Commission-authorized limit of six storage buildings at the Pickering Waste Management Facility. CNSC staff are recommending minor amendments in the proposed operating licence to clarify that no further storage buildings have been authorized.

In its application, OPG requested an increase

to the Commission-authorized DSC storage limit in Storage Building 5. CNSC staff found that OPG's supporting safety assessments met regulatory requirements and are satisfied that OPG has measures in place to safely store up to the requested 1,410 dry storage containers. Continued operation of the Pickering Waste Management Facility during the proposed licence is captured under the proposed consolidated operating licence.

In this section, I will provide an overview of CNSC staff's assessment of OPG's performance at the Pickering NGS and WMF during the current licence period.

CNSC staff assessed OPG's performance during the current licence period, as well as the anticipated impacts of planned activities during the proposed licence period, with respect to the CNSC's 14 Safety and Control Areas, or SCAs. CNSC staff continuously verify OPG's compliance by using a variety of activities, including reviews and assessments of submissions and event reports, inspections, and compliance assessments.

CNSC staff are satisfied with OPG's performance at the Pickering NGS and WMF and are satisfied that OPG remains qualified to perform the planned activities during the proposed licence period. The following slides present highlights from a selection of SCAs. Details on all 14 SCAs are provided in CNSC staff's CMD.

OPG has maintained satisfactory performance in the operating performance SCA at the Pickering Nuclear Generating Station and Waste Management Facility during the licence period. In advance of submitting its application, CNSC staff requested that OPG implement new and revised CNSC regulatory documents, or REGDOCs, and CSA Group standards to support continued operation of the facilities. To date, CNSC staff are satisfied with OPG's progress in implementing these regulatory documents and standards. Most will be implemented prior to the start of the proposed licence period, and further details are contained within the draft *Licence Conditions Handbook* included with CNSC staff's CMD.

With respect to reporting requirements, OPG reports in accordance with REGDOC-3.1.1 at the Pickering NGS and REGDOC-3.1.2 at the Pickering WMF, both of which are reflected in the draft consolidated operating licence.

CNSC staff are satisfied that OPG monitors and controls contamination levels and radiation doses as low as reasonably achievable. Requirements for the Radiation Protection SCA are found in the Radiation Protection Regulations. CNSC staff have found that OPG's radiation protection program meets CNSC regulatory requirements. In its application, OPG indicated that it will use the existing radiation protection program for the planned

activities during the proposed licence period, such as the proposed refurbishment of Units 5 to 8. OPG also intends to incorporate lessons learned from previous refurbishment projects in the planning for radiation protection during the proposed refurbishment project.

As a licensee, OPG is accountable to the CNSC for the health and safety of all workers at the Pickering Nuclear Site. As indicated in its application, OPG has developed a health and safety plan to address the changing hazards that will result from the conduct of the proposed major construction activities across the Pickering Nuclear Site during the proposed licence period. To ensure workers are prepared prior to executing high-risk tasks, OPG plans to use mock-ups and incorporate operating experience from the Darlington NGS refurbishment project. CNSC staff are satisfied that OPG's conventional health and safety programs continue to meet CNSC regulatory requirements and that the health and safety of workers continues to be protected.

Regarding the environment, CNSC staff are satisfied that the environment and the health of people continue to be protected at the Pickering Nuclear Site. OPG has a well-established and robust Environmental Protection Program that includes the preparation of an Environmental Risk Assessment. To support its application,

OPG prepared a Predictive Environmental Risk Assessment, or PERA, in 2025. CNSC staff found that OPG's PERA met regulatory requirements and that most proposed activities are not predicted to result in adverse effects to people or the environment. As we discussed earlier in the presentation, PERA also recommends monitoring and mitigation measures to minimize any anticipated impacts of OPG's proposed activities.

Every five years, CNSC staff publish an Environmental Protection Review Report for the Pickering Nuclear Site. The most recent such report was completed in 2024 and found that the potential risks from releases to the environment are low to negligible. Additionally, in 2025, CNSC staff conducted the most recent CNSC Independent Environmental Monitoring Program, or IEMP, campaign in the vicinity of the Pickering Nuclear Site. IEMP results support CNSC staff's assessment that OPG's Environmental Protection Program is effective.

CNSC staff are satisfied that OPG is prepared to respond to an emergency and has adequate measures in place to mitigate and protect against the risk of fire. CNSC staff have found that OPG's emergency preparedness and fire protection programs meet CNSC requirements. Off-site emergency planning is the responsibility of the Province of Ontario. In 2025, Emergency Management Ontario updated its

Provincial Nuclear Emergency Response Plan, or PNERP.

Finally, OPG is required to ensure that its fire protection assessments accurately capture changes to the design of the facility. CNSC staff will verify that these assessments address any anticipated modifications to fire hazards and fire protection systems that may result from the proposed activities of refurbishment or decommissioning at the Pickering NGS.

CNSC staff are satisfied with OPG's performance in the Waste Management SCA at the Pickering NGS and WMF. CNSC staff have found that OPG's decommissioning and waste management programs meet CNSC regulatory requirements. This includes maintaining preliminary decommissioning plans for the Pickering Waste Management Facility, Pickering Units 5 to 8, and the whole Pickering Nuclear Site, which is linked to the detailed decommissioning plan discussed previously.

OPG's application notes that the planned activities during the proposed licence period will produce increased radioactive and conventional waste volumes. CNSC staff have found that OPG's established programs and procedures for radioactive and hazardous waste management are suitable to identify the appropriate disposition pathway and manage the increased volume of waste.

OPG maintains a consolidated financial

guarantee for all of its facilities that includes segregated funds established pursuant to the Ontario Nuclear Funds Agreement between OPG and the Province of Ontario, as well as the trust fund for the management of used nuclear fuel established pursuant to the *Nuclear Fuel Waste Act*. CNSC staff are satisfied that OPG's financial guarantee, which is based on cost estimates associated with the preliminary decommissioning planning, remains valid, is sufficient for the future decommissioning, and meets CNSC regulatory requirements.

OPG's most recent consolidated financial guarantee for the 2023-2027 period was accepted by the Commission in 2022. As of 2025, financial guarantee requirement is roughly 22 billion dollars and OPG's total guarantee available is roughly 28 billion dollars. Conducting the proposed limited scope decommissioning and refurbishment work is expected to reduce the necessary financial guarantee amount further. It will be captured in OPG's 2027 financial guarantee submission to the CNSC.

CNSC staff have made the Commission aware of past security performance at the Pickering NGS that has been below expectations. Following enhanced regulatory scrutiny, CNSC staff have noted improvements with OPG's performance since 2023. A major change in requirements pertaining to the security SCA are the publishing of new

Nuclear Security Regulations, which replaced the previous regulations in 2025. Requirements for existing sites, such as the Pickering Nuclear Site, will come into force in October 2027.

With respect to planned activities during the proposed licence period, OPG intends to modify security facilities and equipment during the conduct of refurbishment and decommissioning work. This includes the expansion of existing points of entry to accommodate additional workers.

I will now pass the presentation back to Ross Richardson.

ROSS RICHARDSON: Thank you, Daniel. Ross Richardson, for the record.

So in this section, I will provide information on CNSC staff's proposed consolidated operating licence for the Pickering NGS and WMF.

So OPG is requesting that the operating licences for the Pickering NGS and WMF be consolidated into a single Power Reactor Operating Licence. Consolidation of these licences is administrative in nature and would not change CNSC regulatory requirements or CNSC staff's regulatory oversight. All licensed activities, licence conditions and associated requirements for both facilities are included in the proposed licence and *Licence Conditions*

Handbook, which can be found in Appendix C of CNSC staff's CMD. CNSC staff recommend that the Commission consolidate the current power reactor and waste facility operating licences into a single renewed power reactor operating licence and revoke the existing waste facility operating licence upon consolidation.

OPG's application requests a 10-year operating licence ending December 31st, 2036, during which time OPG would complete the refurbishment of Pickering NGS Units 5 to 8 and continue the operation of the Pickering NGS and WMF. CNSC staff are satisfied that OPG's requested licence period is reasonable and are requesting that the Commission issue a 10-year consolidated operating licence.

The proposed power reactor operating licence and associated *Licence Conditions Handbook* are included in Appendix C of CNSC staff's CMD. The draft licence includes new standard licence conditions and wording to align with similar licences, such as a licence condition to conduct a periodic safety review every 10 years and a licence condition to implement an Indigenous engagement program, which is similar to the current operating licence of the Darlington Nuclear Generating Station.

The draft licence also includes a new licence activity pertaining to the storage with surveillance of Units 1 to 4, up to and not beyond the removal of

outbuildings and non-nuclear components, as well as new licence conditions pertaining to the return to service of refurbished units and regulatory hold points.

And so in this section we will briefly look at the CNSC's public engagement on this matter to date.

CNSC staff prepared and published a notice of public hearing and participant funding in July 2025 that was later revised in January 2026. To assist with participation in the regulatory process, the CNSC also offered funding through the CNSC's Participant Funding Program, and roughly $340,000 was awarded to 10 applicants. Additionally, CNSC staff held a public webinar in April 2026 to provide information on OPG's application and the CNSC hearing process to interested members of the public. The deadline to provide interventions on this matter for Part 2 of this hearing is August 12th, 2026.

I will now pass the presentation back to Dr. Alex Viktorov for some concluding remarks.

DR. ALEXANDRE VIKTOROV: Ross, thank you. Alex Viktorov, for the record.

The last couple of slides provide CNSC staff conclusions and recommendations to the Commission.

Based on the information available, as presented in staff CMD and summarized in this presentation, CNSC staff conclude that OPG's application meets the

relevant regulatory requirements, OPG is qualified to carry on the activities proposed to be authorized in the consolidated operating licence, and OPG has made adequate provisions for the protection of the environment and of the health and safety of persons and the maintenance of national security measures required to implement international obligations to which Canada has agreed.

Therefore, CNSC staff are making the recommendations as described on the next slide.

CNSC staff recommend that the Commission renew the Pickering NGS operating licence for a 10-year period and authorize OPG to refurbish and then continue to operate Pickering Units 5 to 8, continue to operate the Pickering Waste Management Facility, and carry out the licence activities listed in Part IV of the proposed operating licence.

Staff also recommend that the Commission consolidate into the renewed power reactor operating licence the licensed activity of the waste facility and concurrently revoke the existing waste facility operating licence; delegate authority as set out in CNSC staff CMD; accept OPG's proposed alternate strategy to reinstate certified staff with respect to licence condition 2.4 for the proposed restart of Unit 5; and finally, authorize OPG to store up to 1,410 dry storage containers in storage

Building Number 5 of the Pickering Waste Management Facility.

Thank you for your attention. CNSC staff are available to take any questions you may have.

THE PRESIDENT: All right. Thank you for that.

So before we do that, we are going to take a short break, 20 minutes, and we will come back to start the questioning of the presentations. Thank you.

--- Upon recessing at 10:26 a.m. /
Suspension à 10 h 26
--- Upon resuming at 10:47 a.m. /
Reprise à 10 h 47

THE PRESIDENT: Welcome back. I will now open the floor for questions from Commission Members to OPG and CNSC staff. As mentioned earlier, there are representatives from other government departments and agencies available to answer questions as well.

Without limiting the questioning, we will group our questions into generally four categories: first, questions arising directly from the presentations we have just heard from both OPG and staff; second, questions about the refurbishment of Pickering Units 5 to 8; then questions around decommissioning of Pickering 1 to 4; and then

finally questions around the Pickering Waste Management Facility.

So I will now turn the floor over to Commission Members and we will start with Dr. Remenda, please.

MEMBER REMENDA: Thank you, everyone, for your presentations. That provided me with some additional clarity. But I wanted to seek some further clarity concerning the management and sequencing of events and activities on site. I am hearing that there is a -- and reading of course -- that there is a wide variety of projects and sub-projects.

And my question first to OPG is, how do you make provisions for the sequencing and management of individual activities? And I know you bundle some projects together to ensure that there aren't clashes, that there aren't concerns about worker safety, concerns about emergency management. So I hope this makes it clear. Great, thank you.

LUCA CECCATO: Luca Ceccato, for the record. I will start off on a high-level perspective on the refurbishment project and go through some things for the Commission to give confidence that we do have this and how we are going to make sure we execute this safely and event-free.

So we are following our outage practices of

onboarding and executing. So this is going to be a large-scale outage with a longer duration. That is the mindset for us. We have completed a large-scale stop of this magnitude at Darlington successfully. That includes running units at the same time, so we have managed all that workforce.

I will share with the Commission this is the eighth refurbishment in 10 years in the province of Ontario, so we have not left any operating experience out there that we haven't pulled in to help us execute this work.

So unlike Darlington, we are going to utilize the existing staff at the site. So all those folks will stay and support the vendor community as we work on that. That is a different lens that has not happened in the province of Ontario.

We are going to be putting a strong emphasis on the one team, and that probably has been heard before, but that was a key to our success at Darlington. So existing staff will be supporting and looking out for the vendor community that will be coming to the site that has not been to the site before.

We've embedded our skilled operations staff into the project bundles. That is something absolutely different than any other previous refurbishments in the

province of Ontario. So the skilled operations staff that are no longer in the control room are supporting our vendor community and the trades as they execute the work. So they're fully embedded to give line of sight to that.

Our vendor community has gone through rigorous process to get qualified to be on site. So highly skilled vendor community that has been trained to OPG standards and have done this before. So we spared no expense to leverage the same vendor community that was successful at Darlington in executing the Pickering refurbishment by bundle, gone through every project-specific to make sure that we utilize that skill set that they have brought successfully to bear at Darlington, and implemented that here at Pickering. I use an example as Aecon doing the retube and feeder replacement box with breathing air. They did that at Darlington, so we brought that expertise over.

And we have obtained the majority of the supervisors from Darlington by name. That's been a major focus of my personal self to make sure that we maintain that lessons learned in those individuals. So it's more than just a process, but bringing those same skilled folks. And they are currently on the site right now with the direct field labour.

We mentioned previously about the Unit 2-3 turbine generator decommissioning. Those folks are the key

players that were successful at Darlington that are helping us out with those evolutions. So the trades are getting experience at the Pickering site, how to get around, getting acclimatized to the units.

We've also got the mock-up training facilities. I'll go in much detail there maybe later on. But we are also utilizing those same skill sets with the REFRESA(ph) building, which is a building we're constructing inside the protected area. So all these key players and supervisory skill sets are in place at Pickering right now with the line of sight of leveraging the lessons learned.

I will mention that in the question, Commissioner, that how are we going to manage all the work on the site. Currently underneath Dave Rogers is one director of Work Management. That individual has line of sight, A, obviously, for the safe operation of the four units to break her open on September 20th. But he also has the refurbishment prereqs under his purview.

So we're all under one umbrella, currently under Dave Rogers. And we don't expect a flick of the switch on September the 20th; we expect a smooth transition. And so we're already amalgamating the two organizations. And so that individual has line of sight to any last-minute safe storage activities that are happening in 2027, minimal decommissioning activities, and obviously

prereqs to get ready for the refurbishment activities.

I will just leave it at this from the lessons learned perspective and give some confidence on how we've utilized the learnings from Darlington and from Bruce Power. We are currently three quarters of the way complete on the facilities and infrastructure project at the site with numerous buildings, parking lots, road excavation, major civil activities, and we are on schedule and on budget. But more importantly, we're about a million and a half hours without a medically treated injury. And the performance is very strong as we head into the refurbishment proper, which again starts on September 20th. The defuel campaign on the prerequisites and understanding the actual refurbishment starts on January 4th with licence renewal based on the licence. I'll leave it there.

MEMBER REMENDA: In both the staff and the OPG CMD, there was mention of an A-B gate, and this was not raised today in the presentation. Maybe you could just provide me some reasoning around not including that piece in the presentation.

KAPIL AGGARWAL: Kapil Aggarwal, representing Nuclear Sustainability Services, for the record.

OPG's strategy for separating Pickering A and Pickering B is to ensure Pickering B can operate independently and reliably after refurbishment. In

practical terms, this means moving the services and control of any equipment needed for Pickering B operations over from Pickering A.

Where controls currently sit only in Pickering A's main control room, new controls will be installed on the Pickering B operating island. Key supporting services such as power, air, and water will be rerouted to Pickering B supplies with critical loads transferred in a planned, staged way.

All systems important to nuclear safety will be complete before the first two units return to service. The remaining non-safety-significant items will be managed through OPG's project portfolio with clear tracking to completion and aligned with Pickering A's safe storage project milestones.

Currently, the A-B gate is conceptually represented as a line on a drawing of Pickering NGS for planning purposes. The aim is to facilitate at an appropriate point in decommissioning distinct organizational and operational controlling authorities and for planning purposes only.

Governance and implementation details specify that the A-B gate line will initially serve as input for design packages for Pickering A and Pickering B's separation during the licence period. Once the A-B system

separation has been achieved, a permanent physical barrier for the A-B gate may be considered to separate the operating island from the active decommissioning island.

Any changes required to establish the potential physical A-B gate will require regulatory notification and will be planned and communicated as required for management system changes, security, and other licensing impacts.

MEMBER REMENDA: So in other words, the A-B gate is a many-months to possible several-year set of actions in order to establish this A-B gate.

KAPIL AGGARWAL: Kapil Aggarwal, for the record. That is correct. It is going to take the licence period to establish the boundaries.

MEMBER REMENDA: So I wanted to thank you for your presentation because that also helped clarify some of the sequencing for me as well. There are a lot of commitments and expectations that are provided in Appendix A. And I wanted to hear about the sufficiency of both human and other resources in order to accept and review and accept those anticipated commitments and expectations.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record. Briefly, we believe we currently have sufficient resources to oversee the activities that will happen at Pickering site during the licensing period. I get this confidence from having a detailed plan for oversight of

what's happening during refurbishment on B site and decommission on A site as well as other activities. So we do have a pretty detailed plan of what we will be spending our time on.

In case of a temporary need, we can draw resources from other parts of the organization, borrow resources from other sites on a short-term basis to support activities. On the longer term, if we see a need, we will develop a business plan to shore up resources. But currently, we project that the current supplement of human and other resources is sufficient.

MEMBER REMENDA: From my reading of the documents that there are still outstanding design decisions to be made, does that complicate your verification plans, or is that something that you are quite accustomed to, to be ready with review of new ideas or new designs or improved or different designs?

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record. CNSC does have a wide variety of technical disciplines, specialists in various disciplines, so we draw on that capability. Our expert in technical support, branch support, technical assessments, or review of design modification -- that's not a unique feature. Similar activities happen at other sites. And depending on the complexity, we may also reach out to outside resources, but

we have experience in this kind of activities.

MEMBER REMENDA: Thank you very much.

I'll pass back to you, President.

THE PRESIDENT: All right. Commission Member Hardie, please.

Good morning and thank you to each of you for your presentations this morning. I'd like to just continue on the questioning from Dr. Remenda with my first question here and that is that there are a number of items here that are being

MEMBER HARDIE: Good morning, and thank you to each of you for your presentations this morning.

I'd like to just continue on the questioning from Dr. Remenda with my first question here, and that is that there are a number of items here that are being asked for by the proponent and included within that proposed licence. And while there's been discussions on each separate item that's been included and the management systems and compliance verification for each one, I would like to hear from OPG and CNSC staff on how you're managing all of these components from a holistic perspective. And so really what I'm asking for is the umbrella, *per se*, to ensure that items don't fall through with the assumption that it's being handled by one group over another. So first I'd like to turn to OPG. Thank you.

KERRY JACKSON: Kerry Jackson, Operations director, for the record. Just looking for some clarification, are you referring to day-to-day activities at the site?

MEMBER HARDIE: I'm looking for understanding sort of your umbrella management system in a sense for how you're going to manage each of these pieces that you've been asking for within this hearing.

KERRY JACKSON: Okay, thank you. Kerry Jackson, for the record.

So OPG maintains a nuclear management system in accordance with the operating licence and the associated *Licence Condition Handbook*, and really that ensures that we maintain and that we're in compliance with all applicable codes and standards. And really, the fundamental objective of the nuclear management system is really to ensure that our nuclear facilities are operated and maintained using sound nuclear safety and defence-in-depth practices to protect our workers, the public, and the environment and to maintain ALARA.

So Pickering through the refurbishment period will fall under the nuclear management system as well as all of our contractors who will be supporting work on the Pickering site will also be required to meet the requirements of our nuclear management system to ensure

that everybody working on the Pickering site is following all the required codes and standards associated with our licence.

MEMBER HARDIE: And when you have different components such as you're going to have your refurbishment piece, which would obviously be included under that, but you also would like to include that waste management component as well too in there, how are you just ensuring that each of those sections are working together from your top-down perspective?

KERRY JACKSON: Kerry Jackson, Operations director, for the record. As part of our nuclear management system, it does provide the framework and organizational structure that provides the oversight to ensure all of the requirements for the licence and the *Licence Condition Handbook* are met.

So with one licence, if that's to clarify what your question is, with one licence for both the Pickering operating station as well as the waste management facility, our nuclear management system framework would have all the required oversight as it does today, but in one spot, to manage one licence for both facilities.

DAVID ROGERS: David Rogers, for the record. There's still one licence holder. That licence holder is reporting to the senior vice-president of Projects but has

a direct line to our corporate and to our chief nuclear officer.

A lot of our same management processes that we use now we'll use during refurb decommissioning and Pickering Waste Management Facilities. They will be adequately staffed and resourced with management and management oversight. And where required, where the activities ramp up, we've actually provided extra staff with extra oversight. For example, conventional health and safety with the refurbishment program significantly staffed up, simply resourced under the same management system. Same with radiation protection refurbishment increasing in staffing there.

And some of our key oversight activities, nuclear oversight, our corporate functional area management system, those systems will still have internal independent oversight as well to ensure that none of this is lost. And our joint look at this in a holistic manner for systems like performance improvement, that system will continue on throughout. Human performance will continue on throughout, and really that defence-in-depth layers to ensure that everything gets covered with a licence will be covered.

JUDE SAMSON: Can I just ask the people who are answering questions to keep their voice up and speaking close to the mic, please. Thank you.

MEMBER HARDIE: Thank you very much for that clarification. Just one final follow-up.

For any type of audit activities or inspection activities that you're doing on that site, is that a general overall OPG piece, or are those functions independent within each of these projects? Are you going to have a waste management one and a decommissioning one and a refurbishment section for that, or do you have sort of one general audit compliance verification, some of those perspectives that are going to have oversight over everything?

DAVID ROGERS: David Rogers, for the record. So from an accountability perspective, that will be with the licence holder. And they may pull on subject matter experts in those areas to support those items. And I'll say more of the local management of those areas that have accountability for those areas. But it will roll up into one licence holder who has accountability for the site.

SARA IRVINE: Sara Irvine, for the record. I do also want to add that we will be relying on our central nuclear oversight organization as well as our central family organizations to have that corporate-wide oversight of all the activities going on at Pickering.

MEMBER HARDIE: Thank you very much for that clarification.

I will just turn to CNSC staff for your comments around what was just said here as well as your own compliance verification viewpoints. Thank you.

DR. ALEXANDRE VIKTOROV: Alex Viktorov. I will start by pointing out that OPG has a mature management system that's been in use, in operation for decades and has been proven to be successful. The same overall management system will continue to apply.

As a representative for OPG mentioned, there is a strong internal oversight within OPG that will monitor.

But from CNSC perspective, we are not so concerned with operational excellence or success in managing projects. We are worried about safety, safety aspects of various activities. We have assessed the programs according to licence conditions that the programs are in place and meet requirements and will continue to meet requirements during the refurbishment decommissioning activities. And we build our plans, oversight plans, knowing what OPG is proposing to do. So our verification activities are reflective of the actual work that will happen during this period. And we also have a capability to adjust our plans based on what we observe from the execution of projects.

MEMBER HARDIE: Thank you very much.

I have one final question, and that pertains to

the global assessment report that was discussed, and we understand that it won't be finalized until August of 2026, so a year from now approximately. And how are we assured that, without that global assessment report finalized, how are we assured that that strong safety performance and those global issues are understood and included in your plan going forward without that piece? If OPG could just start where you're at in those assessments, and then I'll turn to CNSC staff. Thank you.

JEFF MILLER: Jeff Miller, vice-president, Refurbishment Engineering, for the record. So, in the global assessment report currently under development, global issues are prioritized based on their impact on nuclear safety, assigning a safety significance level of high, medium, low, and very low.

The PSR-3 process has resulted in no high safety significance global issues. Seven global issues have been categorized as medium safety significance. The medium safety significance global issues are well understood, and OPG has already built the needed response into refurbishment scope, inspection plans, and follow-up assessments.

There is clear traceability from PSR findings to specific refurbishment actions. These items relate to fitness for service for extended operation, updates to

environmental qualification and life-cycle management plans, service limit assessments for certain Class I components, graded fuel storage, selected single failure improvements, and fire protection enhancements.

Where physical changes or inspections are needed, they have already been captured in the refurbishment scope. For example, work on civil structures, buried piping, emergency water systems redundancy, and fire protection components. Where further technical work is needed, such as updated assessments, long term plans or code gap reviews, that work is also planned under established station programs.

Overall, these medium significance issues are being addressed in a structured way that strengthens confidence in safe, reliable operation following refurbishment.

MEMBER HARDIE: Thank you for that response, and I'll turn to CNSC Staff.

Thank you.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record.

In fact, the Global Assessment Report will be available before Part 2, August this year, and we don't expect any surprises there because we have seen the safety factor assessment so we know the conclusions in various

specific areas, so we don't expect any surprises in the GAR.

Nevertheless, the commitments, opportunities for improvement identified in GAR and integrated implementation plan will be part of the hold points release, so we will follow up on those commitments until their completion.

MEMBER HARDIE: Thank you very much for that.

And I'll correct myself for the record there, August 2026. I'm clearly living in 2025 still, so I will pass it back to the President. Thank you.

THE PRESIDENT: All right. Thank you.

Just before I move it to the next Commissioner, I just want to -- in the spirit of the Commissioner's question, this is a complex undertaking. There are many elements in terms of decommissioning activities that you want to provide the refurbishment of our plant and consolidations of licences that exist.

So just to be clear, a number of the items that are on here constitute regulatory commitments, should we approve this would constitute commitments under the licence. So just to be clear, when you -- and I think the issue was around given the complexity, how do you track all those commitments?

So when a commitment is carried out or any

change to any of those commitments, what would OPG do to ensure that that was communicated and closed off as part of the commitment-making process and the execution? So maybe just talk to us a bit about that and then I'll turn to staff.

SARA IRVINE: Sara Irvine, for the record.

OPG's Nuclear Management System includes a robust process for tracking commitments that are made to the Nuclear Safety Commission. So after today's proceedings and after the October proceedings, we will get from CNSC Staff a letter that says, here's the commitments we think were made through the hearing process. My group then translate that into what's called Regulatory Commitments List. And then whenever there's a commitment that gets closed out, the action is on my group to write a letter to CNSC Staff saying we have completed this, and then they write back and saying yes, it's been completed.

Now, through IIP specifically, if there's a challenge completing something by that agreed to date, we then write to the staff with a request to change the due date for an IIP commitment along with a rationale about why we need to change it, why it's safe to proceed without that, when it will be completed, and what the closure criteria is going to look like. So it's all within our Nuclear Management System that anything that's a

commitment, whether it's in our application, our IIP through today's proceedings or October's, it all gets captured in a list of commitments that then is up to my group to manage and make sure that the organization is completing its commitments on time and as expected.

THE PRESIDENT: All right. Thank you. I was looking for that.

Over to staff. I'm sure we appreciate the closure of commitments, but perhaps there's a trust and verify associated with that, so maybe you could describe that a bit. Thank you.

ROSS RICHARDSON: Yeah, Ross Richardson, for the record, Director of the Pickering Regulator Program Division.

Absolutely. So just to concur with what OPG had said, so following OPG's description of their process, we would then take the outputs of their process and track the commitments at our end as well. We have a number of different methods to track commitments. Primarily would be through their *Licence Condition Handbook*, where we would track commitments there. We have IT tools to track commitments as well through our Regulatory Information Bank software as well at the CNSC. So a number of regulatory tools are available to track these commitments, but I would say the prime tool is the *Licence Condition Handbook*.

The other thing I wanted to mention as well is, during the course of the project, the Status Report on Power Reactors that we'll be presenting to the Commission will be pretty much solely focused on the refurbishment and decommissioning. So the Commission will be receiving detailed updates from CNSC Staff as part of that effort and at that time, of course, if we see commitments perhaps, for example, being dropped or anything of sort of safety significance, we would bring that to the Commission.

So the Commission will be completely informed of the project going forward and, as I mentioned, we also have adequate procedures in place to make sure that licensees are upholding their commitments that they are speaking to today.

THE PRESIDENT: All right. Thank you.

Let's move to Commission Member Berube, please.

JUDE SAMSON: If I could just remind people to speak slowly, please, when they're giving their answers. Thank you.

MEMBER BERUBE: Yes. Good morning, everybody. It's nice to see everybody again.

I'm going to stick more to contextual level this morning and we'll get more into the specifics as we go through item by item this afternoon. But the biggest question I have is, you've asked for consolidated licence

here. Right now it's three separate licences, decommissioning, basically refurbishment activities, typically, and then waste management.

So I've heard some of the reasons that you have for wanting to do this this morning, but what I need to hear from you is why you need this. Not why you'd like it, but why do you need a consolidated licence in this case?

SARA IRVINE: Sara Irvine, for the record.

So I'll start first with why we need the renewed power reactor operating licence. So at the time we last extended the life of Pickering, we had not yet been asked to refurbish the Pickering 5-8 units. That was a decision that had not yet been made. So while the licence does expire in 2028, we're here today because the refurbishment project is ready to proceed starting January 1st. So that answers why we're renewing the PROL early.

At this time, we've chosen to also include specifically decommissioning activities because while we have an approved Detailed Decommissioning Plan, we didn't foresee at the time of the last renewal wanting to bring some of those early pieces of scope ahead. And really, that was meant to keep our staff proficient. There's work they can be doing in the field that ensures that they maintain proficiency while the units are being defuelled and dewatered. Those staff will be in the field working on

non-nuclear components to maintain those skills.

So while decommissioning is not an explicit activity under our PROL, there are approved activities within the Detailed Decommissioning Plan, those windows which will take place during this licence period. Therefore, we wish to have that specific activity within the operating licence.

We don't need a separate specific licence for decommissioning. We can fold all those activities into one operating licence.

The third on why we need to fold the waste licence into the operating licence. One thing we've heard from stakeholders and from Indigenous Nations is this idea of piecemealing our applications. And we've really heard that feedback from them. Obviously, with a refurbishment activity comes a stream of low and intermediate waste. The fuel goes to the bays. So there is a strong element of waste within this licence period. And to ensure there is no piecemealing of activities, we wanted to make sure all the licensed activities captured on the site for the next 10 years are heard during this hearing and are captured in this licence.

So we -- it's not a strict need, but rather than trying to separate these out and parse them out and have three separate hearings where it's a bit hard to

follow the bouncing ball of what we're doing, it really makes sense to fold this all into one licence application. So we're before the Commission talking to the public about all activities at the site over the next 10 years because that's really how the public and Indigenous Nations see the site.

MEMBER BERUBE: So sitting at it from our standpoint, looking at the document from my standpoint particularly, looking at all of these things together, this is an amazingly large and complex issue.

When we had separate licences, we were allowed to look at this in much more detail on each one of these things. We have a finite period of time. So that's a consideration here.

So I'm going to move to the staff and say, you've recommended that this is okay. You've put this to the Commission that you think that's all right that they have one licence. I need to hear why you think it's okay and what the justification is to make that happen and why we should accept that.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record.

I'll start with -- well, a statement that our regulatory framework does not preclude having such, well, rich licences that envelop numerous activities. From our

perspective, we can accommodate all those activities through specifying conditions which are different for units in refurbishment, units in decommissioning and waste through the *Licence Condition Handbook* by identifying the activities that are pertinent to different stages of facility operations.

So yes, we can do it by, again, combining various requirements in the *Licence Condition Handbook*. It makes for a relatively complex one *Licence Condition Handbook*, but we do see efficiency in having one instead of two or three *Licence Condition Handbooks*. It also allows us to build on some synergies in our resource allocation that we can draw from different groups under one licence and *Licence Condition Handbook*.

Again, there also seems to be a move towards consolidating licences. We have an example at Point Lepreau that has a consolidated licence for the power plant and the waste facility. That seems to be working well.

MEMBER BERUBE: Yes, I understand about Point Lepreau, but we are talking a single unit station versus a multi-unit station. This is a much more complex beast, as we are all very well in this room, and the complexity of this project is fairly immense. And I know everybody in this room appreciates that because it is simply a big beast. And the timeline is aggressive, as you know, as

well.

Okay. Well, I'm going to leave that there, and we can probably drill into this as we go along through this process, but certainly give that some thought because it will probably come back again to discuss it at some level.

On a high-level look at this -- this is again for OPG's. We are losing roughly 20 percent of cobalt-60 production here, and you're looking at about two gigawatts of electrical energy being taken off the grid. Maybe you'd like to give us some insight as to what the impact of both of those things are and how you intend to mitigate at least the cobalt-60 side, which is something we'd be concerned about.

SARA IRVINE: Sara Irvine, for the record.

Mr. Chase Amyot is going to speak to the cobalt aspect and then another member of our team will speak to the lost generation.

CHASE AMYOT: Good morning. Chase Amyot, Nuclear Decommissioning Director, for the record.

OPG has moved cobalt production to Darlington, increased capacity at Darlington while refurbishment is happening at Pickering. Following refurbishment at Pickering, OPG plans to do one more harvest that we have contracts for currently in this licence window.

MEMBER BERUBE: So you're actively going to

harvest as you are emptying the reactors at this point, take the rods out, and basically that should hold over until Darlington comes online? How is that going to work?

CHASE AMYOT: Yeah. The cobalt will be removed -- sorry. Chase Amyot, for the record.

Cobalt will be removed from the units on the Pickering side and be moved to the irradiated fuel base for harvesting and then transport to the vendor for use in the medical industry.

SARA IRVINE: And Sara Irvine, for the record.

I just want to remind the Commission as well that within the past couple of years, you've reviewed cobalt generation at Darlington. So it was in anticipation of losing that window of opportunity to generate cobalt at Pickering that we moved operations of cobalt harvesting to Darlington.

So that is expected to take up the shortfall for those years when the three units that produce cobalt at Pickering are shut down. That production will be made up at Darlington.

MEMBER BERUBE: Yeah, my concern was that it's actually been thought about and you're dealing with it in the background, and you are, so that's fine.

DAVID ROGERS: Dave Rogers, for the record. I'll cover the generation aspect.

So the Independent Electricity System Operator and OPG and other generating utilities have known this date is coming. They have a robust plan to ensure generation and secure grid reliability for the province of Ontario. And OPG has been a key stakeholder in that in providing some of our other assets to provide that electricity during this period.

MEMBER BERUBE: So IESO is handling that. Really outside of our scope, but it's something to consider. It's fairly large.

Other question is -- and just again contextually speaking, right now how many people do you have on site and what do you think the maximum number of people you're going to have on site when all of these things are up and running should the licence be granted?

LUCA CECCATO: Luca Ceccato, for the record.

Currently on the site, we have roughly around 2,200 staff. We do anticipate at the peak of execution, which is mid-2028, approximately 3,600 to 3,700 staff. And so all our logistics -- we actually have an empowered logistics teams looking at all parking, security, egress and ingress, lunchroom facilities, pre-job brief areas, all based on that number that I have shared.

MEMBER BERUBE: So there's a fairly significant training part of this. You mentioned basically you're

going to do some simulation work -- not simulation work, but mock-up work. And are you going to do that at Darlington site, on site where it is now, or do you intend to move that somewhere or what do you intend to do here?

LUCA CECCATO: So Luca Ceccato, for the record.

A bit of both, I would say. I can break down -- I know I mentioned it in my initial comments.

So Unit 2 at Pickering, we're utilizing the facilities there. Obviously, the turbine generator program for the decommissioning, that's proving very beneficial. We have a mini-hatch mock-up, and that is essentially one of the more challenging evolutions on Unit 5. So the team is in on Unit 2 to execute that work in a training environment. The retube tool platform and the ball screws, we're looking at doing that in Unit 2. And finally on Unit 2, the boiler welding on the primary side.

So we're setting up Unit 2 as a training facility for the trades to get in there and get acclimatized and remove the challenges as we move into Unit 5.

There's a training and mock-up facility of 200,000 square feet, for those that know Pickering, across from the casino at Pickering. That facility will resemble what we know to be the Darlington Energy Complex. And instead of having one big mock-up, we have several small

ones. And that was based on OPEX from Darlington, but also from Bruce Power. So we can train more people more specifically on reactor face work than we could on the big mock-up that we had at the Darlington Energy Complex.

So that is being -- those are being built right now at the training mock-up facility at Church Street in Pickering.

And the Darlington Energy Complex continues to be a vital area for us to do training, so welding, feeder nozzle welding training. And also out in the parking lot, for those that have driven down the 401, you'll start to see a dome mock-up. So the boiler removal series through the engineered access points through the dome, we are actually doing a second major mock-up of that in the parking lot to go and execute that and get ready for the real evolution in Unit 5.

So those are just some of the things that we are doing to ensure that the staff is trained adequately for a safe and quality execution of the project.

MEMBER BERUBE: And you're currently working on that right now in the background. Is that correct?

LUCA CECCATO: Luca Ceccato, for the record.

All those things are going on right now.

MEMBER BERUBE: Okay. So I'm going to move over to CNSC Staff and ask you.

Obviously, you're seeing some of this activity happening in the background, concurrent activity. I would like to get your feedback on what you're seeing in terms of the preparation for potentially a new licence being issued at this point based on what you're seeing, what your level of confidence is in terms of the simulator aspects as well as the mock-up aspects and in terms of the training aspects.

An additional 1,200 headcount on site is a fairly significant undertaking. Just give us some perspectives on what you're seeing right now and what your level of confidence is.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record.

CNSC Staff have been engaging with OPG on this file for a long time since the announcement of proposed refurbishment. Certainly, we have looked at the training opportunities and the use of mock-up facilities that exist at the Darlington Energy Centre, which was quite beneficial for execution of refurbishment at Darlington or any other facilities.

Again, the situation is slightly different. There are units, shutdown units, that can be used for many of those preparatory activities to rehearse in an environment that's very close to the actual situation at

Units 5 to 8.

We also looked at the experience at other sites. Again, it will be not the first refurbishment, and the industry have accumulated significant experience in executing the equipment work, but also managing contractors. Again, it's quite known that personnel coming from non-nuclear industries have different perception of culture or safety culture, so that's a known area where we would pay attention to.

But again, knowing that this can be done safely successfully give us confidence that OPG Pickering will manage successfully. But we will have our oversight in those activities as we always had it as a refurbishment project.

And I don't know if my colleague will want to supplement.

ROSS RICHARDSON: Ross Richardson, for the record.

So just again, I'll start just perhaps at a high level is that CNSC staff have been engaging with OPG on the proposed refurbishment project for the past couple of years. We have bi-weekly meetings with OPG, where OPG provides us with additional information on specific project execution.

One of the specific topics were the training

and mockup. OPG has provided submissions to CNSC staff describing the mockup facilities. CNSC staff do have a plan to visit some of these mockup facilities to observe some of the key evolutions of the refurbishment project going forward.

So again, as part of our regulatory oversight, so far what we've seen to date is we're satisfied with what OPG is doing so far, but I guess we'll continue to proceed as the project -- if the project advances subject to Commission decision.

MEMBER BERUBE: I think that handles most of the big topics I want to get to, but we'll get to more specifics this afternoon, thanks. Back to the President.

THE PRESIDENT: All right, Commission Member McEwan, please.

MEMBER McEWAN: Thank you, Mr. President. Thank you for the presentations. As I was listening and I read the documents, this is a huge project. I think it has come clear today just how large and ambitious this project is. And I think you will be increasing staff on site, you said, by about two-thirds, ish.

Are you going to be able to get enough trained staff? Are you going to be able to find the expert people to come in and do this? Because most of the tasks that will be taking part on site are probably not

general-purpose individuals. You'll need highly trained and skilled staff people. And recognizing you're able to re-access some of the people from Darlington, do you have confidence that staffing will remain not a problem over the course of the project?

LUCA CECCATO: So Luca Ceccato, for the record.

Yes, I do have confidence. We are right now in probably the most demanding part of the refurbishment project, which is the definition phase, with project managers, leaders, design folks. And so we've got the resources required.

The second part of that answer is the trades. And so, I personally am on the executive committee of EPSCA, which is the Electrical Power Systems Construction Association. Did go through the commercial agreements with all the trade unions. There's 16 in Ontario. And we've got a very competitive collective agreement.

That's one thing right now. We used to have what I would characterize as competition with the industrial sector. And so, we are now very competitive with that. But also we have a lot to offer to the trades of long-term work programs and the culture set at OPG.

And so I, as part of EPSCA, the Electrical Power Systems Construction Association, I'm actually the vice president, I have been to most of the trade halls.

They are very well aware of the demands of their distinct unions, and they're ready. They're ready and they've committed that they will have a line of sight and we will have the trade resources required. That can always change with a change in economy, etcetera. But right now, very confident.

And I'll share in closing that the key trade group is the Boilermakers on this refurbishment project. And we do absolutely have the trade resources for that group. We actually brought in about 600 apprentices over the last two years. So the team's very focused on bringing in new expertise and new people to the trade. But I'm confident that we'll have enough trades and resources to execute this project.

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

Just to go back to a couple of the comments. The first, oversight of our vendor partners or contractors starts before they arrive on site. There's a rigorous process to ensure that their safety performance is in line with what we would expect at an OPG site.

Once our contractors are on site, there is no difference in the safety culture. We expect really one standard, one team in how we perform work. And with that comes an oversight function, specifically learnings from

Darlington around our Construction Centre of Excellence Department, who's solely focused on providing oversight of our vendor partners, ensuring that they're integrated with our OPG team and that all expectations and standards are being met.

The last part of the question, I think you asked about training as well. All of our staff are required to be trained and qualified and meet the requirements of our licence to perform work on site. And with that comes, again, a rigorous onboarding process for our contractors, where before they arrive on site, they are required to complete -- successfully complete all of the required training to ensure that they have that qualification for the work that they will be performing on site.

And our OPG team, our Construction Centre of Excellence, as well as our senior leadership team, ensures that this is completed and that they're following the requirements that are expected of them while they're at the Pickering site.

MEMBER McEWAN: Thank you.

So you're confident you'll have the staff, so for the next 10 years there will be -- if I look at it -- three possible disruptions to the local communities. You'll have increased staff coming in and out, you'll have,

I would guess, a significant amount of extra trucking of materials in and out, and you will have the potential for hazardous materials being moved in and out or around the staff.

How do you actually manage the expectations of the community in managing that traffic flow, in managing the increase in noise management, trucks, to ensure that they're not disrupted in any significant adverse way?

LUCA CECCATO: So Luca Ceccato, for the record.

I'll start with that we've been very much aligned with the City of Pickering and Durham Region. Those are the two entities that I've learned in this job that are key components to our success.

So they'll fully engage with us, A, helping us, but also to communicate to the public of our endeavours. Things like the road infrastructure upgrades, a lot of them reach out into Durham Region. Brock Road, there's a new right-turn lane being constructed on the Brock Road Bridge that goes onto the 401. All those things have been formally communicated to the municipality and the residents of Pickering.

And the other part of this is those two regions, the City of Pickering and Durham Region, have been helping us in how we can best execute upgrades to the systems to better facilitate the traffic coming on and off

the site.

And as I mentioned to Commissioner Berube, we know that number and they are very well aware of that number. We've done assessments on what that looks like for traffic. Essentially, Monday to Friday at 3:00 to 5:00 is the peak of people leaving the site. And so we've done analysis on that to where best to add more infrastructure projects. So again, if you go to the site, you'll see a lot of work outside of the protected area on the roads, the bridges, etcetera, to help facilitate that.

MEMBER McEWAN: Noise going sort of outwards. I mean, my memory of the site is that you've got a dormitory community not a million miles from the site itself. Is there going to be increasing noise for them? Are they going to be getting, I don't know, dust or other irritants that are likely to create problems, friction for the project going forward?

LUCA CECCATO: Peaked at this moment currently with the structural upgrades in the civil construction activities outside of the protected area. Again, strong communication with the community, making sure that we give timelines of when the noise will be highest, and then obviously when the work will be completed to give the residents an understanding of when this project will -- you know, the ebbs and flows of the activity on their roads

will be clearly communicated.

SARA IRVINE: And Sara Irvine, for the record.

I'd just like to pass this discussion to Ms. Cammie Cheng who can speak to how our Environmental Assessment addresses some of those concerns like noise.

CAMMIE CHENG: Cammie Cheng, OPG Director for Nuclear Environment Project Support, for the record.

As part of our predictive environmental risk assessments, we looked at all the different activities that would come through the refurbishment and decommissioning activities in the next licensing period, and dust and noise was one of the aspects that we did consider.

In our modeling that was conducted for the PERA, there was very conservative assessments around the number and types and amounts of equipment and vehicles that would be coming and going around the site and in and out of the site. And we looked at the very peak hypothetical if all that equipment was being used at the same time scenario and ran that through our noise models.

And with that very conservative assumption, there were only minor exceedances in terms of noise guidelines. And the conclusions in the PERA were around if we -- and that is having not accounted for any additional mitigation measures, as well as noise management plans that would be in effect.

So with the addition of mitigation measures, which all of the constructors who are doing work on site will have, they will have noise management planning as part of their work before that work starts, as well as any other mitigation measures around timing for activities and even additional noise modeling.

If we have refined numbers that more accurately depict what the project plans are going to be around equipment and vehicle and activities, that will be sufficient to address any adverse impacts that there could be with the very minimal amount of noise exceedances that we're expecting.

MEMBER McEWAN: Thank you.

Staff, do you have any comments on how this is likely to play out? Are you comfortable with the strategies? Are you comfortable with the training? Are you comfortable with the community communication component?

DR. ALEXANDRE VIKTOROV: I will ask our colleague Melissa Fabian Mendoza to specifically explain how CNSC evaluates the predictive environmental risk assessment.

MELISSA FABIAN MENDOZA: Melissa Fabian Mendoza, Director of the Environmental Risk Assessment Division, for the record.

I can confirm that CNSC staff have reviewed the

predictive environmental risk assessment from OPG and have found that it did take into account impacts of physical stressors such as noise. So the environmental risk assessment considered predicted noise as it may impact both wildlife as well as surrounding populations and compared it to appropriate and conservative benchmark values.

Overall, the conclusion from CNSC staff's review is that the noise impacts from the proposed activities would not cause a significant adverse effect on wildlife or the surrounding population.

That being said, should the project proceed, OPG would be required to implement mitigation measures which they have already outlined in the predictive environmental risk assessment, as well as doing ongoing monitoring to confirm that those predictions are accurate if the project proceeds.

MEMBER McEWAN: And you're comfortable with, if I can call it, the hassle factor of increased traffic through the community into the Pickering site?

DR. ALEXANDRE VIKTOROV: Well -- Alex Viktorov, for the record.

That is somewhat outside of our mandate, but the impression we have is that the community or the municipality is very strongly supportive, and OPG has put in plan the adequate measures to improve the infrastructure

so that the hassle would be minimized, but the question would be probably directed better to the community.

MEMBER McEWAN: Mr. President, thank you.

THE PRESIDENT: All right, thank you for that. Commission Member Hopwood, please.

MEMBER HOPWOOD: Thank you, and thanks everybody for both the presentations and a lot of good answers that have been helpful to me as I've been listening.

I do have a few more questions. So I'll start with following up the discussion about the organization and how the organization of this complex series of projects and programs works. I'm conscious that it's not only complex, it involves some things that haven't been done before, where we've got units shut down, units operating, and units being refurbed, I think theoretically all at the same time, at some point during the next 10 years.

So what I wanted to hear about the organization, you have to have a complicated organization. Your org charts are quite detailed, but I also see that they've already changed since the original application in 2025, last year. The most recent supplemental material sent shows some adaptations of the org chart.

So I'll start by asking about conflict resolution. You have different people with different

responsibilities. There is the license holder, but there's also a number of other people who are involved in taking responsibility for individual activities, along with corporate.

So if there is a discovery issue, let's say, which involves a challenge to the project for refurbishment or decommissioning, or a conflict between operation and refurbishment, what's the conflict resolution mechanism? It's not just about authority, but what's the process that helps you come to the appropriate decision when there's a conflict?

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

I will start with the org chart transition to provide some clarity. After all four units safely shut down in September, on October 31st transitioning to November 1st, the license holder will transition from the current site VP to the VP of Operations and Refurbishment.

In simplistic terms, all of the site, all of the staff on site will essentially transfer over to the enterprise projects organization. So for the staff at site, every employee at site, they know where they're headed. That was the commitment we made to staff. That by June 30th they know what their role will be in refurbishment. So for the staff, although there's an org

chart change, they know where they're headed and the jobs that they will be performing.

We do recognize that when all units are shut down and we're heavy into the project phase, OPG is already looking ahead based on industry benchmarking of what it takes to re-establish a station organization as we island Unit 5 as our first unit return to service. That is the next major transition point.

Our benchmarking from industry has said that we will stand up that station organization usually around six months prior to new fuel load, as we move through our return to service to ensure that we can meet all of the requirements and we're ready to have both the refurbishments. So as you alluded to, we will still have three units in refurbishment while we have one unit in an operating state.

We've referred to this as our lead out plan, but really the lead out refers to transitioning from four units in refurbishment to one unit operating and the remaining units in refurbishment. This plan is currently in progress with five key pillars around organizational readiness, work integration, proficiency, standards, programs and processes, as well as operational focus and equipment reliability.

So you will see the station organization, which

is referred to in the CMD, begin to be developed just prior to new fuel load of our first unit on Unit number 5.

As we work through that process, as was already discussed, there are a number of oversight forums which we will leverage to ensure that we are successful and compliant with our regulatory requirements.

We do have a World Association of Nuclear Operators readiness review as part of our restart, which will take into account, what does that organization look like as you transition from a construction state into an operating state? We have our Nuclear Safety Review Board oversight, our independent nuclear oversight, as well as our centre-led functional management division, CFAM division, which will also provide oversight as we progress through the transition into a restart.

As far as disputes, can I just get some clarity on what you're asking of it?

DAVID ROGERS: Dave Rogers, for the record. Thank you.

We currently institute a lot of processes. The example you gave was, let's say you have a discovery item, a broken item. First stage is a director's scope review board. So the directors on project and operations will be represented there. They'll look at the item, decide if there's a technical thing. We have engineering

decision-making forums, which is a very structured, rigorous process on making engineering decisions.

If it's more of an operational decision, and future operations -- impacts future operations, we have an operational decision-making meeting, which the VP of Plant Ops has the final say in that decision.

Should it get escalated, you've got your senior vice president of the projects, and certainly with input from the operational voice there to ensure that it is appropriate for future operations.

And if it's a big project decision, that can get escalated to the Chief Nuclear Officer and the Chief Nuclear Engineer. And if it's really large, that would include the Chief Executive Officer of OPG as well. So many defence-in-depth layers of decision-making appropriate to the decision that's on the table.

MEMBER HOPWOOD: Thanks, I appreciate that, and I'll pass it over to the CNSC staff, just if you have any comments on these processes.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record.

Yes, we can confirm that OPG has developed a management system including the resolution of conflicts and making decisions with consideration of various factors. From the CNSC perspective, what we expect is that the

safety first priority of safety in executing projects or resolving conflicts, so that's an aspect that we will be monitoring. But from what we observed in the current operation, the processes are adequate and they do prioritize safety over operational aspects.

MEMBER HOPWOOD: Thank you. Thanks.

Coming to this in a slightly different way, and knowing that the level of complexity of the site is going to increase, under the HR SCA I read about some of the considerations of monitoring and assessing performance, a lot of discussion about observation and coaching by supervisory staff. I would like to hear how you get feedback coming the other way, from the staff on the front line, the crews. Can you give me some feedback on how you incorporate the front-facing workforce, the feedback they give, because they are the people who have to cope with the work on the spot.

KERRY JACKSON: Yes, thanks. Kerry Jackson, Operations Director, for the record.

To answer the first part of your question, you touched on the observation and coaching program, which is really fundamental to OPG. It's a program that provides an opportunity to see leading indicators, both strengths and areas that may require some additional focus, and then communicate these observations and learnings to our staff

to improve in various areas, either leverage something that's going really well, or in some cases put some additional controls and mitigating actions in place if it looks like there's a potential drift in standards.

The staff have accepted our observation and coaching program. They are part of it. They perform observations and the coaching with oversight from our leadership teams across the site.

The second part of your question was around feedback. So observation and coaching is one opportunity to leverage a conversation between two individuals around performance on how a particular task was performed. However, that's not our only opportunity to hear from staff. We have various forums at our site, town halls that are held regularly. With a number of changes that are happening at site, we have been leveraging our communications team in the town halls to hear from our staff on site on what their concerns are and be transparent about what our resolutions are for concerns.

Some of those items around infrastructure, as anybody would have understood, is top of mind around a number of people coming to site, and we leverage those forums to be able to obtain that feedback and communicate what we're doing in a transparent manner back to our staff.

Another opportunity that we use is our nuclear

safety and security assessment, which is an assessment that is performed every five years, last done in 2022, with annual check-ins with the staff where they have an opportunity anonymously to provide their feedback in how we're doing in nuclear safety and security. Those responses are rolled up and reviewed by our senior leadership team, who then put actions in place to address concerns, but also communicate some of the things that we're doing well. That last assessment was performed in 2022.

And the last comment I would make on the nuclear safety and security assessment, as well as hearing concerns from staff, is that when I say OPG staff, that's all people who are on OPG sites. So our contractors are integrated within our forums. They provide feedback just as any OPG employee would, and they are also included in our nuclear safety culture assessment so that we can hear from everybody on site what their concerns are and ensure that we can communicate any challenges that are raised.

MEMBER HOPWOOD: Thanks very much. I appreciate that feedback.

Just one other question, and it's really to follow up some of the discussion already on the increased numbers on site. So, it's quite a lot more people, as we heard, that are going to be on the site at the peak, and an

awful lot more traffic, including a lot of heavy traffic. How might that affect and how might you incorporate that into emergency planning, where you may have to evacuate the site or you may have to take actions outside the site area?

SCOTT PRESTON: Good morning. My name is Scott Preston. I'm OPG's Director of Emergency Management and Fire Protection.

To the question around, how do we safely manage that increased level of staff on site and evacuate them, we do demonstrate that we can evacuate staff from the Pickering site through what we call early dismissal drills, in addition to regular assembly and accounting drills through our emergency response organization. Those drills, including the early dismissal drills, will continue throughout the refurbishment period and beyond.

I think it's important to note that with the site in refurbishment, once the units are defueled, there's a fairly significant reduction in the nuclear safety risk, and the types of events that could be possible are unlikely to lead to an evacuation being a realistic public protective measure. However, as I said, we will continue to demonstrate this to ourselves and maintain that proficiency in the unlikely event that dismissing staff from site is required.

One thing I'll add is to say OPG performs

evacuation time estimates regularly as a requirement under our licence. That's an estimate of how long it takes to evacuate different areas under different conditions. Our analysis does conclude that the overall evacuation time estimates are minimally sensitive to increases in the numbers of staff on site. We did some sensitivity analysis and included well over a thousand additional vehicles during that period. And by "minimally sensitive", that's characterized as less than a 15-minute increase across all scenarios to evacuate those areas around the station.

MEMBER HOPWOOD: Thanks, I appreciate that. I will just pass it back in case staff have any comments on that.

DR. ALEXANDRE VIKTOROV: It's Alex Viktorov, for the record.

Not much to add to what was shared by OPG. I'll only add that CNSC staff will continue to conduct evaluations and inspections of the drills and activities related to preparedness for emergency response.

MEMBER HOPWOOD: Thank you, and back to the President.

THE PRESIDENT: All right.

Commissioner Lacroix, please.

MEMBER LACROIX: Well, thank you very much. Most of my questions to OPG have been answered, so I'll

turn to staff for my question.

We all recognize that this is a very complex and ambitious project. For the first time, on the same site and simultaneously, four reactors will be refurbished and four reactors will be in the process of being decommissioned. And when I look at all the safety programs that have been designed and implemented thus far, they were done so for an operating nuclear generating station. And I wonder, are these programs robust enough, appropriate for such a project? Do you foresee any challenges or loopholes?

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record.

We haven't identified any loopholes so far. We do acknowledge the point that there will be complexity in the various projects being executed at the same time. I will point out that the decommissioning during this 10-year period will be rather limited. It will be only non-nuclear decommissioning, so the reactor facilities themselves will not really be decommissioned. It will be some systems that are not nuclear removed just to minimize the risk and maybe free space for various refurbishment activities, as well as demolition of buildings outside of the plant proper.

But we do acknowledge that the situation with four units shut down and being in simultaneous

refurbishment is somewhat different. We do have experience with two units at Bruce being refurbished at the same time, Units 3 and 4. It does put some additional complexity, but not too much.

So again, that's an area that we will keep an eye on, but we do believe that the programs themselves are robust enough to accommodate it. The complexity, however, and the intensity of work will grow.

MEMBER LACROIX: And do you have enough resources within CNSC to follow this project?

DR. ALEXANDRE VIKTOROV: Alex Viktorov.

We believe we do. There was a time when we were preparing to reduce staff at the Pickering site because of a unit shutdown. We anticipated all units will be shut down, so we were planning for a gradual coasting down of the number of people at the site. Again, after the Province announced the intent to refurbish units, we revised our plans to assess the number of people who will be there, and we currently believe it's quite appropriate to keep the current supplement of inspectors and program officers as we have, with the understanding that we have some flexibility to bring people from other sites or, if we have excess of resources, we can allocate them to other activities. So by and large, we believe we are resourced adequately.

MEMBER HOPWOOD: OPG, anything to comment?

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

While we understand the complexity of a large-scale project, what I would add is Pickering -- as an eight-unit site, six units recently operating, Pickering is regularly in multiple outages occurring at the same time, and our programs and how we execute work is really designed around this scale of work that's taking place.

While we recognize there will be some adjustments to our programs in leveraging operating experience from the Darlington refurbishment project, we don't see large-scale changes to how we perform work. We've been successful with performing multiple outages at the same time -- a one-site, one-team approach -- and we will continue to leverage that success with adjustments as required to ensure that we're meeting all regulatory requirements as we work through refurbishment and decommissioning.

SARA IRVINE: And Sara Irvine, for the record.

I would just like to add some context that, while decommissioning activities are asked for in this licence, it's a very small set of work packages that are within our overall detailed decommissioning plan. It's really related to stabilization and a very limited amount

of risk reduction work on the Pickering 1-4 side where we're taking out non-nuclear. So it's a very small proportion of the overall work. So it's not like there's a big decommissioning work program on one side and a big refurb on the other; it really is a small scope on the decommissioning side.

MEMBER LACROIX: Thank you very much.

Back to you, President.

THE PRESIDENT: All right, thank you.

So over to me for some questions. So we will get into the refurbishment questions this afternoon, but broadly, I guess the decision made decades ago by OPG not to refurbish the Pickering facility led to some significant changes in maintenance practices, and even hiring, certainly in investment. So the world has changed, imperatives have changed, and you are coming forward with this request and this proposal. And I appreciate that the condition assessments have been done, there's been a fair bit of work. You can't look at everything, you can't see everything. As you progress forward, what accommodations have you made for discoveries, things that you had not anticipated? I think you spoke briefly about the mechanisms to technically review, but in terms of risk management, what are you expecting, what have you sort of set aside, if you will, as a contingency for discovery as

you move forward?

DAVID ROGERS: Dave Rogers, for the record. I will touch on a few things, President.

The last number of years, you mentioned our maintenance practices. We drove down maintenance backlogs significantly in the last number of years. And so that had been a focus for us, that plant reliability aspects. For the refurbishment project, discovery work, things like that, we do have a substantial sustaining portfolio that is significant capital other than refurbishment that we will be allocating for some of those other projects that didn't make the refurbishment scope, and also some discovery work.

One difference about this refurbishment, I'll say, is the involvement of the site maintenance staff in the refurbishment activities. And they will also be working on backlog and, I'll say, fix-it-now opportunities to ensure that safe, reliable operation on Unit 5 return-to-service and the subsequent return-to units. So that staff is fully engaged during the refurbishment period.

And it will also have oversight. We do have, I'll say, an operationally focused return-to-service Director that is providing that operational focus on the current backlogs or discovery items and things like that during the refurbishment period. And it will go through

the normal processes of scope additions and things like that and use of maintenance resources to tackle it during refurbishment.

There will also be a section of Engineering that will be focused on the plant systems that are shut down or in layup. So the layup monitoring during refurb, the system shut down to ensure that they're safe and reliable on return to service. And the systems that aren't being refurbished will go through the normal system health processes and equipment reliability, plant reliability processes during refurbishment.

So we are confident that we'll be able to have that reliable plant on return to service.

THE PRESIDENT: All right, just to pick up on that last comment. So you are intending to carry out your system health reviews throughout the project? Is that correct?

DAVID ROGERS: That is correct.

THE PRESIDENT: Okay. All right, thank you. That is helpful.

Staff?

ROSS RICHARDSON: Ross Richardson, for the record.

So as mentioned previously, we have already developed a refurbishment compliance plan, which took

lessons learned in OpEx from previous refurbishment projects. As part of the planning for development of that plan, we did allow, I'll say, slack in the plan, if you will, so we don't over-plan, so that we have enough time to perform, be more risk-informed, and improve the overall efficiency of our staff.

I did want to mention as well, in terms of -- the staffing has come up a few times, so just for context, we do have 20 people working directly under myself, fully dedicated to the oversight of the Pickering site, in combination with the hundreds of specialists that support us overall. So we'll have, as I mentioned, 20 people fully dedicated to this. As mentioned, in terms of the overall plan, again, we don't over-plan; we want to be risk-informed, we want to be efficient with our resources, but we do leave slack in the plan so that we can follow up on things as they come forward. I'll leave it at that.

THE PRESIDENT: All right, thank you. I think we will leave it there for now. We will come back for the next round of questions. We will take an hour's break for lunch. Thank you.

--- Upon recessing at 12:09 p.m. /
Suspension à 12 h 09

--- Upon resuming at 1:10 p.m. /
Reprise à 13 h 10

THE PRESIDENT: All right, welcome back.

One aspect of our regulatory regime is that while the CNSC is the lead regulator for most projects associated with the nuclear industry, we don't work on our own. And I just will note that there are a number of ministries and other agencies who are joining us, I think all of them virtually, today. But I'd just like to give voice to them and ask.

So we'll start maybe with Fisheries and Oceans Canada. Just if I might give you the floor and ask you about your oversight of this facility and of the project that we're considering. And so I believe there's somebody from Fisheries and Oceans Canada. I'll turn the floor over to you for comments around your role with this licensee.

JUDE SAMSON: If there is somebody from Fisheries and Oceans Canada, maybe raise your hand in the Zoom feature.

THE PRESIDENT: All right, how about the Ministry of Labour, Immigration, Training and Skills Development? Are you with us today?

SUZANNE THORNHAM: I am, thank you.

THE PRESIDENT: All right, go ahead, please.

You're muted. We cannot hear you.

SUZANNE THORNHAM: Muted on my side.

THE PRESIDENT: Yes, we hear you now. Thank you.

SUZANNE THORNHAM: Okay, perfect. Suzanne Thornham with the Ministry of Labour, Immigration, Training and Skills Development. I'll start first and then I'll pass it over to my colleague Chris. I'm the acting manager of the Radiation Protection Laboratory.

And our role in this is that we do the environmental monitoring around Ontario's nuclear reactors. So we just do routine monitoring of air, soil, water, food products, milk to ensure that there's no adverse -- that everything is as expected, that there's no increased radiation in any of those items. We're also involved -- oh, sorry, go ahead, Chris.

CHRIS BOCCINFUSO: No, go ahead, Suzanne.

I was going to say good afternoon. My name's Chris Boccinfuso. I am the senior manager for the Construction Health and Safety Program.

The Construction Safety Program in Ontario is responsible for doing proactive and reactive inspections related to worker health and safety. So during refurbishment, were there to be problems or accidents that need to be reported, our inspectors would be doing the

investigations into those. And we would also be involved proactively going in and doing inspections based on operational needs.

THE PRESIDENT: All right. Thank you for that. Maybe turn to staff to talk a little bit about the relationship with this agency as well.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record. CNSC does have a memorandum of understanding with various departments and ministries. And we build on them to share responsibility and avail ourselves to different resources. In particular, with Ministry of Labour, we often cooperate in conducting inspections related to conventional health and safety as well as when an incident occurs that relates to worker safety.

That cooperation is fruitful for us. Again, we draw on the experience, and we share resources. We also work together when an actual incident occurs involving occupational health. So that's the primary responsibility of the ministry.

THE PRESIDENT: All right. Thank you very much for that.

I understand that Fisheries and Oceans Canada is with us now and I'll give you the floor as well to talk about your relationship with us and your interest in the project.

SARA EDDY: Hi, Sara Eddy. Can you hear me all right?

THE PRESIDENT: Yes, we can hear you great. Thank you.

SARA EDDY: Great. Thank you. Sara Eddy. I am the regional manager of the Fish and Fish Habitat Protection Program in Ontario and Prairie Region.

So Fisheries and Oceans Canada has a role in terms of reviewing proposed projects under the *Fisheries Act* for harmful alteration, disruption, and destruction of fish habitat and also for the death of fish. So that is a regulatory role that we have worked on together with this Board and CNSC. We have in the past issued an authorization for the ongoing operations of Pickering nuclear, and we will review the refurbishment proposal under a separate file and determine what impacts to fish and fish habitat may relate from that project.

THE PRESIDENT: All right, thank you. Thank you very much for that.

Emergency Management Ontario, I believe you are with us also.

RANDY REID: Yes, good afternoon. Randy Reid, director, Nuclear, Radiological, and Scientific Response Programs for EMO. And with me today is also my colleague Michael Munro.

Michael, would you introduce yourself, please?

MICHAEL MUNRO: Good afternoon. Michael Munro, for the record. And as Randy suggested, we are from the Ministry of Emergency Preparedness and Response for Emergency Management Ontario.

THE PRESIDENT: All right. I guess we will leave it there. I will just remind the Commissioners that if there are questions that relate to that mandate, they're available for answering questions. So thanks, all of you, for doing that.

Let's then move on to questions about the refurbishment of Pickering 5 to 8, and we'll start with Commission Member Hardie, please.

MEMBER HARDIE: Thank you. Before we get into some of the technical pieces for the refurbishment, I did want to have a question revisiting the Section 20.5.4 of REGDOC-2.2.3, which is what requires that period of work under supervision, where returning certified staff with active certified staff prior to returning to active status, so that there needs to be that supervision piece. And because Pickering units, there will be none operated, you are applying then for there to be no certified staff. I'm going to read my notes here. Since there will be no certified staff still in active status to provide that necessary supervision, there is a proposed group of core

certified staff to use work performed during the time of the over-poisoned guaranteed shutdown state as a substitution for that supervision period.

And so I would like to first ask OPG, how would compliance -- first of all, does this work performed during that OPGSS meet the requirements needed from supervised staff, or will there also need to be additional training and supervision once the CNSC regulatory hold point is released to proceed beyond that?

KERRY JACKSON: Kerry Jackson, Operations director, for the record. So first I'd just like to go back. OPG is currently in compliance with REGDOC-2.2.3, certification of reactor facility workers. As was stated earlier, all Pickering 5-8 units will be defuelled at the start of the refurbishment. Therefore, OPG has provided an alternative approach to meet the requirements of subsection 20.5.4, work under supervision, prior to the Unit 5 restart, which is our first unit that is expected to return to service. The CNSC has commented on OPG's plan with reference within their Commission Member Document as well.

As far as the plan itself, so our certified staff will remain at Pickering throughout the refurbishment period supporting the refurbishment, and they will be performing operationally focused roles through that time.

So a number of the duties of a certified staff

will continue as they are today with the authorization of work, administration of work protection as well as performing operational reviews for the activities that will be performed at site.

Our certified staff will also continue to train in our full scope simulator as they do today, practising and being evaluated on various event scenarios, as well as skills training to maintain proficiency. Our training plan has been updated. We do have a five-year training plan, which has been updated to reflect the continuing training aspect of all of our certified staff at Pickering throughout the refurbishment, and also has been updated to include return-to-service-type activities.

As part of your question earlier around the time period that OPG proposes to use, we do intend to use the period from over-poisoned GSS, seeking approval prior to surrender. We do intend to assign a group of certified staff prior to new fuel load to ensure that that specific group will be the group that will oversee Unit 5 restart. And all of the training and skills that they would take through that period will be current to what they are performing today to maintain their licence.

Part of the reason we chose that specific period under the work under supervision, in the control room, there are a number of activities that are taking

place during that time that will reflect the skills required for a trained certified operator to perform as they work through a number of commissioning dynamic-type activities which would be required for certified staff to perform during that period.

MEMBER HARDIE: Thank you for that.

So regulations and laws and technologies are obviously going to change over the time of the proposed licence. And how do you propose handling that in order to ensure that staff are remaining competent and adequately trained?

KERRY JACKSON: Kerry Jackson, Operations director, for the record.

Currently, our certified staff do train. There's a set training requirement for their position. They will maintain that training throughout the refurbishment period.

Refurbishment is introducing some design modifications that will take place. As part of our modification process, there is a training analysis that is required. And all of our staff will be required to ensure that they've completed all of the necessary training to ensure that they can both meet the requirements of the existing design basis, which they're continuing to do today, as well as meet the requirements of the new design

requirements for some of the proposed modifications.

For the remaining staff at the OPG site, their training will be updated to reflect any changes that should occur for any regulatory changes. That is part of our systematic approach to training analysis for implementation. And that training is updated, which is what we do today. If there's a change in governance or regulatory changes, our training is updated to reflect that. And staff who hold those qualifications are required to take that training to maintain their qualification.

MEMBER HARDIE: One final question before I turn to staff. Was there ever any consideration in using the Darlington site as a possibility for obtaining that under-supervision certification? And if that was considered, why or why not was that not chosen as an alternative?

KERRY JACKSON: Kerry Jackson, Operations director, for the record.

I would say, yes, it was considered. The challenge with the Darlington site is it is a separate certification for their operators. It is a different design concept.

We do leverage our fleet operations team to provide support with observation and coaching and maintaining standards. However, as far as the

certification goes, it is specific to the Pickering 5-8 site specifically.

And I can draw on an example with Pickering 1-4 end of commercial operations. A group of our Pickering 1-4 certified staff are currently in our Pickering 5-8 certification program and co-piloting in our main control room right now, looking to obtain their certification prior to refurbishment start.

MEMBER HARDIE: Thanks very much.

I will turn to CNSC staff. And I would just like to add the question in there about how would compliance verification look for that proposed alternative if it was accepted?

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record. I will start and then I will ask Lucie Désaulniers, director of Personnel Certification Division, to elaborate.

But I will only add from my perspective that should any requirements change during the period of refurbishment or while units are shut down, there will be an adequate, meaningful implementation time for OPG to catch up and make sure that they can meet the requirements if they change. So it wouldn't be a surprise. There would be time to prepare for any different requirements.

Now, specifically with regards to the situation

with Unit 5 and maintenance certification, I will ask my colleague to elaborate.

LUCIE DÉSAULNIERS: Good afternoon. Lucie Désaulniers, director of the Personnel Certification Division.

With respect to the oversight that we would do for this situation, of course we will have the regular meetings with the licensee to make sure that we explain our requirements and they are fully understood. But also we would do inspection and desktop reviews, more specifically of their methodology, which task will be chosen for the work under supervision to make sure that they are meaningful tasks where the candidate could demonstrate some of the competency that they would usually demonstrate during work under supervision, like demonstrate their knowledge and skills of course, but their proper execution of approved procedures, the ability to recognize, diagnose, and respond to abnormal conditions, adherence to station standards, conservative decision-making and effective communication.

So we would make sure that the program would include those verifications and we would of course verify the qualification of the proposed supervisors that OPG will propose and making sure that they have the qualification to supervise correctly the operators that want to be active

again.

Another tool that we have in our oversight is that we have a regular work under supervision field inspection that we could also use. And of course, we are retaining the regulatory hold point until the readiness is demonstrated, c'est à dire until OPG demonstrate that the staff are qualified to proceed.

Thank you.

MEMBER HARDIE: Thank you very much.

Just one follow-up question to your response, and that is, are there any critical scenarios that you feel would not be addressed within that over-poisoned guaranteed shutdown state that OPG may need to also find some way to address in terms of certification and training?

LUCIE DÉSAULNIERS: Yes, thank you. Lucie Désaulniers, for the record.

That will be part of our discussions and the information that we will ask the licensee to provide to make sure that all critical scenarios are covered. But the ones that are already identified in the regular training will be covered through their continuing training. They will continue training as they are right now. So we will, of course, make sure that nothing is overseen.

MEMBER HARDIE: Thank you for that.

And before I pass it back to the President, one

final question to wrap up. We have talked a little bit earlier about guaranteeing worker capability and for the shutdown itself. Are there any concerns with OPG about staff turnover during that period of not being in that active state, in that operational state, and potentially losing critical workers?

KERRY JACKSON: Kerry Jackson, Operations director, for the record.

OPG uses a 10-year workforce planning strategy to look at upcoming demands across the fleet. Pickering refurbishment, since it was being reviewed as a possibility, immediately looked at our demand and actually started our certification program from where it had paused momentarily from Pickering end of commercial operations. So from a certified staff perspective, we have individuals who are currently working through the program leading into being able to support both the shutdown and, if approved, the refurbishment of Pickering.

We've continued to map out our certification training plan and our non-licensed operator plan to continue with that 10-year outlook, looking at both succession planning from within, as typically our certified staff do come from within at Ontario Power Generation, and also looking at potential attrition for retirements.

So based on that, our staffing and hiring

requirements, we have the sufficient demand that's required to both safely shut down the units and continue on into refurbishment.

MEMBER HARDIE: Thank you very much. I will just allow staff if they have any comments they'd like to add. Thank you.

DR. ALEXANDRE VIKTOROV: Assuring that there is enough, a sufficient number of trained staff would be one of the prerequisites for hold point removal, so this aspect will be verified.

MEMBER HARDIE: Thank you. I will pass it back to the President.

THE PRESIDENT: All right. Commissioner Hardie is kind of raising some issues for me for follow-up, so maybe before I move on, I just want to get some clarity.

There are a number of design changes, physical changeouts that are going to take place. You've talked about migrating control systems that are at Pickering A that Pickering B will require for separating the two power plants. That's likely to change, perhaps, the design, the look of the control room.

So just walk me through your thinking here around operator certification, the changes that you're making to the simulator, and the expectations around retraining or training certified staff to interface with

those new changes. So just want to get a sense of the thinking that you're doing here.

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

I'll first start with the modifications. Any planned modification that will have an impact to the main control -- well, any planned modification as part of our modification process has a training analysis that is required that goes with that.

For any changes to the main control room that would impact our certified operators who are in the control room, those changes will be replicated in our full-scope simulator and they will be trained on those modifications in the full-scope simulator. So they'll have that training prior to the restart of Unit 5. I think that was your first question.

The second question I took as around changes potentially to our minimum complement, recognizing that we are moving some controls from our Pickering 1-4 control room to the Pickering 5-8 control room. So upon shutdown of all four units and the de-poise of our vacuum building, the minimum complement requirements for our Pickering 1-4 operators who are currently positioned in the 1-4 control room will no longer be required.

Through the refurbishment program, once all

units are shut down and the vacuum building is de-poised, our first phase would be to move those controls over to the Pickering 5-8 main control room. So upon Unit 5 restart, you'll no longer require a minimum complement in the Pickering 1-4 control room.

The Pickering 5-8 control room will not have a change to their minimum complement based on the controls that are being added, as it's inside the existing design basis and our certified operators will be trained on those controls. It's a very minimum amount of controls related to the vacuum building controls.

Minimum complement as a whole, if I kind of walk the team through that, we do expect that as units are defueled, that will change the design basis. So as each unit is defueled, we expect that there would be a reduction in the number of certified staff required in the main control room.

Very similar to the process that we followed for Pickering 1-4. When the first unit was defueled, their minimum complement decreased. When the second unit was defueled, their minimum complement continued to change. We'll follow the same process for Pickering 5-8.

The difference for Pickering refurbishment is we will continue to staff our certified staff in the main control room 24/7 on our Alpha to Echo shift

crews -- that's our 24/7 shift crews -- throughout the refurbishment period to facilitate work and also it gives an opportunity to provide observation and coaching on the standards that are required in the main control room with panel operations, responding to alarms and ensuring that their training is being executed as we would expect.

THE PRESIDENT: All right. Thank you for that.

Any comments by staff?

ROSS RICHARDSON: Ross Richardson, for the record.

Just to add to what OPG has said and just in terms of the CNSC Staff's perspective on that -- and Dr. Viktorov alluded to it earlier in terms of staffing levels, but training is obviously a key input that we look at as well prior to removing any regulatory hold point.

So as part of the process specified in the *Licence Condition Handbook*, we will be developing a return to service protocol as well, and that will include specific training aspects and there will be specific training plans that need to be reviewed by staff and verified before we release that hold point.

THE PRESIDENT: All right. Thank you for that. That's helpful.

All right. Let's move over to Commission Member Berube, please.

MEMBER BERUBE: Yes, I just want to get into some of the turbine generator basically refit operations writ large for you to go through this. You've mentioned that you're going to get an extra 30 megawatts at each one of these units. Maybe you could just run us through some of the logic on how you propose to actually make that happen. It's fairly significant.

And also the control packages for the stator and the rotor as well and the proposed refit of the control room or what control and operations are changed as a direct result of that refit. So if you could just run us through that logic, that'd be good.

JEFF MILLER: Jeff Miller, for the record.

So the increase in megawatts from the turbine generator project is approximately 30 megawatts per unit. That is primarily due to improved efficiencies with the turbine spindles, the HP and LP turbines. There is additional gains from replacement of our moisture separators.

MEMBER BERUBE: So on the thermal side, that's where you're getting the extra push?

Okay. And are you changing the actual control systems for speed control on the unit at the same time or is that being -- is that the same thing, that you intend to move forward with that?

JEFF MILLER: Jeff Miller, for the record.

So the control system is being upgraded to a digital control system. The logic, essentially, is similar with the same sort of trips and safety functions for the control system.

MEMBER BERUBE: And if we look at the actual stator and the rotor actual rewind, are you putting a new control system on that as well, and how do those two actually tie into the main control room? Does it change the actual layout of the control panels and the way the operators have to actually operate that system?

JEFF MILLER: Jeff Miller, for the record.

I'll have to take an undertaking and I'll get back to you with those specific details.

MEMBER BERUBE: And while you're digging that up, actually, because, actually, it's important, the other side of it, of course, is you're looking at replacing the DCCs, the digital control computers, and as that feeds into the digital control computers, that's quite significant for plant safety. We're going to need to know basically how that's integrated into the new DCCs.

And maybe you could -- I don't know if you were prepared to talk about the DCCs today, are you? No? Yes, no?

JEFF MILLER: We can talk.

MEMBER BERUBE: If we don't get to them today, then -- okay. Let's talk about the DCC refit and the logic that you're looking at there and what you intend to do.

JUDE SAMSON: Just for the record, I want to be clear about the undertaking that we're taking and when we're expecting to receive that -- when we might expect to receive that information.

JEFF MILLER: Jeff Miller, for the record.

We will get back to you in two hours the details on the turbine control.

JUDE SAMSON: So that's details on the control logic with respect to the stator rewind as well as impact on the control room.

And in two hours, so later in the --

JEFF MILLER: Later this afternoon.

JUDE SAMSON: Okay.

MEMBER BERUBE: So are you prepared to talk about the DCCs now? Is that possible?

The DCC refit. You're going to replace them and basically -- I guess right now you're using an emulator already, so you intend to replace the entire system? Just go through the general details of that, please.

JEFF MILLER: Jeff Miller, for the record.

So we will be replacing the DCCs beginning with Unit 6 during the refurbishment outage. So Unit 6, 7 and 8

will be replaced, following up with Unit 5 in the first 3351 planned outage.

The specific technical details, could you repeat the question that you're looking for?

MEMBER BERUBE: Well, I mean, this is pretty -- very specific, but I mean, the plant works on what we call PID control, right? I mean, it's basically old analog systems, and it's all routed into the control system. The old computers basically were handling that in the way that they were handling it.

But you actually replaced these computers already at one point with emulators, right, to basically emulate those machines. And so what I need to know is -- there's a bunch of new information that's coming from the turbine generator, which is fundamental to the actual operation of the plant for safety, and it's got to tie into your digital control computers. I'm just wondering, you know, how (a) you're going to do that, and (b) how are you going to ensure that that is working properly and is safe before you start bringing this thing online? Because it's a critical part of it.

JEFF MILLER: Jeff Miller, for the record.

So we are in the midst of detailed design right now for the digital control computer replacement. The design work is based on OPEX, industry OPEX. The team has

done extensive benchmarking with Bruce Power and other utilities.

The commissioning will address all of the integration between the upgrades to the turbine control system as well as the changes to the DCC. So we will have a very detailed commissioning specification to ensure that all of the design requirements that are laid out in the modification process are addressed and met with sufficient margin.

MEMBER BERUBE: Well, I'd like to see a little bit more on this for Part 2, if that's possible. So it would be an undertaking to get a look at how you're going to implement that because I have no visibility to it.

And also I'd like to talk to the staff here at this point. But this is coming online. There's a fair amount of changes in this. And I just want to know how you're going to go through and actually make sure that's working properly and that everything's good to go.

DR. ALEXANDRE VIKTOROV: Alex Viktorov.

And I'll ask my colleague, Sophie Gingras, to elaborate.

But I'll start with pointing out that, of course, Pickering is an old facility with many I&C systems being outdated. And we do expect that OPG will have to replace and update many of those systems. So certainly an

area of attention.

And I'll ask my specialist colleagues to elaborate what kind of oversight we're planning to do.

We seem to have a bit of difficulty finding the right specialist at this time.

MEMBER BERURBE: I think we've got someone ready to speak to it.

MELANIE RICKARD: Ready to speak to it briefly. Good afternoon. Melanie Rickard. I'm the Director General of the Directorate of Assessment and Analysis.

We absolutely do look at the upgrades to the electrical system and the I&C through refurbishment projects. And as was mentioned, we have extensive experience in that oversight.

Unfortunately, we do not have any experts available this afternoon to elaborate on those activities. So we, too, could take an undertaking if that's what you would like.

MEMBER BERUBE: Yeah, that's fine. I didn't expect that you had full detail at this point. But I just wanted to bring it up because certainly we need to chase that one down. It's fairly significant.

JUDE SAMSON: Just for the record, is that to be followed in Part 2, Dr. Berube?

MEMBER BERUBE: Yeah, I think so. We can't get

to it today. It's probably too much detail. They don't have that detailed design yet.

JUDE SAMSON: Thank you.

MEMBER BERUBE: Another question for OPG that I've got, and it's got to do with aging management, particularly, the structures themselves are getting mature. And I'd just like to get into how you're actually dealing with the concrete aging management there.

What are you doing for testing, validation that the concrete's holding up well? We're talking about extending this another 30 years, which is a long, long time. Some of that stuff's been there for well over 50 years now. So how are you actually going through and validating the concrete and the structure is okay, sound?

And then the other side of that is also cable management in terms of the actual cable aging. When you start looking at 100 years on an electrical cable, the insulation has got to be checked on a regular basis for suitability to make sure that it's working.

So these two things are the biggest questions I have. If you could just give us some detail on both of those.

JEFF MILLER: Jeff Miller, for the record.

Regarding concrete containment structures and aging, so OPG has a comprehensive aging management program

aligned with REGDOC 2.6.3 to demonstrate ongoing fitness for service and to support long-term reliable operation. We manage aging of Pickering's concrete containment structures, so the reactor building, vacuum building and the pressure relief duct through a longstanding program of regular leak rate testing and periodic inspections that meet CSA N287.7 requirements. The test results are trended over time and reported to the CNSC, and they confirm the current performance is within regulatory limits with sufficient margin.

And importantly, our inspections do not show signs of accelerated concrete aging affecting structural integrity. Where leakage trends occur, the main contributors are typically non-structural items such as seals, gaskets and joint materials. So refurbishment project will, therefore, focus on renewing these aging components and taking advantage of improved access to inspect and repair areas that are difficult to reach during normal operation. Post-refurbishment, containment integrity will be improved.

MEMBER BERUBE: And of course, you're looking at all penetrations in and out of the actual -- I guess in this case would be the vacuum building as well as all the reactor buildings and vaults as well?

JEFF MILLER: That is correct.

MEMBER BERUBE: And any input on the cable management and aging issues that may be there? I don't know, so that's why I'm asking the question.

JEFF MILLER: So Jeff Miller, for the record.

We do have an aging management program. We've actually done some prerequisite testing leading into the refurbishment project to help define the scope. We have structured our program. It's based on a risk-based approach, so broken down into families of different cables, high, medium and low voltage cables, as well as the environment that they're located in.

So we've bound our testing program based on the worst-case conditions for those respective cables, and that will define the results of our refurbishment scope.

We also have lots of information based on our environmental qualification program that has testing data and qualification data with respect to the qualified life of the cables in our station.

MEMBER BERUBE: So do you have any idea right now how much you're going to have to replace as part of this, or are you still trying to figure that out?

JEFF MILLER: Jeff Miller, for the record.

We are working through the results of our prerequisite cable testing to define the scope. We don't anticipate at this point in time a significant number of

cables needing to be replaced, and this is based on industry OPEX as well, where even south of the border for life extensions down in the States, the utilities have not embarked on a mass cable replacement based on their testing and analysis. So we don't anticipate to be an anomaly when it comes to that.

MEMBER BERUBE: And you said you have to replace some of them. What about the high voltage cables? Which ones are you actually looking at that look like they may need to be replaced?

JEFF MILLER: So the high voltage cables. Other cables of concern are ones that -- submerged cables, for example, that have unique potential degradation mechanisms.

MEMBER BERUBE: Okay. That gives me an idea what you're up to, so that's very good for now.

And I'll pass it back to the President at this point.

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

I just wanted to go back briefly to a couple of your earlier questions. The first one I wanted to respond to was changes to the main control room with some of the changes associated with the turbine controls and the DCCs.

The first with the DCCs, the DCCs' first

replacement will be for unit number 6. Unit number 5 will come later. The changes for our control room personnel will be, I'd say, quite minimal for any changes to them. There will be no additional actions that they would be required today. It would be mainly upgrades to monitors. The real change is in the CPU.

As part of that design change, all of our design changes, which include changes to the turbine generator equipment, it does follow an engineering change control process. Recognizing the complexity as part of that process, there are reviews that trigger any interfacing systems that are required to be looked at to ensure that all of the systems that are required to speak together can speak together to ensure that safety is maintained and that we are inside the regulatory requirements.

From a main control room perspective for our turbine generator controls, we are expecting mainly, I'll say, visual controls will be the main change in the same location on the panels as they exist today. But as part of our engineering change control process, that modification will be installed and tested on a simulator and our certified staff will train on that prior to using that in the main control room.

So there are some changes that are still being

worked out through the design process, but certainly, as part of our review and design process, all of those interfacing systems will be understood, known, tested to ensure that it can meet our safety requirements.

SARA IRVINE: And Sara Irvine, for the record.

Just building on what Mr. Jackson said, that fundamental of the ECC process where it captures all those elements for changing something on the turbine generator control and how that propagates to the main control room, that's at the heart of the undertaking that we took.

So if that answer satisfies the Commissioner, we'll consider that complete.

MEMBER BERUBE: Yeah, it gets me some of the detail that I'm looking for. But I mean, like, fundamentally what my concern here is you've got some mature operators that have been there for a long time. They're not going to go away maybe. Maybe they are, I don't know.

But fundamentally speaking, if the control room changes quite a bit, then the impulse is to use what you know instead of what you've been taught recently. And so the more change you make, the higher the probability of problems, as you're probably well aware. So this is my concern.

So if you're emulating the analog systems in

place already with the digital systems, then it's not really a problem. If you're doing something radically new, maybe it is. I don't know. So this is why I'm asking.

And back to the President.

THE PRESIDENT: All right, thank you.

Commissioner McEwan, please.

MEMBER McEWAN: Thank you, Mr. President.

I'm hoping you can help me understand the deep-water intake. It's discussed a lot in the document, and there's very little detail of what actually is involved in this.

It sounds a complicated engineering process. How complex, how much disruption, how essential is it that it is in place to decide on the modifications, if you need any, for restarting Unit 5? Or if you don't go ahead with that and you go ahead with your backup plan, does that change what you need in place to bring Unit 5 back into service?

LUCA CECCATO: So, Luca Ceccato, for the record.

Just to give some perspective on the deep-water intake, and I know I talked about it this morning, but a little more clarity on that. So we do expect it to yield several benefits, improve thermal efficiency, support for increased electrical output about four megawatts per unit

with the deep-water intake in place, minimizing of biofouling events.

The deep-water intake fish impingement will be comparable currently with what we have in place with the surface intake because we do have a net to divert the fish. So that's going to be comparable.

Right now we are currently in the detailed design phase with construction and planning going on in parallel. So we are actually in the field executing. We have moved 36 dry storage modules that Mr. Richardson had previously talked about, 180 tonnes, event-free.

So the site is getting ready to do -- and what we're preparing the site for is a launch shaft to go down and do this tunnel. And as mentioned before, the parameters vary between 10 metres and 10.5, about a kilometre and a half out in the lake.

But currently we are seeing cost increases on the tunnel as we progress through detailed design. The financial pressures are exceeding, in some cases, our business case, and it's putting pressure on that whole overall 26.8 envelope. So currently, again, we are still progressing in the field with design and execution phasing and preparation.

But before our second hearing in October, we will have what we called, and we talked about this earlier,

about a director scope review, where we bring everybody together. That would include the CNO, the Chief Project Officer, and may even include the CEO of the company. To get what -- we've got the team formally coming together to optimize the design on this tunnel. And with that output and that product, we will make a determination on whether to proceed forward.

And so, again, we do have people out of role. Currently my vice president for the turbine generator and deep-water intake operations projects and the leads from the vendor community are together working through this to get us the most optimal solution. So we can make the right decision for the business and to be able to safely operate Pickering for the next 38 years.

MEMBER McEWAN: And how complex is the engineering to actually go down there and do it?

LUCA CECCATO: I'll probably ask Mr. Miller to give us some colour on that.

It is a hydraulic analysis, before I hand it over to Mr. Miller, that we're progressing through. So a lot of hydraulic analysis has brought up some concerns that have added complexity and size to the tunnel. And that's some of the things that the team is working through right now.

But I'll hand it over to Mr. Miller for some

more colour on that.

JEFF MILLER: Jeff Miller, for the record.

So as Mr. Ceccato indicated, the team is working with our vendors to finalize the design. And one of the critical factors to consider is the hydraulic process through the intake shaft down into the forebay to ensure that the tunnel will supply water for all events, considering low lake level events, for example, and not impact our existing ability to source water from the forebay.

So it's an iterative process, as Luca, Mr. Ceccato mentioned, that we are working to converge on an optimum design solution, not just the size of the tunnel, but also what the intake structure on the lakebed looks like.

MEMBER McEWAN: So deciding the deep-water intake or not, does that have implications on how you would design and plan the restart of the first unit?

SARA IRVINE: Sarah Irvine, for the record.

I know the deep-water intake is of great interest. It's an important scope item. So I'd like to pass it to Mr. Dave Rogers for a bit of context about deep-water intake before we take specific questions.

DAVID ROGERS: Yes, for the record, Dave Rogers.

So just to be clear, our surface water intake meets all regulatory requirements now. The deep-water intake does not actually add to our safety analysis case, probabilistic or deterministic safety analysis. What it does do is it does provide that operational margin, let's say.

But for Unit 5 return to service, Unit 5 return to service does meet all regulatory requirements and safety margins with the current surface water intake that we currently use. And so that's basically what we're looking at there.

Our desire is to do the deep-water intake. If that tie-in comes at a later time, we will put the units in the appropriate condition, appropriate safe state to do that tie-in at a later time during the refurbishment period.

So just to recap, surface water intake meets all requirements perfectly safe for the restart of Unit 5 and subsequent units.

MEMBER McEWAN: Thank you.

So do you have a date for the finalization of the design and the decision?

LUCA CECCATO: Luca Ceccato for the record. Our final decision will be in September and it will be prior to the second hearing.

MEMBER McEWAN: Thank you.

Staff, do you have any comments on the relative advantages, or disadvantages, or risks of the deep-water intake as opposed to the current surface water intake?

DR. ALEXANDRE VIKTOROV: So Alex Viktorov, for the record.

I will concur with the view that the surface water intake allows Pickering Station to meet requirements. So from a purely safety perspective, safety can be achieved with surface water intake. However, deep-water intake does offer from our perspective some advantages and benefits.

I'll ask my colleague Melissa Fabian Mendoza to elaborate on those.

MELISSA FABIAN MENDOZA: Melissa Fabian Mendoza, for the record.

So from CNSC's technical perspective, considering environmental aspects, there are advantages from the deep-water intake and OPG has assessed this as part of their predictive environmental risk assessment that they've submitted as part of this licence application. So that predictive environmental risk assessment specifically identifies that a deep-water intake would be expected to significantly reduce impingement and entrainment in comparison to the current situation. And there would be either consistent or slight improvements as it relates to

thermal effects as well.

So overall, CNSC staff have reviewed the predictive environmental risk assessment, which considers the implementation of the deep-water intake, and agree that it does have benefits from an environmental risk perspective.

In terms of the ongoing operation of the surface water option that OPG has referred to, we would need to see additional assessment of that to demonstrate their conclusions on that.

SARA IRVINE: Sara Irvine, for the record.

We would like to add a bit more context that I think is relevant to that assessment.

So the construction of a deep-water intake structure, similar in concept to Darlington, is proposed at Pickering during the requested licence period. And we know that drawing water from deep below the surface ensures a stable supply of clean, cold water to the forebay.

The installation of the deepwater intake is expected to yield several benefits, which include improved thermal efficiency and support for increased electrical power output. This deep-water intake will also reduce the risk of biofouling events, which, unmitigated, can lead to the reduced water flow rates and reduced electrical output.

Some environmental impacts could be mitigated

by a deep-water intake, fish in particular. We expect the impact of fish impingement with a deep-water intake to be similar to that of the currently employed fish net at Pickering. As well, the deepwater intake is one way to improve our climate change resiliency and reduce generation vulnerability due to extreme weather events such as heat waves and higher lake temperatures.

The detailed engineering and construction plans are ongoing, and project planning and infrastructure work continues while OPG is finalizing the project strategy. OPG has and will continue to thoroughly engage with the Michi Saagiig Nations as planning advances.

If certain challenges cannot be resolved within acceptable technical and financial parameters, we'll continue to use the existing deep surface -- sorry, existing surface water intake to sustain its operation over the life of the station. The existing surface water intake remains a safe and viable option to supply cooling water at Pickering. OPG maintains the screen house in a condition that meets its original design requirements for continuing safe and reliable operation while meeting all regulatory requirements.

And just to be clear, OPG's preferred option is to complete the DWI intake project during the next licence period, and the deep-water intake is not needed to meet any

regulatory requirements, and no part of the deep-water intake is credited in deterministic or probabilistic safety analysis. The deep-water intake does not constitute a safety improvement opportunity, it's an enhancement only. We will continue to proceed DWI and manage challenges through our established project governance and risk management process.

Thank you.

MEMBER McEWAN: Thank you very much.

Mr. President.

THE PRESIDENT: All right. Commission Member Hopwood, please.

MEMBER HOPWOOD: Thanks, thank you. A couple of questions on some details of the refurbishment work.

To start with, the work to do one of the major tasks, which is to replace the steam generators. And I heard that you're doing work to create a mock-up of the hatch in the containment building's roof that will be used.

I'm aware that with a significant number of steam generators, you're going to have quite a complicated manoeuvre inside the reactor buildings to manoeuvre the steam generators out of the hatch and the new ones back in. Is that part of the mock-up process or simulation process? That's going to be quite a critical manoeuvre, so I want to hear how you're going to prepare for and rehearse for that

manoeuvre, just to use it as an example.

JEFF MILLER: Jeff Miller, for the record.

I recognize that there's probably a lot of interest in the boiler removal evolution, so I'll just walk through a summary, and then we could address any specific questions that you may have following that.

OPG is committed to planning and executing the boiler replacement program safely and with quality. The Pickering refurbishment project will include the replacement of all 48 boilers to support safe and reliable long-term operation.

The overall program includes boiler manufacturing, removal of the original boilers, installation of new boilers, containment dome, engineered access point and restoration, installation and inspection of the new boilers, and temporary on-site storage of the removed boilers. The work is being planned and executed with a strong emphasis on nuclear safety, environmental protection, contamination control, and the use of proven global nuclear refurbishment practices.

OPG has conducted benchmarking trips to Canadian and international utilities that have performed boiler replacement and has incorporated individuals and contractors with expertise in planning and executing boiler replacement. The new boilers are being manufactured in

Ontario. The design has been approved and production for Unit 5 and Unit 6 is underway.

The replacement boiler design incorporates enhanced inspection, access, preheater improvements, and additional tubes to support performance requirements. Manufacturing quality assurance includes baseline inspections and controlled production processes.

Site preparation will include removal of interferences and installation of a heavy lift crane. Access for boiler removal and installation will be provided through two engineered openings in each unit's containment dome. Concrete core sampling will be performed once the units are in the guaranteed shutdown state to verify concrete strength.

The dome access work will follow a controlled engineered process supported by a mock-up program to validate the cutting and restoration methods. A high-pressure water technique will be used for precise concrete removal. The radiation protection team will monitor each dome opening and procedures will be in place to manage opening and closing of the engineered access points.

Following installation, the dome openings will be restored to full containment capability using engineered formwork and new concrete to recreate the dome geometry. A

containment pressure test, an integrated leak rate test, an examination of the structure will be performed in accordance with all applicable codes and standards. Successful testing and inspection is a prerequisite to declaring containment in service and starting up the first unit.

MEMBER HOPWOOD: Thank you. That's helpful. It gives us a good example of the work that's needed to do that.

A couple of kind of follow-up questions that arise from this, and perhaps relate to refurbishment but also to waste management. Because you're going to be removing these steam generators, that's now an extra burden at the end of the life of this facility.

Looking far ahead, we now have an extra number of steam generators that will be located on site and soon, you know, eventually to be followed by other materials quite possibly.

So my question is really how congested is the site? How much space is there on site for the manoeuvring? You're going to have a structure that will contain this kind of removed materials along with the fuel channel materials and so on. Do you have assurance that you have enough room on the site to both store these materials and manoeuvre large objects around the facility?

JEFF MILLER: So Jeff Miller, for the record.

Before I pass it to Ms. Rambukkana to talk about the waste aspect, I'll say in the south yard where the boiler replacement work is taking place, we do have dedicated routes that the steam generators will take to travel from the reactor building once removed to the PCSS. And that has all been planned and pre-approved.

So I'll pass it over to Ms. Rambukkana to talk about the waste aspect.

HEATHER RAMBUKKANA: Good afternoon. For the record, my name is Heather Rambukkana, Director of Operations and Maintenance and Nuclear Sustainability Services for the Eastern Region.

In terms of the steam generators, yes, the Pickering Component Storage Structure is being built to store the 48 steam generators associated with the proposed refurbishment project. And as part of those activities, we have learnings at the Western Waste Management Facility from overhauling steam generators from Bruce Power, Bruce A. And the learnings that we have from that are being applied to this refurbishment project.

We actually do have a pilot project right now undertaken on the Bruce A original steam generators where we're embarking on volume reducing these large objects to really make sure that the environmental footprint is

minimized. And so, the learnings from that pilot would then be applied to this project.

So in summary, in nuclear sustainability services and at OPG, we don't accept waste as inevitable. We are taking full responsibility for the material that does get generated and look at the entire lifecycle of the material and are finding solutions to successfully continue to volume reduce and really minimize this footprint. So although they'll be internally stored in the Pickering Component Storage Structure, we do anticipate in the future that we'll be able to leverage the learnings from the pilot that we have undertaken to really have a positive impact and really minimize that footprint.

MEMBER HOPWOOD: Thanks very much.

Just I'll pass it back to the staff in case you have any comments about this overall important aspect of the work, you know, the removal and then the storage of these large objects. Any comments?

DR. ALEXANDRE VIKTOROV: Alex Viktorov. I will ask my colleagues to speak about our assessment of removal of steam generators, of boilers and then dealing with waste.

But if you allow me, I will come back to the previous question on the deep-water intake. It's important to recognize that staff recommendations to the Commission

were given assuming that deep-water intake is part of the refurbishment scope. So our assessment of environmental impacts as well as safety performance assumed that deep-water intake is in the scope. Whether safety requirements would be fully met without this element of the refurbishment, that was not part of our assessment. Again, our recommendation assumes that deep-water intake is within the scope.

Now I will ask my colleague in Engineering groups to comment on our assessment of how steam generators could be removed through openings in the containment building.

Cédric, if you can take the mic?

CÉDRIC ANDROUET: It's Cédric Androuet, Civil Engineering Specialist in the Engineering Design Assessment Division, for the record.

So we have been looking at this project very closely related to the impact on the concrete structure from a structural integrity perspective and from a leakage rate perspective as well. So there has been a number of activities that have been started at OPG. Those include additional inspections in addition to the traditional inspections that are being done through the aging management plan.

JUDE SAMSON: Sorry, could I just -- we're

having a little trouble hearing you. If you could just keep your voice up a little bit and just speak a little bit louder, I think it will be okay.

CÉDRIC ANDROUET: Sure, can you hear me better now?

JUDE SAMSON: Yes, thank you.

CÉDRIC ANDROUET: So Cédric Androuet, for the record.

So I was saying that there's been a number of activities that OPG has started to carry on related to that project and we have been looking at those very closely. Some of them include some additional inspections that have been carried out to have a detailed mapping on the condition of those reactor buildings and in particular of the domes to identify the locations of potential cracks, *et cetera*.

So this helps establishing the baseline condition, and there are plans to do additional inspections during the activities and after the activities. There are plans to monitor the critical areas for unexpected deformation, stress concentrations, etc. There are a number of analytical models that are being developed at OPG to evaluate what would be the impact should this licence be granted, what would be the impact of those openings on the overall behaviour of those reactor buildings in terms of

structural integrity and for the restoration part so that there are some challenges from a technical standpoint that are related to ensuring basically that there will be a monolithic behaviour of the structure after restoration and that the leakage rate will still be below the acceptance criterion, the safety analysis limit. And those are being dealt with currently in the detailed design engineering portion of the project.

Some aspects that are important are related to the use of non-shrinkage concrete, for example, for the restoration to ensure a good bond between the old part and the new part of the concrete structure. There are some couplers that are planned to be used to join the old rebars and the new rebars, for example. Those are being looked at very closely, and one important part is at the end of the project, at the end of the restoration, there would be a pressure test and a leakage rate test that would be performed to ensure that the leakage rate would be below the safety analysis limit.

MEMBER HOPWOOD: Thanks. What I heard was work is underway to complete the necessary work to provide assurance that this is going to be a successful operation from the point of view of leak rate resistance and integrity of the dome. So I would like that we do hear more about this when it comes to Part 2 Hearing. I think

it would be worth getting a further update when it comes to Part 2.

I am going to have a couple more questions. Just more about some of the safety work that's been done around the scope of the refurbishment project. One item that I saw, it's under SCA Number 6, Operating Performance, was noted that there were a series of trips and setbacks over the last five years. My notes tell me there were 21 trips and/or setbacks in five years. So that's an example of OpEx that I would expect to be addressed in the scope of the refurbishment, that it would make sense that that would be considered as a piece of important OpEx. Can you comment on how you reacted to that little bit of background there, please?

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

I think what you're referring to are the metrics associated with unplanned transients in the Commission Member Document.

MEMBER HOPWOOD: That's correct.

KERRY JACKSON: So really, all of our staff at OPG are trained specifically -- I'll go back to our certified staff. Unplanned transients by their nature are unplanned. However, our staff are trained and qualified to respond to a number of design-basis and beyond-design-basis

type events. And in all of the events within the licence period, there were no nuclear safety consequences associated with those events.

What I will say is there is a combination of what can cause these events. Some of them are related to -- you know, we were discussing biofouling, which can cause a reduction -- an operator to take action or a reduction in reactor power, and some of them may be equipment-related as well.

That being said, from an operator response to all of these events, following any type of transient, there is a post transient review that takes place, which is a very formalized debrief, which is full of a number of different work groups from within OPG, including the fleet, to look at a number of aspects: How did the equipment respond? How did staff respond to the event? And, what changes would we make as a result of any transients?

All transients are tracked as part of our corrective action performance improvement program, with actions associated to drive improvement where there's an area to improve. So you will see that we have some transients over the reporting period, but our trend has been improving over that reporting period, based on a number of factors, one of them being our focus on equipment reliability, which has been a focus for the nuclear

industry to make improvements in that area, and we're seeing the benefits of the improvements in that area.

DAVID ROGERS: Dave Rogers, for the record.

One thing I would like to add some colour to what Mr. Jackson said is our corrective action program is very robust. And when we have a transient, like it was mentioned in the report, we go through a root cause analysis. And if it's equipment related or there's a single point of vulnerability with a particular equipment that failed that have caused it, we will institute modifications to provide redundancy so that we don't have that single point of vulnerability. And we do have a few of those in refurbishment scope and sustaining scope as well.

Just for clarification, I want to make very clear on the topic about deep-water intake and surface water intake, if ever at any time we did not meet regular requirements, we would not run. We would only run when it's safe, and that is how we operate. So I just want to make sure that's clear.

Thank you.

MEMBER HOPWOOD: Back to the President.

THE PRESIDENT: All right.

Commission Member Lacroix, please.

MEMBER LACROIX: Well, thank you very much for

this very interesting discussion. I do have follow-up questions concerning the reactor dome and the steam generators. So my questions are addressed to OPG.

Simple question -- was the original reactor dome designed to be removed at one point in its lifetime?

JEFF MILLER: Could you repeat the question?

MEMBER LACROIX: Yes. Was the reactor dome -- the original reactor dome -- was it designed to be removed at one point?

JEFF MILLER: Jeff Miller, for the record.

The steam generator replacement through containment is not an industry first. It has been done at other utilities, Watts Bar being one example of successful boiler replacement during, through containment. We have benchmarked a number of utilities, both nationally and internationally, that have successfully completed steam generator replacement. The restoration activities is utilizing a similar contractor that was used down at Watts Bar to build that OpEx into it.

Your question around, was the original reactor building specifically designed to be opened at a later date? I don't have that specific detail on me, but I could bring that back.

MEMBER LACROIX: Yes, I would like to know. And once you remove the reactor dome, you will take

concrete samples. These samples will be analyzed. What if the results of this analysis reveal that there is premature aging? What's next?

JEFF MILLER: Jeff Miller, for the record.

We are taking samples of the containment dome, as stated. We are not expecting the results to be unfavourable. We have taken samples of our safety-related and non-safety-related concrete containment structures in the past and confirmed that the condition of the concrete is as expected. We have not seen any indications of accelerated aging of our concrete structures. We have a very comprehensive periodic inspection program aligned to CSA N285.4, and we report those results annually to the CNSC. As I stated earlier, the leakage trends that we are seeing are more related to the seals and the non-concrete components, and that's where our focus primarily is during refurbishment, is to replace those to restore and to improve containment integrity.

SARA IRVINE: Sara Irvine.

Just going back to the previous item that Mr. Miller said he'd bring back, we've confirmed that, no, the domes are not designed to have this happen once during their life, --

MEMBER LACROIX: Okay.

SARA IRVINE: -- but others in the industry

have done it, so we're relying on both our rigorous engineering change control process, the industry experience on how this was done to use the smallest possible access points. It's not really removing the entire dome; it's an access port on the dome.

MEMBER LACROIX: Okay.

SARA IRVINE: So I just want to clarify that. And then post-restoration, it will receive rigorous post-maintenance testing in terms of pressure testing, and that kind of testing goes on over the life of the reactor, the second life of the reactor, for the CSA standards. So we would be retesting it at prescribed intervals to ensure that that remains integral over the second life of the plant.

MEMBER LACROIX: So it's going to be a very delicate intervention.

SARA IRVINE: I wouldn't say delicate. It does require some hydro cutting, but it is being mocked up at our Darlington Energy Complex. We do have a mock-up of the dome, and this is being exercised to ensure that our staff are proficient, to ensure our procedures are adequate, to ensure the technique is rigorous. So it is -- I would say it's not a delicate technique, but it is being practiced at a mock-up facility to ensure that the team is ready to execute.

MEMBER LACROIX: And once the dome is put in place -- back in place, I presume that you will test for leakage and make sure that there is no leak.

SARA IRVINE: Sara Irvine.

That's correct. That testing methodology has been established and it will be tested to ensure it is integral prior to the unit being restarted.

MEMBER LACROIX: Thank you very much.

KERRY JACKSON: Yes, sorry, Kerry Jackson, Operations Director, for the record.

Just for clarity, we are not removing the dome on the reactor building, but we are going to use two engineered hatches within the dome on the reactor building.

MEMBER LACROIX: Okay.

KERRY JACKSON: Just for that clarification. And then as was already stated, when the work is complete, those engineered openings, or I'll call them holes within the dome, will be repaired and tested to all the required codes and standards.

MEMBER LACROIX: What is the size of this hole?

JEFF MILLER: Jeff Miller, for the record.

Each of the holes will be 15-and-a-half feet in diameter.

MEMBER LACROIX: Okay, okay. And within the dome itself, there will be how many holes?

JEFF MILLER: There will be one -- Jeff Miller, for the record.

There will be one hole over the east bank of boilers and one engineered access point over the west bank of boilers.

MEMBER LACROIX: Okay, okay. That's good. Thank you.

Now back to the boiler itself, the steam generator itself. Over the last 60 years, there's been quite a progress in material science. And one of the things is that the new steam generator will be made of material with a reduced amount of stresses. And the aging of these steam generators is also affected by the chemistry corrosion. So there is a problem of design or a challenge of design, and also a challenge of chemistry that is the operation itself. So how will you make sure that the operation of these newly designed refurbished steam generators will be equal to the operation of the steam generators themselves?

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

I will start with our chemistry control process and surveillance program that we have in place today, which will continue for restart as well.

For a number of our systems within the power

plant, there are strict requirements on the testing and monitoring of the chemistry associated with each of those systems, one of those systems being the boilers. The chemistry of the boilers are monitored at an increased frequency, shiftly-type frequency, reviewing parameters to ensure that they are within the required specifications. And if anything is trending in a direction or outside of those specifications, we have procedures and actions in place to take the necessary actions associated with the trend that was not in the correct direction.

We do have a lot of operating experience associated with boiler chemistry, and as part of our performance improvement program and our corrective action space, we've evaluated that and made those changes into our chemistry control program, operating procedures and surveillance program to ensure that our boiler integrity meets all the requirements for the codes and standards.

MEMBER LACROIX: Okay. So you will make sure that you will reduce the problem of pitting?

KERRY JACKSON: Correct. The chemistry control process, if we're within spec, would eliminate or reduce any chances of corrosion or degradation of the boilers.

MEMBER LACROIX: Okay.

JEFF MILLER: Jeff Miller, for the record.

I would just add, there have been additional

enhancements made to the design and material selection of the new boilers with a focus on minimizing the risk of tube degradation and known degradation mechanisms observed in other CANDU and PWR pressurized water reactor boilers. The replacement boilers will have Inconel 690 tubing, which is a significant enhancement to the corrosion resistance properties compared to the existing 1L 400 of the original boilers.

And another change is the support structure within the new boilers will be manufactured from 410 stainless steel which, compared to the carbon steel in the original boilers, provides better resistance to flow-accelerated corrosion.

MEMBER LACROIX: Okay, good.

One last question. The whole purpose of refurbishment is to make sure that Pickering is able to meet the increased electricity demand for the next 30 years. And when I look at the general problem from nuclear fission to the delivery of the kilowatt hour to the customer, there is a whole supply chain. And when I read the submissions, a lot of the emphasis was put on the boilers, the fuel, the reactor cores, the piping systems, and so on, essentially on the fluid flow.

And one thing in the supply chain that is of concern to me, and this question was already partly

addressed, is what I call the plant's electrical infrastructure. Over the lifetime of the newly refurbished reactors, the plant's electrical infrastructure will be modified, will have to be modified, will have to evolve.

And how will OPG make sure that it will be in sync with the new technological developments over the next 30 years? What I'm saying here is that will the plant's electrical system be equal to the newly refurbished reactor core and steam generators?

JEFF MILLER: Jeff Miller, for the record. Our scope of refurbishment is based on comprehensive system condition assessments and component condition assessments, which factor in aging management and life-cycle management plans, system health to define the scope for refurbishment.

The electrical equipment, as you mentioned, is part of that. These are normal or I'll say established OPG processes that inform the scope of refurbishment based on component condition and fitness for service.

Where components are projected to require refurbishment or replacement, we do that proactively to ensure that there is adequate margin in order to support the design requirements for that particular system, structure, component to replace them before we experience any sort of unanticipated degradation or equipment failures.

MEMBER LACROIX: Okay. Let me turn to staff. So I would like you to comment on the reactor dome holes, on the steam generators, and on the plant's electrical system.

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record. I think we already covered the aspect of cutting holes or engineering openings in the reactor building's domes. Again, just to reconfirm that we expect and actually require that testing will be done after those openings are closed to ensure leak--tightness of the containment boundary. If there are any more detailed nuances, we can ask our specialists to elaborate.

Regarding the chemistry control following all the changes made, I'll ask a specialist, Sophie Gingras, to elaborate on this aspect.

SOPHIE GINGRAS: Good afternoon, my name is Sophie Gingras. Can you hear me?

MEMBER LACROIX: Yes, I can.

SOPHIE GINGRAS: Yes, thank you.

So the chemistry control program at Pickering is considered acceptable right now. And during refurbishment, there is not going to be like chemistry control program activities outside of normal. And when OPG is back to operation, then the OPG chemistry control program will resume as usual.

There are a lot of parameters that OPG sends us every three months as part of REGDOC-3.1.1, and so far we have not found any issues with their chemistry control program.

MEMBER LACROIX: Thank you.

And what about the electrical system?

DR. ALEXANDRE VIKTOROV: Again, of course, station condition assessment includes all system hydraulic, electrical, any other, and we do expect that OPG has identified both limitation imposed by aging obsolescence design restrictions as well as opportunities for safety improvements.

And let me check who is on our side could elaborate. Nadine El Dabaghi, please.

NADINE EL DABAGHI: Hi, good afternoon. For the record, my name is Nadine El Dabaghi, Systems Engineering Division.

So I concur with what's been said. Part of the PSR and the component condition assessments, the electrical components are all going to be evaluated, and we review that as well as we receive those component condition assessments. OPG would evaluate if any need would need to be replaced or there's additional maintenance that needs to be performed.

It's an ongoing process and a living document.

So and if there's any gaps identified, for example, the standby generators were identified to be replaced, so they would be replaced. So we do inspections and we follow up to ensure that everything is performed as per requirements.

MEMBER LACROIX: Thank you very much.

Back to you, President.

THE PRESIDENT: All right, just before I move on, just an element that was identified on the chemistry side. Like, I appreciate that the components are going to be replaced, but just maybe for OPG to talk about the system of layup and protecting equipment while the units are shut down, just to assure us that there's no degradation going on during that period.

KERRY JACKSON: Kerry Jackson, Operations director, for the record.

We will still have our chemistry control in place during the shutdown. When all the units are shut down, we're still in the guaranteed shutdown state. We still have fuel in the core while we're defuelling, so we will still maintain, as per our requirements, our chemistry control program.

In addition to that, through the refurbishment period, even once all of the cores are defuelled, we still have some systems, auxiliary systems in operation, which we will still maintain to ensure that they're within chemistry

spec to ensure that they maintain their requirements and their availability so that they're ready for restart and don't experience any degradation through the shutdown layup period.

THE PRESIDENT: All right, thank you for that clarification.

Commission Member Remenda, please.

MEMBER REMENDA: So I had a whole bunch of good questions to ask you about the DWI, but I will hold those on the assumption that that will be coming back to us in Part 2.

So I do want to ask a question that was just to me curious, and I'm sure there is a reasonable explanation. I think it was a climate change resilience report determined that there was some reinforcement needed in the Unit 7 control room exterior wall. If I got that right, could someone please explain to me why only Unit 7? And thank you, I'd appreciate from OPG an answer to that.

DIVYA KAKUZHYIL: Divya Kakuzhyil, senior manager, Nuclear Safety, for the record.

The reinforcement of the Unit 7 control equipment room wall was not identified as a part of the climate resilience assessment. It was mentioned there, but it was driven as a safety improvement opportunity. The high wind probabilistic safety assessment currently meets

the OPG safety goal, but it is a desired objective of the risk and reliability program that it meets the administrative safety goal, which is 10 times lower. So as part of the assessment, it was determined that if we strengthen the metallic clad portion of the Unit 7 control equipment room wall, it would offer better resistance, and we would be able to meet that requirement and meet that administrative safety goal.

Now your question was why are we not doing it for Units 5, 6, and 8. The reason is they're already sufficiently protected, and the assessment validated it. So no additional actions had to be taken for those walls.

MEMBER REMENDA: Thank you. I appreciate the answer, and it was simple and quick.

And I'll pass back to the President.

THE PRESIDENT: All right, thank you.

I just note that, OPG, that you may have some further elaboration on Dr. Berube's question, so maybe I'll give you a chance to provide any input that you want to make. If not, maybe I got that wrong.

JEFF MILLER: Jeff Miller, for the record. This was in relation to the undertaking regarding changes to state or control logic and any impact the state or changes may have on the main control room layout. So the answer is no, no change in the main control room panel

layout just from the rewind activities.

We do have a new excitation system, and this involves having field operators engaged at the exciter panel in the field during startup to support synchronization activities. The rewind on the rotor is primarily driven by aging, and improvements are made to handle higher current, allowing for a megawatt increase and better cooling.

MEMBER BERUBE: Yeah, that makes sense to me. So I'm good with that. As long as we're not seeing major changes on the control room panel, that's my primary concern.

THE PRESIDENT: All right, thank you for that.

So just a couple of questions from me specific to the refurbishment. The first is can you elaborate for me a bit your D2O management strategy just given the amount of water, the heavy water that's tritiated that's got to be moved around the plant and the state changes of the unit. So maybe just walk me through what you're doing to ensure that the environment isn't impacted and that you've got sufficient storage.

KERRY JACKSON: Kerry Jackson, Operations director, for the record.

Our D2O management plan through refurbishment, there's I'll say a detailed plan which has been reviewed

and signed off by all the relevant stakeholders within OPG. The process through refurbishment is to defuel and then subsequently dewater the reactors.

We have reviewed all of the space that we have on site to hold D2O, and our D2O management plan sufficiently captures all of that inventory to be maintained on site. That's a combination of both our common services area, which is an area that currently is used to transfer D2O-grade water to, as well as the use of our Pickering 1-4 helium storage tanks, which are also approved for use of D2O-grade water.

As part of a contingency, if required, we also have the tritium removal facility at Darlington, where we can ship as part of storage. However, right now that will mainly be used for processing to improve, reduce the curie content of our moderator waters prior to Unit 5 restart, which will support improved tritium reduction upon Unit 5 restart.

THE PRESIDENT: Okay, thank you. So what I get from that is that you feel you have adequate storage, you've got contingencies in case you need to transfer water off site, but you're not expecting that.

KERRY JACKSON: That is correct.

THE PRESIDENT: All right, thank you.

I'd like to take you to slide 21 in your

presentation this morning, and just an elaboration on these safety improvement opportunities and specifically what those safety opportunities will improve. So from core cooling to containment to external hazard protection, just give us an indication of what, for example, diesel firewater makeup is there to do for us.

DIVYA KAKUZHYIL: Divya Kakuzhyil, senior manager, Nuclear Safety, for the record.

So OPG will be implementing safety improvement opportunities during Pickering B refurbishment to further improve the PSA results. As I mentioned before, all the Pickering PSA elements meet the OPG safety goal. And as I mentioned before, all of them also meet the administrative safety goal except for high wind. But once we implement the strengthening of the Unit 7 control equipment room wall, we should be able to meet that as well. However, we are estimating that our risk estimates will be further improved once the safety improvement opportunities will be implemented during the refurbishment period.

Now, some of the SIOs that will be implemented during refurbishment are -- so I will highlight it based on the function -- so for improving the cool function, firewater makeup. So an alternate beyond-design-basis emergency water supply will be established for the Pickering Units 5 to 8. So how it's going to be done is

connecting the diesel water pump system to the existing emergency service water infrastructure. So you have a whole separate source of water supply now.

The next way to improve the cool function is emergency water system EWS valve duplication. We currently have some single point vulnerabilities which we will eliminate by duplicating flow-path supplies to the boilers, heat transport system, and moderator. So new flow paths will be created. We are improving redundancy, defence in depth, and will improve the overall reliability of the emergency water system.

We are also providing a direct flow path from the EWS pumps to the moderator, and this will mitigate -- right now, water is being used from the Pickering 1-4 side. And since Pickering 1-4 will be decommissioned and safe stored, we will be getting this flow path using the EWS pumps to the moderator. So we'll have a diverse and independent supply post refurbishment. And this will be an alternate source of supply to the moderator.

Another SIO to improve the cool function is improving the design of the low pressure service water strainer system. And this will improve the post-accident reliability of the low pressure service water.

Moving on to the contain. So there's two major

modifications or SIOs that's going to be completed. One is the calandria vault overpressure. This is especially important in mitigating beyond-design-basis events. It improves our safety margin. It will install a large relief capacity on the end shield calandria vault to improve the margin to calandria vault overpressure. So this will mitigate the progression of the accidents. It will buy us more time from accidents from progressing from beyond design basis to severe accidents.

And the installation of the containment filtered venting system, and this will also mitigate beyond-design-basis accidents and it will relieve containment overpressure and provide a filtered relief pathway to enhance containment integrity. And once this SIO is credited in our level two probabilistic safety assessment, we will definitely gain additional margin, as was seen when this SIO was implemented at Darlington.

And the last SIO related to the external hazard protection is the one I just spoke about regarding the reinforcement of the Unit 7 control equipment room wall. And the external hazard we're talking about here is high wind, windborne missiles as a result of high wind, when things become projectiles and come and hit that wall. We don't want the control room to be destroyed. So now with the strengthening of that wall, we mitigate and we bypass

those sequence of events.

So once these safety improvement opportunities when implemented at Pickering as part of refurbishment will further decrease the risk estimates for Pickering B PSA and will make Pickering safer.

THE PRESIDENT: All right, thank you for that.

Just a thought from me. You're talking about design-basis and beyond-design-basis accident. Just to put it in English, for those who aren't safety analysts, you're lessening the potential for impact to the public for these events, and maybe even, as you say, delaying the possibility of release. I think that's what I heard in layperson's terms. But okay, thank you for that.

And just turn to staff for your comments on the description and what's being done here.

DR. ALEXANDRE VIKTOROV: Alex Viktorov.

I concur is these safety improvement opportunities will reduce the likelihood of significant accidents and significant releases to the environment and thus mitigate or eliminate risk to the public. I'm also sure that probably the significant improvement opportunities, but not the only ones. Again, we encourage OPG to look for opportunities to improve radiation safety, environmental safety, and not just nuclear safety.

THE PRESIDENT: All right. Thank you for that.

Then perhaps one last area for me is around the containment strategy. You've talked about taking the vacuum building out of service. So just perhaps describe the approach you're using around ensuring containment within each of the reactor buildings.

You know, you're removing -- or you're creating access ports within the domes themselves. How are you containing, you know, the potential for release of radiation? How are you managing containment through that process?

Again at a high level. I don't need the details. But just to get an understanding of how you're going to manage it. Thanks.

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

Starting with the dome cutting or openings, that will take place after the source term has been removed from the reactor building. So this is post-defuel, dewater. As part of our normal radiological monitoring program that we have at OPG at Pickering, there will be routine monitoring and sampling of areas inside the reactor building just looking for any potential changing conditions, but we don't anticipate any based on the fact that we've removed the source term.

With the openings in the dome, we will continue

to monitor, but we'll also have negative airflow. So in simple terms, air will come from the outside into the building and it will exit through a monitored pathway.

As discussed, there are engineered hatch coverings. Should we see a condition or inclement weather or something that could cause any challenges, we can close those hatches per our approved procedures to ensure that nothing getting into the reactor building is required, but also allows good contamination control practices should we see any challenges, which of course we're not anticipating.

SARA IRVINE: Sara Irvine, for the record.

If you'd like to hear more about how we'll protect our own people plus the environment during this evolution, we have both Mr. Raph McCalla and Ms. Lauren Corkum available to respond.

THE PRESIDENT: Yeah, I think it would be helpful to look at the additional monitoring that's being invoked.

There's a change in the configuration of the plant that I think is material, and we'd like to hear about that. Thank you.

SARA IRVINE: So perhaps Ms. Corkum first, and then Mr. McCalla.

LAUREN CORKUM: Lauren Corkum, Director of Site Support at Ontario Power Generation, for the record.

Ontario Power Generation has a robust radiation protection program. We will be utilizing that program throughout refurbishment. All work on refurbishment will be done under OPG's radiation protection program.

For this work in particular, all workers will be, will be -- oversight of this activity will occur by the Radiation Protection Department. Health physicists will identify radiation exposure permits so that that -- so work to remove boilers, cut appropriate lines, that will be administratively controlled by a radiation exposure permit providing instructions. Radiation protection technicians will be providing briefings to workers involved.

And in addition to that, we will have extensive monitoring in place. So that includes the use of portable instruments as well as dosimetry. OPG conducts and monitors dosimetry through TLD badges. We also utilize electronic personal dosimeters, which provide a real-time and alarming function to prevent any situation where dose rate or dose would exceed expected values.

This has allowed OPG's radiation protection program to keep workers safe throughout Darlington refurbishment. We are very happy to report that throughout refurbishment, we collected lessons learned throughout our activities. And the DNRU-1 refurbishment was actually the lowest dose refurbishment for any CANDU unit that has been

performed across the world to date. So we're very happy to share that.

I will turn it over to Mr. McCalla to go over how we are going to be monitoring emissions.

RAPHAEL McCALLA: Raphael McCalla, Director of Environmental Health and Safety, for the record.

OPG has a very extensive radiological emissions monitoring program. It complies with all the requirements of CSA N288.5, which is effluent and emissions monitoring of Class 1 nuclear facilities, as well as REGDOC 2.9.1, Environmental Protection, Environmental Principles, Assessment, and Protection Measures.

OPG's goal is to ensure that our emissions are as low as reasonably achievable. And this is achieved through the establishment of administrative barriers, which is used alongside the assessment of the actual monitoring and the data that we actually gather from that process to ensure that we are not exceeding the environmental -- I mean the actual emissions limit that we actually have, which is to ensure that we don't exceed the licence limit of one millisievert per year.

And the way we do that is by using these barriers, which are the derived release limits, as well as action levels. And the derived release limit is the release of radionuclides at a rate which would result in

the most affected critical group receiving a dose of one millisievert over a year. With respect to environmental action levels, an environmental action level is a statistical analysis which results in us setting a limit which is just slightly above our normal emissions rate.

And by using those two barriers, we're able to actually monitor on a weekly basis the releases and compare that against a portion of that derived release limit, which is actually proportioned throughout the year. So with those barriers in place, we're able to assess how we're performing with respect to our emissions.

Now, with respect to the actual boiler activities, these barriers will remain in place. And what we will be doing is we will actually have stack fans that we're going to put in place to actually try to ensure that the emissions basically will remain in containment as opposed to actually going out of containment. And we're also going to be monitoring -- have monitors in place before we open up the hatches to understand what is the potential emissions that we could actually have from the site.

So by using all of that information, we're actually able to decide when is the best time to actually open the hatches to actually conduct this activity. And we will have thresholds in place which we will actually assess

against the actual conditions within the boiler containment to decide when's the appropriate time.

Utilizing all that information and going back to our actual emissions, what they are today in terms of our dose, they're actually very low. And we don't anticipate that it's actually going to be much higher than what it is actually today. So you know, they're very low in comparison to the limit and we anticipate they will remain in that domain, so to speak.

And finally, the way we're actually going to be able to verify that is going back to our actual environmental monitoring program which exists today. And if you look at our dose that we have seen over the operations of the Pickering site, they're also very low. So we'll be able to verify at the end of the day whether or not those emissions that will result from opening up the hatch periodically -- whether or not that is having an impact on the environment or the public at large. And those will actually be determined and reported through the normal environmental monitoring program that we have today.

THE PRESIDENT: All right. Thank you.

I don't know if staff have any comment they wish to make on this one, but over to you.

DR. ALEXANDRE VIKTOROV: OPG, there was facets of your question. Can you maybe clarify for us where you

were looking for more information?

THE PRESIDENT: I think we'll be speaking more about this further. You know, it really is around the extent of the planning around the spread of contamination, protection of workers.

I mean, it was addressed at a high level. I'm sure there'll be more to be discussed later. Just thought if there's a comment on the oversight that you're providing, that was all I was looking for.

DR. ALEXANDRE VIKTOROV: Yeah, we have specialists who are ready to elaborate on various details of radiation protection, environment protection but, by and large, we are satisfied with the strength of our program at OPG, and we are sure they will be effective in monitoring impacts from activities during refurbishment.

THE PRESIDENT: All right. Thank you very much.

We're going to take a 20-minute pause now and come back to focus on the decommissioning of 1 to 4. Thank you.

--- Upon recessing at 3:00 p.m. /
Suspension à 15 h 00

--- Upon resuming at 3:25 p.m. /
Reprise à 15 h 25

THE PRESIDENT: All right. Welcome back.

We're going to move to questions around decommissioning, but before we do that, I'll just turn it back over to staff for some comments they might like to have.

RAMZI JAMMAL: It is Ramzi Jammal, for the record.

I would like to start with the fact that the CNSC has its regulatory philosophies based on performance-based regulation, not prescriptive because we've been hearing quite often with respect to the deep-water intake, it's not a regulatory requirement, but we will accept and we will continue the discussion on Part 2.

But there is one thing I would like to clarify on the record with respect to the predictive environmental risk assessment and the statement that the fact that the conclusion that the proposed deep-water intake will draw deeper, cooler water from Lake Ontario and is expected to significantly reduce fish impingement and entrainment as well as reduce potential biofouling events with attached algae.

So from staff's perspective, the recommendation is the DWI will be considered as part of refurbishment on the life extension of this site. Even though we conduct environmental review on a five-year cyclical basis, it's very important for us to look ahead with respect to the operational performance of the site, not for the short term but for the long term.

So that is the statement I would like to make on record with respect to the conclusions arising from the PERA, so that it is on record with respect to accuracy on the significant reduction of fish impingement.

Thank you for that.

THE PRESIDENT: Okay. I think I heard the same thing from OPG so I think we'll leave it there, then, and we'll move on.

SARA IRVINE: Let me just check with Ms. Cammie Cheng.

Do you have further context?

CAMMIE CHENG: I do. Cammie Cheng, for the record, Director of Nuclear Environment Project Support.

The predictive environmental risk assessment for the Pickering Refurbishment Decommissioning and Continued Operations considers several aspects that could be introduced from the deep-water intake project. How the predictive environmental risk assessment was structured is

that it did look at previous assessments that were done, which include environmental risk assessments based around current operations and current conditions, existing conditions, as well as previous predictive effects assessments that were carried out for the Pickering Safe Storage Project. And it looked at where there could be incremental impacts introduced by the new project activities that were being proposed.

With respect to the deep-water intake, some of these impacts included the construction impacts around the tunnelling and other construction activities that would be taking place in the site and into the lake. And those impacts were assessed as part of this risk assessment.

If deep-water intake were not to proceed, these current impacts would be conservative, as they are looking at conditions that would not be present if the deep-water intake was not constructed.

In terms of environmental benefits, the predictive environmental risk assessment describes the deep-water intake as being able to reduce fish impingement significantly compared to the existing surface water intake. But this is compared to the surface water intake without mitigation measures that are currently in place. And this example being the fish diversion system, which is put there to prevent and reduce the amount of impingement

that otherwise the surface water intake would be operating with.

The deep-water intake system would also be beneficial with respect to reduction of the potential for debris runs, including algae, zebra mussels and silt, and could also contribute to climate change resilience in terms of drawing deeper cooler water that could result in slightly cooler discharge temperatures.

However, with respect to the risk assessment itself, it does not credit that cooler discharge and conservatively assumed that the discharge temperatures would be the same as what the current operational conditions are. And that was the basis for carrying out the assessment, and so the conclusions around the discharge temperature would not change regardless of whether or not the deep-water intake project proceeded.

THE PRESIDENT: All right. Well, I'm sure we're not done talking about the subject, so I appreciate it.

You know, in a previous life, I tried to run that plant in late August, early September, and if I was an operator, still an operator, I think I know where I'd get on this issue, but.

All right. Thank you very much.

So let's move on to other questions, and let's

start with Dr. Berube on decommissioning Pickering 1 and 4, please.

MEMBER BERUBE: Let's start with OPG. And maybe if you could, you can bring us up to speed on where you are right now decommissioning Units 1 and 4, just to give us a basis of what your current activities are there, and then I'll have some questions about spent fuel handling and where all that's going right now.

CHASE AMYOT: Chase Amyot, Director of Program Management for Decommissioning, for the record.

Currently the DDP, or Detailed Decommissioning Plan, was accepted last June 2025. Decommissioning is a project sponsor of the decommissioning work, and the execution is done by the project organization. Currently we are removing conventional materials from Units 2 and 3 as well removing some of the unneeded outbuildings that are no longer needed to support the A-side.

So currently, the team is removing systems, no structures, just systems and components, and that is -- 99 percent of those materials would be clean materials or non-contaminated and sent for conventional recycling.

MEMBER BERUBE: And where are you on 1 and 4 right now in terms of your process?

CHASE AMYOT: I'll start and maybe turn it over to operations. But Chase Amyot, for the record.

Units 1 and 4 are going through safe storage from now until the end of 2028. So they're working through end-stating systems no longer needed to support the units. Units have been defueled and dewatered. We are continuing to move fuel out of the fuel bays and basically removing systems no longer needed to be supported. So mostly systems around conventional heating, lighting and supporting the fuel bays are still running.

But maybe I'll turn it over to Mr. Jackson.

KERRY JACKSON: Kerry Jackson, Operations Director, for the record.

I think Chase captured it well. Units 1 and 4 have been defueled and dewatered. They're currently progressing through end-stating remaining systems, which entails, I'll say, fuse pulls, safe-stating those systems, tracked through our change control process, which is all part of our managed system process. And then once we progress into a shutdown vacuum building de-poised state, that will provide the operational conditions which would then allow us to take on some of the additional work that we discussed today, such as some of the vacuum building control movements and other changes.

MEMBER BERUBE: So I just want to spend a little bit of time on spent fuel.

Basically, we've already loaded up, I guess,

the fuel bay right now in A-side is full, or where is that? And the auxiliary bay as well. Where are we there? And basically, because it's related to, actually, the refurb, you're going to defuel, dewater four more units. So that gives you the capacity problem here with fuel and moving it around.

So I just want to get into that dynamic, how you're going to shift that fuel. What are you doing right now with 1 and 4's fuel? And what are you doing with the stuff that's already ready to go to long-term storage? And just give us some ideas of where you are in that process.

KAPIL AGGARWAL: Kapil Aggarwal, Nuclear Sustainability Services with OPG.

Our used fuel program will continue through this period. We will continue to take fuel from the wet bays and load them into dry storage containers, process them and safely store them in our facilities. And that won't stop through this period.

Obviously, as we've talked about today about all the coordination that's required, this will be part of that work planning and coordination as well.

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

And just to add a little bit more context. So the team is planning for both the removal of fuel from the

Pickering 1-4 fuel bays as well as the defueling from the planned refurbishment.

So happy to say through our planning activities that we have been now loading approximately 80 dry storage containers a year. Last year we were successful with that and are on track for similar loading this year. And really, that supports both potential decommissioning as well as the potential refurbishment.

So the rate that we're loading allows for sufficient bay space for the defueling of the four units associated with refurbishment, as well as the planned decommissioning of the fuel bays supporting the Pickering 1-4 site.

MEMBER BERUBE: So I'm just trying to understand how you are going to daisy-chain the fuel movement because there's quite a bit of movement in this process. So you're going to basically, right now is 1 through 4's fuel bay full or is it, where is it? And basically, as you're moving stuff out of the auxiliary fuel bay, I mean, and --

KERRY JACKSON: So I can answer that. Kerry Jackson, Operations Director, for the record.

So all of the Pickering 1-4 fuel that was in the units is currently in the Pickering 1-4 irradiated fuel bay. That will be processed, put into containers and

eventually stored in our dry storage container location.

On the Pickering 5-8 side, we currently have enough space for 3.5 reactor cores. We're expecting to have four full cores by September of this year. And as Ms. Rambukkana had already mentioned, that came from planning that started ahead of time to ensure that we're meeting a rigorous schedule to move irradiated fuel from the Pickering 5-8 irradiated fuel bay on a delivery schedule within our dry fuel storage containers to the dry fuel storage facility. So we're on track to have room for four cores by September of this year.

MEMBER BERUBE: So you've got no capacity issues, then?

KERRY JACKSON: No capacity issues. That is correct.

MEMBER BERUBE: Okay. That's what I need to know.

Back to the President.

THE PRESIDENT: Commission Member McEwan, please.

MEMBER McEWAN: No questions, Mr. President.

THE PRESIDENT: Commissioner Hopwood.

MEMBER HOPWOOD: I'll follow up with a question with regard to the spent fuel as well, which is the defective fuel, and how are you going to manage the

transfer of that fuel. It's something that is, I think, novel, but we know that's been experienced in other jurisdictions.

Could you tell me a little bit about the work you've done to demonstrate that canned damaged fuel can be stored in the dry spent fuel containers?

KAPIL AGGARWAL: Kapil Aggarwal, for the record.

So the defective and damaged fuel work is being conducted right now. It is a project in which we are developing the appropriate tooling to be able to conduct this operation at the Pickering site. It is leveraging OPEX that we -- or operating experience that we've been involved with with Hydro-Québec. They went through a canning process for defective fuel several years ago, and we were intimately involved in that to ensure we understood the equipment and the process in which we could can defective and damaged fuel.

And the analysis, in order to put it into a DSC, that is being -- that's part of the project scope, is to ensure that the safety of the container as well as the fuel remains once the fuel is loaded into that DSC.

MEMBER HOPWOOD: Just follow it up a little bit more because at Hydro-Québec, I think the containers are a different design than your own. So I'm guessing that the

analysis proceeds through a sort of feasibility study first to see if approximately this is a practical proposition.

Have you gotten through that stage or are you still kind of wondering what kind of container you're going to need?

KAPIL AGGARWAL: The intent right now is for the fuel to be canned and be put into a DSC. Now, to your point, Hydro-Québec would not have put it into a DSC, but that's the analysis and the feasibility piece that is being worked on now.

MEMBER HOPWOOD: Right. Okay, thanks.

That's all my questions. Back to the President.

THE PRESIDENT: All right.

Commissioner Lacroix, please.

MEMBER LACROIX: I have no questions.

THE PRESIDENT: All right.

Commissioner Member Hardie.

MEMBER HARDIE: Thank you.

Well, I have two questions, and the first one is to OPG, and that is, are there any lessons that were learned from the continued maintenance of Units 2 and 3 in safe storage that can be applied to the decommissioning activities that are going towards Units 1 and 4, or proposed for Units 1 and 4?

CHASE AMYOT: Chase Amyot, for the record.

Yes, lessons learned from Units 2 and 3 Safe Storage Project in the early 2000s have been incorporated into the Units 1 and 4 Safe Store plans. So detailed planning, work plans have been included as part of the lessons learned. Some of the lessons learned, I could pull up. I just need a minute to pull those.

MEMBER HARDIE: If you have a few that you could pull up, I don't mind going to my next question and then giving you a chance to just pull that up for the response.

My second question just had to do is, are there any concerns with workers and the potential handling of contaminated materials, whether that be radioactive or that be just potential contaminants of concern?

LAUREN CORKUM: Lauren Corkum, Director of Site Support at Ontario Power Generation, for the record.

I'll begin and then I'll pass it on to Mr. Aggarwal for additional context. But in general, Ontario Power Generation maintains a Joint Committee for Radiation Protection. So this is a meeting that's held with radiation protection workers and union leaders across the organization.

We take that time and opportunity to identify any challenges that are being raised by union membership,

similar to any that you've identified that could be happening. We have not to date seen examples where people are actively expressing that particular concern. And I think a portion of that is because of the training that's provided at Ontario Power Generation, and also the expertise that we have in our radiation protection department as a continuous source of people to go to throughout, if you do have any questions or concerns about radiation protection.

One thing to note is that we will maintain certified responsible health physicists. We're actually planning to increase the number of responsible health physicists who are certified at the Pickering site to help support and ensure that we have that two-way dialogue to ensure that those concerns are raised and addressed.

So just to close off, we do have forums for those conversations to occur, and we do have expert staff, certified staff by the CNSC who help with addressing those concerns.

And I'll pass it along to Mr. Aggarwal if you have anything else to add.

KAPIL AGGARWAL: Kapil Aggarwal, for the record.

I think Ms. Corkum covered it off quite well. I would just say that we have a rigorous work planning

process right across OPG Nuclear, and so any work or activities that are happening in the station are adequately planned with ensuring appropriate personal protective equipment and barriers are in place to ensure the safety of the workers.

And so through that process, it ensures that there's healthy dialogue between the workers and the supervisors to ensure all of the hazards are identified and either prevented or mitigated through work planning.

MEMBER HARDIE: Thank you.

So you had mentioned earlier that with the decommissioning activities that are proposed, that 99 percent of what you anticipate to be moving or dealing with would be non-radiological in nature. What about that 1 percent in terms of what you're handling? And my next question is, is there potential for there to be greater than 1 percent that you just perhaps haven't discovered yet?

CHASE AMYOT: Chase Amyot, for the record.

The 1 percent that I alluded to would be related to the heavy water upgrader plant. That is about 450 cubic meters of metal materials that we can dismantle and then most likely volume reduce even further before interim storage.

KERRY JACKSON: Yeah, Kerry Jackson, Operations

Director, for the record.

Just to add to what Chase was saying, on our 1 to 4 units, as stated earlier, the main source term has been removed. They've been dewatered and defueled. As per any normal outage that we perform at OPG, if a system is to be opened and it's characterized as a system that would potentially have contained radioactive material, we would use all of our radiation protection planning and assessment practices to plan and execute that work.

As stated, the majority of the source term has been removed, but this is a normal practice within any OPG outage to open up a system that would have contained or had possible contamination. And we would have all the radiological controls in place to perform that work, whether it's the 1 to 4 side of the station or anywhere on our site.

KAPIL AGGARWAL: Kapil Aggarwal, for the record.

If I can just add, that 1 percent that's being referred to would be considered low level. And so, we safely manage low-level waste on a daily basis for many, many decades. And so we have lots of experience on how to safely manage this material, and we would enact and follow what we know.

MEMBER HARDIE: Thank you very much. That was

what I was looking for.

I just will turn back. Do you have the lessons learned there?

CHASE AMYOT: Chase Amyot, for the record.

I don't have a comprehensive list, but we can submit later or bring back the next part. But we do have experience from vac dry, benefits of doing the vac dry early and the dewatering units and where water was left and things like that. So there are some benefits and lessons learned that have been incorporated into the existing storage and surveillance plans -- or sorry, safe store project.

MEMBER HARDIE: Thank you. We don't necessarily need to do an undertaking for a list, but let's just speak to it in Part 2 that we may talk about lessons learned at that point. Thank you.

And before I finish up, I'll just allow CNSC staff to provide any comments if they wish to add anything. Thank you.

ROSS RICHARDSON: Ross Richardson, for the record.

The only thing I'd wanted to add is, you know, similar to the refurbishment compliance plan, we also have a safe storage compliance plan that's been developed and we are executing. One of the specific areas that we focused

on was the asbestos removal that's currently happening on Units 2 and 3.

We also met with the Ontario Ministry of Labour as well and discussed OPG's asbestos abatement program and the Ministry of Labour was satisfied, more than satisfied with OPG's program and so we're satisfied in that nature.

That's the one element that comes to mind in terms of our oversight on the safe storage side. I'll leave it at that.

MEMBER HARDIE: Thank you very much for that.

I'll pass it back to the President.

THE PRESIDENT: All right, thank you.

Just on the subject of lessons learned, you've alluded to this a few times, not only with regards to decommissioning and so on, but also around refurbishment. It would be helpful to us to have some specifics, and I appreciate there's a program, but perhaps that's something we can focus on in Part 2 as you provide your update.

So my questions relate to the end state. You spoke earlier about the transition in staff and keeping some of the maintenance staff engaged through, perhaps working on the decommissioning plan as the work dries up, if you will, on the operating units.

So what's the end state over the 10-year period that you're proposing? What would be the end state? What

would be the impact on the footprint of the Pickering site? What buildings would be removed? Just give us a sense of the physical change that would be expected through the decommissioning activity on 1 to 4.

CHASE AMYOT: Chase Amyot, for the record.

Majority of the non-nuclear component removals would happen within the turbine hall and turbine auxiliary building. So from a physical outside the plant perspective, you wouldn't see much change. There would just be much less materials and systems on the inside.

From outside the plant, we are removing, plan to remove the upgrading plant A towers and associated buildings, as well as we have removed two scaffold sheds, a scaffold shed, an insulation shed that were on the lake side of the Pickering A. There are some other small auxiliary little structures or buildings that on the A side to be removed. They're detailed in the decommissioning plan, but they would be outbuildings like compressed gas tank storage areas and things like that.

THE PRESIDENT: Okay. So none of the major components, transformers, large visible pieces of equipment would be removed over that time period?

CHASE AMYOT: Chase Amyot, for the record.

Large visible pieces of equipment that are removed would be like the turbines inside of the turbine

hall. Units 2 and 3 have been removed and transported offsite. But yeah, outside the facility you would not see significant changes.

THE PRESIDENT: Okay. Just as well in terms of any sub-soil removal, is there ground disturbance at all that's proposed for this decommissioning activity?

CHASE AMYOT: Chase Amyot, for the record.

Ground disturbances are part of future planning envelopes. The current removal of buildings would take the buildings down to grade and leave their concrete foundation or poured pad behind. And those poured pads and concrete foundations would be dealt with in future planning envelopes outside this licence period.

THE PRESIDENT: Okay, maybe just one more follow-up then, and this sort of bleeds into the next area. It's intended to be non-radioactive, so what steps and measures -- there's a lot of material to be moved out -- are you taking to ensure that that doesn't find its way into the public?

CHASE AMYOT: Chase Amyot, for the record.

The decommissioning program, just like operations, and refurbishment, and safe storage, follow the RP practices of verifying materials before they leave site. So everything is checked twice per the program, and then it is placed into the proper waste stream, whether it's likely

clean, low-level waste, or if it goes into the hazardous waste materials or potential metal components that go to interim storage as well.

THE PRESIDENT: Okay, maybe over to Staff around your oversight of those activities, please.

ROSS RICHARDSON: Ross Richardson, for the record.

Sorry, just to confirm oversight of the decommissioning activities?

THE PRESIDENT: The disassembly, the removal of equipment.

ROSS RICHARDSON: Yeah, absolutely. So perhaps I may turn this over actually to, we do have our Power Reactor Site Office Supervisor, Mr. Cody Krasnaj, on the line that can provide a bit more context and details in terms of our oversight on the 1 to 4 side, our compliance plan, and what's been executed to date.

So perhaps, Cody, I can turn it over to you in terms of our oversight of the 1 to 4.

CODY KRASNAJ: Good afternoon, Cody Krasnaj, for the record. I'm the Power Reactor Site Office Supervisor at the Pickering Site Office.

As Ross alluded to earlier, we do have a safe storage specific compliance plan which looks at activities that could be completed under planning envelopes A and B

for outbuildings. So our oversight is mostly captured through that plan where we focused on conventional health and safety inspections, rigging and lifting as an example.

Ross mentioned asbestos abatement and as well as monitoring of the IFBs, making sure that their heat exchangers and pumps are still being maintained while fuel remains in those activities or in those fuel bays.

We also, like the 5-8 side, have our inspectors at site that conduct surveillance activities and meet with OPG bi-weekly, specifically on the decommissioning work. So we stay up to speed on that work that's happening and can take field visits to make our own observations and follow up on any issues.

We'll continue to inspect, especially as phase two of end-stating starts to happen, which has been spoken to once the vacuum ability is deployed and some of those additional end-stating activities will take place. So we'll maintain our oversight during this period.

THE PRESIDENT: All right. Thank you.

Let's move to questions on Pickering Waste Management Facility, and we'll start with Commission Member McEwan, please.

MEMBER McEWAN: Thank you, Mr. President.

The Waste Management Facility, or the two parts of it, seem to be on relatively exposed land in that it's

right down to the water, there's the channel going between the two. So in the part of the assessments of climate change and excess weather caused by climate change, could you perhaps describe the ways in which that exposure is reviewed and the assessment is built in?

And the second part of it, have you built a tsunami into the assessment of risk? I understand that it's not going to be a seismic cause to the tsunami, but they do occur in the Great Lakes. They may be up to three, four metres. Does that have any implications for the way in which you would plan for the future of the storage site?

SEVANA BEDROSSIAN: First, can you hear me?

JUDE SAMSON: Yes, we can.

SEVANA BEDROSSIAN: Sevana Bedrossian, Senior Manager for Performance Engineering, for the record.

So, we have completed a climate change resiliency assessment for the Pickering Waste Management Facility, and we did look at the various systems and structures, categorizing the ones that are safety-related. And we did look at the various hazards that would be applicable through using a similar methodology to the one that's used for the hazard assessments in our probabilistic safety assessments.

As per the conclusions of those reports, with it -- and the report has been submitted to CNSC staff for

review -- they do conclude that our various structures, including the dry storage containers, the dry storage modules, the waste cell containers, are all of them resilient and robust against these various hazards, including the external hazards such as seismic or high wind.

And the conclusion was that they would be able to withstand these hazards as they are right now, including as they undergo regular inspections as part of our reliability and aging management program as well.

MELISSA FABIAN MENDOZA: Melissa Fabian Mendoza, for the record.

So CNSC staff have requested a climate change resilience assessment from OPG for the Pickering Waste Management Facility. We did receive that from them fairly recently, so we are still reviewing it and we'll have more to share on the outcome of our review at the Part 2 hearing.

But just at a high level, we recognize that OPG's conclusions in that report is that the PWMF is resilient to changing climate conditions and that they've taken into account the guidance that we outlined to consider in their assessment, including the EPRI guidance, so the Electric Power Research Institute guidance, as well as guidance from Environment and Climate Change Canada.

MEMBER McEWAN: What degree of climate change excess is built into this? And I mean, just a simple question, we've read a lot in the papers recently about the El Niño, becoming a super El Niño. Do these type of effects build into that type of assessment?

MELISSA FABIAN MENDOZA: Melissa Fabian Mendoza, for the record.

So for the climate change resilience assessments for the generating station as well as the Pickering Waste Management Facility, they've considered climate changes over the next roughly 40-year period. And this considers potential changes to climate hazards considering both a medium and a high emission scenario. So the high emission scenario would be kind of the worst case climate change scenario based on international standards.

MEMBER McEWAN: And a final question just on, if you like, the exposure of the storage facilities. Are CNSC staff satisfied that the security arrangements that -- and this is a very simple question at this stage, I think -- security arrangements that have been built into the multiple engineering activities that are going on, are sufficient to cope with the expectations of a secure site?

DR. ALEXANDRE VIKTOROV: Alex Viktorov.

Very simply, yes, we are satisfied. The facility itself and dry storage containers are assessed to

meet the requirements and we do conduct inspections at the waste facilities. And we have our Security Specialist if you would like a more elaborate answer.

MEMBER McEWAN: There will be new site entrances, there will be new people coming on board, so the security environment has changed since you last would have reviewed that.

DR. ALEXANDRE VIKTOROV: Yeah, of course, we are certainly aware that there will be more people on site, more traffic, and new entry points. So security aspects are being assessed as part of preparation for refurbishment and future life.

MEMBER McEWAN: And so your current -- because again, in the CMD you say, "have been" satisfactory. Not an overt statement they will be satisfactory in the future.

DR. ALEXANDRE VIKTOROV: We're making a statement that OPG will be able to continue to make adequate provisions for safety and security, and safeguards.

MEMBER McEWAN: Thank you.

Mr. President.

THE PRESIDENT: All right.

Commission Member Hopwood, please.

MEMBER HOPWOOD: I just have a question with regard to the dry storage containers, which is really to

better understand or better get your detail about the change to the fifth building. So you've increased the capacity of that building, I think, to 1,420 units, but that doesn't help me understand how many years or how long the building will be able to accommodate the continuing move into dry storage. Could you just kind of go through the strategy that led you to that number?

KAPIL AGGARWAL: Kapil Aggarwal, for the record.

Ms. Rambukkana will be answering that question, but maybe I'll just open by saying that the current licence has provisions to build two buildings, SB5 and SB6, for a total capacity of 1,200 DSCs. The change here is to build one big building with a capacity of 1,410 DSCs, and I'll let Ms. Rambukkana elaborate on that.

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

So as part of OPG's commitment to both project excellence and operation excellence, the one building that is being constructed in the footprint that we had proposed for two has gone through a detailed design process. And through that detailed design process for this Storage Building 5, is what it's named, we determined that the maximum footprint for the area could accommodate some additional dry storage containers, an additional 210

compared to the operational limit that is currently in our licence, which is 1,200 dry storage containers.

And so as part of that process, we're seeking to increase the number. There is sufficient space for 1,200 dry storage containers. But through ongoing communications and engagement activities that we've had with the Indigenous nations and communities, as well as the public, we are listening and acting on that feedback and building in some additional contingency until disposal measures are implemented and in service. For used fuel, that is the Nuclear Waste Management Organization's deep geological repository.

So really, there is sufficient capacity at the site to be able to have this operational flexibility. But we do want to have some additional contingency measures in place related to ensuring that there is sufficient storage abilities when the deep geological repository comes in service. So that is planned right now for 2043. And with the 1,200 dry storage containers, we would be able to continue to operate until then. With the proposed maximizing the space of the 1,410, this allows us that additional operational flexibility.

MEMBER HOPWOOD: Okay, thank you. Now I understand the reason behind the change, so I appreciate that. Thanks very much, and back to the President.

THE PRESIDENT: All right, thank you.

Commission Member Lacroix, please.

MEMBER LACROIX: I have a follow-up question concerning the dry storage container, and that's for OPG.

I read that during the proposed licence period, OPG intends to come up with a new design for lid-to-base welding for the DSC, and it says that it is supposed to improve efficiency and increase DSC production. So could you clarify what you mean by "improve efficiency" -- with regard to what? And "increase DSC production" -- production of DSCs or production of containing waste?

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

So that is correct. OPG does plan to implement a design change to the dry storage container design lid and base. And so this version is called a Mark III, which really means that this is the third iteration of the dry storage container. So we've made some modifications in the past and are continuing to look at innovations and ways that we can improve to support operational excellence, and one is to simplify. This change would allow us to simplify the type of weld that's applied to the lid and the base, and that's called a V-groove weld. And this change is based on the 30 years of lessons learned that we have with

handling and welding the lid and base together for used fuel.

And really when we talk about efficiencies, yes, it would essentially allow us to process more using the same equipment that we have available to us today. So it is an innovation, it continues to perform a containment function for separating the used fuel from the environment, but ultimately, yes, it will still be the same number of fuel bundles, 384, that go into a dry storage container.

MEMBER LACROIX: I understand. Thank you for the clarification.

Back to you, President.

THE PRESIDENT: Commission Member Remenda, please.

MEMBER REMENDA: I just wanted to follow up with some questions concerning the foundation for these buildings that I understand now are under construction. I would like to know, what is the geological foundation on which these buildings are based? And OPG if you could provide that?

KAPIL AGGARWAL: Kapil Aggarwal, for the record.

So the buildings that are being constructed -- and I will assume you're talking about SB5 in particular, the storage building for used fuel -- it's

going through a rigorous design process through our engineering change control modification, and the base is designed accordingly, similar to some of our other buildings. I won't have the details on the geological conditions, and perhaps we can take an undertaking to get that back to you for Part 2.

MEMBER REMENDA: I don't think we need to do an actual undertaking, but certainly I would like to hear more about the foundation conditions underneath these buildings that will be holding extremely heavy loads. Thank you.

Back to you, President.

THE PRESIDENT: All right. Well, I do think OPG's undertaking to get more information -- they're not in a position to do that today, so that sounds like an undertaking to me. So all right, we'll accept that.

Commissioner Member Hardie, please.

MEMBER HARDIE: Well, thank you.

I just wanted to go back to OPG and ask, with the construction of the Storage Building Number 5, as well as that encapsulating of the damaged or defective fuel that is currently stored, would there be any changes in radiological hazards that have been identified?

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

So we looked at this, and maybe I will start

and I will turn it over to my peer, Ms. Corkum, to talk about the radiological piece as well as Cammie, Ms. Cheng from Environment.

But in terms of the Storage Building 5 and dealing with damaged and defective fuel, our feasibility assessment looked at how many bundles could actually be stored in a dry storage container. And due to the size of the encapsulated bundle, we wouldn't be able to fit as many bundles in a dry storage container. So the actual dose rate from a container that would contain damaged or defective fuel would actually be less, or predicted to be less, through that feasibility study.

In terms of Storage Building 5, yes, we do need to and we have completed the safety analysis for that building. And overall, during normal operations, as well as any postulated accident scenarios, that we would still be a fraction of the regulatory limits in terms of dose to workers, public, or the environment.

And maybe now I will turn it over to Ms. Cheng to talk about any environmental impacts.

MEMBER HARDIE: Just before I do turn it over, I would just like to add to my question whether that proposed additional capacity to, in addition -- I believe it was 250 storage containers that may have been added; I don't have the number exactly in front of me, so if someone

has it for the record, that would be great -- but if you could include that within your response, as well, to whether that impacts the radiological hazards as well?

Thank you.

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

So perhaps I will start and then talk about, yes, the addition of the 210 dry storage containers to SB5 has been looked at. And ultimately, again, the dose rates are negligible from that increase.

But I will turn it over to Ms. Cheng for details from an emissions perspective.

CAMMIE CHENG: Cammie Cheng, for the record.

I have a very short addition there, really. I think Ms. Rambukkana covered most of it. But in terms of the predictive environmental risk assessment, we do include the contributions to dose that are estimated from the various facilities within the Pickering Waste Management Facility, and that includes SB5. And there were safety assessments conducted for SB5 previously, as well as for the current 1,410 dry storage containers. And actually, the dose rates that are used encompass those that are estimated for the 1,410. It does not increase the overall dose at the fence line of the property, and so there's no change to those conclusions.

LAUREN CORKUM: Lauren Corkum, for the record.

Just to add one additional point, we will be utilizing environmental TLDs around this building to confirm that our assumptions and what we've identified in our analysis does indeed meet what we're seeing. We will also be following our routine monitoring surveying program throughout this building. So those are the ways that we'll be validating our modelling. Thank you.

MEMBER HARDIE: Thank you, and I am sure we will talk about some of that validation in Part 2 as well. Thank you.

I will just turn to CNSC staff, if they have any comments they would like to add? Thank you.

DR. ALEXANDRE VIKTOROV: I will seek help from our Specialist in Waste and Decommissioning, Ms. Dranga, perhaps.

--- Pause

VLADIMIR KHOTYLEV: Vladimir Khotylev, Reactor Physics --

DR. ALEXANDRE VIKTOROV: We will go to Mr. Khotylev.

VLADIMIR KHOTYLEV: Thank you. Reactor Physics and Thermalhydraulics Division, for the record.

THE PRESIDENT: Could you speak up a bit? It's very difficult to hear you.

VLADIMIR KHOTYLEV: Vladimir Khotylev, Reactor Physics and Thermalhydraulics Division, for the record.

As OPG clarified, they performed systematical review and update of safety analysis report to account for increased capacity of Building 5. CNSC staff reviewed updated safety analysis report and concluded that OPG followed the well-established process and they evaluated all important aspects of the change. The review of CNSC staff confirmed also that changes in proposed capacity will not impact results of credible accidents and malfunctions. Our review confirmed what OPG established, and they established that bounding credible accident will remain valid, the same accident which is currently included into final safety analysis report for the facility.

Thank you.

MEMBER HARDIE: Thank you very much for that response.

I will pass it back to the President.

THE PRESIDENT: All right.

Commission Member Berube, please.

MEMBER BERUBE: I've got a couple of questions on this. For OPG, let's go there first.

You've been evaluating basically putting six-year cooled fuel into casks, and I'm wondering if that is the current plan for Building 5, versus the usual 10.

Could you bring us up to speed on that?

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

So the six-year-old fuel -- our licence is very specific on once a dry storage container is processed, where it can be stored. And so under the existing licence, it allows us, once the dry storage container is processed, to then move it to Storage Building 3 in a very specific area of that building, which is in the middle of the building. And so we are not seeking to change that under relicensing. So any fuel that is loaded that has been cooled for less than 10 years, but more than six years, would continue to be stored in Storage Building 3 at the Pickering Waste Management Facility.

MEMBER BERUBE: So that is still under evaluation then for later on? There are no current plans at all to move to Building 5, is what I am hearing.

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

That is correct.

MEMBER BERUBE: That concludes my questions. Thank you.

Back to the President.

THE PRESIDENT: All right, just a couple of follow-ups for me.

You've made extensive preparations and plans to build out capacity for storage of used fuel and components removed during the refurbishment. Just clarify for me the extent to which the increased activity, decommissioning and otherwise, will increase the amount of off-site transfer of radioactive material to the Western Waste Facility. Maybe just walk me through what is expected and what changes would occur to that.

Thank you.

HEATHER RAMBUKKANA: Heather Rambukkana, for the record.

I will start and talk about the refurbishment and operations and maintenance type waste, and then I will turn it over to my colleague, Mr. Amyot, to talk about the decommissioning waste.

But from a refurbishment project perspective, the majority of the reactor core components are actually going to remain on site in the Pickering Component Storage Structure. And so we have been innovative and implemented learnings from previous refurbishment projects, and are actually volume reducing through specialized tooling in the reactor containment area, to volume reduce pressure tubes and calandria tubes and then store them in specialized containers, retube waste containers they're called, at the Pickering component storage structure.

We talked earlier about the steam generators, the 48 steam generators that will also remain on site and be stored in this building. So the majority of the refurbishment waste, particularly around reactor core components and other large metal objects like the boiler, will actually remain on site.

From a perspective of other operations and maintenance activities and in this 10-year period, we're also returning the units to service. We have anticipated a storage volume from a low-level and intermediate-level waste perspective for what would need to occur at a licensed facility. And so after processing low-level waste, which we do on a regular basis, we see a volume reduction of low-level waste of over 60 per cent. And after those volume reduction activities, the stored volume is predicted to be over the next 10 years around 6,500 cubic metres of low-level waste.

From an intermediate-level waste, which is a much smaller volume -- and overall in Canada, it makes up about 7 per cent of radioactive waste -- the amount of non-reactor core components is estimated over the next 10 years to be just under 400 cubic metres of low-level waste.

And then finally, the piece around the high-level waste, that stays on site. And over the next 10 years during the licensing period, as I mentioned earlier,

it is approximately 80 dry storage containers worth of fuel per year, which is approximately 30,000 bundles that will be handled and in that case stored on site.

And now I'll turn it over to Mr. Amyot to talk about the decommissioning.

CHASE AMYOT: Chase Amyot, for the record. Earlier, I mentioned about the low-level waste and the 454 cubic metres that will follow the standard low-level waste streams that will go for volume reduction and metal processing and then for interim storage.

As part of the decommissioning plan, just sort of the overall picture, decommissioning is looking to wrap up most of our activities before refurbishment would start. So we have about 150 people on site at the max of decommissioning over this licence period. During the refurbishment window, we will shrink down to a very small organization to just continue with managing the fuel bays and the storage or surveillance plan. So it's a fraction of the 150 people.

As part of the waste that's coming out between now and beginning of refurbishment, should it get approved, 85 per cent of the materials, so about 26,500 cubic metres of metals will be removed. So far to date, I know I'm going to mix up volumes here, but 1.2 million metric -- sorry, 1,234 metric tons of clean metals have

been removed from site so far as part of the decommissioning activities at Pickering. And over the course of this licence period, that will add up to about 26,500 cubic metres of metal removed.

So that kind of gives you an idea of volume removed off site. There is about 7 per cent of the volume would be asbestos and that goes to hazardous landfill for disposal.

THE PRESIDENT: Okay, I probably missed it, but if there was that kind of material in your -- was that in your submission, those numbers? I don't recall if those were in your submission, but if they were not, it would be useful to have that in context for Part 2, just so we have an understanding.

And on the refurbishment side, I appreciate what was said and the numbers that were provided as estimates. Could I just get a clarification? In terms of reactor components, I think the comment was made that it was most of the components. Is there a part that is not the most, a small part, that would end up elsewhere? And I presume it's all in Western Waste. I assume that's where it's headed.

HEATHER RAMBUKKANA: Heather Rambukkana, for the record. So from our reactor core component, I'll give you an example. So the end fittings, we are separating the

intermediate-level waste from the low-level waste. So the inboard end fittings will remain on site. Those have been characterized through our waste characterization program as intermediate-level waste. So they will be cut to size and stored in retube waste containers in a Pickering component storage structure. The outboard end fittings are actually -- through waste characterization and we know these learnings from other refurbishments -- are low-level waste. And so in that situation, we are going through the effort to ensure that the low-level waste and intermediate-level waste does get separated. And so yes, the outboard end fittings would end up going to a licensed facility that can take low-level waste, such as the Western Waste Management Facility.

THE PRESIDENT: Okay, all right. I anticipate some public interest in the volumes and the context. And I appreciate that there's been a ton of work done in reducing volumes, and we've seen those results. But I think there's interest, so I think it would be useful to provide that.

And I just turn to staff around their observation, their thoughts, their analysis, or their view on what we've just heard about. Thank you.

JOHN THELEN: Thank you, and good afternoon. My name is John Thelen, and I'm the acting director of CNSC's Waste and Decommissioning Division, a division

that's responsible for regulatory oversight of OPG's activities at their current Pickering Waste Management Facility under their current licence.

So OPG does have in place a waste management program to oversee the management of all current and planned future volumes of hazardous and radioactive waste. This program is being maintained and staff verify compliance through our inspections, desktop reviews and reporting. And staff, just to summarize what was said earlier, staff are satisfied that OPG does have established plans, procedures, and processes in place for managing all of its wastes, including the waste discussed today from proposed Units 4 and 5 to 8 refurbishment at the Pickering component storage structure that is under construction.

THE PRESIDENT: All right. Thank you for that. So again, just would be interested in having those numbers for Part 2 so that we can address those.

And I have another question that has nothing to do with waste. It's around the organizational changes that are contemplated as OPG moves to refurbishment and other activities. Can you describe to me -- again, very high level -- the change management plan that's in place? You know, you've talked about new roles for individuals, potentially new skills, new work assignments. So how are you preparing, at a very high level, the staff for that

evolution that's going to take place fairly quickly?

KERRY JACKSON: Kerry Jackson, Operations director, for the record. Just to reinforce, so all of the departments at Pickering right now do have a change management plan in place that details the transition within each of their own work groups, with senior leadership oversight monitoring the progress of those transition plans.

One thing to clarify, although as workers transition, I'll say from the operating org chart to the project's organization, which would occur post-shutdown, the OPG workers will primarily be performing the same tasks that they do today. For example, a maintenance control technician will still be performing control tech duties; a non-licensed operator will still be performing non-licensed duties.

Our commitment to those staff were to ensure that they knew exactly what they were doing as part of our change management process so that they could focus on first priority, which is the safe, reliable operation of the units until we shut down, but also be thinking ahead and understanding what skills and qualifications might be required if there was anything to change or additional in order to perform their role in refurbishment. There really is no change there, but we want to ensure that folks have

that understanding and comfort of knowing what they would be doing during the refurbishment period.

But for clarity, really, the OPG folks will be performing the same skills that they would perform today on the Pickering 5-8 units.

As part of that, we did talk a little bit about oversight and the voice of operations. So as the licence holder changes from Mr. Rogers to the current DOM, we still want to make sure that there's a very deliberate operational conscience to ensure that the site is maintaining all of the regulatory compliance standards that would be required for safety and then to ensure that we maintain that as we progress into the next phase of our transition plan is where we stand up a station organization which would look quite similar as it does today.

We want to ensure that that station organization is stood up and ready prior to new fuel load for Unit No. 5 and that they will progress into the startup of Unit 5 so that we can have a safe startup and run Unit 5 islanded while the enterprise project organization will continue what they're planning for right now is Unit 6, 7, and 8 refurbishment.

And as Unit 6 looks to come back online, that station organization will continue to grow. And as 7 comes back online, that will continue to grow as well and so on

and so forth until you have a Pickering station with four operating units of which the enterprise organization, projects organization will be moving on to other things.

THE PRESIDENT: All right, thank you. That is helpful clarification. These organizational changes can be significant, and aligning people is an important part of doing a quality job.

We have organizational specialists and folks that are looking at this. Do you have any comments to make on, as we watch the changes that are taking place?

DR. ALEXANDRE VIKTOROV: Alex Viktorov, for the record. We do have specialists in the human performance and organizational performance domain. I don't know if anyone is online to comment on this aspect. Do we have anyone from Management System Division? Lisa Hagen --

THE PRESIDENT: I think we have got clarification from OPG. I am satisfied. Just obviously it's part of our surveillance plan and --

DR. ALEXANDRE VIKTOROV: Of course, yes. We do perform a lot of oversight activities regarding the management system, that's an overarching area, cross-cutting areas that we always look, regardless of the topical area, so of course that's part of our oversight.

THE PRESIDENT: Okay, thank you very much for that. I think that is my questions on this topic. I will

turn to the Commissioners and see if there is any residual questions they want to ask, so I will go from left to right.

MEMBER BERUBE: I think I am going to reserve the rest of my questions for Part 2.

MEMBER HARDIE: No additional for me.

MEMBER LACROIX: Yes, I do have a snap question. I think that we have discussed this matter in the past, not today, but in the past. And I have read that according to the periodic safety report number three, and my question is addressed to OPG, there is a plan to address the leakage from the Pickering irradiated fuel bay and a plan to develop and to correct this fault.

And first of all, I would like to know what sort of leakage are we talking about, and what is this plan all about?

JEFF MILLER: Jeff Miller, for the record. The fuel bay leakage is internal to the station and is bounded. The leakage is through the bay's internal liner system into the designated interspace and is captured in interspace sumps. So bay water does not leak externally, therefore it does not pose an uncontrolled environmental release risk.

Nuclear safety is maintained through multiple layers of monitoring and controls. The bay level is continuously monitored, so there's alarms; there's operator

checks and procedural controls to ensure that the water shielding is maintained above the spent fuel. The leaked water is pumped back into the bay in a closed loop, and there's adequate pumping capacity as well as system redundancy with only one pump required for the bay.

Environmental protection is verified through monitoring. There is quarterly groundwater sampling, including tritium as a key indicator, which shows no adverse trends attributed to these bays.

And OPG also has an approved project to repair the liners. This project includes comprehensive liner inspection, a definition which will define the repair scope, and execution of repairs is coordinated with the refurbishment defuelling constraints. So inspection work is planned for 2026-2027 timeframe, with repairs scheduled in the 2030 to 2032 timeframe.

MEMBER LACROIX: And what is the volume, the rate of leakage in terms of litres per hour, per day?

JEFF MILLER: Jeff Miller, for the record. The monitored leakage rate is on the average of about 1.5 cubic metres per hour. Per hour.

MEMBER LACROIX: Per ...?

JEFF MILLER: Per hour.

MEMBER LACROIX: Oh, I see. Okay, thank you very much.

THE PRESIDENT: Okay, Mr. Hopwood?

MEMBER HOPWOOD: Sorry guys, I have got two questions. I will make them quick, and I will encourage quick answers too. So I don't need a long answer, but I think that there are a couple of points that I do want to check in with you.

First of all, on the financial guarantee, the choice to refurbish Units 5 to 8 means that there will be a greater level of radioactive waste and other wastes to deal with at the end of the station life. So I would expect that, other things being equal, the financial guarantee would have to increase in some way. On the other hand, I have seen also that you have reasons to be able to have decreased the financial guarantee. Maybe you can walk me through that to confirm that you have taken into account the changes.

CHASE AMYOT: Chase Amyot, for the record. Yes, as part of refurbishment and extension of Pickering, OPG did review the financial guarantee and submit it to the CNSC for review and acceptance. The financial guarantee was reviewed and accepted by the CNSC staff and deemed sufficient.

MEMBER HOPWOOD: Just any comments from the staff?

DAVID ROGERS: Dave Rogers, for the record.

Your short answer is more time, value of money, earns interest. That fortifies the financial guarantees.

MEMBER HOPWOOD: Thank you, that is helpful.

And just quickly, any comments from the staff?

DANIEL MacDONALD: Yes, Daniel MacDonald, Pickering Regulatory Program Division, for the record. OPG did provide an assessment of their financial guarantee to CNSC staff. CNSC staff found that it was acceptable. OPG's financial guarantee does remain sufficient for the future decommissioning of the Pickering nuclear site. Their current financial guarantee covers a period from 2022 to 2027. OPG will come to the Commission with a full revised financial guarantee package for the next five-year period probably sometime later this year or next year.

MEMBER HOPWOOD: Thank you, that is good.

And my other question really is regarding incorporating OpEx and other industry experience into the refurbishment project. I am conscious that your application and your submissions talk quite a lot about getting OpEx from Darlington, from the refurb that you carried out there. The Pickering units are different. They have containment buildings as opposed to the Darlington model. So I would think that you may be able to collect good OpEx from other CANDU refurbs from some of the offshore Canadian units. Can you just tell me if you have

been approaching or gathering information from beyond the Darlington experience?

LUCA CECCATO: So, Luca Ceccato for the record. Short answer is yes. We do have some interface with Point Lepreau, which obviously did it years ago. We do have lessons learned from Units 1 and 4, from that return to service evolution, although not to the same magnitude, but a return to service nonetheless. And certainly with Bruce Power.

And so we are not just, you know, on Teams meetings collaborating, but we are actually sending folks to the site. Bruce Power has been very instrumental in helping us on the Pickering as we were instrumental in their success from our Darlington success. So there's a clear organization supported with all the past refurbs to gain all the lessons learned on top of our already robust process from Darlington, but didn't want to leave anything to chance in Ontario. As I mentioned at the onset, eighth refurbishment in Ontario, and we're taking all the learnings. We haven't left a stone unturned to go after all the OpEx.

MEMBER HOPWOOD: Thanks very much.

And back to the President.

MEMBER HOPWOOD: All right. Commissioner McEwan, anything further? No? Okay.

THE PRESIDENT: All right. So I want to be clear on this point, and then we'll conclude our questions. So the Commission's assumption is that OPG will proceed with the deep water intake, as this was the preferred option reflected in OPG's application. And we've heard that it formed the basis of CNSC staff's assessment of the application.

Recognizing that interventions in this proceeding are due on August 12th, 2026, and Part 2 of the hearing is scheduled for the week of October 5th, should OPG decide not to proceed with the DWI, it must inform the Commission of its decision on or before August 1st of 2026. This date is driven by the need to be fair to all participants in this proceeding, including Indigenous nations and communities, and to give them sufficient time to respond to any potential changes to OPG's application and CNSC staff's assessment of those changes.

So with that, it concludes our questions in Part 1 of this public hearing for the application from Ontario Power Generation for the authorization to refurbish the Pickering Nuclear Generating Station and renew its Pickering Nuclear Generating Station and Waste Management Facility licences.

I will now turn the floor to OPG for final remarks. Thank you.

DAVID ROGERS: Dave Rogers, for the record. President Tremblay, Members of the Commission, Members of the CNSC, those watching, thank you for your time and consideration for OPG's application. As you heard today, Pickering continues to operate safely and reliably with strong performance across all safety and control areas. In all that we do, whether operating our nuclear units, waste facility, refurbishment or decommissioning activities, we do in accordance with approved programs, regulatory requirements, and industry best practices, protecting workers, the public, and the environment.

In closing, OPG remains qualified to carry out the licensed activities and will continue to protect health, safety, security, and the environment while meeting national and international obligations. We remain committed to engagement with Indigenous rights holders, our host community, and the public. We look forward to discussing OPG's application and hearing from intervenors during Part 2 of this hearing in October. Thank you.

THE PRESIDENT: All right, thank you very much for that, for your remarks and presentations.

I'll just turn to CNSC staff for any final remarks they'd wish to make.

DR. ALEXANDRE VIKTOROV: Thank you, President Tremblay. CNSC staff did our best to assess the

information submitted by OPG in the application as well as information that we collected through our site activities. So we stand by our conclusions and recommendations. Of course, we also take into account any additional information that will be submitted before Part 2 and confirm or modify our recommendation. Again, appreciate everyone's support for this hearing. Thank you.

THE PRESIDENT: All right, thank you.

Thank you to everyone for your presence and participation today. We look forward to seeing you in Part 2. Please contact the Registry should you have any questions about participating in or attending Part 2 of the hearing. I'll now turn the floor to the Deputy Registrar for a few final remarks.

JUDE SAMSON: As previously mentioned, this public hearing will continue during the week of October 5th, 2026, in Ajax, Ontario. Virtual participation will also be available. Public is invited to participate in writing, by oral presentation or both in Part 2 of the public hearing. Persons who wish to intervene must file their requests with the Commission Registry by August 12th, 2026.

Thank you, merci, et bonne fin de journée.

--- Whereupon the hearing adjourned at 4:44 p.m. /

L'audience est ajournée à 16 h 44

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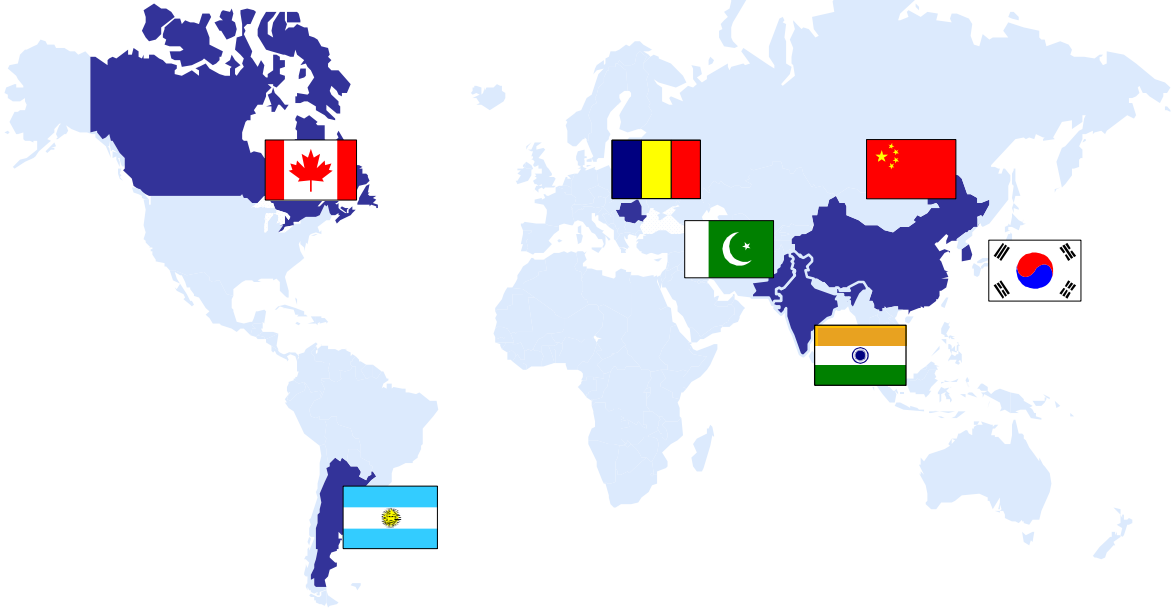
COG-JP-4499-025-R1

K-410122-REPT-0010 R02

Whole-Site Risk Considerations for Nuclear Power Plants



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Whole-Site Risk Considerations for Nuclear Power Plants

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Source of Funding: The work reported in this document was funded by the COG Joint Project Agreement titled, *CANDU Whole-Site Probabilistic Safety Assessment* under the joint participation of *Bruce Power, Canadian Nuclear Laboratories, Korea Hydro and Nuclear Power, New Brunswick Power, Ontario Power Generation and Societatea Nationala NUCLEARELECTRICA.*

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Quality Program: The work reported herein performed under the Kinectrics Quality Manual which is consistent with CSA Z299.1-85 *Quality Assurance Program – Category 1* and the relevant sections of CSA N286-05 *Management System Requirements for Nuclear Power Plants*.

Whole-Site Risk Considerations for Nuclear Power Plants

Report No. COG-JP-4499-025-R1

September 2017

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Acknowledgement

The author acknowledges with thanks the feedback on a draft version of this report provided by Paul Lawrence of Kinectrics Inc. and Ben Hryciw of Amec Foster Wheeler.

Executive Summary

The nuclear power industry has, from its inception, made safety the highest priority in the design, construction, operation, and decommissioning of nuclear power plants. A long-standing fundamental safety principle is that a nuclear power plant should pose only a very small incremental risk to the health and safety of the surrounding population. This principle is inherent in the way nuclear power plants are operated and is a fundamental element of the Canadian nuclear regulatory framework. Utilities have a comprehensive suite of programs which ensure that nuclear power plants are operated well within regulatory limits. Utilities would not operate a nuclear power plant, nor would the Canadian Nuclear Safety Commission allow it, if the risk to the surrounding population was unreasonable. The reasonableness of the risk is a value judgment based on extensive qualitative and quantitative information, including confirmation that regulatory limits and goals are met.

In recent years, the term "whole-site risk" has become the subject of discussion at licensing hearings in Canada and in the international nuclear community, particularly as it relates to multi-unit nuclear power plants. The discussion has centred around whether the safety assessments of multi-unit stations account for the "whole-site risk", meaning the overall risk associated with all sources of radioactivity on the site and all potential hazards. This report describes how the whole-site risk associated with a nuclear power plant is characterized and how it is managed by utilities to ensure that the level of safety remains acceptable.

The report describes the 14 Safety and Control Areas that are assessed by the Canadian Nuclear Safety Commission to determine whether regulatory requirements are met and whether the overall risk associated with a nuclear power plant, that is, the whole-site risk, is judged to be acceptable. The Safety and Control Areas span all sources of radioactivity on a site and all aspects of normal operation as well as postulated accidents.

The Safety and Control Areas consider the results of environmental monitoring programs, which are contained in public reports issued annually by Canadian nuclear utilities. All plants report that releases to the environment are well below regulatory limits and therefore pose insignificant risk to the public. As part of the industry's commitment to openness, the results of annual environmental monitoring are available to the public. The Safety and Control Area assessments also confirm that regulatory dose limits will be met for postulated accidents that are within the plant Design Basis. Meeting these limits ensures very low incremental risk to the surrounding population should such an accident occur.

The Safety and Control Areas also include consideration of very low probability accidents, referred to as Beyond Design Basis Accidents, and how Probabilistic Safety Assessment is used to determine whether established risk goals for such events are met. The report describes how Probabilistic Safety Assessment explicitly accounts for the presence of multiple reactors in a station, and illustrates how the results can be presented to understand the risk associated with different types of hazards.

In summary, the report demonstrates that whole-site risk, while not described in terms of a single number, is an important consideration by utilities and the regulator when assessing the overall safety of a nuclear power plant. The report describes the means by which whole-site risk is evaluated, with particular attention paid to how the results of

Probabilistic Safety Assessment contribute to understanding the risks of different types of hazards and the aggregate risk associated with multiple reactors at a site.

1 Introduction

The nuclear power industry has, from its inception, made safety the highest priority in the design, construction, operation, and decommissioning of nuclear power plants. A long-standing principle that has guided utilities and the regulator is that a nuclear power plant should pose only a very small incremental risk to the health and safety of the surrounding population. Utilities would not operate a nuclear power plant (NPP), nor would the Canadian Nuclear Safety Commission (CNSC) allow it, if the risk to the surrounding population was unreasonable.

In recent years, the term "whole-site risk" has become the subject of discussion at licensing hearings in Canada and in the international nuclear community, particularly as it relates to multi-unit nuclear power plants. The discussion has centred around whether the safety assessments of multi-unit stations account for the "whole-site risk", meaning the overall risk associated with all sources of radioactivity on the site and all potential hazards. This report describes an approach to characterizing whole-site risk for operating NPPs, including multi-unit stations. In particular, the potential for large off-site releases of radioactivity is addressed as it is a key part of the discussion of whole-site risk. This report was prepared for CANDU Owners Group (COG) Joint Project 4499. The approach described in this report leverages results from the COG Joint Project, and builds on the framework described in a COG report on this topic [1].

The following section discusses the key terms "safety" and "risk". Section 3 describes how whole-site risk is assessed in the context of the Canadian nuclear regulatory framework. Section 4 discusses how Probabilistic Safety Assessments (PSA) of severe accidents are performed for Canadian NPPs and how the results can be documented to provide insights into the risk of large off-site releases from NPPs. Conclusions are stated in Section 5.

2 Meaning of Safety and Risk

It is important to be clear on the meaning of terms such as "safety" and "risk", or more specifically "adequate safety" and "reasonable risk", when discussing the topic of whole-site risk and how it relates to the licensing of NPPs in Canada. These terms are discussed in this section.

An objective of assessing whole-site risk is to holistically demonstrate that a nuclear site with a single-unit, or an interconnected multi-unit NPP, or a group of single units within the exclusion zone[1], is adequately safe and does not pose an unreasonable risk to the surrounding population or the environment. However, the word "safety" does not have a simple definition [3], [4]; it is not something that can be simply measured to determine its adequacy. The CNSC's website describes how the CNSC defines safety [5], and refers to a federal court decision which states:

> *"...safety is not measured. It is judged and it is judged according to an assessment of an acceptable risk: ... An acceptable risk is essentially a value-based proposition determined by policy and/or by those authorized by governments to judge safety and/or by those exposed to the risk."*

[1] CNSC REGDOC-3.6 [2] defines site as: "With respect to nuclear facilities, the area within an exclusion zone where one or more nuclear facilities and all associated support structures and systems are located."

The statement that safety is not measured but rather is judged is consistent with the Nuclear Safety and Control Act (NSCA), which includes the following relevant statements (emphasis added):

- NSCA Section 3: The purpose of the Act is to provide for "**the limitation, to a reasonable level** and in a manner that is consistent with Canada's international obligations, **of the risks** to national security, the health and safety of persons and the environment…"
- NSCA Section 9: Objects of the Commission: "to regulate...in order to... **prevent unreasonable risk**, to the environment and to the health and safety of persons..."
- NSCA Section 24(4): "No licence shall be issued, renewed…unless, **in the opinion** of the Commission, the applicant…is **qualified to carry on the activity**…and…will…**make adequate provision for the protection of the environment, the health and safety of persons**…"

The CNSC has recently published a glossary of terminology [2]. The terms "safety" and "whole-site risk" are not defined in the glossary, but "risk" is defined as:

> *The chance of injury or loss, defined as a measure of the probability and severity of an adverse effect (consequence) to health, property, the environment or other things of value; mathematically, risk is the probability of occurrence (likelihood) of an event multiplied by its magnitude (severity).*

When judging the acceptability of risk, guidance can be taken from the fundamental safety goal stated in CNSC REGDOC-2.5.2 [6], the regulatory document on the design of nuclear power plants. The goal is that an NPP should not impose a "significant additional risk to the life and health of individuals". Although this REGDOC applies to new NPPs, the principle is universal and applies to operating plants as well.

A recent document issued by the US Nuclear Regulatory Commission (NRC) [7] aligns with the Canadian approach to risk and safety. The NRC document states:

> *In the context of NRC regulation, safety means avoiding undue risk or, stated another way, providing reasonable assurance of adequate protection for the public in connection with the use of source, byproduct and special nuclear materials.*
>
> *Quantitative (absolute) risk estimates serve as an important measure of plant safety, but do not embody the full range of considerations that enter into the judgment regarding adequate protection. The judgment regarding adequate protection derives from a more diverse set of considerations, such as acceptable design, construction, operation, maintenance, modification, and quality assurance measures, together with compliance with NRC requirements including, license conditions, orders, and regulations.*

It is clear from the above discussion that in the Canadian (and US) nuclear regulatory context, "adequate safety" and "reasonable risk" are value-based judgments made by the authorized body. In Canada, utilities are responsible for operating nuclear power plants in a manner that ensures adequate safety and reasonable risk, and the Commission Tribunal makes a determination of whether these objectives have been achieved. In making this determination, the Commission Tribunal takes into account input from the utility, CNSC staff, as well as other interested parties including those who may be exposed to the risk, through the licence hearing

intervention process. These judgments are based on a broad set of qualitative and quantitative information, as discussed in the next section. There is no requirement to reduce the input to the Commission's licensing decision to a single number. In fact, it can be argued that by requiring the Commission to "form an opinion", and in accordance with the principle that safety is judged and not measured, the NSCA prohibits licensing decisions from being based solely or primarily on a formulaic or simple numerical approach.

The implication is that whole-site risk should be defined and evaluated in a way that supports the value-based approach to safety determination and licensing decisions and takes into account both qualitative and quantitative information. That is, rather than attempting to characterize whole-site risk by a single number or even a series of numbers, the approach should be to provide quantitative and qualitative information to support a judgment on whether the whole-site risk is limited to a reasonable level. This is further discussed in the context of an NPP site in the next section.

3 Whole-Site Risk

Canadian nuclear utilities and the CNSC take an integrated approach to the evaluation of safety, consistent with a value-based approach as discussed in the previous section. The utilities as the licensees are responsible for safety. The CNSC oversees licensed activities and confirms that the requirements of the NSCA, and other applicable requirements, are met.

CNSC staff's oversight and evaluation of licensee performance is documented in annual regulatory oversight reports on NPP safety performance. These reports are available on the CNSC's public website, and they evaluate NPP safety across 14 Safety and Control Areas (SCAs), comprising 73 sub-topics as illustrated in Figure 1. The figure shows that the evaluations of the SCAs performed by CNSC staff are inputs to the Commission's licensing decisions. The evaluations assess both qualitative and quantitative information.

The CNSC staff evaluates NPP performance regularly and reports on it annually for each of the 14 SCAs. The most recent report available is for 2016 [8]. The results are shown in Figure 2 and indicate that all Canadian NPPs performed at a Satisfactory or Fully Satisfactory level for each SCA. The ratings in the CNSC's oversight report reflect a utility's effectiveness in managing the safety of and risk associated with its NPPs. Each utility has formal programs in place to ensure effective performance in each SCA. Achieving effective performance ensures that adequate safety is achieved and that the overall risk associated with the plant is limited to a reasonable level, as required by the NSCA.

The Satisfactory or Fully Satisfactory Integrated Plant Rating at the bottom of the figure is an indication that the overall risk associated with each NPP site is limited to a reasonable level, consistent with the requirements of the NSCA. That is, the Integrated Plant Rating is interpreted to mean that the whole-site risk is considered by CNSC staff to be acceptable. It is noteworthy that the whole-site risk is judged to be acceptable without requiring the total risk to be numerically quantified. This approach reflects a value judgment that depends on the specific circumstances, and is consistent with that described in Section 2.

In terms of the risk of off-site releases of radioactivity, there are several important indicators included in the SCAs and assessments of each station's performance that support the overall conclusion that the risk is acceptably low, as follows:

> **Actual off-site releases** – Actual releases of radioactivity from each NPP site are monitored through a comprehensive environmental monitoring program. The results

reflect actual safety performance in terms of off-site releases of radioactivity and so provide a quantitative input to the evaluation of whole-site risk. Bruce Power, Ontario Power Generation (OPG) and New Brunswick Power (NB Power) operate NPPs in Canada[2], and each utility reports annually on the results of its environmental monitoring program. As an example, Bruce Power's report for 2015 [9] documents actual releases of radioactivity from the Bruce site for the year and calculates the associated dose to the public. The report states that the maximum dose to an off-site individual was 2.89 µSv, which is 0.29% of the legal limit of 1,000 µSv. Moreover, the report states that 2015 was the 24th consecutive year that the calculated dose was less than 10 µSv, which is the value regarded as the lower threshold for health significance. Similar results are reported for OPG [10] and NB Power [11], with OPG reporting that public dose in 2015 from both the Pickering and Darlington stations was about 0.1% of the limit, and NB Power reporting that public dose from the Point Lepreau Generating Station was less than 0.1% of the limit. In summary, actual releases of radioactivity from Canadian NPP sites are extremely low and have not had any adverse health effect on the public.

Potential future off-site releases – The potential for future off-site releases is evaluated for normal operation and for abnormal events and accidents.

- Normal operation – The potential for future releases during normal operation is assessed based on past experience, the physical condition of the plant and expected aging effects, adherence to deterministic limits established for normal operation, and programs in place for managing radiological waste and for monitoring and mitigating releases to the environment. These factors are all considered in the SCAs identified in Figure 1 and shown in Figure 2 to be Satisfactory or Fully Satisfactory.

- Anticipated operational occurrences (AOOs) and design basis accidents (DBAs) – The potential for releases to occur as a consequence of AOOs and DBAs[3] is assessed based on the maintenance program that keeps plant equipment fit for service so as to prevent accidents due to equipment failure, operational testing to confirm that mitigating systems are reliable, operation within specified limits, and deterministic safety analysis (DSA). The DSA shows that any resultant off-site doses will be within specified limits and therefore will pose negligible incremental health risk to the off-site public. These factors are all considered in the SCAs identified in Figure 1 and shown in Figure 2 to be Satisfactory or Fully Satisfactory.

- Beyond design basis accidents (BDBAs) – These are accident scenarios that are lower probability than design basis accidents and which potentially have greater consequences. The plant design may not have been intended to cope with them but mitigating provisions in place will help to limit the consequences. PSA is performed to assess plant robustness and to quantify the likelihood of events that can lead to severe core damage and large off-site releases. PSA results are used to identify potential safety enhancements and to inform procedures and guidance aimed at mitigating BDBAs. Such procedures and guidance are in place to use available plant equipment and new equipment, such as Emergency Mitigating Equipment (EME) that was installed or is in the process of being

[2] The Gentilly-2 plant is permanently shutdown.
[3] Accidents, even those with low frequency, the plant is designed to cope with such that off-site doses are within regulatory limits.

installed based on lessons learned from the Fukushima accident in 2011, to prevent an accident from escalating to a severe accident (refer to Section 4.1 for more details). Utility staff is trained on these procedures and guides, and exercises are performed to ensure the response will be effective. All aspects of BDBA evaluation are considered in the SCAs identified in Figure 1 and shown in Figure 2 to be Satisfactory or Fully Satisfactory.

In summary, the potential for future off-site releases is evaluated holistically, covering normal operation, abnormal occurrences, design basis accidents, and beyond design basis accidents. The experience to date, programs in place, and the supporting safety analysis, demonstrate that the potential is low, and this conclusion contributes to the Integrated Plant Rating.

The discussion in this section demonstrates that the overall safety of an NPP is assessed through an evaluation of actual experience, the fitness for service of equipment, the effectiveness of programs, and the safety analysis, including both DSA and PSA. The outcome of this assessment that the Integrated Plant Rating is Satisfactory or Fully Satisfactory means that the whole-site risk is limited to a reasonable level, confirming that the fundamental safety objective is met.

An additional observation is that only very low probability scenarios, i.e., those in the BDBA category, have the potential for significant off-site releases. Actual experience demonstrates that off-site releases during normal operation are inconsequential. It also shows that there have not been abnormal occurrences or accidents with significant off-site consequences. The SCA ratings show that the predicted consequences of such events will be within prescribed limits, meaning there will not be a significant risk to the health of individuals in areas around the site. The conclusion is that the plant design, operation and maintenance ensures that the likelihood of significant off-site consequences is of very low probability and is therefore associated with highly unlikely events in the BDBA category. The next section focuses on BDBAs and how the risk associated with severe accidents is evaluated.

4 Risk of Severe Accidents

4.1 Qualitative Assessment

A defence-in-depth approach to design, operation, safety analysis, and accident management ensures that accidents that could lead to a large off-site release of radioactivity are very unlikely to occur. The defence-in-depth concept provides multiple, overlapping barriers to achieve the desired result of very low probability of an accident with significant off-site consequences.

The levels of defence can be described as follows (quoted text is from Reference [6]):

Level 1 - "The aim of the first level of defence is to prevent deviations from normal operation, and to prevent failures of structures, systems and components (SSCs) important to safety." This is achieved through the use of high quality materials and equipment, good maintenance and inspection, and operating within predetermined limits.

Level 2 - "The aim of the second level of defence is to detect and intercept any deviations from normal operation in order to prevent AOOs from escalating to accident conditions and to return the plant to a state of normal operation." This is achieved through effective control systems that are routinely exercised, and through procedures that operating staff is trained on including refresher training.

Level 3 - "The aim of the third level of defence is to minimize the consequences of accidents by providing inherent safety features, fail-safe design, additional equipment and mitigating procedures." This is achieved through the provision of systems specifically designed to mitigate accidents. These systems are routinely tested, and staff is trained on their operation.

Level 4 - "The aim of the fourth level of defence is to ensure that radioactive releases caused by severe accidents are kept as low as practicable." This is achieved by an effective containment system, together with the new EME whose purpose is to prevent an accident from progressing to a severe accident or to mitigate a severe accident should severe core damage occur.

Level 5 - "The aim of the fifth level of defence is to mitigate the radiological consequences of potential release of radioactive materials that may result from accident conditions". This is achieved through effective emergency management provisions which are regularly tested.

As noted in Section 3, Canadian nuclear utilities have made or are in the process of completing safety improvements which, in part are a result of lessons learned from the Fukushima accident. These improvements are aimed primarily at defence-in-depth Levels 3, 4 and 5, and further reduce the already low risk of events that may lead to a large off-site release of radioactivity. The specific improvements vary among the stations but generally include the following areas:

- Provision of additional make-up water for fuel cooling;
- Pressure relief for heat sinks such as the shield tank and for the calandria vault;
- Additional hydrogen mitigation;
- Portable electrical power supplies for key equipment and instrumentation and control;
- Updated Severe Accident Management Guidance which incorporates multi-unit effects and effective use of Emergency Mitigating Equipment, including for irradiated fuel bays; and
- Incorporation of severe accident and multi-unit scenarios in emergency planning and exercises.

The information presented in Section 3 indicates that all of the elements of defence-in-depth, including the Fukushima-related improvements, have been evaluated and constitute a high degree of safety for all Canadian NPPs. This provides a high level of confidence that the likelihood of a severe accident is very low, and that the NPPs are robust in guarding against a wide range of different hazards, including internal events, fires, floods, seismic, high wind, malevolent acts, cyber security threats and other hazards.

4.2 Quantitative Assessment Using PSA

As described in Section 3, Deterministic Safety Analysis is used to confirm that postulated accidents within the plant design basis meet regulatory dose limits. Deterministic analysis may also be used for some BDBA events to confirm that the consequences meet even the conservative dose limits for design basis events. For such events, quantification of the risk is not required.

For postulated BDBAs that can result in a severe accident, quantitative assessment is performed using PSA techniques. Quantitative goals for operating reactors have been established to assess the PSA results. These goals are:

- The Severe Core Damage Frequency (SCDF) should be less than 1 in 10,000 per reactor per year. Severe core damage is defined as extensive physical damage of multiple fuel channels due to overheating leading to loss of core structural integrity. This is consistent with the definition of core damage in Reference [2].
- The Large Release Frequency (LRF) should be less than 1 in 100,000 per reactor per year. A large release is defined as exceeding 10^{14} Becquerels of Cs-137 released to the environment.

These goals are surrogates for higher level qualitative goals, such as the goal described in Section 2, that a nuclear power plant should pose only a small incremental risk to the life and health of the surrounding population. Meeting the quantitative PSA goals helps to assure that the higher level goals are met. Consistent with a value-judgment-based approach to risk, the goals are not treated as hard limits, and if the PSA goals are not met or even if the margin to the goals is small, an evaluation is performed to determine whether additional measures should be taken to confirm that the risk of a severe accident and a large off-site release is limited to a reasonable level.

Consistent with international best practice, the PSAs for Canadian NPPs start with identification of initiating events that could potentially lead to fuel failures and severe core damage. This includes internal equipment failures and other hazards such as fire, as well as external hazards such as seismic events, tornados, etc. Screening is performed to identify those hazards that require more detailed evaluation. The hazard categories typically selected for quantitative PSA evaluation are:

- Internal[4] failures while the reactor is operating at power;
- Internal failures while the reactor is shutdown;
- Internal fires;
- Internal floods;
- Seismic events; and
- High winds.

An important aspect of the PSA is that it explicitly accounts for the potential for an initiating event, and for additional failures caused by the initiating event or that occur randomly, to affect more than one reactor unit in a multi-unit station. Therefore, the PSAs performed for multi-unit NPPs in Canada are, in effect, what are referred to as MUPSAs (Multi-Unit PSAs). This is a critical element of the PSAs because it explicitly takes into account interactions among units and phenomena that affect all units when calculating the PSA results. Although PSA results are expressed on a per reactor basis, the results take into account multi-unit effects and interactions that contribute to the SCDF and LRF.

For example, if an initiating event were severe enough, it could potentially cause severe core damage in all 4 units at a 4-unit station. The release of radioactivity from all 4 units is therefore considered when determining whether the Large Release threshold is exceeded. In order to not exceed the threshold, the release per unit can only be one quarter of the threshold value (assuming all units behave the same). This imposes greater safety constraints compared to

[4] An internal event is any event that proceeds from a human error or from failure of a structure, system or component. This includes equipment failures and other hazards such as a fire in the station. External events are an event of natural or human-induced origin that originates outside the site and whose effects on the reactor facility are considered as potentially hazardous. These definitions are consistent with Reference [12].

only considering the units individually. In this example, if only individual units were considered the release from a single unit could be a factor of four greater and still not exceed the Large Release threshold. This example illustrates how the current PSAs for multi-unit NPPs in Canada are MUPSAs and also how the per reactor PSA goals are applied in a multi-unit context.

These concepts are discussed in a recently issued Commission Member Document (CMD) that was presented to the Commission Tribunal in August 2017 [13]. The CMD includes a report prepared by CNSC staff titled "Regulatory Role of Probabilistic Safety Assessment" and a third-party report titled "Probabilistic Safety Assessment - Safety and Regulatory Framework". Both reports in the CMD are consistent with the concepts described thus far in this report. The following discussion elaborates on how the results of PSA can be documented and evaluated to gain further insights into NPP safety.

Table 1 shows how PSA results can be presented to facilitate understanding the relative risks associated with different hazard categories and the margins to the PSA goals. The results in the table do not correspond to those for a specific NPP but are typical of PSA results for a multi-unit Canadian NPP. The table shows the SCDF and LRF for each hazard category as well as the summed results for all internal events and comparison to the PSA goals.

Table 1 has the following attributes:

- The top half of the table shows the SCDF and LRF for hazard categories associated with internal plant failures.
- The results for At Power events are aggregated results covering a wide range of equipment failures, such as pipe breaks, control system failures, electrical system failures, etc. The results include initiating events that affect only a single reactor, as well as events that affect multiple units in a multi-unit station, such as loss of external electrical power.
- Events while the reactor is shutdown are assessed explicitly, to obtain insights for the plant when a reactor is in an outage state. For a multi-unit station, there can be combinations of reactors operating and shutdown, so expressing the results on a per reactor per year basis facilitates understanding of individual reactor behaviour in the two states.
- Results are shown separately for internal flood and internal fire. These events can affect multiple reactors simultaneously. In addition, since an internal fire may be initiated by an event that is not an equipment failure, the uncertainty in the initiating event severity as a function of frequency is different compared to that for equipment failures. Therefore, it is appropriate to present the results separately.
- The SCDF and LRF are summed for all internal hazards assessed with PSA and compared to the PSA goals on a per-reactor basis. The result is an indication of the incremental risk to the public associated with a large release of radioactivity due to internal plant failures. This result can be used with actual release results for normal operation and safety analysis of DBAs to assess the overall risk of the NPP. In the example in Table 1, there are significant margins to the PSA goals, which is typical for Canadian NPPs.

The bottom half of the table (labeled External Events) presents results separately for the two external event categories assessed with PSA: seismic events and high wind events. The first and third of the four rows in this part of the table show PSA results for initiating event

frequencies as low as 1 in 1,000 years, corresponding to events that may occur over the life of the plant and those that are unlikely to occur. The table shows that in this example the contribution to SCDF and LRF is very low. Expressing the results in this manner provides insights into the robustness of the plant for external events that are plausible and is a means to confirm that the plant poses no significant incremental risk for plausible external events.

The second and fourth rows in this part of the table present results for more unlikely external events, including those with frequencies as low as 1 in 10,000 years. Table 1 shows that these example PSA results for external events are well below those for internal events and that there are large margins to the PSA goals. For such very low initiating event frequencies for external events, it can be expected that the event itself would directly impact the safety of the surrounding population irrespective of the NPP. In such circumstances, the table is a means to convey that the LRF risk is small despite the elevated risk to the public directly associated with the initiating event, and hence the qualitative safety goal described in Section 2 is met.

It is reiterated that these results account for multi-unit effects such as scenarios where all 4 units in a station are affected and contribute to the release of radioactivity to the environment.

4.3 Whole-Site PSA Considerations

Further insights into the robustness of multi-unit NPPs with respect to severe accidents and large off-site releases of radioactivity can be obtained by comparing the PSA results expressed on a per reactor basis to results expressed on a station basis. The discussion in this section focuses primarily on LRF as it is a more direct indicator of risk to the public than SCDF. It is the primary indicator used by utilities to assess and manage the risk associated with severe accidents.

Through the Joint Project described in Section 1, COG members have developed the following approach to aggregating PSA results to express the LRF on a per station basis for a given hazard. For a station comprising 4 reactor units, for example, the station LRF for a given hazard category is obtained as follows:

LRF per station = 4 x LRF per reactor for initiating events that affect a single unit only

+ 2 x LRF per reactor for initiating events that affect two units simultaneously (this term accounts for all possible combinations of two-unit events)

+ 1 x LRF per reactor for initiating events that affect all units simultaneously (three-unit sequences are very few and are lumped in with four-unit cases)

The underlying basis for this formula is that initiating event frequencies are established on a per reactor basis independently of how many reactors are in a station. Therefore, if a random event affects only a single reactor unit, the probability that the event will occur in a multi-unit station is the probability per reactor multiplied by the number of reactors. For example, if the predicted frequency of occurrence of a small pipe break in the reactor coolant system is 1 in 100 years per reactor, then the predicted frequency of a small pipe break in a 4 unit station is 4 in 100 years. Therefore, for event sequences that affect only a single reactor and which result in the LRF threshold being exceeded, the LRF for the station is 4 times the LRF calculated for the event sequence (the first term in the above equation). The other two terms account for events that can affect combinations of two, three or all four reactors at a time.

Figure 3 demonstrates a method for graphically showing PSA results aggregated on a per reactor basis and separately on a per station basis. The results are typical for a multi-unit

Canadian NPP but do not correspond to a specific station. The "Unit Aggregation" results show the LRF aggregated for all event sequences for the given hazard category, expressed on a per reactor unit per year basis taking into account multi-unit effects that can affect the per reactor LRF. These results are the same as shown in Table 1 and for external events reflect initiating event frequencies as low as 1 in 10,000 years. The "Station Aggregation" results for each hazard category are based on the formula described above.

Figure 3 is a means to show whether a particular initiating event category dominates the calculated LRF. In this example, it is internal fire. The figure also characterizes modeling conservatism and uncertainty in the PSAs for the different hazard categories. This information is provided at the bottom of the figure, below the horizontal axis. In this context, modeling conservatism refers to assumptions in the accident progression modeling, especially for sequences that lead to severe core damage in multiple units. Uncertainty refers primarily to uncertainties associated with characterizing the initiating event severity as a function of initiating event frequency plus uncertainties associated with failure probabilities of mitigating provisions.

This information is useful when considering how to interpret and aggregate the results. For instance, common cause hazards such as internal fire, seismic events and high winds are modeled more conservatively and have larger uncertainty than at-power internal events, as shown in the figure. There are two primary reasons for this difference:

(i) There is greater uncertainty in the initiating event severities for fire, seismic and high winds for low frequencies compared to internal equipment failures. For the latter, the equipment is either functional or not. There is also greater availability of relevant component failure data for frequency quantification of equipment failures. For common cause hazards, the severity for low frequency events is extrapolated from experience with less severe but higher frequency events, so there can be substantially greater uncertainty in the initiating event severity at low frequencies. For example, it is difficult to quantify with a high degree of certainty what a 1 in 10,000 year wind is, or a 1 in 10,000 year earthquake. Sensitivity analysis is used to assess the change in SCDF or LRF to variations in event severity for low frequency initiating events.

(ii) For common cause hazards, all units in a station can be affected. Simplifying conservative assumptions are made that the accident progression is identical in all units. This typically leads to predictions of the earliest and most severe core damage and release of radioactivity, and adds a degree of conservatism beyond that in analysis for events that are limited to a single unit.

Figure 3 indicates that differences in modeling approach and uncertainty among the PSAs for different hazard categories may not change the insights gained from PSAs, in that internal fire still dominates the risk profile in this example. Similarly, consideration of station LRF versus per reactor LRF also does not change the insight that internal fire dominates the risk in this example. Nevertheless it is important to consider differences in modeling approach, conservatism, uncertainty and station aggregation to ensure a complete understanding of the PSA results for different hazards, and the figure provides a means to communicate this information.

Additional ways of presenting PSA results to facilitate risk insights are provided in Appendix A. The appendix includes examples with increasing amounts of detail and with broader scope. The second figure in the appendix presents a template for describing all factors that affect LRF for a site, including non-reactor sources of radioactive material, the effect of uncertainties, the impact of multiple units on a site, and the relative risks associated with different categories of hazards.

5 Conclusions

This report has described how the whole-site risk of NPPs is evaluated in the Canadian regulatory framework. The report presents information used by utilities and the CNSC to confirm that the risk of Canadian NPPs is limited to a reasonable level, as required by the NSCA. The report also describes the role of PSA and PSA goals in assessing whole-site risk, and describes methods for documenting and communicating PSA results to convey whether the goals are met as well as safety insights from PSA.

The key conclusions of this report are:

- The level of safety and reasonableness of the whole-site risk associated with a nuclear power plant site are evaluated using qualitative and quantitative information grouped into 14 Safety and Control Areas and 73 sub-areas.
- The actual risk to the public and environment associated with operation of Canadian NPPs has been negligible, based on the measured releases of radioactivity to the environment.
- The predicted risk associated with the continuing operation of Canadian NPPs is evaluated based on predicted releases during normal operation, abnormal occurrences, design basis accidents and beyond design basis accidents.
- Only low probability beyond design basis accidents have the potential to release large quantities of radioactive material to the environment. The risk associated with such accidents is evaluated using PSA techniques taking into account mitigating factors such as post-Fukushima design enhancements and emergency management improvements.
- PSAs performed for Canadian NPPs with multiple units are Multi-Unit PSAs in that they explicitly account for the presence of multiple units in a station, even though results are typically expressed on a per reactor basis.
- PSA results for different hazard categories for single and multi-unit stations can be documented in a way that provides meaningful insights into the margins to PSA goals, into the hazards that dominate risks, and into how the risk changes when results are expressed on a per station basis compared to a per reactor basis. Examples are shown in Figure 3 and in the additional examples shown in Appendix A.

In summary, whole-site risk is not expressed as a single number or a series of numbers. Consistent with the requirements of the Canadian regulatory framework, whole-site risk comprises qualitative and quantitative information that facilitates a value judgment of the reasonableness of risk. This information includes quantitative information on the risks of large releases of radioactivity expressed on both a per reactor basis and a per station basis for stations comprising multiple reactor units.

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Table 1 - Typical PSA Results

NOTE: The results in this table are representative of a Canadian multi-unit NPP but do not correspond to a specific station.

Hazard Category	PSA Results (per reactor per year)	
	SCDF	LRF
INTERNAL EVENTS		
At Power	5E-6	7E-7
During Shutdown	1E-6	-[1]
Flood	5E-7	-[1]
Fire	4E-6	2E-6
Total for Internal Events	1.8E-5	2.7E-6
Margin to PSA Goal	Factor of 5.6	Factor of 3.7
EXTERNAL EVENTS		
Seismic Events (up to 1 in 1,000 y frequency)	<1E-8	<1E-8
Extremely Unlikely Seismic Events (up to 1 in 10,000 y frequency)	1E-6	4E-7
High Winds (up to 1 in 1,000 y frequency)	<2E-6	<2E-7
Extremely Unlikely High Winds (up to 1 in 10,000 y frequency)	7E-6	7E-7

NOTES:

1. In this example set of results, a dash (-) in the table indicates that the result was not quantified because the contribution to risk is very low. For Flood events, the SCDF may be about an order of magnitude less than for other hazard categories, so the LRF will also be a small fraction of that for other hazard categories.

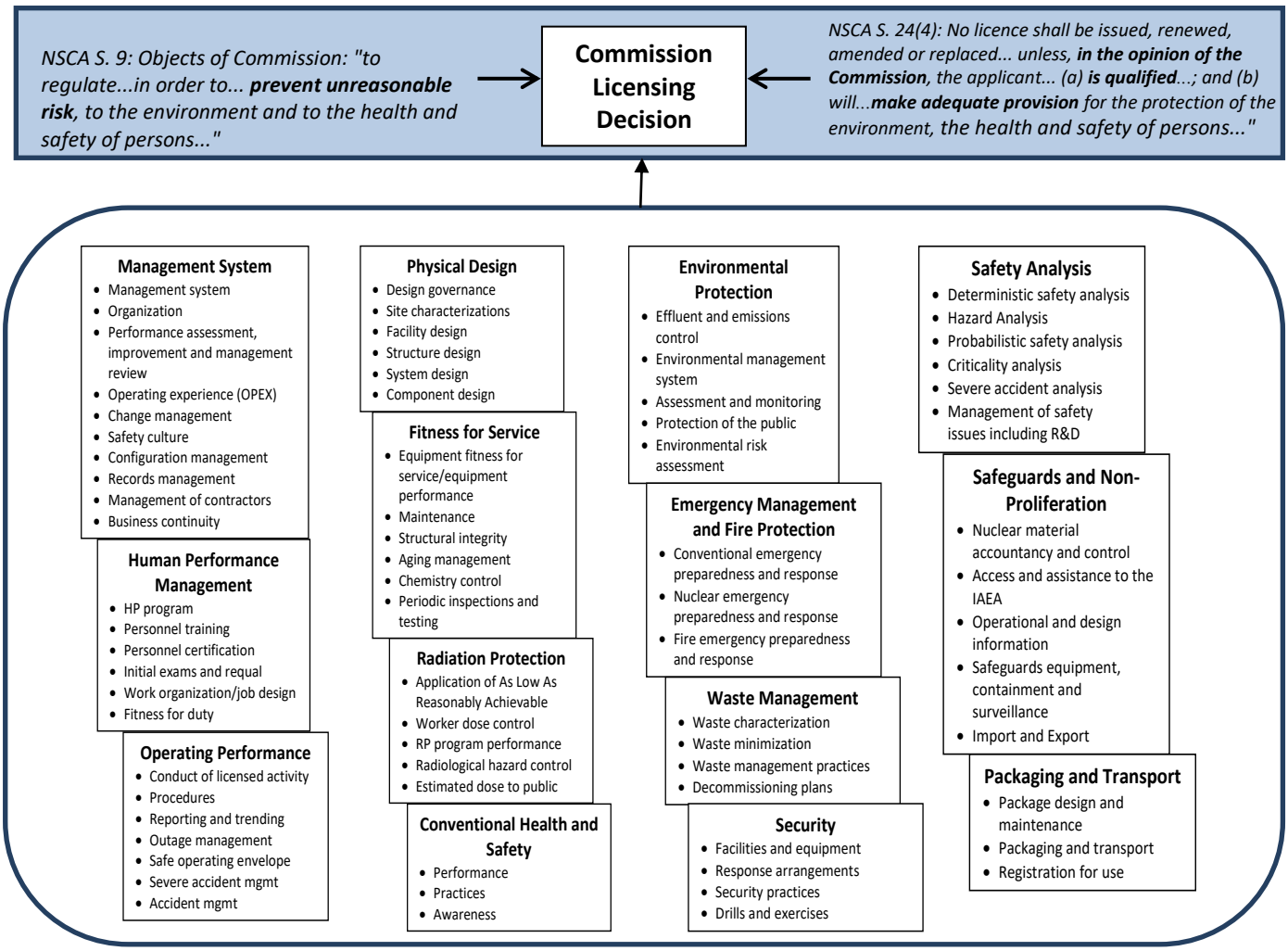


Figure 1 - CNSC Safety and Control Areas

SA = Satisfactory
FS = Fully Satisfactory

Safety and control area	Bruce A	Bruce B	Darlington	Pickering	Point Lepreau
Management system	SA	SA	SA	SA	SA
Human performance management	SA	SA	SA	SA	SA
Operating performance	FS	FS	FS	FS	SA
Safety analysis	FS	FS	FS	FS	FS
Physical design	SA	SA	SA	SA	SA
Fitness for service	SA	SA	SA	SA	SA
Radiation protection	FS	FS	FS	SA	SA
Conventional health and safety	FS	SA	SA	FS	FS
Environmental protection	SA	SA	SA	SA	SA
Emergency management and fire protection	SA	SA	SA	SA	SA
Waste management	FS	FS	FS	FS	SA
Security	SA	SA	SA	SA	SA
Safeguards and non-proliferation	SA	SA	SA	SA	SA
Packaging and transport	SA	SA	SA	SA	SA
Integrated plant rating	FS	SA	FS	FS	SA

Figure 2 - CNSC Staff Assessment of Canadian NPP Performance 2016 [8]

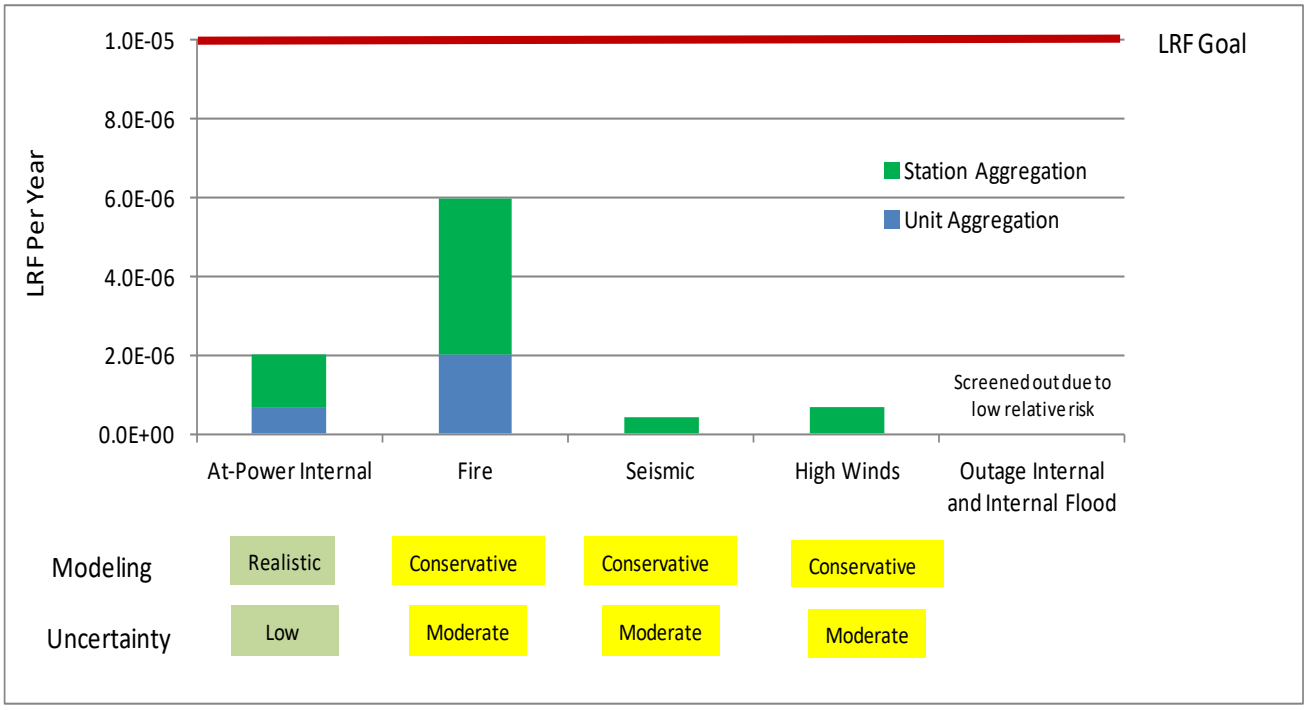


Figure 3 - Example PSA Results for Large Release Frequency

NOTE: The results in this figure are representative of a Canadian multi-unit NPP but do not correspond to a specific station.

Appendix A Additional PSA Results Presentation Approaches

PSAs produce a large amount of information that needs to be distilled and documented in order to effectively communicate the results and the risk insights. Section 4.3 of the main text presents one format for presenting PSA results. This appendix presents additional means of displaying PSA results to provide increasing levels of detail. These are examples only and are provided to illustrate that there is no one best way to present the results. The presentation should be tailored to the audience in a way that the results can be best understood, without being superficial or misleading.

Figure A-1 is similar in layout to Figure 3 but in this case the black horizontal bar represents the per reactor per year LRF calculated without accounting for the presence of other units. The additional information in the figure provides the following additional insights for this example set of results:

- The proximity of the black line to the unit aggregation value for Internal Fire indicates that fire events that lead to LRF are typically events that affect only single unit, because there is little difference between the single unit result and the result taking into account the impact of and on other units.
- The proximity of the black line to zero for seismic and high wind events indicates that external events that contribute to the LRF typically affect all four units.

Including the additional information in the figure helps to understand the extent to which a given hazard affects only a single unit or multiple units in its contribution to LRF. This insight would be used to evaluate additional mitigating provisions should such provisions be required.

Figure A-2 demonstrates a method to provide even more comprehensive information. This figure places Figure 3 in the main text into a broader context and summarizes additional aspects of the whole-site risk of large off-site releases. Figure A-2 is adapted from an approach developed for the Electric Power Research Institute and described in Reference [14]. The key features of this figure are:

- It incorporates Figure 3 and includes text that summarizes the implications of the PSA scope, methodology, conservatisms and results.
- It shows whether PSA goals are met, and margins to the goals, for the full spectrum of internal events.
    - It provides a means to discuss potentially significant hazards not assessed with PSA. In this example, events associated with spent fuel bays are addressed through deterministic safety analysis and are in the process of being screened to determine which spent fuel events, if any, for which a PSA would provide additional safety insights.
- For external events, the figure provides a means to show that:
    - In this example, unlikely external events (those with frequencies up to 1 in 10,000 years) are small contributors to LRF. These results are shown in the embedded figure.
    - The section on Safety Margin includes results of sensitivity analysis for external events with frequencies less than 1 in 10,000 years. The purpose is to confirm that there isn't a significant increase in SCDF or LRF for frequencies just beyond 1 in 10,000 years. In this example, the SCDF for seismic events does not increase significantly until the severity increases beyond that associated with

frequencies lower than 1 in 30,000 years, confirming the robustness of the plant. Similarly, for high wind events, the SCDF only doubles, and remains small, even when events with frequencies as low as 1 in 100,000 years are considered. These results are representative of a multi-unit CANDU NPP and demonstrate the robustness of the NPP to even more extreme but unlikely postulated external events.

- The figure provides a means to illustrate the risk-dominant hazard category and whether it changes when results are expressed on a per station basis compared to a per reactor basis. In this example, internal fires dominate the risk profile. Although the PSA goals are met in this example, the figure draws attention to the risk-dominant hazard category and would prompt consideration of whether further safety improvements are warranted.
- For stations of similar but not identical design, such as Bruce A and B, the bar chart in Figure A-2 could be adapted to show PSA results for the "A" and "B" stations side by side, facilitating an understanding of the differences in results between the two stations. Similarly for Pickering, which has 2 operating “A” units and 4 operating “B” units in one station, the figure provides a means to distinguish results for the two “sides” of the plant.

Overall, Figure A-2 is a means to present PSA results in a way that provides a broader range of risk insights on a single page. The figure illustrates the change in risk insights when results are expressed on a per station basis compared to a per reactor basis. It provides a means to discuss the sensitivity of PSA results to a change in cutoff frequency for external events, and it also provided a means to integrate discussion of hazards not assessed with PSA with discussion of PSA results.

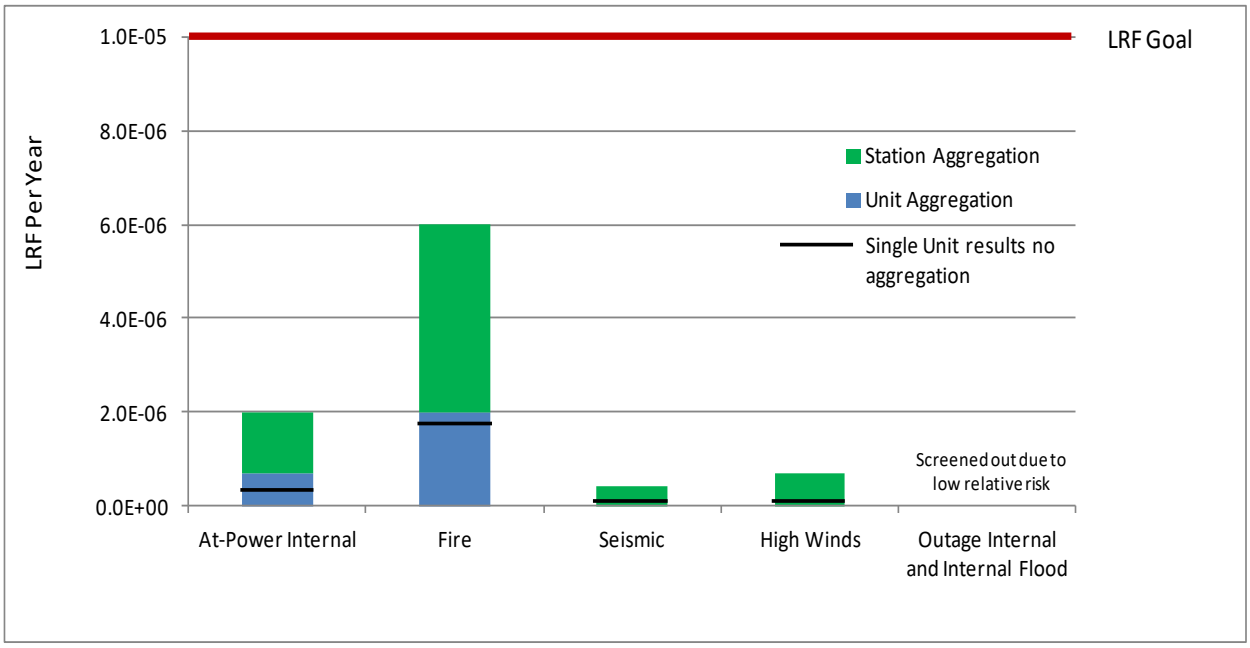


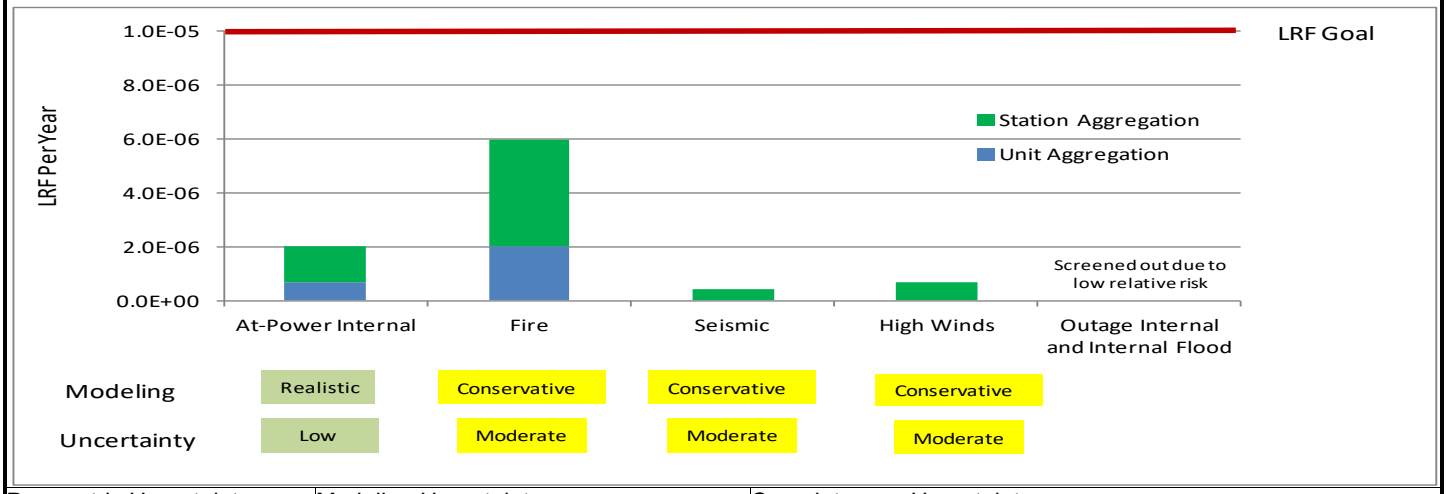
Figure A-1 - Alternative Presentation of Aggregated PSA Results

Example Whole-Site PSA Results for Large Release Frequency

Purpose

The purpose is to summarize the safety assessment based on the Large Release Frequency (LRF), defined as the frequency of accidents causing a release to the environment greater than 10^{14} Becquerels of Cs-137.

PSA Results Summary



Parametric Uncertainty	Modeling Uncertainty	Completeness Uncertainty
• Mean values represented in results	• Low uncertainty for internal failures at power and during outage • Conservative assumptions for fire initiating events • Conservative modeling for external hazards seismic and high winds	• Comprehensive scope/screening of internal and external reactor hazards. PSA performed for hazards not screened out. • External hazards cutoff frequency 1E-4/a. Margins assessed for lower frequency events. • Spent fuel events addressed deterministically and in the process of being screened for PSA.

Overall Results Characterization

- Individual hazard category results meet the LRF PSA goal of 1E-5 per reactor per year
- Sum of internal events LRF = 3.E-6/reactor-year, which meets the LRF PSA goal of 1E-5/a
- LRFs for seismic and high winds are small in comparison to aggregated internal events LRF

Multi-unit considerations: Fire hazard category dominates both single unit and multi-unit results.

Defence-in-Depth Characterization

- Defence-in-Depth improvements implemented or in progress based on Fukushima lessons learned, including fire protection improvements such as early smoke detection.

Safety Margin Characterization

- No vulnerabilities identified. No risk dominant sequences identified in Fire PSA.
- Seismic results do not increase significantly until severity exceeds that associated with 1 in about 30,000 year frequency.
- High wind SCDF only doubles and remains small even for event frequencies as low as 1 in 100,000 years.

Performance Monitoring

- Utility governance ensures potential design or operational changes are evaluated for impact on PSA results.
- Utility governance requires review if average or instantaneous PSA goals are exceeded.
- PSAs are updated every 5 years or sooner if major changes occur.

Integrated Decision-making Inputs

PSA Goal Met?	Defence-in-Depth	Safety Margins	Multi-Unit Implications	Performance Monitoring
Yes	*Confirmed*	Dominated by Fire hazard category	Fire still the dominant hazard category. No additional insights.	Annual average and instantaneous LRF monitoring

Conclusions

- Individual hazard categories and the sum of results for internal hazards meet the LRF goal.
- Large margins for external hazards with frequencies 1 in 10,000 years or greater.
- Fire dominates LRF. Results credit EME and improvements in fire detection equipment. There are no risk-dominant sequences.

Figure A-2 - Whole-Site PSA Results Summary

Appendix B Acronyms and Abbreviations

ALARA	As Low As Reasonably Achievable
AOO	Anticipated Operational Occurrence
BDBA	Beyond Design Basis Accident
CMD	Commission Member Document
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
DBA	Design Basis Accident
DSA	Deterministic Safety Analysis
EME	Emergency Mitigating Equipment
LRF	Large Release Frequency
MUPSA	Multi-Unit PSA
NB	New Brunswick (Power)
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NSCA	Nuclear Safety and Control Act
OPG	Ontario Power Generation
PSA	Probabilistic Safety Assessment
SCA	Safety and Control Area
SCDF	Severe Core Damage Frequency

CHAPTER 16

Regulatory Requirements and Licensing

prepared by
Dr. Victor G. Snell & Dr. Nikola K. Popov

***Summary**:*

This chapter covers the overall aspects of nuclear reactor regulation, with emphasis on Canada, but also covering other regimes such as the United States, the United Kingdom, and Europe. It explains the need for regulation; how regulators work; how they are structured; and the types of requirements they set. Canada, the United States, and the United Kingdom are used as specific examples in these areas, with detailed material on Canadian licensing processes, requirements, and guides.

***Disclaimer:** This Chapter captures the elements of nuclear power reactor regulation as of ~2013, when the Chapter was drafted. Although regulatory concepts change slowly, detailed licensing requirements can change fairly frequently, and some of the details in this Chapter could be superseded by the time the book is used. The reader is advised to consult the most up-to-date regulatory documents if the intended use at the time is critical. The information in this Chapter is current as of late 2012 or early 2013.*

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1 Introduction

The purpose of this chapter is to:

1. describe the need for, and concepts underlying, nuclear regulation
2. cover specific requirements in Canada in some detail, and examples from other jurisdictions where the differences are instructive.

Whereas the first aspect covers "why", the second aspect covers both "why" and "what".

1.1 Overview

Regulatory oversight of nuclear power plants is an essential aspect of the application of the technology, albeit not generally considered a specific engineering discipline. Regulation affects reactor design and operation by setting requirements at both a high level (safety goals, system requirements, operations performance metrics, etc.) and in more detail, and through regulatory review of design and operation.

Section 2 looks at overall regulatory requirements. It begins by explaining the need for regulation. It then looks at generic aspects such as regulatory structure and independence, which tend to be broadly similar in advanced nuclear-power-capable countries. It discusses how a good regulator works through drafting standards and guides and through independent review, but emphasizes that the operator is ultimately responsible for safety. It describes licensing milestones (site licence, construction licence, operating licence, decommissioning licence, and licence to abandon), which are common to most countries.

In Section 3, different regulatory approaches are briefly contrasted, varying from risk- or goal-driven at one end to highly prescriptive at the other.

Section 4 gives a number of case studies of regulatory behaviour (in the same spirit as Chapter 13, of learning from real experience)—not to suggest that the regulator was primarily responsible for an event, but that weaknesses in the overall regulatory approach led to inattention on the part of the industry to certain risks.

Section 5 discusses a recent tool for facilitating the licensing of a new design: a pre-project licensing review.

Section 6 gives some international and national examples of organizations which influence or apply regulatory requirements. Specifically, we summarize the consensus developed by the International Atomic Energy Agency (IAEA) as reflected in its guides and touch briefly on regional organizations. Then we discuss examples of prescriptive (United States) and non-prescriptive (United Kingdom) approaches.

Safeguards are described in Section 7. Security is also covered briefly in Section 8.

Sections 9 and 10 cover the Canadian approach. This can be considered as a balanced approach, mixing prescriptive requirements and overall goals. The mandate and organization of the Canadian Nuclear Safety Commission (CNSC) is presented in Section 9. The structure of Canadian Nuclear Regulations and the CNSC's Regulatory Documents and Regulatory Guides is summarized. Certain key Regulatory Documents affecting design and operation are presented in more detail. The licensing process in Canada for both existing and new nuclear power plants is described in Section 10; this includes a summary of the environmental assessment process.

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Three supplementary sections follow. Section 11 gives some problems for the student to work through. Section 12 is an Appendix giving some examples of U.K. Safety Assessment Principles as typical of a non-prescriptive approach. Section 13 is an Appendix listing the information that is typically required in Canada when applying for construction and operating licences for nuclear power plants.

Reactor regulation is a very broad field, and we can cover only the essentials in this chapter. Most sections in the text list references to documents or published papers that the interested reader can pursue for more detail: see Section 14 for the list of references. Section 15 concludes the chapter with a glossary of terms.

1.2 Learning Outcomes

The goal of this chapter is for the student to understand generically:

- The need for regulatory oversight of nuclear technology
- How a regulatory agency is structured
- How it fulfills its responsibilities
- The regulatory process and milestones
- The nature of a regulatory review
- Enforcement of regulatory requirements
- Common threads in international approaches to regulation
- Different approaches to regulation (prescriptive, goal-based, mixed)
- Concepts behind safeguards and security.

In addition, the student will gain a familiarity with the organization and mandate of the Canadian regulatory organization (the Canadian Nuclear Safety Commission, or CNSC). The student will understand the Canadian licensing process for both new and existing reactors, as well as acquiring an overall knowledge of CNSC's most important requirements and guides.

2 Generic Regulatory Requirements

This section covers regulatory requirements and processes in general, supported by selected examples. Later sections give details for Canada.

2.1 Why is a Regulator Needed?

Nuclear power is an inherently hazardous and complex technology. Society regulates such technologies, especially if they are relatively new—non-nuclear examples include prescription drugs and air travel. Regulatory oversight of a technology can encompass most or all of the life cycle, including site selection, design, construction, commissioning, operation, decommissioning, and site abandonment. In addition, fuel fabrication and management of spent fuel (should it leave the plant site) are also regulated, although these aspects are not covered in this chapter.

Regulators are authorized by the government—usually the national one—and have strong legal powers to set requirements and to intervene in the technology they regulate, e.g., to order a drug to be withdrawn from the market, or to order a nuclear power plant to make changes. They can use sanctions (fines or prison) on both corporations and individuals in case of egregious non-compliance.

An "ideal" regulatory framework balances the risk of the regulated technology against those of

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similar technologies, so that social risks are rationalized. This hardly ever happens because the mandate of regulators is usually restricted to a specific technology. Nuclear power tends to be regulated more strictly (i.e., lower risks and more risk aversion) than comparable technologies. There are many reasons for this which we do not have space to explore, but which have been touched on briefly in Chapter 13, and which include voluntary versus involuntary risk, perceived risk versus benefit, association of radiation with atomic weapons and cancer (the "dread" factor), and public unfamiliarity with the technology. The interested reader is referred to the literature on this subject. [Starr1969] and [Slovic1987] are good places to start.

One of the key functions of a regulator is that through the *potential* of independent verification of *any* aspects of a licensee's work, the regulator influences *all* a licensee's internal processes, because any piece of safety-related work can be audited. In particular, the simple *existence* of regulatory oversight stops the licensee from internally accepting deficient safety cases or accepting too readily that their assumptions are right and that they have thought of everything. In other words, oversight encourages the building and maintaining of an adequate safety culture. Regulatory oversight provides a critical independent challenge to a licensee's premises, assumptions, and positions and affects far more than the actual items that a regulator reviews.

2.2 What a Regulator is Not

A good regulator does not remove the onus for safe plant design from the designer or that for safe plant operation from the operator. *This is a key concept.* The ultimate responsibility for safe plant operation lies with the licensed operator. A regulator sets the framework and the rules to be followed, but these do not guarantee a safe plant. A deficient regulatory structure, or a regulator that is not sufficiently independent from government, may contribute to an accident by inadequate oversight, or inadequate requirements, or by being so prescriptive that the operator relinquishes the responsibility to maintain a questioning attitude. However, an operator is not absolved by a weak regulator.

Conversely, a good regulatory structure defines a climate of openness, fairness, and high expectations of safe performance that operators will internalize. In addition, a good regulatory structure ensures that international regulatory trends and approaches are readily adopted and implemented by a national regulator.

In any case, it is impossible for a regulator to check everything. This has two implications:

- there is an implicit understanding that once a licence has been given, the licensee will follow the rules and that the regulator is there to verify selected activities from time to time as it sees fit, usually in some risk-informed manner, as discussed in Section 2.5. This is called "trust but verify"[1]. See Section 2.3.3.
- a regulatory "pass" does not guarantee that the plant is safe, just as a corporate audit does not remove the responsibility of the corporation to exercise fiduciary responsibility, and just as such an audit may miss mistakes.

As stated in [IAEA2000]: "The prime responsibility for safety shall be assigned to the operator. The operator shall have the responsibility for ensuring safety in the siting, design, construction,

1 This is a Russian proverb (Доверяй, но проверяй) originally applied to US-USSR relations during the Cold War.

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commissioning, operation, decommissioning, close-out or closure of its facilities, including, as appropriate, rehabilitation of contaminated areas; and for activities in which radioactive materials are used, transported or handled." When an operator takes over a plant, it formally assumes most of the responsibility for design safety. In other words, the "design authority" responsibility is transferred from the technology vendor to the operator. In fact, the operator is required to satisfy itself that the design is adequate *before* assuming operating responsibility. Although technical aspects of the design authority can be delegated in part to the vendor, informed oversight cannot be delegated, and the operator must remain at least an "intelligent customer". In any case, this does not absolve the design organization from its safety responsibilities.

2.3 How a Regulator Exercises its Responsibilities

2.3.1 Scope

Generally, the authority of a regulator is enshrined in national legislation, which gives it the ability to permit (or deny) siting, construction, commissioning, operation, decommissioning, and site abandonment of nuclear facilities, and to control the use of radioactive material. Without a licence for a particular activity, it is illegal to proceed.

Typically regulatory authority covers more than just nuclear power plants (NPPs), including any significant nuclear activity (uranium mining, radioactive sources, research reactors, medical and industrial uses of radiation, radioactive waste disposal, etc.). However, this chapter focusses only on NPPs.

Most nuclear regulators also take on enforcement of national policy on safeguards (Section 7) and on security (Section 8) as applied to nuclear power and represent the national government in international fora on nuclear safety.

2.3.2 Nuclear power plant licensing

The steps in the licensing process, through which a regulator can exercise authority, each involve submission by the applicant of information required by the regulations (Section 10). The required information is discussed in more detail in Section 10.1 for Canada; this section gives an overview.

Typically, there are up to eight major approval milestones for a nuclear power plant. In each, the operator submits the required documentation; the regulator reviews this and (normally) grants a licence to start the milestone activity[2]. The regulator continues to review and monitor progress as the milestone is implemented. Typical milestones are:

- Environmental assessment (including site evaluation)
- Site preparation licence
- Construction licence
- Initial fuel loading
- Low-power commissioning

2 Federal, provincial, and even municipal government bodies may require many other regulatory approvals, which are outside the scope of this chapter.

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- Operation
- Decommissioning
- Site abandonment (i.e., return to unrestricted usage).

In addition, most regulators perform on request a preliminary review of an NPP design even before a construction licence application, to reduce the risk of licensing issues arising during the project stage when they are difficult and expensive to address. In Canada, this is called a pre-licensing vendor design review [CNSC2012]. In the United States, it is called design certification [USNRC2012], and in the United Kingdom, generic design assessment [HSE2008]. This topic is discussed in Section 5. Note that there are differences in the depth, scope, and implications of these approaches in different countries.

A regulatory review is usually performed against a set of published standards, requirements, and "expectations" (typically through laws, regulations, and guides). Laws are laws of the land, set by the national government; regulations are requirements set by the government (which in some countries are also laws of the land); and guides are non-mandatory "good practices" issued by the regulatory agency. The licence itself is a legal instrument and typically contains additional specific requirements that must be met. In addition, the regulator may require, or the operator may volunteer, compliance with other national standards. In Canada, these are developed by the Canadian Standards Association (CSA). Every country has its own set of requirements, but now there is more and more international coordination, initially after the severe accidents at Three Mile Island and Chernobyl, and now post-Fukushima[3].

Usually, there is some risk basis for regulatory requirements, which are generally a mix of risk-informed and prescriptive requirements based on past experience, as discussed in the accident case studies in Chapter 13.

In many countries, but not all, regulatory deliberations and decisions are made openly, with many opportunities for public input.

Given the power of the regulator, most regulators set up checks and balances to remain fair. Most regulators are structured somewhat like a corporation, with a Commission at the top supported by a small dedicated group of people, and a large separate specialized technical staff which assesses the safety case in detail and presents its conclusions and recommendations to the Commission. The Commission makes the final decision, not the staff. The Commission reports directly to, or is accountable to, the national government in a way that is independent of the proponent. Some regulators have independent advisory committees (e.g., the United States Nuclear Regulatory Commission (USNRC) has an Advisory Committee on Reactor Safeguards (ACRS) consisting of highly respected technical experts). Such committees are independent of the staff, so that technical inputs to the Commission come from more than one source [USNRC2012a]. This is a "typical" description—there are differences in the organization of regulators in different countries because the regulatory structure is the responsibility of each national government.

National and international peer reviews of regulators use the yardstick of international best practices to encourage improvement, e.g., as implemented by the IAEA through its Integrated

3 Post-Fukushima refers to the changes in nuclear safety regulations and requirements in response to the nuclear accident in the Fukushima Dai-ichi NPP in Japan in March 2011.

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Regulatory Review Service (IRRS).

2.3.3 Ongoing oversight

Granting of a licensing milestone is only part of a regulator's scope. Regulators also monitor, through documentation review and site inspections, the construction, commissioning, operation, and decommissioning of nuclear power plants. Trend indicators are used as input to decisions to take regulatory action. In particular, assessment of the safety culture in different parts of the nuclear infrastructure is an important indicator of the overall health of the nuclear safety structure and organization (for a discussion of safety culture, see Chapter 13). An operating licence renewal or major plant changes provide opportunities (or requirements) for the operator to perform an in-depth re-assessment of the plant against modern standards and to close any gaps where it is practical to do so.

Internationally, it is becoming common for periodic safety reviews (PSRs) to be performed, typically every ten years [IAEA2013]. The purposes of these are to:

- assess the cumulative effects of plant aging and plant modifications, operating experience, technical developments, and siting aspects, and
- assess plant design and operation against applicable current safety standards and operating practices.

PSRs are particularly useful if the licence renewal period is long (40 years or more in many countries).

More routinely, regulators perform audits on key aspects throughout the nuclear-facility life cycle, effectively "sampling" performance. Negative audit findings in an area will often trigger broader and more intrusive regulatory audits, until the regulator is satisfied that non-compliances are isolated, or conversely that regulatory action is warranted to ensure that any systemic deficiencies are addressed.

2.3.4 Nuclear emergencies

A regulator has a special role in nuclear emergencies. Although practices vary from country to country, the regulator is *not* expected to take control of a station during an emergency or to direct the operators. However regulatory approval may be needed for such things as planned containment venting. Instead, the regulator is often responsible for, or heavily involved in, emergency (off-site) planning and in coordinating the various governmental bodies involved in public notification, sheltering, prophylactic iodine medication, and evacuation. In some cases, the regulator may be either *de jure* or *de facto* the spokesperson to the public.

2.4 How a Regulatory Review is Done

The following are some of the main tools that a (generic) regulator uses to perform an independent review of a licence application[4] [IAEA2002]. They include:

4 Often the Safety Report is equated with the documentation required for a licensing milestone (called the safety case in some countries), but this is not correct—many more documents are reviewed, as described in Figure 14. The Safety Report as a minimum consists of a detailed design description, a deterministic safety analysis, a summary of the probabilistic safety assessment, a description of programmatic aspects such as quality assurance

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- Issuing requirements and guidance for the scope, content, and quality of the safety case. This ensures that the licensee or proponent is aware beforehand of what is required and desired. Typical material includes high-level requirements such as safety goals or dose limits ([HSE2006], [CNSC2008]); lower-level requirements such as the scope of safety analysis [CNSC2008b]; and guidance on acceptable methods [CNSC2012b]. Most jurisdictions also issue requirements on design (e.g., [HSE2006], [CNSC2008], [USNRC2012b] with various levels of prescriptive detail.
- Issuing Review Guides for regulatory staff use so that their reviews are consistent and complete. Some of these are published ([HSE2002], [USNRC2012c], [CNSC2013]), while others remain internal.
- Performing an initial acceptance review of the safety case, for completeness.
- Detailed specialist review of the safety case, using such tools as:
  - Line-by-line review using the Review Guides;
  - Independent calculations using the same computer codes as the licensee, or using independent codes;
  - Expert elicitation, including use of outside panels or individuals and independent analysis by consultants;
  - Assessment against written international standards (e.g., IAEA), industry standards (e.g., those from the Canadian Standards Association (CSA)), or against the licensee's own procedures;
  - Quality assurance review and audit, including management and safety culture;
  - Review of the completeness and appropriateness of the research and development support for any step in NPP implementation, including design, site licensing, etc.;
  - Visits to the site;
  - Consultation with international bodies and fellow regulators;
  - Independent experiments or independent analysis of industry experiments;
  - Questions arising from the above to the licensee and discussion and review of the answers.
  - Extensive technical meetings with the proponent.
- The staff review results are consolidated.
- The position is communicated to the Commission in formal meetings, at which the licensee has the opportunity to present its position. The meetings are public, along with the information that is reviewed and discussed at the meetings. It is the Commission that gives the approval to proceed with the relevant milestone.

2.5 Regulatory Independence and Social Policy

Regulatory bodies generally report to a Minister in Cabinet, or to Parliament through a Minister (as is the case in Canada), or to a similar level of the national government. The reporting is usually independent of the industry at the government level [IAEA2000]. Personnel movement between industry and the regulator is fairly common, especially in smaller countries, and is

and human factors, and demonstrations of compliance with regulatory requirements. See [USNRC, 1978] and [IAEA, 2004] for examples of the format of a Safety Report; the former is far more widely used. Details of the probabilistic safety assessment are usually submitted along with the Safety Report.

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beneficial because it ensures that regulatory staff have deep hands-on knowledge of the technology they are to regulate. Often personnel movement occurs internationally, which helps internationalize and harmonize regulatory experience.

At the top level of the regulator, the Commission Chairman, and usually the Commission members, are appointed by the government, and therefore the process is political (in the sense of reflecting the will of the people through their elected representatives). The government provides ultimate oversight of the regulatory body, but rarely formally intervenes in licensing decisions.

Regulatory decisions must balance risks, not eliminate them (risk elimination is an impossibility in almost any technology). Moreover, although most regulators say they evaluate only safety and do not explicitly acknowledge benefits, in fact they have a responsibility to account for the fact that an activity may have significant net social benefit. In this sense, regulation sets (or reflects) social policy. Hence, regulatory approval means that the project poses an *acceptable* level of risk. Very few regulators use *only* risk as a regulatory tool (i.e., based just on the probabilistic safety analysis (PSA)); see Chapter 13 for a discussion on PSA. Many regulators use or accept a risk framework to sharpen decisions—this is called *risk-informed decision making* (RIDM) and is a combination of PSA insights and traditional deterministic requirements. This is a broad topic—for more detail, see [Apostolakis2004], [Bujor2010], [CSA1997], and [USNRC2002]. In certain cases, usually when a retrofit is being considered, a cost-benefit case prepared by the proponent may be considered to ensure that the cost is not disproportionate to the benefit; see [HSE2001], [CNSC2000].

Finally, regulators must have adequate resources and experience, but they do not have to replicate all the resources of the designer or operator; cf. our example of a financial audit. To ensure adequate financial resources, regulators are given strong government support at the early stage of their development, whereas later most of their daily operations are on a cost-recovery basis from the licensees.

The above description applies to countries with an established nuclear industry. A country just embarking on a nuclear program, but with no nuclear infrastructure, faces special challenges in developing appropriate expertise and organizations ([IAEA2012b], [Popov2012]).

Four mechanisms have been used. These are not exclusive and are often used in combination.

1. The typical mechanism in the past has been to start with a research reactor and to use it to develop staff for design, operating, and regulatory functions from that base. This route was followed by Canada, where the power reactor program benefited from staff trained in the research reactors (NRU, NRX) at Chalk River. Indeed, the Atomic Energy Control Board (AECB), established in 1946 under the Atomic Energy Control Act, had responsibility for making regulations *and* for the Chalk River research project until Atomic Energy of Canada Limited (AECL) was created in 1952 under the AECB. Two years later, the Canadian Government formalized the separation of the designer/operator of the Chalk River site from the regulatory agency [Sims1980].
2. A second mechanism has been to use the vendor of a power reactor to help create the required infrastructure in the country purchasing the reactor by technology transfer and training of operating personnel. The regulatory staff in the purchasing country are developed under an agreement with the regulator of the vendor's host country. This can go as far as "importing" experienced regulatory staff.

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3. A third mechanism is to use IAEA documents and assistance. This organization has developed a comprehensive set of requirements and guides covering all aspects of commercial nuclear applications, including the organization and management thereof. The IAEA also gives assistance to countries developing a nuclear program, such as training courses in nuclear regulation [IAEA2002c].
4. A fourth mechanism is to hire *technical support organizations* (TSOs) to provide technical assistance to the developing regulatory body. These are neutral official organizations which are either part of the regulatory body or are separate, for example a national laboratory. See [IAEA2010] for recent developments.

Figure 1 from [IAEA2011] gives an idea of the steps and timescale involved in a new national nuclear power program.

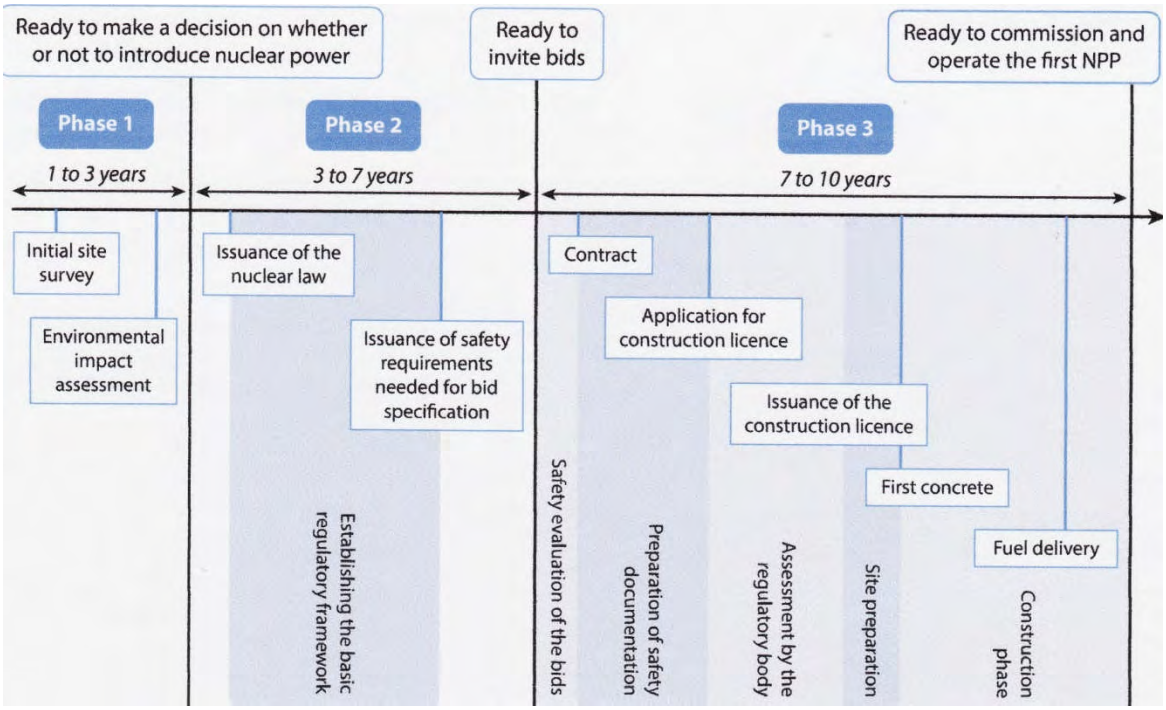

Figure 1 Steps in the development of a nuclear power program

3 Regulatory Approaches

The approaches taken by regulatory agencies across the world tend to fall into four general categories (based on the proven approach used by the regulator in the past; of course, this assertion is oversimplified, and many regulators allow alternative approaches that can be a mix of the four listed below):

1. A largely risk-based approach (e.g., the early U.K. Farmer curve [Farmer1967], and Argentina [Barón1998])
2. A risk-informed, largely non-prescriptive approach with specification of goals or results, not the process (early Canada, some early IAEA, current U.K.);
3. A risk-informed, but more prescriptive approach (current Canada, IAEA);
4. A highly prescriptive approach, with specified detailed requirements (e.g., USNRC and the U.S. Code of Federal Regulations).

Each approach has its advantages and disadvantages.

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A risk-based approach is logical, quantifies and balances risk, and leaves major design and operating decisions up to the proponent, who decides how to meet the risk goals. It thereby accommodates and even fosters innovation and evolution. On the other hand, it can lead to uncertainty in licensing because demonstration of numerical risk can be subjective or limited by lack of knowledge. It may not account sufficiently for previously unknown failures, which might be addressed more effectively by a more prescriptive defence-in-depth approach. In particular, cross-linked or common-cause failures (see Chapter 13) are difficult to identify *completely* in a PSA, so that the contribution to risk from these causes can be underestimated.

A mixed approach, which uses both risk-based and prescriptive approaches to various degrees, captures most of the benefits (and some of the disadvantages) of both extremes and generally has served Canada and the United Kingdom well. In particular, it is effective in providing defence against accidents not previously identified, without too much misallocation of resources to truly rare low-risk events.

A highly prescriptive approach in principle leads to well-defined licensing rules and more regulatory certainty and tends to produce an apparently conservative design. However, it can shift the burden of proof of safety from the designer/operator to the regulator. If the designer/operator is required mainly to follow detailed rules set by the regulator, and if the regulator misses something, part of the responsibility for safety has shifted subtly from the proponent to the regulator. Furthermore, de-emphasis of risk leads to some resource misallocation to severe but rare events, as opposed to more frequent ones with *apparently* milder consequences. We shall cover a case study of this situation (Three Mile Island) in the next section.

Much of the world, especially those countries using designs originating from U.S. light-water reactor (LWR) technology, follows the U.S. regulatory approach, largely because the regulatory approach was adopted along with the LWR technology.

4 Case Studies

This section considers a few case studies of regulatory behaviour. Where possible, the regulatory agency's self-evaluation (or that of an independent body) is used, rather than the authors' opinions. The case studies are presented not as a criticism of a particular organization, but as lessons learned about effective regulatory behaviour, as was done for design and operational safety in the case studies in Chapter 13. Some of these examples are also covered in Chapter 13, but with an emphasis on design and operation, rather than regulation.

Success is marked by the *lack* of accidents and "near misses", and just as an accident cannot be laid solely at the feet of a regulator, neither can the lack of accidents be solely ascribed to the quality of regulation. As Three Mile Island, Chernobyl and Fukushima were studied from the technical point of view in Chapter 13, we shall now look at these accidents from the regulatory point of view.

4.1 Three Mile Island

The TMI sequence of events was covered in Chapter 13. A minor event escalated through a small and unrecognized loss of coolant into a partial core melt which was contained in the vessel. Two commissions were struck to carry out an independent review.

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The President of the United States appointed a Commission headed by John G. Kemeny. Their report [Kemeny1979] was very broad. Their concentration was on human aspects: "The equipment was sufficiently good that, except for human failures, the major accident at Three Mile Island would have been a minor incident." Of interest here are their findings on the U.S. regulatory structure at the time, which are summarized as follows:

- The regulations were too voluminous and complex, required immense effort for compliance, and equated compliance with safety;
- The preoccupation with the most severe accident (the largest-break LOCA) took attention away from more likely, but slower-developing accidents, which were therefore not analyzed in depth;
- There was too much preoccupation with equipment performance rather than human performance;
- There was no requirement to look beyond the single events specified by the USNRC, for example to multiple failures;
- The role of systems classified as "non-safety-related" in causing accidents was not recognized;
- There was no systematic way of evaluating prior operating experience or looking for patterns;
- There were serious deficiencies in internal communication in the USNRC.

An additional review by the Nuclear Regulatory Commission Special Inquiry Group [Rogovin1980] covered very similar ground, albeit with more comments on the USNRC internal decision-making processes.

Subsequent to TMI, and based on the findings of the Kemeny Commission, severe accidents, emergency preparedness, plant performance monitoring, and human factors were given far more prominence on the regulatory side. Insights from PSA were used more widely, not only to reveal severe accident vulnerabilities, but also to flag requirements that attracted inordinate resources compared to the risk they mitigated. The industry formed cooperative groups such as the Institute of Nuclear Power Operations (INPO) to promote high standards of safety and reliability through plant evaluations, training and accreditation, events analysis, and information exchange and assistance.

4.2 Fukushima

The Fukushima Dai-ichi sequence of events was covered in Chapter 13. A beyond-design-basis tsunami caused a loss of all electrical power in four units, resulting in severe core damage or melting and possible containment failure in the three reactors that contained fuel.

With respect to the Japanese regulator (the Nuclear and Industrial Safety Agency (NISA)), the National Diet of Japan Nuclear Accident Independent Investigation Commission [Japan2012] concluded that "root causes were the organizational and regulatory systems that supported faulty rationales for decisions and actions", and in particular that the inadequacy of the design basis for Fukushima was known to both the utility and the regulator, but was not acted on. The regulator lacked separation from the utility. The Commission recommended formation of a new regulatory body which would be independent, transparent, professional, consolidated, and proactive.

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The Japanese Government also established the "Investigation Committee on the Accident at Fukushima Nuclear Power Stations of Tokyo Electric Power Company", an independent committee largely composed of academics. Their final report [Japan2012a], although more technical than the Diet report, raised the same issues and called for reorganization of the regulatory body. They also noted NISA's attention to short-term rather than long-term issues. In addition they focussed on NISA's performance after the event, in emergency measures off-site, and in public communication.

As of this writing (early 2014), the Japanese regulatory body has been reorganized and renamed the Nuclear Regulation Authority (NRA).

Regulators worldwide have applied the technical lessons learned, as outlined in Chapter 13. In terms of national response, in particular, the radiation threshold levels that were used for evacuation were based on the Linear No Threshold (LNT) hypothesis, which implies an unrealistically high level of risk for individual doses of radiation <100 mSv, although no effects on human populations have actually been observed at that level. Although there were no deaths due to radiation from the accident (and there are not expected to be), the evacuation itself resulted in many unnecessary deaths [Tanigawa2012].

4.3 Chernobyl

The Chernobyl sequence of events was covered in Chapter 13. A test which involved running down the main reactor coolant system (RCS) pumps while at low power ended up in a power transient due partly to the voiding of the core, but mainly to the *reverse* action of the shutdown system, destroying the reactor core and bypassing the containment.

The International Safety Advisory Group (INSAG), which provides independent advice to the IAEA Director-General, performed very extensive fuellings of the Chernobyl accident [IAEA1992]. This reference includes as an Appendix the *Report by a Commission to the USSR State Committee for the Supervision of Safety in Industry and Nuclear Power*, which covered the technical facts and the role of the regulatory structure. It noted that:

- There were known violations of the safety standards and regulations in force at the time in the design of Chernobyl Unit 4, but the design was approved and authorization given for construction by all the relevant authorities and regulatory bodies.[5]
- This deficiency resulted from a lack of a well-organized group of experts endowed with its own resources, rights, and responsibilities for its decisions;
- The USSR State Committee for the Supervision of Nuclear Power Safety could not be regarded as an independent body because it was part of the same state authority responsible for the construction of nuclear power plants and for electricity generation.
- The regulatory bodies had no legal basis, no economic methods of control, and no human and financial resources.
- The operating organization did not have ultimate responsibility and decision-making authority for safety.

5 Note added by authors: there are always outstanding regulatory issues when a plant is given a licence—what is important is that the process to determine which issues are acceptable should be an open and robust one.

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4.4 Olkiluoto

It is not just accidents than can reveal a regulatory problem. Cost and schedule overruns because of poor communication between the vendor/operator and the regulator can be very damaging to a nuclear build project. This has been exemplified by the construction of the third nuclear power plant unit at Olkiluoto in Finland. The reactor is based on the European pressurized-water reactor (EPR) and is the first of its kind to have started construction. It was meant to start operation in 2009, but as of this writing (early 2014), it is projected to come into service in 2016. This delay has been accompanied by a cost overrun of almost double the turnkey price of ~3.5 billion euros.

Because the matter is currently in legal dispute, the parties have not published much objective analysis of what went wrong. However, the Finnish regulator (STUK) did comment publicly on several occasions, listing lessons learned as follows, from [STUK2006], [Laaksonen2008], [Laaksonen2009], [Tiippana2010], [IAEA2012]:

- "too ambitious original schedule for a plant that is first of its kind and larger than any NPP built earlier;
- inadequate completion of design and engineering work prior to start of construction;
- shortage of experienced designers;
- lack of experience of parties in managing a large construction project;
- world-wide shortage of qualified equipment manufacturers".

The operator TVO, and particularly the vendor AREVA, have offered their own perspective, in which one of the causes mentioned is evolution of Finnish regulations during the construction process, which introduced additional uncertainty into the licensing process.

Of particular interest to the thrust of this chapter is the need for mutual understanding between the regulator and the operator on how the regulatory practices are to be applied:

"The licensee and the regulator need to discuss early enough on how the national safety requirements should be best presented in the call for bids.

- just making reference to national requirements and regulatory guides is not adequate to ensure that requirements are correctly understood by vendors".

With respect to the last point: In Finland, before a project is committed, the Government makes a "Decision in Principle", for part of which the regulator reviews the basic design requirements and main safety features of each proposed alternative design and gives a formal preliminary safety assessment. STUK confirmed in this assessment that "no safety issues can be foreseen that would prevent the proposed plant(s) from meeting Finnish nuclear safety regulations". Although the regulator can apply caveats to this statement in terms of changes that would be required, it is not based on a very detailed review.

5 Pre-Project Licensing Review

The risk to a project from delays associated with uncertainties in the regulatory legal process and requirements has been recognized for decades, spurred by cost and schedule overruns in nuclear plant construction in the United States in the 1970s. The solution adopted in many countries has been a form of pre-project licensing review which is done before project commitment. Typically, a vendor or utility requests a review of a proposed design against the

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regulatory requirements of the country in which it is to operate. Such a review ensures that the vendor understands the regulatory requirements and enables regulatory questions to be flushed out and answered earlier. The review is very thorough and is similar in many ways to the construction licence review, but focussing mainly on design details rather than operation. The format of such a review, and the extent to which it is binding on the regulator, depend on national legal and cultural practices. We describe briefly three examples.

In Canada, a designer can request a Pre-Licensing Vendor Design Review, which [CNSC2012]

> "…is to inform the vendor of the overall acceptability of the reactor design…

A Pre-Licensing Vendor Design Review evaluates whether:

- the vendor understands Canadian regulatory requirements and expectations;
- the design complies with, as applicable, CNSC regulatory documents RD-337[6], Design of New Nuclear Power Plants or RD-367, Design of Small Reactor Facilities, and related regulatory documents and national standards;
- a resolution plan exists for any design issues identified in the review".

The review is not binding on the CNSC (i.e., the CNSC can request repeated in-depth reviews on the same topics during construction licence review, and CNSC review results may be subject to change from pre-licensing to the construction licence review).

In the United States, Standard Design Certification is a legal process covered by Part 52 to Title 10 of the Code of Federal Regulations [USNRC2012]. The design submitted and the review are both comprehensive, although some detailed design normally performed in the project phase may be more conceptual (e.g., details of control and instrumentation design). When completed, the product is a standard reactor design certified by the Commission, independent of a specific site, through a rulemaking action (Subpart B of Part 52). This rulemaking action can certify portions of a reactor design for 15 years.

Separately, a licensee can apply for an *early site permit* for approval of a site for one or more nuclear power facilities. This is also separate from the filing of an application for a construction permit or a *combined licence* (see below) for the facility. A specific design is not required for an early site permit, but the design eventually chosen must fit within the safety and environmental parameter envelope requested and specified in the early site permit.

As a third option, a licensee can request a combined licence, which is a combined construction permit and operating licence for a nuclear power facility.

A licensee can use any combination of these tools. For example, a licensee can obtain an early site permit based on the environmental and safety performance envelope of a set of designs; can obtain certification of a specific design that meets the envelope; and can then ask for a combined construction and operating licence for the certified design on the approved site. Standard design certification and early site permits are legally binding, subject to conditions (specified during the certification process) to be met by the detailed project design.

In the United Kingdom, the Office for Nuclear Regulation (ONR) and the Environment Agency

6 Note that the CNSC document titles are in the process of changing; this document has been renamed and revised as REGDOC 2.5.2. Other CNSC regulatory documents are being revised and renamed as REGDOCs.

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(EA) offer a *generic design assessment* (GDA) of new reactors. The purpose is [HSE2008] an "assessment of the safety case for a generic design, leading to issue of a design acceptance confirmation if the outcome is positive". It consists of four phases: the first is preparation of the safety case by the proponent, and the next three are reviews by ONR and EA at increasing levels of detail, namely: 1) fundamental safety overview; 2) overall design safety review; and 3) detailed design assessment. A design acceptance confirmation is then issued, subject to possible conditions and exceptions, if the design is considered acceptable. Once completed, a GDA ensures that within the scope of the design, ONR review results are binding. ONR may extend the review during the construction licence review, but the results of the prior GDA remain binding.

6 International and National Examples

6.1 International Organizations

A number of international bodies have a quasi-regulatory role. Regulation is still a *national* legal responsibility, and therefore these organizations do not have formal authority. However, they influence governments, national bodies, and nuclear organizations, and if a country formally adopts their guidance, it takes on the force of national law or regulation.

As before, we give an overview, and the reader may find out more in the references.

6.1.1 IAEA

The origins and purposes of the International Atomic Energy Agency (IAEA) are described in Chapter 13, to which the reader is referred.

Its current document structure with respect to nuclear safety and regulation includes a hierarchy of Principles, Standards, and Guides, as shown in Figure 2, from [IAEA2012a]. Although these have no legal or regulatory force unless they are formally adopted on a national level, they are seen as setting minimum international requirements for safety and regulation. The structure and nomenclature in Figure 2 are self-explanatory, with Safety Fundamentals and Safety Requirements being mandatory (if adopted) and Safety Guides being recommendations and guidance on how to comply with the Requirements. As of this writing (early 2014), they have been formally adopted in countries such as China and the Netherlands; are used directly to establish national requirements in Canada, the Czech Republic, Germany, India, Korea, and the Russian Federation; are used as a reference for review of national standards by most nations; and are also used as inputs by international organizations such as the Western European Nuclear Regulators' Association (WENRA) (Section 6.1.2).

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Figure 2 Hierarchy of IAEA safety documents

Of particular relevance to this Chapter, IAEA report GS-R-1 [IAEA2000] deals, among other things, with overall regulatory structure and organization. It is supported by a number of more detailed guides.

Table 1 IAEA guides related to regulation

Report Number	Title
SSG-16	Establishing the Safety Infrastructure for a Nuclear Power Programme (2012)
GS-G-1.1	Organization and Staffing of the Regulatory Body for Nuclear Facilities (2002)
GS-G-1.2	Review and Assessment of Nuclear Facilities by the Regulatory Body (2002)
GS-G-1.3	Regulatory Inspection of Nuclear Facilities and Enforcement by the Regulatory
GS-G-1.4	Documentation for Use in Regulating Nuclear Facilities (2002)
GS-G-1.5	Regulatory Control of Radiation Sources (2004)
GS-G-3.5	The Management System for Nuclear Installations (2009)
SSG-12	Licensing Process for Nuclear Installations (2010)
SSG-16	Establishing the Safety Infrastructure for a Nuclear Power Programme
GS-G-4.1	Format and Content of the Safety Analysis Report for Nuclear Power Plants (2004)

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Some of the key principles reflected in these Requirements and Guides are:

- a regulator's purpose is the verification and assessment of safety in compliance with regulatory requirements;
- it has sufficient resources to perform its job;
- it is effectively independent from other organizations that could influence its decisions;
- it has qualified and competent staff;
- its management system is aligned with its safety goals;
- it may solicit expert advice, but does not delegate decisions;
- it communicates professionally with interested parties and with the public;
- regulatory control is stable and consistent;
- it authorizes all facilities unless specifically exempt;
- authorization is supported by a demonstration of safety submitted by the applicant;
- regulatory review occurs before authorization and over the facility lifetime;
- review is commensurate with risks;
- it inspects facilities;
- it has legal enforcement powers and can require corrective action;
- it issues, reviews, and promotes regulations and guides;
- it maintains records;
- it promotes a safety culture (see Chapter 13);
- it ensures that emergency preparedness arrangements and emergency plans are in place.

Finally, there is a growing international consensus on high-level numerical safety goals. Safety goals are defined and discussed in Chapter 13 (Section 4), and the reader is referred there for details. Almost all regulators who specify safety goals do so implicitly or explicitly on the following basis: the goals are set so that the predicted health effects of normal operation and accidents in nuclear power plants are small compared to other social risks. Although IAEA itself has not specified numerical safety goals, its senior advisory body (INSAG, in [IAEA1999]) has proposed the following targets:

- For existing nuclear power plants:
  - Frequency of occurrence of severe core damage < 10^{-4} events per plant operating year;
  - Probability of large off-site releases requiring short term off-site response < 10^{-5} events per plant operating year.
- For future nuclear power plants:
  - Frequency of occurrence of severe core damage < 10^{-5} events per plant operating year;
  - Practical elimination[7] of accident sequences that could lead to large early radioactive releases.

These targets have been used by many other international and national bodies.

7 i.e., if it is physically impossible for the conditions to occur or if the conditions can be considered with a high degree of confidence to be extremely unlikely to arise [IAEA, 2004a]

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Note also that the frequency of the design basis earthquake (DBE) has recently been changed in most jurisdictions from 10^{-3} / year to 10^{-4} / year, meaning that the DBE that new plants are required to withstand is significantly more severe.

6.1.2 MDEP and WENRA

There has been an impetus for several years to "harmonize" nuclear regulations, much as aircraft regulations are common internationally. The ideal is that a reactor design licensed in one competent jurisdiction would be licensable anywhere because the requirements would be the same. In practice, licensing requirements are closely linked to design type, and therefore harmonization has made slow progress because of the wide variety of reactor types requiring different regulations at the detail level.

An initiative called the "Multinational Design Evaluation Programme" (MDEP) was established in 2006 to:

- enhance multilateral co-operation within existing regulatory frameworks;
- encourage multinational convergence of codes, standards, and safety goals;
- facilitate the licensing of new reactors.

It currently includes nuclear regulatory authorities from 13 countries. Its work includes co-operation on safety reviews of specific reactor designs. National regulators retain sovereign authority for all licensing and regulatory decisions.

The Western European Nuclear Regulators' Association (WENRA) was formed in 1999 to develop a common approach to nuclear safety and to be able to review nuclear safety independently in countries applying for membership in the European Union (EU).

One of their earliest reports [WENRA2000] surveyed the state of nuclear safety and regulation in countries applying for EU membership. The report covers the status of the regulatory regime and regulatory body and of nuclear power plant safety. For the former, issues such as independence of the regulatory body, resources, and expertise were priority topics.

Because the predominant type of reactor in Western Europe is the LWR, it is somewhat easier to propose harmonized requirements. WENRA has adopted common top-level requirements for any reactor to be built in Western Europe [WENRA2010]. These are largely consistent with the IAEA Safety Fundamentals [IAEA2006a]. More detailed requirements are given in the WENRA "Reference Levels" publications, e.g., [WENRA2007], for reactor safety.

Finally, as requested by the European Union (EU) after the accident at Fukushima Dai-ichi in March 2011, WENRA proposed [WENRA2011] that a series of analytical "stress tests" be done on all EU nuclear power plants to show their robustness against extreme natural events. Unlike the traditional approach of analyzing design basis accidents (DBAs) and beyond design basis accidents (BDBAs) (see Chapter 13), WENRA proposed an analysis assuming:

> "sequential loss of the lines of defence in a deterministic approach, irrespective of the probability of this loss. In addition, measures to manage these situations will be supposed to be progressively defeated."

In other words, design and accident management measures were to be assumed to fail one after the other and the consequences at each step tabulated to understand the plant's robustness. The initiating events conceivable at the plant site included earthquake, flooding and other

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extreme natural events. The consequential loss of safety functions included loss of electrical power, including station black-out, loss of the ultimate heat sink, and a combination of both. Each EU country (or country applying for EU membership) completed these stress tests for its own plants and reported the results back to the European Nuclear Safety Regulators' Group (ENSREG), an independent expert body formed by the European Commission, for evaluation.

6.2 National Regulatory Organizations

Although each country operating a nuclear power plant has a national regulatory agency, the overall approach taken generally follows that used in the countries which have designed and deployed the major types of reactors: Canada for pressurized heavy-water reactors (PHWRs), the United States for LWRs, and the United Kingdom for gas-cooled reactors. In this section, we give a brief description of the regulatory approach in the latter two examples; Canada is covered later and in more detail in Sections 9 and 10.

It is impossible in one section of one chapter to cover the full breadth of regulatory philosophy and practices in several jurisdictions. Rather, we focus on the overall philosophy rather than the mechanics of licensing and give references so that the interested reader can find more detail.

6.2.1 USNRC

6.2.1.1 Background

In the United States, the Atomic Energy Act of 1946 established the Atomic Energy Commission (AEC), transferring atomic energy from military to civilian control (see [Buck1983] for a history). Its early years were devoted to nuclear weapons production and research, including the military submarine program, whose reactors became the inspiration for the U.S. PWR commercial power reactor design. The revised Atomic Energy Act of 1954 gave the Atomic Energy Commission the responsibility for regulating and licensing commercial atomic activities. Research, military use, regulation, licensing, and development of the nuclear power industry were all under the same organization, and although it set up separate divisions, the conflicts of interest became more and more apparent, until in 1974 the regulatory functions were transferred to the newly-formed U.S. Nuclear Regulatory Commission (USNRC), which began operations on January 19, 1975.

6.2.1.2 Organization

The USNRC has had enormous influence on regulatory practice in the world because the wide-spread use of LWR technology was accompanied by the LWR regulatory philosophy. Indeed, because almost all the reactors licensed in the United States have been PWRs or BWRs[8], the regulatory requirements tend to be specific to these designs.

The USNRC is headed by a five-member Commission appointed by the U.S. President for staggered five-year terms. The President designates one member to serve as Chairman. The Commission formulates policies and regulations governing nuclear reactor and materials safety, issues orders to licensees, and adjudicates legal matters brought before it. The Executive

8 The United States have also licensed a fast-breeder power reactor (Fermi 1) and high-temperature gas-cooled reactors (Peach Bottom and Fort St. Vrain).

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Director for Operations (EDO) carries out the policies and decisions of the Commission and directs the activities of the program offices.

The Commission is supported by a large technical staff, as well as powerful advisory committees who can challenge both the proponent and the staff, e.g., the Advisory Committee on Reactor Safeguards, or ACRS. Typically, the ACRS consists of well-known and respected scientists and experts, often world-renowned professors drawn from universities. It has its own independent support staff. There is no current equivalent in Canada, although at one point the Reactor Safety Advisory Committee (RSAC), which later became the Advisory Committee on Nuclear Safety (ACNS), had a similar role with the then-AECB, albeit on a smaller scale.

The actual trial-level adjudicatory body of the USNRC is the Atomic Safety and Licensing Board Panel (ASLB). The Panel is composed of administrative judges who are lawyers, engineers, and scientists, and administrative law judges who are lawyers. It is chaired by the Chief Administrative Judge. The Panel conducts all licensing and other hearings as directed by the Commission, primarily through individual Atomic Safety and Licensing Boards or single presiding officers appointed by either the Commission or the Chief Administrative Judge. Individual licensing boards conduct public hearings on contested issues that arise in the course of licensing and enforcement proceedings and uncontested hearings on construction of nuclear facilities.

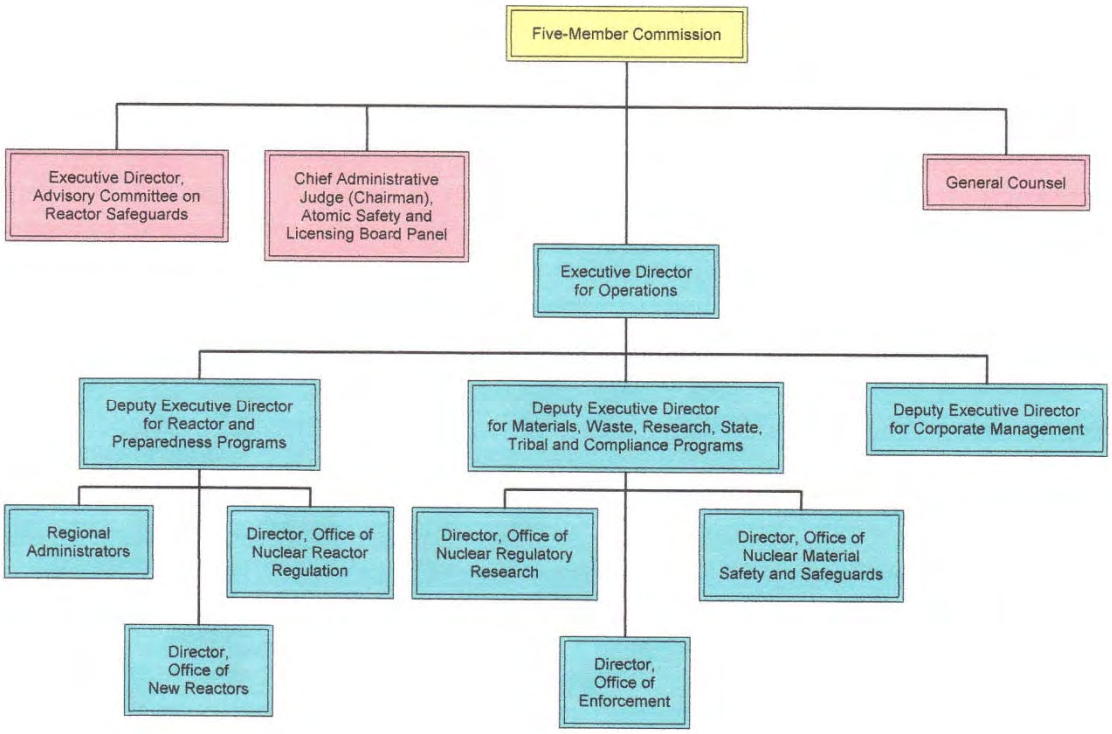

Figure 3 Simplified organization chart of the USNRC

Figure 3 shows a highly simplified organization chart of the USNRC. The five-member Commission (yellow) is supported by the ACRS, the ASLB, and legal counsel, shown in red. The large technical staff (turquoise) is headed by the Executive Director for Operations. For nuclear regulatory purposes, the United States is divided into four regions; the four regional offices conduct inspection, enforcement, and emergency response programs for licensees within their

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borders. USNRC also has a large and powerful R&D Division which conducts experimental and analytical work of very broad scope, along with development of their own computer tools, independently of utilities and design organizations.

6.2.1.3 Regulatory approach and documents

As we saw with the IAEA and shall see with other regulators, regulatory documents form a hierarchy, with legislation at the top, government regulations and licences next (both of which are enforceable by law), mandatory regulatory requirements set by the regulator next, then optional regulatory guides provided by the regulator, and finally guidance to regulatory staff themselves on how to review a submission. Figure 4 shows such a generic structure.

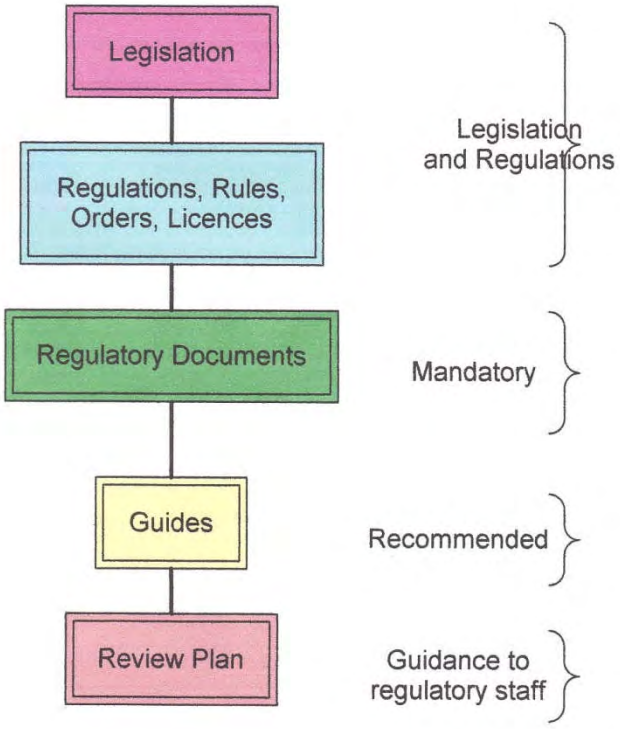

Figure 4 Generic hierarchy of regulatory documents

6.2.1.3.1 Consolidated legislation

In the United States, nuclear energy-related activities are governed by the Atomic Energy Act of 1954, as amended, and the Energy Reorganization Act of 1974, as amended (which created the USNRC). The interested reader is referred to [USNRC2011] as updated, which consolidates all nuclear-related legislation and executive orders and runs well over 1000 pages.

The Nuclear Safety and Control Act in Canada [GovCan2013], by contrast, has 54 pages. [Ahearne1988] is an excellent source of information on the origins of the U.S. regulatory approach and its differences from that used in Canada. Although the work is dated, it is recommended reading to gain an understanding of the differences. One interesting difference in practice is the length of a plant licence; typically, in the United States, it is up to 40 years, whereas in Canada the utility must reapply about every five years.

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6.2.1.3.2 Code of Federal Regulations

The technical requirements are covered in the Code of Federal Regulations Title 10 (10CFR), which corresponds to the second box from the top in Figure 4. The requirements are binding on all persons and organizations who receive a licence from the NRC to use nuclear materials or to operate nuclear facilities. Of particular interest are 10CFR Part 50 [USNRC2012d] and 10CFR Part 52 [USNRC2012].

10CFR Part 50 covers requirements for the traditional domestic licensing of production and nuclear power facilities. Much of the main text is legal and procedural, and these aspects will not be discussed further here. Even a detailed technical discussion of 10CFR Part 50 is well beyond the scope of this chapter. We select a few examples to give an idea of the approach.

Section 50.46 states acceptance criteria for emergency core cooling systems for light-water nuclear power reactors. The requirements are in the form of numerical analysis values rather than safety objectives. For example, it is an overall requirement in water reactors to prevent excessive oxidation of the sheath because if it becomes brittle, it can fragment on rewet, and the debris can block fuel cooling. Moreover, at high temperatures, the Zircaloy-water reaction becomes autocatalytic, so that temperature prediction is difficult. Finally the reaction produces hydrogen, which is a source of energy to containment if it burns. In Canada, these requirements are stated as objectives, and it is up to the proponent to justify how they are met. However, in 10CFR Part 50, the acceptance criteria can be quite explicit (from 50.46):

> "(b)(1) Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200° F.
> (2) Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph, total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred, but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a prediction of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding, the circumference does not include the rupture opening.
> (3) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react."

Although this example illustrates a difference in regulatory philosophy (and not all criteria are this detailed), it is also related to design differences. In PWRs, if the metal-water reaction becomes autocatalytic, there is no nearby heat sink to mitigate it, and it can quickly affect a

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large part of the core. In CANDU (as discussed in Chapter 13) the nearby moderator can remove heat from the fuel through the pressure tube and has the effect of tempering the metal-water reaction rate.

As another example, consider Appendix A to 10CFR 50, the General Design Criteria [USNRC2012b]. This Appendix consists of 64 basic requirements which underpin much of the technical approach to safety in the United States. Compliance is mandatory. Although high-level, many are focussed on LWRs, as stated in the introduction:

> "These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units."

Many of the GDCs *are* generic; for example

> *"Criterion 16—Containment Design*. Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

Some are clearly specific to LWRs, for example:

> *"Criterion 11—Reactor Inherent Protection*. The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity."

The underlying purpose of Criterion 11 is that a fast reactivity excursion in an LWR, caused for example by a rod ejection, must be compensated for (not necessarily terminated, as discussed in Chapter 13) by negative (Doppler) feedback since (because of the short prompt neutron lifetime) engineered shutdown systems may not be able to act quickly enough (see Chapter 13). It is often misinterpreted as a blanket requirement for a negative void reactivity.

At the same level in Figure 4 are plant licences, rules (U.S. term for regulations), and orders.

6.2.1.3.3 Commission Policy Statements

Commission Policy Statements are an example of Regulatory Documents (third box down in Figure 4). For example, in 2011, the Commission issued a Policy Statement on Safety Culture, setting forth its expectations for industry to promote a positive safety culture. Such notices are published in the U.S. Federal Register.

Given the detail in 10CFR, USNRC staff have less need to issue *mandatory* regulatory documents.

6.2.1.3.4 Regulatory guides

Regulatory guides (fourth box from the top in Figure 4) provide guidance to proponents on implementing USNRC regulations, on techniques used by NRC staff in evaluating specific prob-

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lems or postulated accidents, and on data needed by staff in its review of applications for permits or licences. Although nominally optional, exceptions taken to regulatory guides in the United States (and most other countries) require thorough justification and significant extra regulatory review.

6.2.1.3.5 Standard review plan

Many regulators now have a standard review plan (SRP, bottom box in Figure 4) written by the regulator and aimed at regulatory staff performing reviews of proponent applications. This approach was first developed in the United States and is now being adopted in other countries, including Canada. The purpose of the SRP is to ensure the quality and uniformity of staff safety reviews and to make the review process more transparent. For this reason, the SRP is public in the United States, and proponents can see in advance the criteria against which their submissions will be reviewed and the level of information needed; see [USNRC1987].

6.2.1.4 Licensing process

It is beyond the scope of this chapter to describe the U.S. licensing process in detail. The following description is highly abbreviated; the reader is referred to [USNRC2009] for more detail.

We have already described in Section 5 aspects of an alternative licensing approach in the United States governed by 10 CFR Part 52 that includes options for:

- Early site permits that enable an applicant to obtain approval for a reactor site without specifying the design of the reactor(s) that could be built there;
- Certified standard plant designs which can be used as pre-approved designs;
- A licence which combines a construction permit and an operating licence with conditions for plant operation.

However, at the time this chapter was written (early 2014), all operating plants in the United States have been licensed under a two-step process defined in 10CFR50:

- A construction licence for a particular plant design at a particular site. This process includes submission of relevant information on design, safety analysis, intended operation, the site, the environment, and emergency planning; review by NRC staff, who issue a Safety Evaluation Report summarizing the anticipated effect of the proposed facility on public health and safety; and a public hearing, conducted by the Atomic Safety and Licensing Board (see Section 6.2.1.2).
- An operating licence. This process includes submission of final design and safety information by the applicant and review by the USNRC, who prepare a Final Safety Evaluation Report. A public hearing is not mandatory or automatic, but can be requested by members of the public whose interests might be affected by the issuance of the licence.

In parallel, the licence applications are reviewed by the ACRS (see Section 6.2.1.2), who provide the Commission with independent expert advice.

The entire process is remarkably open, with all documents and correspondence related to the

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application placed in publicly accessible paper and electronic repositories[9]. This includes the plant Safety Report, the NRC Safety Evaluation Reports, letters between the NRC and the applicant, records of meetings, and other documents. The NRC also holds numerous public meetings during the licensing process.

In addition, conditions may be placed on the licence, which have the same legal force as the licence itself. A generic set of conditions is described in 10CFR § 50.54, "Conditions of licenses". Many areas are covered, e.g., quality assurance, minimum staffing levels, licence amendment or revocation, responsibility of the operator, process for making changes, security, and emergency planning.

Finally, the USNRC has the authority to impose penalties for violations of regulatory requirements. It uses three primary enforcement sanctions:

1. Notice of Violation: This identifies a requirement and how it was violated and normally requires a written response.
2. Civil Penalties: A civil penalty is a monetary fine up to $130,000 per violation per day.
3. Orders: Orders modify, suspend, or revoke licenses or require specific actions by licensees or persons.

6.2.2 United Kingdom

Unlike the United States, the United Kingdom has a very different legal structure, and the regulator has had to license a wide variety of reactor types (gas-cooled reactors, fast reactors, heavy-water reactors, and light-water reactors). This task has been accompanied by a less prescriptive approach.

6.2.2.1 Organization

The Office for Nuclear Regulation (ONR) is responsible for all nuclear regulation in the United Kingdom. It was formed on April 1, 2011, as an agency of the Health and Safety Executive (HSE). ONR combined the safety and security functions of HSE's former Nuclear Directorate, including Civil Nuclear Security, the U.K. Safeguards Office, and the Radioactive Materials Transport team. In turn, HSE is a non-departmental public body (NDPB), which is defined as a "body which has a role in the processes of national Government, but is not a Government Department or part of one, and which accordingly operates to a greater or lesser extent at arm's length from Ministers" [UK2009]. It is sponsored by the Department for Work and Pensions. At the top level, it has a Board, which also contains its Executive Management Team. The work is organized along program lines, as shown in simplified form in Figure 5.

On April 1, 2014, ONR became a Public Corporation under the 2013 Energy Act. ONR is now the enforcing authority for licensed sites in Great Britain, as well as the licensing authority.

9 Some records are not disclosed, e.g., records which must be kept secret in the interest of national defence or foreign policy; internal personnel records; trade secrets and commercial or financial information; records compiled for law-enforcement purposes. For a comprehensive list, see 10CFR § 9.17, "Agency records exempt from public disclosure".

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6.2.2.2 Regulatory philosophy

One of the underpinnings of U.K. nuclear safety philosophy is the concept of "so far as is reasonably practicable", or SFAIRP. This was introduced in the Health and Safety at Work Act of 1974 [UK1974], Section 2-(1), and applies to all industry, not just nuclear:

> 2.-(1) It shall be the duty of every employer to ensure, so far as is reasonably practicable, the health, safety, and welfare at work of all his employees."

For this concept, HSE uses the term As Low as Reasonably Practicable (ALARP) and states that [HSE2006]:

> "for assessment purposes, the terms ALARP and SFAIRP are interchangeable and require the same tests to be applied. ALARP is also equivalent to the phrase 'as low as reasonably achievable' (ALARA) used by other bodies nationally and internationally".

ALARA is discussed in Chapter 12. However, the implementation and emphasis in regulatory practice of ALARA and ALARP can be very different. ALARP places the onus on the nuclear-plant designer and operator to justify why further improvements to all aspects of safety cannot be practicably achieved, and in the authors' experience goes considerably beyond ALARA in its scope.

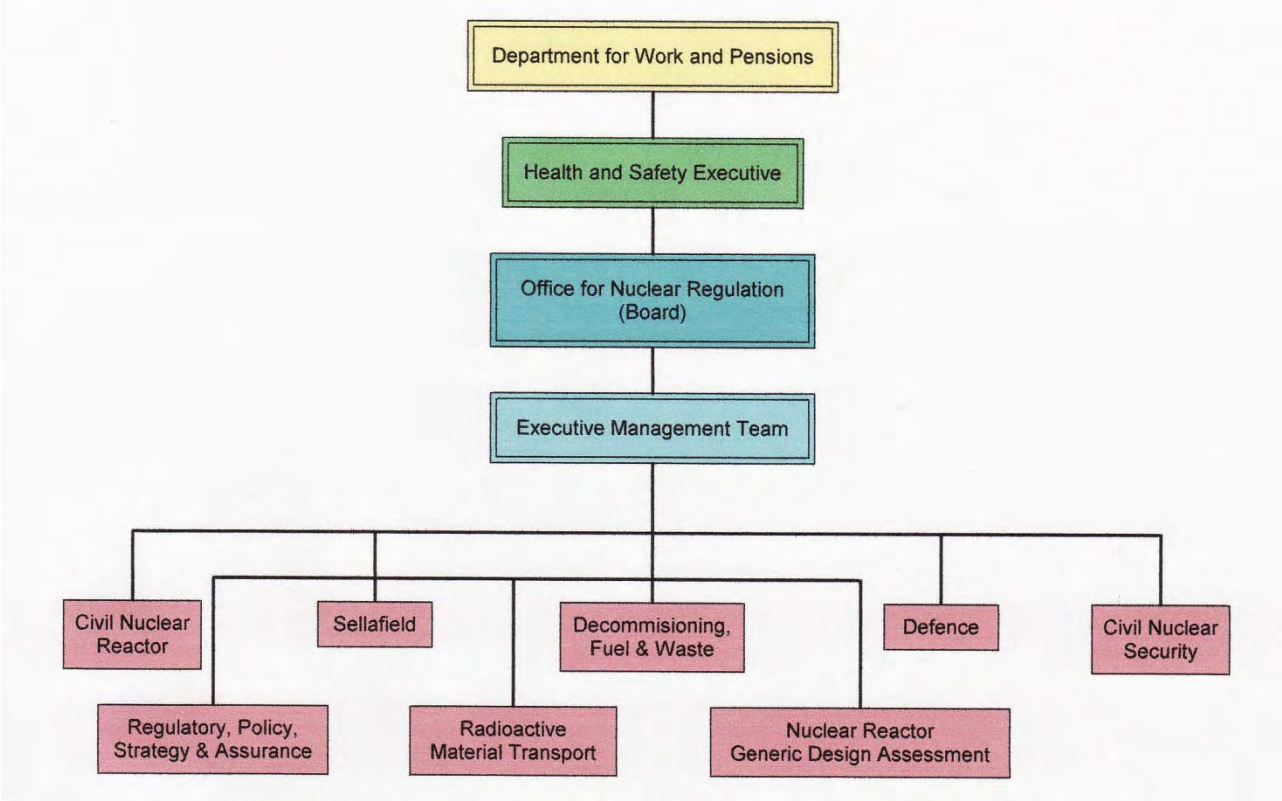

Figure 5 Structure of the ONR

A second underpinning is the concept of acceptable risk. In 1967, Farmer [Farmer1967] proposed a risk-based criterion for use in siting a nuclear reactor. Figure 6 shows his proposed iodine-131 release curve in an accident, versus reactor-years between releases (i.e., the inverse of accident frequency). The basis of the curve was the predicted health effects of an accident, which (if the criterion was met) would be small compared to other social risks. The flattening at the top was to minimize the frequency of small releases due to their nuisance value.

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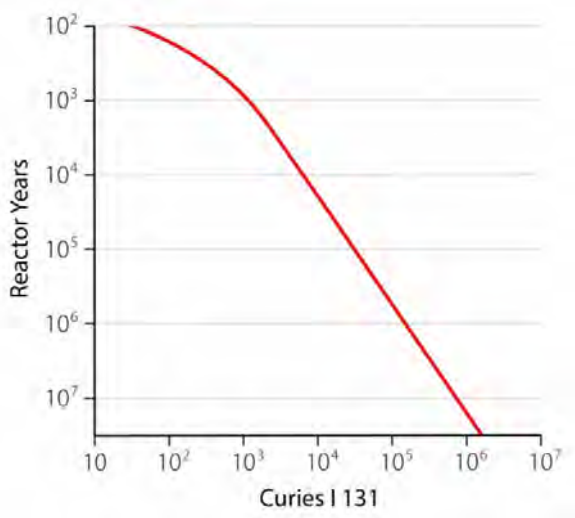

Figure 6 "Farmer" curve

6.2.2.3 Tolerability of risk

This approach developed in the United Kingdom into a risk-based approach along with high-level requirements. These are described in three key documents:

1. The Tolerability of Risk from Nuclear Power Stations [HSE1992];
2. Reducing Risks, Protecting People [HSE2001];
3. Safety Assessment Principles (SAPs) [HSE2006].

The concept evolved from a go/no-go risk line as proposed by Farmer to three distinct regions as shown in

Figure 7. At the top, the risk of the activity is too high regardless of the benefits. At the bottom, the risks are broadly acceptable, and no further reduction is usually required unless reasonably practicable. The middle region of tolerable risk is typical of the risks from activities that people are prepared to tolerate to secure benefits. ALARP would be applied to outcomes in this region.

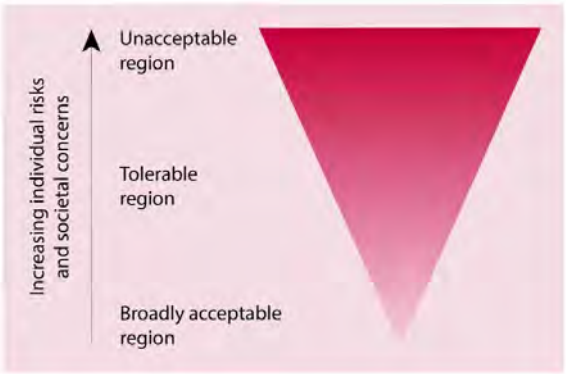

Figure 7 HSE concept of tolerable risk

The risk boundaries are set as follows:

Boundary between "tolerable" and "unacceptable" regions for risk entailing fatality:
Worker: 1 in 1000 /y

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Member of the public: 1 in 10,000 /y
This is called the Basic Safety Limit (BSL).
Boundary between "broadly acceptable" and "tolerable" regions for risk entailing fatality:
Worker: 1 in 1,000,000 /y
Member of the public: 1 in 1,000,000 /y
This is called the Basic Safety Objective (BSO).

ALARP is applied in this region between the BSL and the BSO to see if it is reasonably practicable to reduce the risks below the BSO.

These risk targets are applied to all industries in the United Kingdom. For nuclear power, they are translated in the SAPs to BSL/BSOs for normal operation and accidents. The latter become numerical acceptance criteria and targets for the plant PSA. For example, for off-site doses, Table 2 gives typical frequency/dose targets.

Table 2 U.K. frequency / dose targets for accidents off-site

Frequency dose targets for accidents on an individual facility – any person off the site	**Target 8**

The targets for the total predicted frequencies of accidents on an individual facility, which could give doses to a person off the site, are:

Effective dose, mSv	**Total predicted frequency per annum**	
	BSL	**BSO**
0.1 – 1	1	1×10^{-2}
1 – 10	1×10^{-1}	1×10^{-3}
10 – 100	1×10^{-2}	1×10^{-4}
100 – 1000	1×10^{-3}	1×10^{-5}
> 1000	1×10^{-4}	1×10^{-6}

The reader should consult [HSE2006] for the full set of numerical targets.

6.2.2.4 Safety assessment principles

The SAPs consist of a set of high-level principles, followed by a large number of technical principles covering the plant life cycle. Although they are expected to be followed by a proponent, they are not legally mandatory (unless they refer to a legal requirement), and are officially written to guide regulatory staff in their assessment of a safety case. They are generally consistent with IAEA Safety Requirements. Most SAPs are followed by a brief explanation or commentary.

It is outside the scope of this chapter to cover these principles in depth (SAPs consisted of 140 pages as of 2006). Appendix A (Section 12) shows some examples of SAPs.

6.2.2.5 Technical assessment guides

The SAPs are supplemented by a series of Technical Assessment Guides (TAGs), which provide more detailed guidance to regulatory staff and fill a similar role to the U.S. Standard Review Plan. Like the latter, they are public.

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6.2.2.6 Nuclear site licence conditions

Finally each licensed plant has a series of 36 Standard Conditions attached to its licence, covering design, construction, operation, and decommissioning [ONR2011]. For example, a number of conditions cover changes to the plant during construction and operation and the need for HSE approvals. Because they are attached to the licence, they give HSE a powerful compliance tool.

7 Nuclear Material Safeguards

IAEA and CNSC material has been used throughout this section and referred to specific documents where appropriate. *Italic* fonts have been used to indicate where statements from IAEA or CNSC documents have been used essentially verbatim.

Previous sections have focussed on the "why" of regulatory approaches and have been relatively high-level. The remainder of this chapter provides more detail on safeguards and security and on the Canadian regulatory approach. It is intended to be a guide to the practitioner in finding and understanding relevant requirements, and for Canada, it provides a concise and unique regulatory roadmap. Because regulatory requirements can affect design and operation in a significant way, regulators and the IAEA spend much time making each sentence as precise as possible, much as one does in drafting legislation. For that reason, we have avoided paraphrasing regulatory requirements as much as practicable, instead quoting directly from the relevant documents.

Nuclear material safeguards are enforced around the world by the International Atomic Energy Agency (IAEA). IAEA safeguards are measures through which the IAEA seeks to verify that nuclear material is not diverted from peaceful uses. States that have signed an agreement with the IAEA (which are then considered to be nations committed to the peaceful use of nuclear energy) accept the application of safeguards measures through the conclusion of safeguards agreements with the IAEA [IAEA2007]. There are various types of safeguards agreements, but the vast majority of countries have agreed not to produce or acquire nuclear weapons and to place all their nuclear material and activities under safeguards to enable the IAEA experts to verify that they are complying with the agreements. The origins of the IAEA and of safeguards are discussed in Chapter 13.

7.1 Objectives and Evolution

IAEA safeguards implementation started more than 50 years ago and has been subject to continuous evolution and improvement [IAEA1998], mainly from lessons learned over time.

Safeguards have gone through three major phases [IAEA1998], [IAEA2002d], [IAEA2007b]. The first phase began in the late 1950s and 1960s as nations started to trade in nuclear plants and fuel. The safeguards at that time were designed mainly to ensure that this trade did not lead to the spread of nuclear weapons. The second phase, from 1971 to 1991, reflected a growing perception that, "pending nuclear disarmament, world security is better served with fewer rather than more nuclear weapons and nuclear weapon states". The third and most recent phase, from 1991 to 1997, consisted of a far-reaching review designed to remedy shortcomings in the 1971 system. The review culminated in approval of a significantly expanded legal basis of IAEA safeguards. Figure 8 shows highlights of safeguards evolution from 1991 to the present time.

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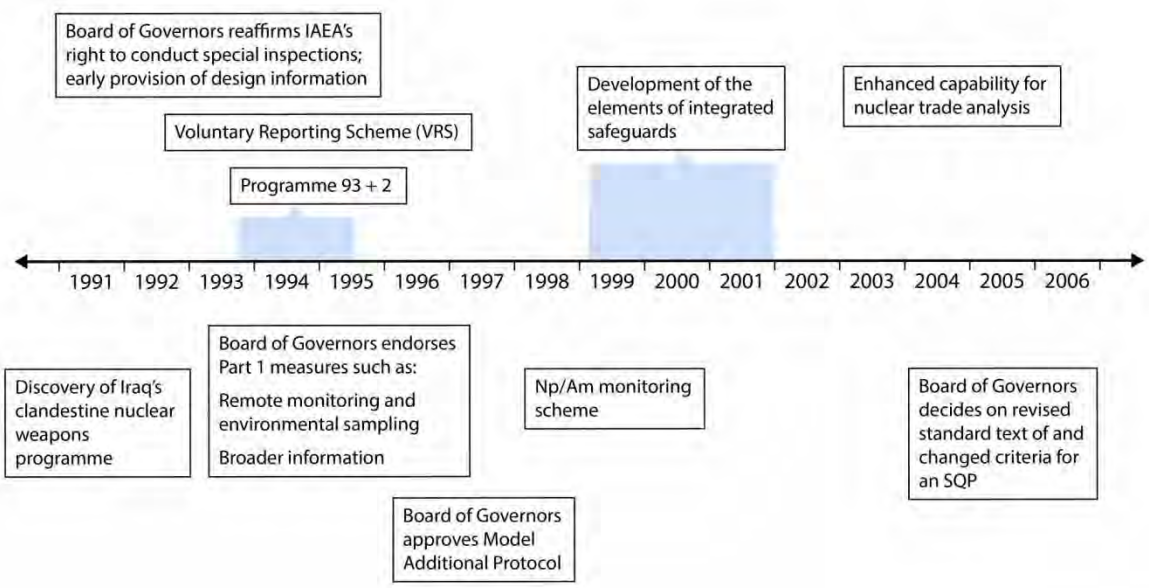

Figure 8 Safeguards state evaluation is a continuous process [IAEA2007b]

One should distinguish between the aims that safeguards seek to achieve in the non-nuclear-weapon and in the declared nuclear-weapon states. The safeguards objectives in the non-nuclear-weapon states are:

- To provide assurance that non-nuclear-weapon states party to comprehensive safeguards agreements are complying with their obligations not to acquire nuclear weapons or other nuclear explosive devices;
- To deter states that have renounced nuclear weapons from acquiring unsafeguarded nuclear material[10]. Safeguards must therefore be effective to detect within a reasonable time the diversion or clandestine production of nuclear material;
- To verify certain facilities and material in some non-nuclear-weapon states with significantly developed nuclear programs, but which have not yet concluded comprehensive safeguard agreements, and therefore are at high risk of proliferation activities. These states are placed under safeguards to prevent them from acquiring nuclear-weapon capability.

The objectives of safeguards related to the formally recognized nuclear-weapon states are:

- To commit not to use nuclear energy to further any military purpose;
- To refrain from any explosive use of nuclear energy;
- To discourage any proliferation of nuclear material that is potentially useful for developing nuclear weapons;
- To encourage widespread acceptance and implementation of safeguards.

10 Note that "nuclear material" includes all fissionable isotopes that can be used as nuclear fuel or explosive material, whereas "non-nuclear material" includes all material of strategic importance, such as heavy water, zirconium tubes, etc.

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The technical aim of safeguards is to use technical means to detect and verify safeguards implementation and to provide for reporting of deviations.

For further background, see [IAEA1983], [IAEA1998a], [IAEA2011a], [IAEA2011b], [IAEA2007b].

7.2 Safeguards Treaty Detection Fundamentals

The Treaty on the Non-Proliferation of Nuclear Weapons (the Non-Proliferation Treaty, or NPT) is a cornerstone of the nuclear non-proliferation regime and the basis for implementation of safeguards. It was opened for signature in 1968 and entered into force in 1970. By December 2001, the NPT was in force in 187 states. In 1995, the NPT was extended indefinitely. For details, see [IAEA2002d].

The key requirements of the NPT are:

- Not to receive and transfer any nuclear weapons or nuclear explosive devices, or control of such weapons or devices directly or indirectly;
- Not to manufacture or otherwise acquire such weapons or devices;
- Not to seek or receive any assistance in development or manufacture of such devices;
- To accept IAEA safeguards on all fissionable material in peaceful nuclear activities;
- Not to provide source or special fissionable material, or equipment or material especially designed or prepared for the processing, use or production of special fissionable material to any non-nuclear-weapon State;
- To sign safeguard agreements with the IAEA either individually or together with other States within 18 months of the date on which the State deposits its instruments of ratification of or accession to the NPT Treaty
- Allows each State party to the NPT Treaty to develop research, production, and use of nuclear energy for peaceful purposes and to facilitate and participate in the fullest possible exchange of equipment, materials, and information for the peaceful uses of nuclear energy.

7.3 Safeguard Approaches, Concepts, and Measures

The key requirement for a successful safeguards program is to collect sufficient information from various sources [IAEA2007]. The main sources of safeguards information are:

1. State-supplied information (under the obligation of the safeguards treaties and agreements)
    a. Nuclear material accountancy and facility design
    b. Voluntary reported information
    c. Additional protocol declarations
2. Key results of verification activities
    a. Inspection
    b. Design information verification (DIV)
3. Other information coming from open and other sources

In the first and second phases of the safeguard evolution process (until the early 1990s) the focus of the safeguards program was limited to the Comprehensive Safeguards Agreement protocols, which included:

    a. Fuel enrichment facilities

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b. Fuel fabrication facilities
c. Nuclear reactors
d. Spent fuel storage
e. Reprocessing facilities.

The Generic safeguards approach specifies the IAEA goals and safeguard activities for a reference plant. The Facility safeguards approach is prepared for a specific facility, developed and adjusted to specific facility characteristic features and activities. The State-level safeguards approach is developed for a specific state, encompassing all nuclear material, nuclear installations, and nuclear activities.

The integrated safeguards approach, which is a variation of the State-level safeguards approach, was developed as an optimum combination of all safeguards measures available to the IAEA under comprehensive safeguard agreements and additional protocols. Starting in the early 1990s, the safeguards program was extended, and is known today as Broad Overview of Expanded Safeguards Coverage Protocols, which in addition to the above targets, include the following:

a. Equipment manufacturers
b. Mining and milling
c. Conversion facilities
d. R&D centres.

The nuclear material accountancy within the framework of IAEA safeguards begins with the nuclear-material accounting activities by facility operators and the state system of accounting for and control of nuclear materials. *The IAEA implements nuclear accountancy at various levels [IAEA2002d]:*

1) At the Facility Level by:
   a) Dividing operations into nuclear material balance areas (MBAs);
   b) Maintaining records on nuclear material quantities in each MBA;
   c) Measuring and recording all transfers of nuclear material from each MBA to another area or facility;
   d) Periodically determining quantities of nuclear material in each MBA;
   e) Closing the material balance over the period between two successive physical inventory takings and computing the material unaccounted for (MUF);
   f) Providing for a measurement control program to determine the accuracy of calibrations and measurements;
   g) Testing the computed MUF against its limits or errors in the accidental loss of gain;
   h) Analysing the collected accounting information to determine the cause and magnitude, and taking lessons learned for future improvements.
2) At State Authority Level
   a) Preparing and submitting nuclear material accounting reports to the IAEA;
   b) Ensuring that nuclear material accounting procedures and arrangements are adhered to;
   c) Providing for IAEA inspector access and co-ordination arrangements;
   d) Verifying facility operators' nuclear material accountancy performance.
3) At IAEA Level by:
   a) Independently verifying nuclear material accounting information in facility records and State records, and conducting activities as per safeguard agreements signed with the

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State;

b) Determining the effectiveness of the State safeguards program;
c) Providing statements to the State about the IAEA verification activities.

7.4 Safeguards Design Implications for Nuclear Power Plants

Most safeguards attention is devoted to nuclear power plants (NPPs) ([IAEA1998a], [IAEA2014]).. NPPs are subject to a large flow of nuclear materials. At both off-load refueled LWRs and on-load refueled PHWRs, the entire facility constitutes one MBA. Therefore, all activities at the NPP site are under the safeguards monitoring program.

Figure 9 shows the elements of safeguards in a generic NPP facility. The top route in Figure 9 shows the real material storage areas and transfer routes, including the possibilities for diversion as dashed lines. The middle route in Figure 9 represents the nuclear-material accounting by the operator concerning the activities that have occurred. If this declared record of storage status and transfers accurately reflects the real inventories and transfers, then there can be a high degree of confidence that diversion activity has not occurred. The bottom route in Figure 9 represents the IAEA activities in monitoring physical activity in the storage and material-transfer areas for accounting verification purposes. Any activity observed that does not correspond to that declared will initiate a pre-planned process to re-establish confidence that the declared record reflects the real inventory. That process might lead to an examination and validation of the entire inventory, an expensive task which should be avoided.

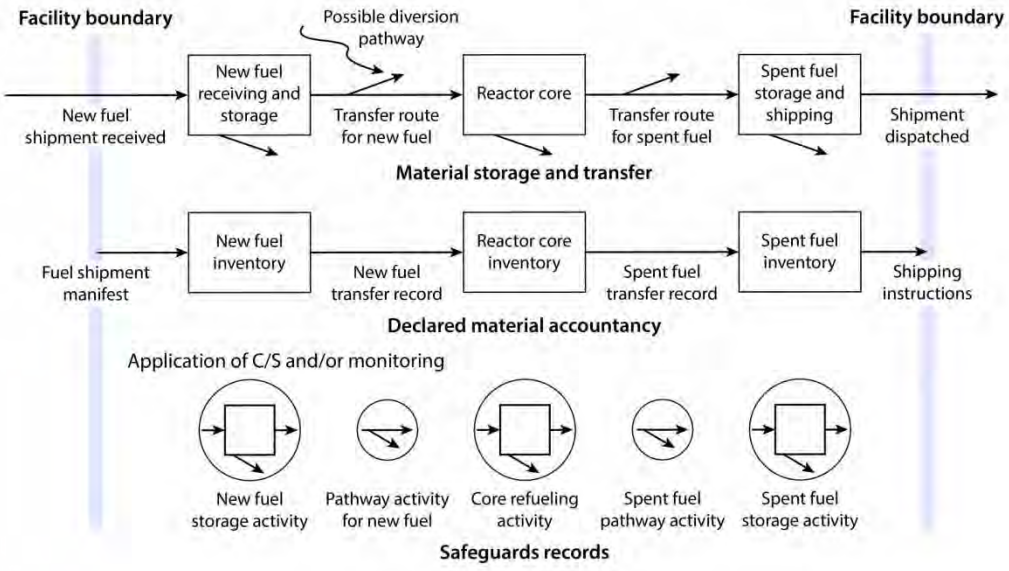

Figure 9 Elements of safeguards [IAEA1998a]

Safeguards approaches, experience, and the associated guidelines vary between off-load and on-load refueled reactors:

a) For off-load refueled reactors, refuelling operations and the associated fuel movements are conducted at low frequency and with large numbers of items moved per refuelling;

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b) For on-load refueled reactors, fuel movements and refuelling operations are carried out at high frequency (daily), but with a small number of items moved per refuelling.

LWRs are refueled off-load, refuelling operations occur with an open vessel, and transfers occur in channels filled with transparent water shielding. This design combination permits direct visual observation. Transparent shielding is impractical for refuelling operations in on-load refueled reactors, and therefore observation by other means is required. In these NPPs, detection of variations in radiation fields from continuously operating instrumentation together with optical surveillance provide the information needed on refuelling sequences for safeguard purposes. Regardless of the NPP design, all fuel is properly labelled by visible and non-visible marking to be read by modern optical and laser devices.

Figure 10 shows the position of optical surveillance devices in the reactor building and the fuel-storage pool building in the LWR reactor design.

Application of the general principles for safeguards is common to all pressure-tube PHWRs. Most IAEA experience in handling PHWRs has been obtained on CANDU reactors.

Figure 11 shows the position of optical surveillance devices in the reactor building and the fuel-storage pool building in the CANDU reactor design.

CANDU NPPs have horizontal fuel channels and are heavy-water moderated and cooled. The refuelling system uses two refuelling machines, one at each end of the core, to insert new fuel at one end and to accept spent fuel at the other. CANDU NPPs require appropriate shielding of the refuelling machines and operations to protect personnel. Consequently, the positioning of the fuelling machines in front of the reactor channels, and the refuelling itself, are carried out by remote control.

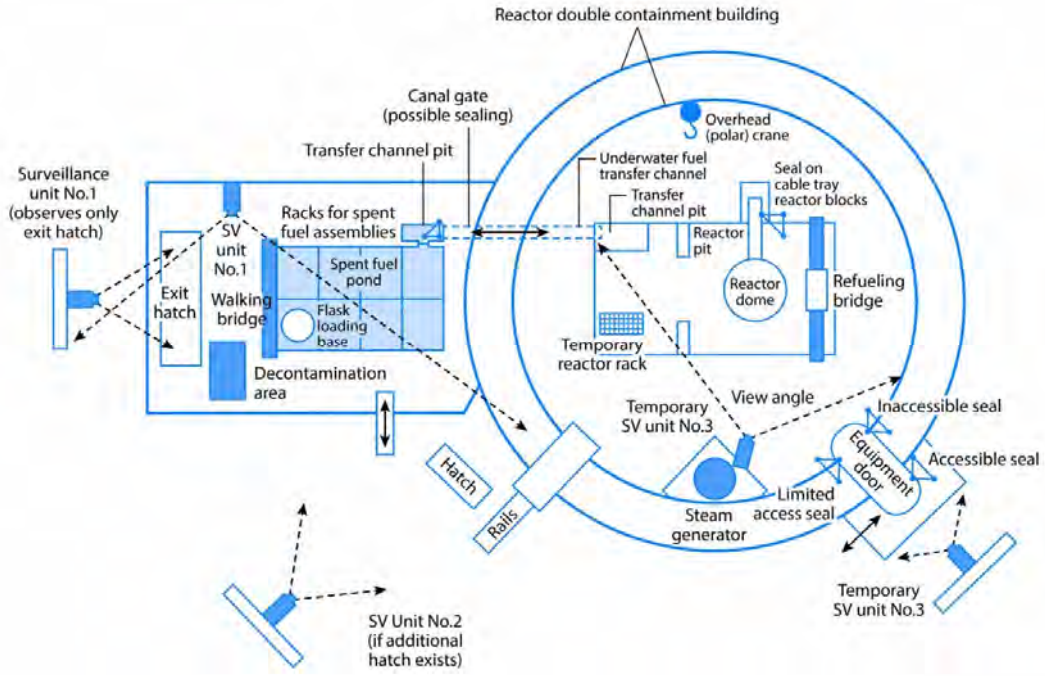

Figure 10 Design aspects of safeguards in LWR reactors (Type II) [IAEA1998a]

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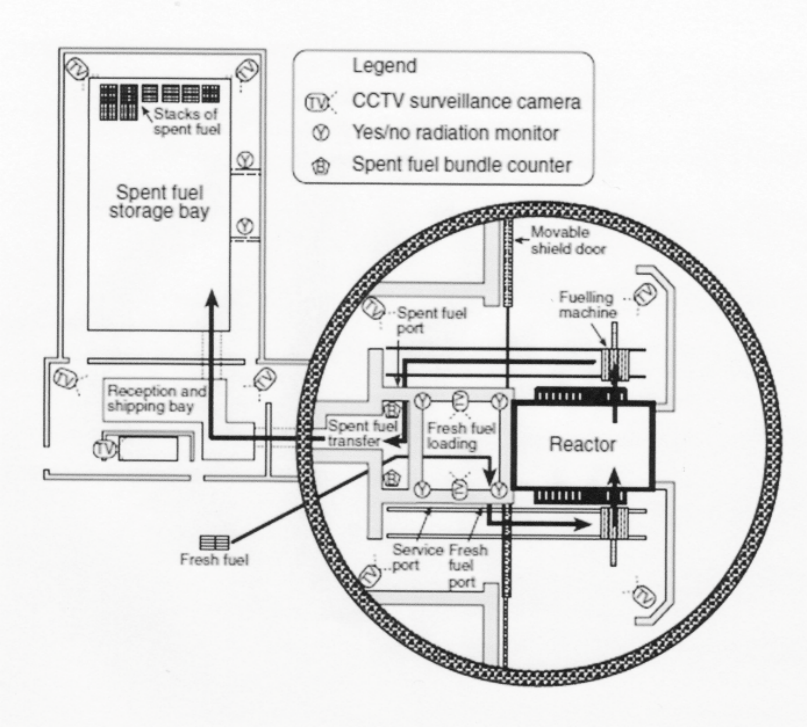

Figure 11 Design aspects of safeguards in CANDU reactors [IAEA1998a]

Item accountancy and maintaining continuity of knowledge using containment and surveillance (C/S) measures form the basis of the CANDU safeguards approach. Direct verification of the operator's declared nuclear material inventory is implemented for fresh and spent fuel. Although fuel bundles in the core remain inaccessible for verification, continuity of knowledge is maintained by a monitoring system when they are discharged. C/S covers the discharge of spent fuel from the reactor and storage in the spent fuel bays to provide continuity of knowledge.

All NPP designer organizations are required to ensure that their designs are safeguards friendly, i.e., the design allows adequate implementation of the safeguards specific approaches. Some of these requirements are listed below (not an inclusive list):

- Requirement for early interaction between the designer and the IAEA;
- Design of barriers which form a part of the C/S system;
- Design of seals for moving nuclear material;
- Design of surveillance techniques;
- Design of fuel identification and fuel verification techniques;
- Detection of unrecorded production or use of nuclear material;
- Transfer for spent fuel from the spent fuel pool to storage or reprocessing.

8 Nuclear Security

Nuclear security has become far more important in the last couple of decades, due largely to changes in the external political environment after the September 11, 2001, terrorist attacks in the United States. Each utility implements security at its sites, and in Canada, regulation of nuclear security is part of the mandate of the CNSC.

Top-level requirements are set by the Nuclear Security Regulations [GovCan2010]. This Act

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covers the security of both certain nuclear materials and nuclear facilities.

Nuclear materials are divided into three categories related to their security impact: for example, a quantity of non-irradiated plutonium >2 kg is Category I; between 500 g and 2 kg is Category II; and between 15 g and 500 g is Category III. Possession in each category requires a licence, and the security obligations for each case are defined in the regulations.

For nuclear facilities in Canada, it is up to the CNSC Commission to establish and update the design basis threat which is taken into account by the utility in determining its security provisions. In addition, the utility must conduct an annual facility-specific threat and risk assessment. Nuclear facilities must be located within a protected area enclosed by a barrier. Category I materials are further protected by an inner-area structure or barrier whose function is to detect and delay unauthorized access before the on-site nuclear response force can successfully intervene.

In addition, the operator must identify *vital areas*. These are areas inside the protected area containing equipment, systems, devices, or a nuclear substance, the sabotage of which would or would likely pose an unreasonable risk to the health and safety of persons arising from exposure to radiation. The operator is then required to implement the appropriate physical protection measures.

The Act also sets out requirements for persons authorized to enter protected and inner areas and for security personnel. The CNSC has supplemented and expanded the Nuclear Security Regulations through a number of Regulatory Documents and Guides. Most of those pertaining to high-security sites such as nuclear power plants contain prescribed information and are not publicly available, and therefore they are not referenced here.

Further details of security regulations, implementation techniques, and verification methodologies are beyond the scope of this textbook and are not available to the general public.

9 Canadian Regulatory Process[11]

The Canadian regulatory process is based on over 60 years of CANDU reactor development and operation in Canada and internationally. Canada has been in the forefront of nuclear power development and use since the very beginning, including the development of its nuclear regulatory program.

The Parliament of Canada first established legislative control and federal jurisdiction over the development and use of nuclear energy and nuclear substances in 1946 with the introduction of the *Atomic Energy Control Act* (AECA), which also established the Atomic Energy Control Board (AECB). Fifty years later, the Canadian Government updated its regulatory requirements, and the Canadian Nuclear Safety Commission was established as a successor to the AECB.

CNSC material has been used throughout this section. We refer to specific documents where appropriate. As noted earlier, we do not try to paraphrase carefully developed regulatory texts. *Italic* fonts have been used to indicate where statements from CNSC documents have been used essentially verbatim.

11 Some of the descriptive material has been drawn from the Web site of the CNSC over the period from July 2011 to January 2013.

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9.1 The CNSC

The Canadian Nuclear Safety Commission regulates the use of nuclear energy and materials to protect the health, safety, and security of Canadians and the environment, and to implement Canada's international commitments on the peaceful use of nuclear energy.

The CNSC develops, maintains, and revises all regulatory requirements and documents under the authority of the Nuclear Safety and Control Act [GovCan2013], which sets the overall legal framework for regulation.

9.2 CNSC Mandate and Organization

This section covers the mandate and role of the CNSC in the Canadian NPP regulatory process. It also covers the key elements of the CNSC organization, starting with the Commission and continuing with the organization of various departments and their role in Canadian regulatory processes.

As discussed in Section 2.3, the CNSC consists of an appointed Commission supported by technical staff.

The CNSC Commission has up to seven appointed permanent part-time members whose decisions are supported by more than 800 staff. These employees review applications for licences according to regulatory requirements, make recommendations to the Commission, and review (and can be delegated by the Commission to enforce) compliance with the Nuclear Safety and Control Act, regulations, and any licence conditions imposed by the Commission.

Under the NSCA [GovCan2013], CNSC's mandate involves four major areas:

- regulation of the development, production and use of nuclear energy in Canada to protect health, safety and the environment
- regulation of the production, possession, use and transport of nuclear substances, and the production, possession and use of prescribed equipment and prescribed information
- implementation of measures respecting international control of the development, production, transport and use of nuclear energy and substances, including measures respecting the non-proliferation of nuclear weapons and nuclear explosive devices
- dissemination of scientific, technical and regulatory information concerning the activities of CNSC, and the effects on the environment, on the health and safety of persons, of the development, production, possession, transport and use of nuclear substances.

The CNSC organization is shown in Figure 12. Note:

- the Regulatory Operations Branch, which is responsible for NPP licensing in Canada;
- the Technical Support Branch, which is home to all the CNSC discipline specialists and experts who review and assess various reactor designs and all information submitted in support of a licence application.

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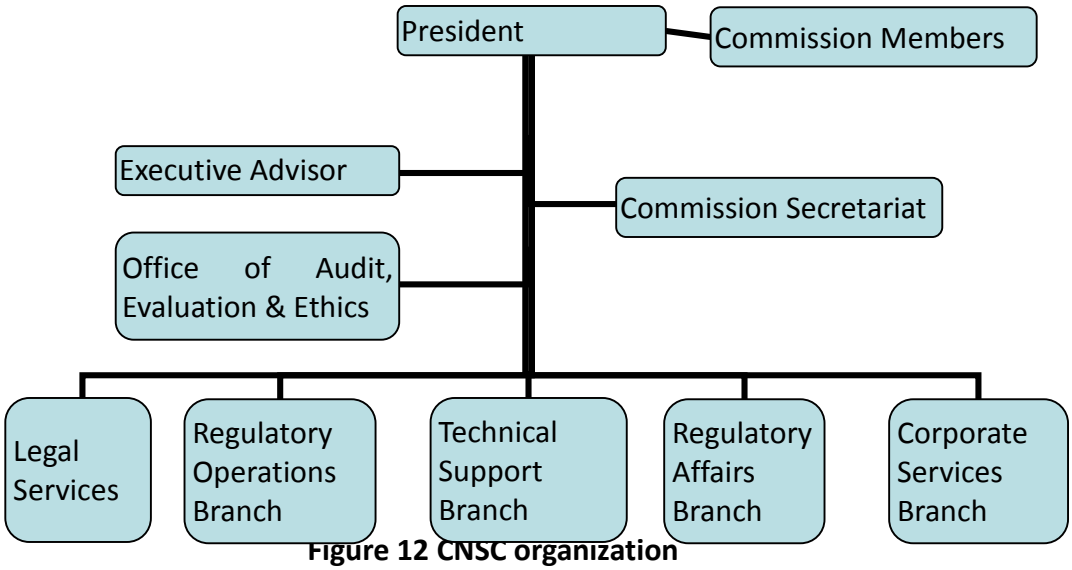

Figure 12 CNSC organization

9.3 Canadian Regulatory Framework and Regulatory Acts

The Canadian regulatory documents are arranged in several levels, as shown in Figure 13, starting at the top level with enabling legislation, then at the middle level with requirements documents, and at the lowest level with guidance documents. The requirements documents can be subdivided into regulations, licences, and regulatory documents (defined in Sections 9.5.1 and 9.5.2).

Three Acts are relevant and are described in the following sub-sections.

9.3.1 Nuclear Safety and Control Act

The Nuclear Safety and Control Act [GovCan2013] is the enabling legislation in Canada, as shown in Figure 13 and as described in Section 9.2.

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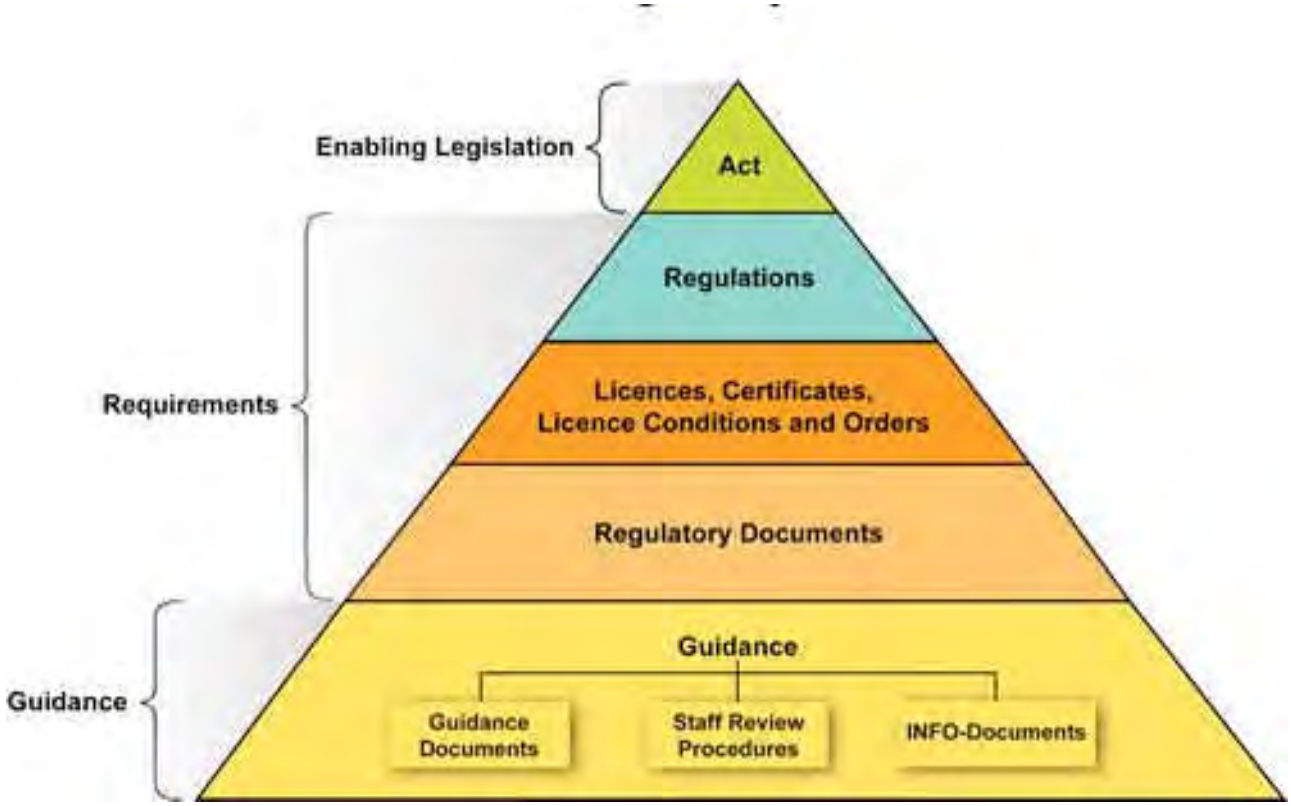

Figure 13 Structure of regulatory documents in Canada

The Act covers the following areas:

- Establishment and organization of the Canadian Nuclear Safety Commission (CNSC);
- Number and role of CNSC members;
- Role, responsibilities, and duties of the CNSC President;
- Directives and powers of the CNSC;
- Use, establishment, and responsibilities of various panels that may be established by the CNSC;
- The decision-making process at the CNSC;
- The scope and process for issuing or revoking various licences to the Canadian industry;
- The responsibilities of the CNSC for maintaining various records and reports;
- Appointment of various specialists, analysts, and inspectors of the CNSC;
- Appointment of designated officers (to whom the Commission can delegate some if its authority to make decisions on its behalf);
- Process for creation of procedures for decisions and orders;
- Process for redetermination and appeal of decisions and orders;
- Scope of regulations and process for their creation and enforcement;
- Provisions for exceptional powers;
- Provisions for offences and punishment;
- Transitional provisions.

Of particular importance is sub-section 24(4) of the Act, which states that:

> "No licence shall be issued, renewed, amended or replaced—and no authorization to

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transfer one given—unless, in the opinion of the Commission, the applicant or, in the case of an application for an authorization to transfer the licence, the transferee:
(a) is qualified to carry on the activity that the licence will authorize the licensee to carry on; and
(b) will, in carrying on that activity, make adequate provision for the protection of the environment, the health and safety of persons, and the maintenance of national security and measures required to implement international obligations to which Canada has agreed."

The next level in Figure 13 is *Regulations*, which are published in the Canada Gazette and include:

- General Nuclear Safety and Control Regulations [GovCan2008]
- Radiation Protection Regulations [GovCan2007]
- Class I Nuclear Facilities Regulations [GovCan2012a]
- Class II Nuclear Facilities and Prescribed Equipment Regulations
- Uranium Mines and Mills Regulations
- Nuclear Substances and Radiation Devices Regulations
- Packaging and Transport of Nuclear Substances Regulations
- Nuclear Security Regulations [GovCan2010]
- Nuclear Non-Proliferation Import and Export Control Regulations [GovCan2010a]
- CNSC Cost Recovery Fees Regulations
- Administrative Monetary Penalties Regulations
- Canadian Nuclear Safety Commission Rules of Procedure
- Canadian Nuclear Safety Commission By-Laws

We have provided references above for some of the key Regulations in the nuclear power-reactor field.

In addition, when required, the CNSC puts forward proposals for new regulations or proposed amendments to existing regulations:

- Regulations Amending the Class I Nuclear Facilities Regulations and the Uranium Mines and Mills Regulation
- Amendments to Nuclear Non-Proliferation Import and Export Control Regulations
- Amendment to Class II Nuclear Facilities and Prescribed Equipment Regulations
- Regulations Amending Certain Regulations made under the Nuclear Safety and Control Act (Miscellaneous Program)
- Amendments to the Packaging and Transport of Nuclear Substances Regulations.

9.3.2 Canadian Environmental Assessment Act

The second relevant Act is the Canadian Environmental Assessment Act [GovCan2012], which covers all aspects of the environmental assessment process that must be followed for a new nuclear energy facility of any purpose and type. It sets out the requirements for addressing all important environmental issues, radiological and non-radiological.

9.3.3 Canadian Nuclear Liability Act

The third Act is the Canadian Nuclear Liability Act [GovCan1985], which regulates the nuclear

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responsibility of various participants in the Canadian nuclear industry. The underlying purpose of the Act is to require nuclear operators to carry insurance to cover liability should an accident occur. The operator is absolutely liable for damage irrespective of the cause of the accident: that is, claimants do not have to prove negligence on the part of the operator—they need only prove that they have suffered damage. Operators must obtain liability insurance for damages up to a maximum of $75 million (but see next paragraph); if the consequences exceed that amount, the federal government appoints an independent Nuclear Damage Claims Commission that will receive claims, assess damages, and recommend the level of compensation that should be paid. The responsibility to pay claims exceeding $75 million rests with the federal government. The Act does not set a limit on what the government pays.

As of this writing (early 2014), Bill C-22 (the Energy Safety and Security Act) has been tabled in the House of Commons to replace the Nuclear Liability Act. It raises the absolute liability limit to $1 billion for both nuclear energy and offshore oil and gas. It is expected that this bill will be enacted by the House of Commons.

9.3.4 Other acts

Other Acts that pertain to some aspects of nuclear power plants include:

- Canadian Environmental Protection Act
- Fisheries Act
- Species at Risk Act
- Navigable Waters Protection Act
- Transportation of Dangerous Goods Act.

9.4 Licences

The third level in Figure 13 also includes *licences*. These are mandatory legal documents which set forth the requirements and limitations for activities undertaken during the life cycle of a nuclear activity. For example, a Power Reactor Operating Licence typically covers (among other items):

- Authorization to operate the nuclear power plant for a specified period;
- Reporting of events to the CNSC;
- Control of the exclusion zone within which no permanent dwelling is permitted and temporary access is controlled;
- References to some of the CNSC Regulatory Documents listed in Section 9.5, which therefore become mandatory;
- References to some of the CSA Standards listed in Table 6, which therefore become mandatory;
- Maintaining minimum operating-crew size and composition, including obtaining operating licenses for key staff;
- Complying with the requirement to use a set of Operating Policies and Principles, including Operating Limits;
- Complying with reactor power limits, including bundle and channel power limits
- Limits on permissible changes without CNSC prior consent. Specifically, this forbids changes which "introduce hazards different in nature or greater in probability than those considered by the Final Safety Analysis Report and Probabilistic Safety Assess-

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ment". Such a clause binds the results of the safety assessment performed by the operator to the licence;
- Using only those fuel-bundle designs approved by the CNSC;
- Complying with requirements for an emergency plan for the facility;
- Complying with the limits on releases of radioactive material in normal operation (derived release limits);
- Following requirements for a maintenance program; and
- Complying with requirements for radiation protection, site security, and safeguards.

The licence is accompanied by, and references, a rather lengthy "Licence Conditions Handbook", which provides:

- compliance verification criteria to meet the conditions listed in the licence;
- information regarding delegation of authority and current versions of documents;
- implementation timelines for specific licence conditions; and
- an explanation of each regulatory requirement specified in the licence.

In effect, the Licence Conditions Handbook extends the scope of the licence.

9.5 CNSC Regulatory Documents

The fourth level in Figure 13 contains the Requirements Documents. They are not *legally* mandatory until and unless referenced in a licence. They are *effectively* mandatory for new designs and are used as an evaluation tool for existing designs.

The fifth level in Figure 13 contains the Guidance Documents. These documents provide direction to licensees and applicants on how to meet the requirements set out in the CNSC's Regulations, regulatory documents, and licences, and the techniques used by CNSC staff to evaluate specific problems or data needed in the review of applications for permits or licences.

The following sub-sections provide a basic explanation of key CNSC regulatory and guidance documents.

9.5.1 Historical classification schemes

The CNSC documents have followed several labelling and numbering schemes that reflect the different periods when these documents were created. We review the historical schemes briefly because many of these documents are still referenced under their original names in licences.

Originally the AECB issued all requirements documents as "R" ("regulatory") documents.

This system was superseded by the following classification:

- "S" ("Standard") documents, which were mandatory
- "G" ("Guidance") documents, which provided guidance
- "P" ("Policy") documents, which provided direction to CNSC staff and information for stakeholders.

More recently, CNSC has used the labels "RD" (Regulatory Document) and "GD" (Guidance Document). Occasionally, CNSC has issued documents as "RD/GD", which means that some parts of the document contain (effectively) requirements and other parts contain guidance. In

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such cases, CNSC staff clarifies when information is mandatory.

The “RD” and “GD” labels have now been replaced by “REGDOC”, as discussed in Section 9.5.2.

In CNSC regulatory documents, “shall” is used to express a requirement, i.e., a provision that a licensee or licence applicant is obliged to satisfy to comply with the requirements of the document. “Should” is used to express guidance, or that which is advised, but not required. “May” is used to express an option or that which is permissible within the limits of the document. “Can” is used to express possibility or capability.

9.5.2 Current classification scheme

The CNSC has recently developed a classification scheme (by subject) for its existing and planned regulatory documents. As of April 2013, all these documents are now organized under three key categories and twenty-five series as follows:

1.0 Regulated facilities and activities

1.1 Reactor facilities

1.2 Class IB facilities

1.3 Uranium mines and mills

1.4 Class II facilities

1.5 Certification of prescribed equipment

1.6 Nuclear substances and radiation devices

2.0 Safety and control areas

2.1 Management system

2.2 Human performance management

2.3 Operating performance

2.4 Safety analysis

2.5 Physical design

2.6 Fitness for service

2.7 Radiation protection

2.8 Conventional health and safety

2.9 Environmental protection

2.10 Emergency management and fire protection

2.11 Waste management

2.12 Security

2.13 Safeguards and non-proliferation

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2.14 Packaging and transport

3.0 Other regulatory areas

3.1 Reporting requirements

3.2 Public and aboriginal engagement

3.3 Financial guarantees

3.4 Commission proceedings

3.5 Information dissemination

The numbering scheme used in this categorization is "REGDOC-x.yy.zz", where "x" is the number of the key category, "yy" is the number of the series, and "zz" is a number which identifies the document uniquely within that category and series. For example, the design requirements for new reactor designs are documented in REGDOC-2.5.2, where the first "2" means the document is in the "safety and control area"; "5" means it is under the "physical design" series; and the last "2" is a sequence number identifying the specific document.

9.5.3 Licensing basis

Many of these documents form the licensing basis of a regulated facility or activity. The *licensing basis* [CNSC2010a] is a set of requirements and documents consisting of:

a) Regulatory requirements set out in the applicable laws and regulations.

- Examples: the Nuclear Safety and Control Act, the Canadian Environmental Assessment Act, the Canadian Environmental Protection Act, the Nuclear Liability Act, the Transportation of Dangerous Goods Act, the Radiation Emitting Devices Act, the Access to Information Act, and the Canada/IAEA Safeguards Agreement.

b) Conditions and safety and control measures described in the facility licence and the documents directly referenced in that licence.

- Examples: the Nuclear Security Regulations, the Nuclear Non-Proliferation Import and Export Control Regulations, the General Nuclear Safety and Control Regulations, the Radiation Protection Regulations, the Uranium Mines and Mills Regulations, the Class I Nuclear Facilities Regulations, the Nuclear Substances and Radiation Devices Regulations, the Class II Nuclear Facilities and Prescribed Equipment Regulations, and the Packaging and Transport of Nuclear Substances Regulations.

c) Safety and control measures described in documents directly referenced in the licence.

- Examples: regulatory documents (such as RD-204, RD/GD-210, S-294, plus others), industry codes and standards (such as CSA N286-05, CSA N285.4, CSA N290.13), proponent- or licensee-produced documents, and any subsequent changes made to these documents in accordance with a CNSC-approved change control process (such as Management System Manual, Limits and Conditions of Operation).

d) Documents needed to support the licence application.

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- Examples are documents describing the design, safety fuellings, and all aspects of operation to which the licensee makes reference; conduct of operations; and conduct of maintenance.

The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity and thus establishes the basis for the CNSC's compliance program with respect to that regulated facility or activity.

We now summarize selected regulatory documents. Note that document numbers and titles are shown in terms of the currently issued documents (and not in terms of the new classification scheme shown in Section 9.5.2).

9.5.4 Design of new NPPs in Canada: RD-337

RD-337 [CNSC2008] sets out comprehensive CNSC requirements with respect to the design of new water-cooled NPPs and provides examples of optimal design characteristics[12]. Excerpts follow:

> The information provided in this document is intended to facilitate high quality design, and consistency with modern international codes and standards, for new water-cooled NPPs. It is recognized that specific technologies may use alternative approaches. If a design other than a water-cooled reactor is to be considered for licensing in Canada, the design is subject to the safety objectives, high level safety concepts and safety management expectations associated with this regulatory document.
>
> To a large degree, this document represents the CNSC's adoption of the principles set forth in International Atomic Energy Agency (IAEA) document NS-R-1, Safety of Nuclear Plants: Design, and the adaptation of those principles to align with Canadian practices. The scope of NS-R-1 has been expanded to address the interfaces between NPP design and other topics, such as environmental protection, radiation protection, ageing, human factors, security, safeguards, transportation, and accident and emergency response planning.

The main components of RD-337 are:

i. Safety objectives and concepts
   This section covers safety objectives in Canada (general, radiation protection, and technical), safety goals, radiation dose criteria, accident mitigation and management, and safety concepts (defence-in-depth, physical barriers, and operating limits and conditions).
ii. Safety management during design
   This section discusses safety during the design process. It includes all important aspects of the design process, including design authority, design management, QA, proven engineering practices, design documentation, etc.
iii. Safety considerations
   This section covers all aspects of safety applications in a design, including the application of defence-in-depth, safety functions, accident prevention and plant safety characteris-

12 While this Chapter was being finalized, CNSC issued an update to RD-337 – called REGDOC 2.5.2 – which adds lessons learned from Fukushima. See [CNSC2014].

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tics, radiation protection principles and acceptance criteria, exclusion zones, and facility layout.

iv. General design considerations
This section contains general requirements for all technical disciplines and reactor engineering areas. It covers:

- structure, system, and component classifications;
- plant design envelope and operating plant states;
- guidance for consideration of all important hazards and initiating events that need to be considered in the design;
- design rules, limits, and reliability for safety (such as common-cause failures, single failures, fail-safe requirements, etc.);
- pressure-retaining systems and components;
- equipment classification (seismic, fire, environmental, etc.);
- instrumentation and control;
- safety support systems;
- civil structures;
- human factors;
- commissioning;
- security;
- safeguards;
- decommissioning.

v. System-specific requirements
This section covers requirements specific to most important systems and functions, including the reactor core, heat-transport system, steam-supply system, means of shutdown, emergency core cooling system, containment, emergency power supply, heat transfer to an ultimate heat sink, emergency heat-removal system, control facilities, waste management and control, fuel handling and storage, and radiation protection.

vi. Safety analysis
The safety analysis sections cover the most important aspects of safety analysis, namely analysis objectives, hazards analysis, deterministic analysis, and probabilistic analysis. The topic is covered in RD-310 in more detail [CNSC2008b].

vii. Environmental protection and mitigation
This section covers environmental protection design and requirements, as well as requirements for released radioactive and hazardous substances.

viii. Alternative approaches
RD-337 is intended to be technology-neutral for water-cooled reactor designs. It is recognized that specific technologies may use alternative approaches.

The CNSC considers alternative approaches to the expectations in this document where:

- The alternative approach would result in an equivalent or superior level of safety;
- Application of the expectations in this document conflicts with other rules or requirements;
- Application of the expectations in this document would not serve the underlying purpose, or is not necessary to achieve the underlying purpose; or
- Application of the expectations in this document would result in undue hardship

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or other costs that significantly exceed those contemplated when the regulatory document was adopted.

One example of an alternative approach is consideration of best estimate and uncertainty analysis instead of a conservative analysis; see Chapter 13.

9.5.5 NPP site evaluation: RD-346

This regulatory document [CNSC2008c] sets out requirements for the evaluation of sites for new NPPs before an application is made for a Licence to Prepare Site and before an environmental assessment determination is initiated.

RD-346 represents the CNSC staff's adoption, or where applicable, adaptation of the principles set forth by the International Atomic Energy Agency (IAEA) in NS-R-3, Site Evaluation for Nuclear Installations. The scope of RD-346 goes beyond NS-R-3 in several aspects such as the protection of the environment, security of the site, and protection of prescribed information and equipment, which are not addressed in IAEA's NS-R-3.

Site evaluation is a process that should precede the submission of an application to prepare a site for the construction of a new NPP. RD-346 is written to serve the broader licensing needs under the Nuclear Safety and Control Act and the Canadian Environmental Assessment Act, and will facilitate a more effective and efficient regulatory review.

The document covers criteria for site evaluation; type and scope of baseline data required in the site evaluation process; evaluation of natural external events; evaluation of external, non-malevolent, human-induced events; security considerations; decommissioning; quality assurance; and the consultation process.

9.5.6 NPP life extension: RD-360

The regulatory document RD-360 [CNSC2008d] covers the regulatory requirements for NPP life-extension programs. The purpose of this regulatory document is to inform licensees about the steps and phases to consider when undertaking a project to extend the life of a nuclear power plant. The main topics are:

- Key elements to consider when establishing the scope of the life-extension project; and
- Considerations to be taken into account in planning and executing a life-extension project.

The document covers the initiation of the life-extension project, preparation of the integrated project plan (which includes environmental assessment, safety assessment, integrated implementation plan and its adequacy, etc.), project execution, and return to service.

A major part of the assessment for life extension requires reviews of the existing plant against modern codes and standards, including, but not limited to, CNSC documents more recent than were in place at the time the plant was originally licensed. Although compliance is not mandatory, gaps must be identified, assessed, and remedied if practical. In addition, the condition of systems, structures, and components (SSC) is assessed, largely to capture the effects of aging and degradation. These assessments may also identify gaps. The entire assessment is collected in a Global Assessment Report (GAR), from which an Integrated Implementation Plan (IIP) is developed. The IIP is a work plan and schedule to fix the identified gaps insofar as is practical

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and risk-effective. Such life-extension assessments are similar to, but much broader in scope than, periodic safety reviews.

9.5.7 Safety analysis for NPPs: RD-310 and GD-310

Safety analysis is one of the most important topics in NPP licensing because it provides the assurance that the plant can be operated safely and that the postulated events can be prevented, mitigated, and managed. The CNSC has issued both the requirements document (RD-310 [CNSC2008b]) and the guidance document (GD-310 [CNSC2012b]).

> RD-310 identifies high-level regulatory information for a nuclear power plant licence applicant's preparation and presentation of a safety analysis. In particular, it establishes a more modern risk-informed approach to the categorization of accidents, one that considers a full spectrum of possible events, including the events of greatest consequence to the public. The CNSC expects proponents and applicants for new reactor licences to immediately apply this regulatory document in new-build submissions. In the context of existing reactors, CNSC expects the licensees to apply this document, in a graduated manner, to all relevant programs in future submissions.
>
> RD-310 explains the objectives of deterministic safety analysis, and in detail covers the requirements for safety analysis. In particular, it covers identification and classification of events following the international practice, i.e., groups them into Normal Operation (NO), Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs), and Beyond Design Basis Accidents (BDBAs), and explains the acceptance criteria. However, most of the details in the document are focussed on the safety analysis methods and assumptions, including methods, data, assumptions, computer code requirements, and required conservatisms in the analysis. The document also covers the required documentation that a licence applicant needs to prepare, and the time line for updating this documentation.

9.5.8 Reliability programs for NPPs: RD/GD-98

Regulatory document RD/GD-98, *Reliability Programs for Nuclear Power Plants* [CNSC2012a], sets out the requirements and guidance of the CNSC for the development and implementation of a reliability program for nuclear power plants in Canada. One of the important aspects of this document is to provide a clear definition of the Systems Important to Safety (SIS) and how these are treated and classified in NPP design.

9.5.9 Reporting requirements for operating nuclear power plants: S-99

The purposes of this Regulatory Standard [CNSC2003] are:

- To help the Canadian Nuclear Safety Commission collect the information that it needs to ensure that a nuclear power plant is operating safely and to verify that the licensee is complying with regulatory requirements;
- To help applicants for operating licences for nuclear power plants design programs for collecting and reporting information in accordance with regulatory requirements; and
- To facilitate CNSC evaluations of the appropriateness, completeness, and timeliness of information reported to the CNSC by operators of nuclear power plants.

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This document is a key document for utilities that operate nuclear power plants because it states the requirements for reporting of operational issues. The document groups the requirements into two groups: reporting unscheduled events, and reporting scheduled events.

9.5.10 Pre-licensing review of a vendor's reactor design: GD-385

Guidance document GD-385, Pre-Licensing Review of a Vendor's Reactor Design [CNSC2012], describes the pre-licensing review process provided by the CNSC for assessing a vendor's design for a nuclear power plant or small reactor. The review considers the areas of design that relate to reactor safety, security and safeguards. For this topic, a licensee or licence applicant is not obliged to satisfy any provisions through regulations or licence conditions, and therefore no regulatory document (RD) accompanies this guidance document.

A pre-licensing review is an optional service provided by the CNSC. The objective of a pre-licensing review is to increase regulatory certainty while ensuring public safety. The review can be undertaken by a reactor vendor prior to an applicant's submission of a licence application to the CNSC and can provide early identification and resolution of potential regulatory or technical issues in the design, particularly those that could result in significant changes to the design or safety analysis.

This review does not certify a reactor design and does not involve the issuance of a licence under the Nuclear Safety and Control Act. It is not required as part of the licensing process for a new nuclear power plant or small reactor. The conclusions of a design review do not bind or otherwise influence decisions made by the Commission, with whom the authority resides to issue licences for nuclear power plants and small reactors. It is different in these respects from the U.S. Standard Design Certification.

9.5.11 Licence to construct a nuclear power plant: RD/GD-369

To request CNSC assessment of a request for a construction licence, a formal application must be submitted to the Canadian Nuclear Safety Commission (CNSC) along with appropriate documentation. This CNSC guidance identifies the information that should be submitted to support such an application.

RD/GD-369 [CNSC2012c] applies to applications for a licence to construct a water-cooled nuclear power plant. It does not presuppose or limit an applicant's intention to follow any particular kind of water-cooled reactor technology. This document follows the format of the IAEA Safety Guide No. GS-G-4.1, *Format and Content of the Safety Analysis Report for Nuclear Power Plants*, but is more specific to Canadian regulatory practice. In following these guidelines, an applicant must submit the appropriate information to demonstrate that it is qualified and will make adequate and reasonable provisions to undertake the activity to be licensed, pursuant to sub-section 24(4) of the *Nuclear Safety and Control Act* and associated regulations.

RD/GD-369 is a combined document that lists and explains the requirements, but also provides guidance on how to address them. The document covers plant description; management of safety; site evaluation; general design aspects and support programs; design of plant structures, systems, and components; safety analysis; construction and commissioning; operational aspects; operational limits and conditions; radiation protection; environmental protection; emergency preparedness; radioactive and hazardous waste management; decommissioning and end-of life-aspects; and safeguards.

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9.5.12 Severe accident management programs for nuclear reactors: G-306

The CNSC Guide on *Severe Accident Management Programs for Nuclear Reactors* [CNSC2006] describes a typical severe accident management (SAM) program for a nuclear reactor. A person who applies for, or holds, a licence to construct or operate a nuclear reactor can use this guide when developing and implementing measures to help:

a) Prevent the escalation of a reactor accident into an event involving severe damage to the reactor core;
b) Mitigate the consequences of an accident involving severe damage to the reactor core; and
c) Achieve a safe, stable state of the reactor and plant over the long term.

Some severe accidents (in particular dual failures, see Chapter 13) have historically been part of the CANDU design basis. However the spectrum of severe accidents that must be evaluated has broadened in the past few decades due to the maturation of probabilistic safety assessment as an evaluation tool and to the real events at Three Mile Island, Chernobyl, and Fukushima. Because operating plants may not have been explicitly designed against extreme events, in the past, severe accident assessment and management looked at the capabilities of existing plant systems to mitigate such events and developed severe-accident management procedures to take advantage of these capabilities. However, especially post-Fukushima, design measures (retrofits) are being implemented specifically to mitigate a spectrum of severe accidents, for example, filtered containment venting.

A Severe Accident Management (SAM) program provides an additional defence against the consequences of those accidents that fall beyond the scope of events considered in the reactor design basis. The establishment of a SAM program should ensure that personnel involved in managing an accident have the information, procedures, and resources necessary to carry out effective on-site actions.

G-306 provides guidance on the goals and principles of severe-accident management, considerations for program development (risk assessment and accident analysis), determination of high-level accident response (preventive and mitigating actions, evaluation of systems and equipment, and assessment of material resources), and SAM procedures and guidelines.

9.5.13 Containment systems for CANDU nuclear power plants: R-7

This regulatory document [CNSC1991] is one of the original series of AECB documents. It was superseded by RD-337, except in areas in which it is consistent with RD-337, but provides more information.

The document describes the design, operating, and testing requirements for the containment system. It includes definition of the containment envelope, dose limits under accident conditions (see Section 9.5.17), structural integrity, leakage criteria, and requirements in several important areas, such as availability, environmental qualification, shielding, status monitoring, and seismic provisions.

9.5.14 Shutdown systems for CANDU nuclear power plants: R-8

This regulatory document [CNSC1991a] is one of the original series of AECB documents. It was superseded by RD-337 except in areas in which it is consistent with RD-337, but provides more

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information.

The document describes the design, operating, and testing requirements for the shutdown systems. It includes definition of the minimum allowable performance standards and requirements in several important areas, such as performance, environmental qualification, availability, separation and independence, actuation instrumentation, status monitoring, and seismic provisions.

9.5.15 Emergency core cooling systems for CANDU nuclear power plants: R-9

This regulatory document [CNSC1991b] is one of the original series of AECB documents. It was superseded by RD-337, except in areas in which it is consistent with RD-337, but provides more information.

The document describes the design, operating, and testing requirements for the emergency core cooling system. It includes definition of the minimum allowable performance standards and requirements in several important areas, such as core cooling, environmental qualification, availability, separation and independence, actuation instrumentation, leakage control, inadvertent operation, shielding, status monitoring, and seismic provisions.

9.5.16 Use of two shutdown systems in reactors: R-10

This regulatory document [CNSC1991b] is one of the original series of AECB documents. It was superseded by RD-337.

The document formalized the requirement for two independent shutdown systems:

“All nuclear power reactors licensed for construction in Canada after January 1, 1977, shall incorporate two independent protective shutdown systems unless otherwise approved by the Board.”

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It required that the two shutdown systems be independent, reliable, diverse, and individually capable of meeting regulatory dose limits[13]. Therefore, it could be assumed that at least one system would operate as designed when required (i.e., there was no need to postulate simultaneous unavailability of both systems after an accident).

9.5.17 Classification of systems, design basis accidents, and dose limits

The selection and classification of accidents for safety analysis, or in other words the definition of design basis accidents, is covered extensively in Chapter 13 and is summarized here only for convenience:

The deterministic safety analysis limits under which all large operating CANDU plants up to but excluding Darlington have been licensed can be found in [Hurst1972]. The spectrum of possible design basis accidents was collapsed into two broad categories: *single failures*, or the failure of any one process *system* in the plant, and *dual failures*, a much less likely event defined as a single failure coupled with the unavailability of either a shutdown system, or containment, or the emergency core cooling system, these constituting the so-called *special safety systems*. For each category, a frequency and a consequence limit was chosen that had to be satisfied. In addition, to deal with the siting of a reactor (Pickering A) next to a major population centre (Toronto), population dose limits were defined for each category of accident.

The limits were as follows (sometimes this is called the Siting Guide):

Table 3 Single / dual failure dose limits

Accident	Maximum Frequency	Individual Dose Limit	Population Dose Limit
Single Failure	1 per 3 years	0.005 Sv 0.03 Sv thyroid	10^2 Sv 10^2 Sv thyroid
Dual Failure	1 per 3000 years	0.25 Sv 2.5 Sv thyroid	10^4 Sv 10^4 Sv thyroid

For the licensing of Darlington, as discussed in Chapter 13, Consultative Document C-6 was used. This expanded the number of accident classes to five, as follows:

13 This is an oversimplification of R-10. R-10 actually specifies that for single failures, "at least" one shutdown system had to meet dose limits, whereas for dual failures, each shutdown system had to be individually capable of so doing. The distinction was removed in later practice, so that each shutdown system alone had to meet requirements for all design basis accidents.

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Table 4 Consultative document C-6 dose limits

Event Class	Expected Frequency per reactor-y [Charak, 1995][14]	Whole-Body Dose (Sv)	Thyroid Dose (Sv)
1	$> 10^{-2}$	0.0005	0.005
2	10^{-2} to 10^{-3}	0.005	0.05
3	10^{-3} to 10^{-4}	0.03	0.3
4	10^{-4} to 10^{-5}	0.1	1
5	$< 10^{-5}$	0.25	2.5

Finally, for new plants (and as an evaluation tool for existing plants), a classification scheme based more on international practice has been used (from [CNSC2008] and [CNSC2008b]):

Table 5 New plants in Canada: dose limits

Event	Frequency	Dose Limit (Sv)
Anticipated Operational Occurrence	$\geq 10^{-2}$ / reactor-year	0.0005
Design Basis Accident	10^{-2} to 10^{-5} / reactor-year	0.020

Plant licences refer to the Final Safety Analysis Report, which in turn references the appropriate classification scheme.

9.6 CNSC Staff Review Procedures

CNSC staff have prepared a set of Staff Review Procedures (SRPs) used during the technical assessment stage of the environmental assessment and site licensing process. They provide instructions to CNSC staff on the conduct of an assessment. They also inform potential applicants and the public about the criteria used to assess environmental impact statements and licence applications for new nuclear power plants. SRPs are not regulatory documents. They are available on the CNSC Web site. Other SRPs are under development.

9.7 CNSC CANDU Safety Issues

All regulators track safety-related issues which are not so pressing as to require immediate regulatory action, but have not been addressed convincingly enough by the operator. Typically, these include knowledge gaps, safety analysis assumptions, and methodologies. Such issues in Canada were formerly called “Generic Action Items” (GAIs); these have now been phased out as a regulatory tool and replaced by the CANDU Safety Issues. Generally, they pertain to more than one operating plant.

14 Expected frequency ranges are not part of C-6; they were used by Ontario Hydro in the licensing of Darlington to classify events not listed in C-6.

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The list of CANDU Safety Issues was developed from:

- The IAEA document, Generic Safety Issues for Nuclear Power Plants with Pressurized Heavy Water Reactors and Measures for their Resolution [IAEA2007a];
- Regulatory oversight of currently operating reactors;
- Results of life-extension assessments;
- Safety issues identified in pre-licensing reviews of new CANDU designs.

They are classified into three categories:

- Category 1 means issues that have been satisfactorily addressed in Canada;
- Category 2 identifies issues that are a concern in Canada, but have appropriate measures in place to maintain safety margins;
- Category 3 issues are a concern in Canada and measures are in place to maintain safety margins, but the adequacy of these measures needs to be confirmed.

For the last, the industry and the CNSC have used a risk-informed decision-making (RIDM) process to identify, estimate, and evaluate the risks associated with each and to recommend measures to control these risks. None of these issues has yielded a risk significance level that required immediate corrective action.

As of August 2013 [GovCan2013a], there are 16 Category 3 issues outstanding, including six associated with large-break loss-of-coolant accidents (LBLOCA) and ten other safety issues not associated with LBLOCA.

The current status of each CANDU Safety Issue is reported publicly at International Nuclear Safety Convention meetings [GovCan2013a].

9.8 CNSC Fukushima Action Items

On March 11, 2011, a magnitude 9.0 earthquake, followed by a devastating tsunami, struck Japan. The combined impact of the earthquake and tsunami on the Fukushima Dai-ichi nuclear power plant caused a severe nuclear accident. Details are presented in Chapter 13.

The CNSC established the CNSC Fukushima Task Force to evaluate the operational, technical, and regulatory implications for CANDU nuclear power plants and the capability of CANDU NPPs to withstand conditions similar to those that triggered the Fukushima accident. It issued its recommendations in the Fukushima Task Force Report [CNSC2011]. In response, the CNSC Staff developed an Action Plan [CNSC2011a], which mandates actions both for the utilities and for the CNSC itself. In these documents, CNSC staff requested utilities to conduct certain activities in the following areas:

a) Safety-review criteria, including identification and magnitudes of external events, adequacy of design-basis-accident analysis, consideration of beyond-design-basis accidents, implementation of severe-accident management, licensees' emergency response plans, nuclear emergency management in Canada, and CNSC regulatory frameworks and processes;
b) Strengthening reactor defence-in-depth;
c) Enhancing emergency response; and
d) Improving regulatory framework and licensing.

This work resulted in a number of Fukushima Action Items that the Canadian industry is putting

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in place.

9.9 Canadian Codes and Standards

The Canadian Standards Association (CSA) is a not-for-profit membership-based association serving business, industry, government, and consumers in Canada and the global marketplace. CSA develops standards that enhance public safety and health, advance the quality of life, and help to preserve the environment.

The objective of the CSA Nuclear Standards Program is to help promote a safe and reliable nuclear power industry in Canada and to have a positive influence on the international nuclear power industry. Although focussing on nuclear power plants, the program provides guidance for other types of nuclear facilities on selected topics such as radioactive waste management and environmental releases. Compliance with CSA nuclear standards is mandatory for those that are referenced in a plant licence and is *de facto* mandatory for other organizations which contribute to design, analysis, and other aspects of plant construction and operation. Table 6 shows selected CSA Standards relevant to nuclear power plants.

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Table 6 Selected key CSA codes and standards

Document number	Title	Creation / Revision or Reaffirmed date
CSA N285.0	General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/Material Standards for Reactor Components for CANDU Nuclear Power Plants	2008
CSA N285.4	Periodic Inspection of CANDU Nuclear Power Plant Components	2009
CSA N285.5-13	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	1998 (2013)
CSA N285.8	Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	2010
CSA N286-05	Management System Requirements for Nuclear Power Plants	2005 (2010)
CSA N286.7	Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants	1999 (2007)
CSA N287.1	General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	July 1993 (2009)
CSA N287.7-08 (R2013)	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	2008 (2013)
CSA N288.4-10	Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	2010 (1990)
CSA N288.5-11	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	2011
CSA N288.6-12	Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	2012
CSA N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	2008
CSA N289.1	General Requirements for Seismic Qualification of CANDU Nuclear Power Plants	2008

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Document number	Title	Creation / Revision or Reaffirmed date
CSA N290.1	Requirements for the Shutdown Systems of CANDU Nuclear Power Plants	1980 (2006)
CSA N290.15-10	Requirements for the Safe Operating Envelope of Nuclear Power Plants	2010
CSA N291	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	2008
CSA N293	Fire Protection for CANDU Nuclear Power Plants	2007
CSA N294	Decommissioning of Facilities Containing Nuclear Substances	2009

10 Reactor Licensing Process in Canada[15]

CNSC material has been used throughout this section and referred to in specific documents where appropriate. *Italic* fonts have been used to indicate where statements from CNSC documents have been used essentially verbatim.

This section covers the licensing process of both new-build NPPs and operating reactors. The emphasis is on new-build NPPs. However, certain aspects of the licensing (relicensing) of operating reactors are also covered. See [CNSC2008a] and [Andrews2012].

The focus is on the Canadian licensing process, but relevant and important aspects of, and differences from, the licensing process in other countries are also covered at a high level.

10.1 Licensing Process

Nuclear power plants are defined as Class I nuclear facilities, and the regulatory requirements for these facilities are found in the *Class I Nuclear Facilities Regulations*. The regulations also require separate licences for each of the five phases in the lifecycle of a nuclear power plant:

i. licence to prepare a site, including the environmental assessment;
ii. licence to construct;
iii. licence to operate;
iv. licence to decommission; and
v. licence to abandon.

There are four major steps in the licensing process:

1. Applicant submits a licence application
2. Environmental assessment

15 Some of the descriptive material has been drawn from the Web site of the CNSC.

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3. Licensing technical assessment
4. CNSC renders its decision.

We summarize these in turn.

10.1.1 Licence application

In Canada, the licensing process begins when an application is received by CNSC. All new licence applications or amendments to existing licences require the approval of the Commission or a CNSC Designated Officer. The Commission is notified when an application that requires a decision from them has been filed.

The preparation of a licence application needs to consider all regulatory criteria as defined by the *Nuclear Safety and Control Act*, and *relevant regulations,* CNSC requirements and expectations, international and domestic standards, and applicable international obligations.

For major resource projects such as nuclear power plants, uranium mines, or fuel processing facilities, a project description is sent to Natural Resources Canada's Major Projects Management Office (MPMO). MPMO is responsible for coordinating the work of all the federal departments and agencies that have a role to play in the regulatory process for major resource projects. MPMO offers licensees a single entry point into the federal regulatory system.

For new nuclear power plants in Canada, three licences are initially required:

1. licence to prepare site, including an environmental assessment
2. licence to construct the nuclear power plant, and
3. licence to operate.

The regulatory process in Canada in principle makes it possible to combine these processes, but the normal route is for an applicant to request them sequentially, i.e., each process is completed when the appropriate licence is issued.

Very often, before a formal licensing process is initiated through an application by the NPP owner or operator, a pre-licensing process is performed by the NPP design organization, as explained in detail in Section 5.

The key element of a successful licensing process is to prepare adequate documentation that meets regulatory requirements and expectations. Figure 14 provides a general idea of the level of detail required during different phases of the licensing process. On the left-hand side, the reactor design process is shown, and on the right-hand side, the level of documentation completeness. From top to bottom, the design and documentation move from the initial stage to completeness. The middle of the diagram provides an indication of the percentage completeness of the design. However, the design and documentation completeness level is discussed and agreed in detail with CNSC at the beginning of a licensing process.

The technology vendor must complete a conceptual design and appropriate documentation before engaging in any licensing activity. After the conceptual design has been firmed up, which means that key design options and characteristics have been decided, a basic engineering program begins to develop a preliminary design. During the basic engineering process, at some point, the technology vendor may decide to engage the regulator in performing a non-binding pre-licensing review. This is not a mandatory process, and technology vendors may or may not decide to request it.

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As shown in Figure 14, from [CNSC2012], following the preliminary design, detailed design is conducted. After the detailed design and appropriate documentation have been prepared, the licensee applies for a construction licence. During the regulatory review for the construction licence, further detailed design activities related to the construction phase may be conducted. After the regulatory body issues an operating licence, the licensee will continue to provide ongoing design support activities.

Section 13 gives as an example the information that must be submitted in Canada for construction and operating licences.

At any time during operation, if a design or procedural change is considered for implementation which deviates from the safety case or from any prior licence issued by CNSC, an approval process for the particular change must be conducted and CNSC approval obtained before implementing the change.

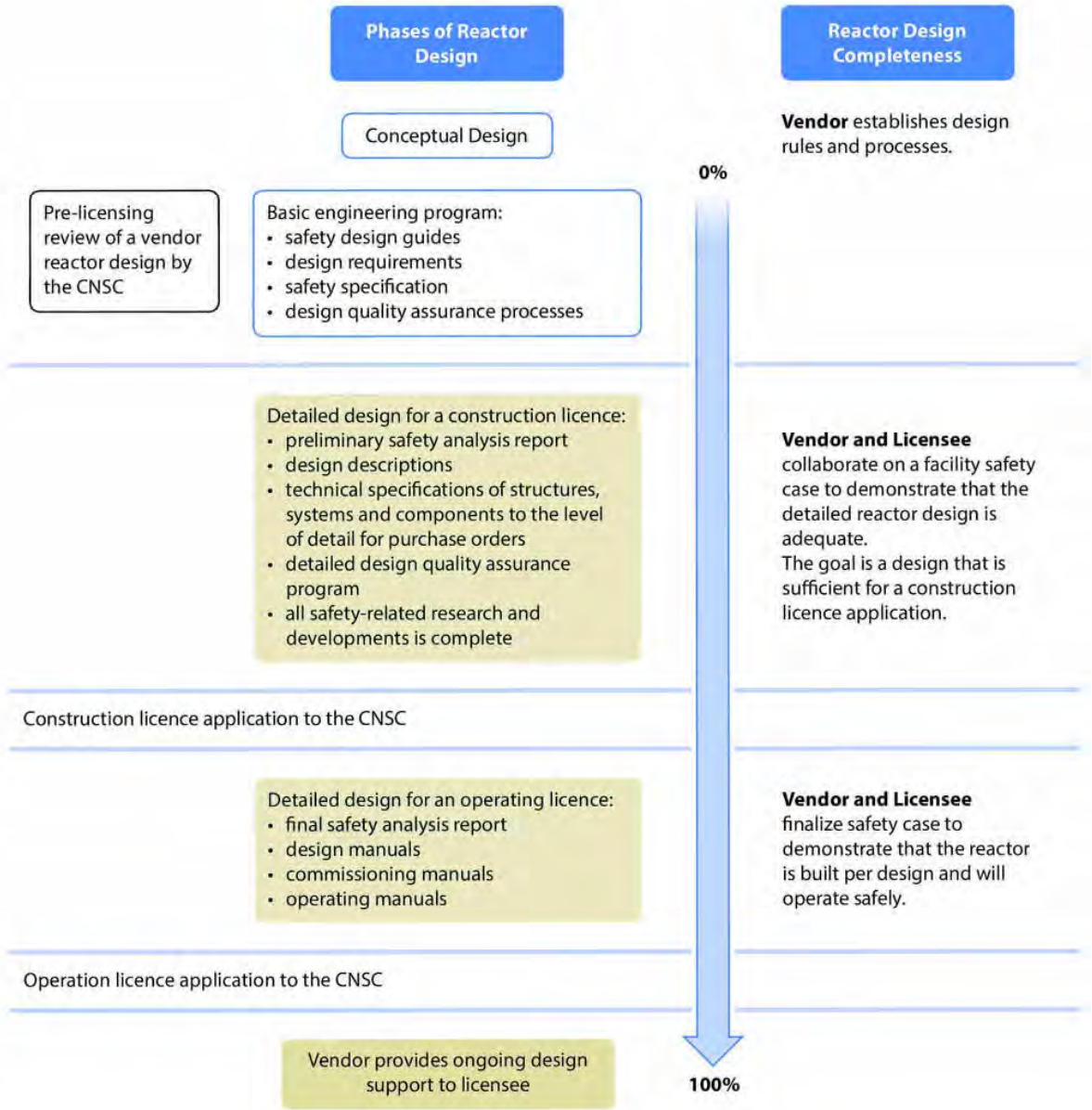

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Figure 14 Design and documentation completeness phases

10.1.2 Environmental assessment

CNSC has obligations and responsibilities under the *Canadian Environmental Assessment Act* (CEAA), which is the basis for federal Environmental Assessments in Canada. See [CEAA2012].

During the application review process, CNSC determines whether an environmental assessment (EA) is required for a proposed project. EAs are used to predict the potential environmental effects of a specific project and to determine whether these effects should be mitigated, and by what means, before a project is carried out. EAs consider environmental components such as:

- air, water, and soil quality;
- noise;
- human health;
- aboriginal interest, physical and cultural heritage; and
- use of land and resources.

EAs provide opportunities for public participation in activities undertaken by the potential licensees, the CNSC, or both, including aboriginal consultations.

Under the CEAA [GovCan2012], there are two levels of environmental assessment: a standard environmental assessment and a panel review. The standard environmental assessment is the default level of assessment applicable to a designated project; the second, higher level of environmental assessment is a panel review. The CEAA imposes time limits for environmental assessments. Most standard environmental assessments must be completed within one year, while panel reviews are limited to two years.

At the beginning of an environmental assessment, CNSC provides project-specific advice and guidance about what should be assessed, such as the scope of the project and the factors that apply to each individual project. Further guidance is available from CNSC for conducting these technical studies.

All applications for new nuclear facilities and some amendments to existing licences require the applicant to undertake an EA and technical studies. After technical studies are completed, the applicant can submit an Environmental Impact Statement (EIS) to CNSC, who will assess the EIS and prepare an EA Report. For major new facilities, the assessment of an applicant's EIS, referred to as an EA technical assessment, and is guided by an Assessment Plan and EIS-related *Staff Review Procedures*. The EA report contains recommendations for an EA decision, including all appropriate mitigation measures and the requirements of a follow-up program. Pursuant to the CEAA, the Responsible Authority (the Commission or a delegated officer) must make its EA decision before any licensing action can be considered for the project.

Only one EA is required for a project, even if that project requires separate licences for different phases (i.e., licence to prepare site, licence to construct, etc.). However, additional EAs may be required if a project changes over time. For a new nuclear power plant, if construction is to be performed at a new site, the EA is conducted before a site licence is issued. For a new nuclear power plant where construction is intended on a site that already has other NPP units, the EA is conducted before site preparation can begin for the new NPP addition.

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10.1.3 Licensing technical assessment

After receipt of a licence application, CNSC staff undertakes a variety of technical assessments according to a prescribed Assessment Plan and licensing-related Staff Review Procedures to ensure that each application complies with its corresponding regulatory requirements. Peer reviews are sometimes used, when additional rigour is required.

The Licensing Technical Assessment considers all regulatory criteria as defined by the Nuclear Safety and Control Act, relevant regulations, CNSC requirements and expectations, international and domestic standards, and applicable international obligations.

At the end of this process, CNSC staff make a recommendation for a decision on the licence application through an integrated assessment report. A recommended compliance plan for each licence is also developed, and the mitigation measures included in the follow-up program, if applicable, are included in the licence.

10.1.4 CNSC decision

The final decision on a licence is made by either the Commission (e.g., for licences for nuclear power plants) or a CNSC designated officer.

When the decision is to be made by the Commission, public hearings may be held to take into account the views, concerns, and opinions of interested parties, including the applicant and interveners. This is an important part of the process of establishing regulatory policy, making licensing decisions, and implementing programs. CNSC staff recommendations are presented, and the applicant and the public are normally given a chance to make statements.

Most decisions involving major nuclear facilities are made through a two-session[16] public hearing process. The first hearing is for presentations by the applicant and CNSC staff, and the second for intervener presentations. Commission members may ask questions in both cases. The applicant and CNSC staff also attend the second hearing and respond to further questioning by the Commission.

Whether made by the Commission or a CNSC designated officer, the decision will take the form of the granting of a licence or certificate, or else a letter of refusal.

The next section provides more information about the EA process for NPP construction on a new site.

10.2 Site Selection and Site Licensing

This section covers the criteria used for appropriate site selection. A number of regulatory documents in Canada and internationally cover this topic, e.g., [CNSC2008c].

The CNSC must be satisfied that it is feasible to perform the site preparation activities in a manner that will satisfy all health, safety, security, and environmental protection requirements. In addition, the Commission cannot issue a site preparation licence unless a decision as a result of the EA has been made, indicating that the project may proceed; however, these approval

16 The first hearing is known as “Day 1” and the second as “Day 2”, although they may last more than one day each.

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processes may proceed in parallel.

The following aspects are considered when evaluating the suitability of a site over the life of a nuclear power plant:

- the potential effects of external events (such as seismic events and floods) and human activity on the site;
- the characteristics of the site and its environment which could influence the pathways of radioactive and hazardous material that may be released; and
- the population density, population distribution, and other characteristics of the region, insofar as they may affect the implementation of emergency measures and the evaluation of risks to individuals, the surrounding population, and the environment.

Under the regulations, an applicant must submit, for any licence, a Project Description of the facility and plans showing the location, perimeter, areas, structures and systems of the facility. An application for a Licence to Prepare Site does not require detailed information or determination of a reactor design; however, high-level design information is required for the environmental assessment that precedes the licensing decision for a Licence to Prepare Site. An application for a Licence to Construct must contain more detailed information about the reactor design and a supporting safety case.

The goal of the CNSC, during the site preparation stage, is to ensure that the site characteristics which may have an impact on health, safety, security, and the environment have been identified, and that these characteristics can and will be taken into consideration in the design, operation, and decommissioning of the proposed nuclear power plant. The technical information arising from the consideration of external events, site-specific characteristics, and supporting assessments is used as input into the design of the nuclear power plant and must be included in the application.

The Canadian Environment Assessment Act, as amended in 2012 [GovCan2012], gives the CNSC decision-making authority on all environmental assessments (EAs) for nuclear projects. Therefore, both environmental assessment and site licensing processes occur concurrently under the authority of the CNSC Commission. This ensures that the information submitted by the proponent can be considered by public and government agencies through a single process and that any appropriate decisions under EA and site licensing can be made by a single body—the CNSC Commission. The Commission process establishes the level of an environmental assessment based on risk and other factors; for example, an application for a site for a new nuclear power plant will almost certainly require a panel review.

The CNSC Commission makes a decision on the EA within the 24-month timeline established for site licensing.

10.3 Construction Licence

During this stage, CNSC undertakes a variety of technical assessments according to a prescribed assessment plan and licensing-related staff review procedures to ensure that each application complies with its corresponding regulatory requirements. The scope and duration of each assessment will vary depending on the type of licence requested. Peer reviews are sometimes used when additional rigour is required.

The licensing technical assessment considers all regulatory criteria as defined by the Nuclear

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Safety and Control Act, relevant regulations, CNSC requirements and expectations, international and domestic standards, and applicable international obligations.

The CNSC reviews the submitted application and judges whether the information contained in the application is acceptable and whether the licence applicant is qualified and will make adequate provision for safety, etc., in carrying out the licensed activity (in accordance with 24(4) of the NSCA). If the CNSC accepts this information, it becomes the reference safety case for the plant and will form part of the licensing basis at the construction licence stage. The information provided with the licence application, including the documents to which the application makes reference, constitutes the construction safety case. The requirements for the safety case are listed in detail in [CNSC2012c].

The information required in support of the application to construct a nuclear power plant includes, for example:

- a description of the proposed design for the nuclear power plant, taking into consideration the physical and environmental characteristics of the site;
- environmental baseline data on the site and surrounding area;
- a preliminary safety analysis report showing the adequacy of the design;
- measures to mitigate the effects on the environment and the health and safety of persons that may arise from construction, operation, or decommissioning of the facility;
- information on potential releases of nuclear substances and hazardous materials and proposed measures to control them;
- programs and schedules for recruiting and training operations and maintenance staff;
- and in general, those safety control areas listed in Section 9.5.2.

After the construction licence application has been received, the CNSC performs a comprehensive assessment of the design documentation, the preliminary safety analysis report, the construction program, and any other information required by the regulations. The assessment focusses on determining whether the proposed design and safety analysis, along with other required information, meet regulatory requirements.

Specifically, the evaluation involves rigorous engineering, scientific analysis, and engineering judgment, taking into consideration the CNSC's experience and knowledge of the best practices in nuclear plant design and operation, as gained from existing power plants in Canada and around the world. This review may take place in parallel with the Environmental Assessment and site preparation licensing process. At the end of this process, CNSC staff makes a recommendation for a decision on the licence application, through an integrated assessment report. Occasionally, the recommendations include proposed changes to the regulatory framework to accommodate evolving nuclear technologies. A recommended compliance plan for each licence is also developed, and the mitigation measures included in the follow-up program, if applicable, are included in the licence.

In addition to reviewing the information included in the application, the CNSC also verifies that any outstanding issues from the site preparation stage have been resolved. The CNSC staff's conclusions and recommendations from these reviews are documented in reports submitted to the Commission; the Commission then makes the final decision on the issuance of the construc-

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tion licence. As noted earlier, the Commission will not issue a licence unless it is satisfied that the applicant will make adequate provisions to protect health, safety, security and the environment, and to respect the international obligations to which Canada has agreed. As such, it is the responsibility of the applicant to show that there are no major safety issues outstanding at the time the Commission considers the application for a construction licence.

During the construction phase, the CNSC carries out compliance activities to verify that the licensee is complying with the NSCA, with associated regulations, and with its licence. Such compliance activities focus on confirming that plant construction is consistent with the design, that the licensee is demonstrating adequate project oversight and confirming that quality assurance requirements are met, and that the licensee is respecting any requirements of the EA follow-up program.

In the latter part of construction, regulatory attention turns towards the inactive commissioning program (without fuel loaded) and associated activities, whose purpose is to demonstrate to the extent practicable that all systems, structures, and components function in accordance with their design specifications.

10.4 Operating Licence

When applying for a Licence to Operate a nuclear power plant, it is the responsibility of the applicant to demonstrate to the CNSC that it has established the safety management systems, plans, and programs that are appropriate to ensure safe and secure operation. Information required by the CNSC in support of the application for a licence to operate includes, for example:

- a description of the structures, systems, and equipment at the nuclear power plant, including their design and operating conditions;
- the final safety analysis report; and
- proposed measures, policies, methods, and procedures for:
  - commissioning systems and equipment;
  - operating and maintaining the nuclear facility;
  - handling nuclear substances and hazardous materials;
  - controlling the release of nuclear substances and hazardous materials into the environment;
  - preventing and mitigating the effects on the environment and on health and safety resulting from the operation and subsequent decommissioning of the plant;
  - assisting off-site authorities in emergency preparedness activities, including assistance to deal with an accidental off-site release; and
  - nuclear security.

A typical application [OPG2013] would also cover:

- Management system, including safety culture;
- Human performance management;
- Operating performance and corrective action program;
- Fitness for service;
- Probabilistic safety analysis;

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- Radiation protection;
- Conventional health and safety;
- Emergency management and fire protection;
- Waste management;
- Safeguards;
- Packaging and transport;
- Community relations and public information;
- Financial guarantees;
- Nuclear liability insurance;
- Open action items.

In addition to assessing the information included in the application to operate the nuclear power plant, the CNSC also verifies that any outstanding issues from the construction licensing stage have been resolved.

The CNSC staff's conclusions and recommendations from these reviews are documented in reports submitted to the Commission, which then makes the final decision on issuance of the operating licence.

The Licence to Operate will enable the operator to begin active commissioning. The purpose of the commissioning activities is to demonstrate that the plant has been constructed in accordance with the design and that the systems, structures, and components important to safety are functioning in accordance with their design specifications. The initial operating licence is typically issued with conditions (hold points) to load nuclear fuel, permit reactor start-up, and operation at power in steps up to the design rating of the plant. All the relevant commissioning tests must be satisfactorily completed before the hold points can be released.

During the subsequent long-term operation of the plant, the CNSC carries out compliance activities in order to verify that the licensee is complying with the NSCA, associated regulations, and its licence terms. If the compliance activities identify any non-compliance or adverse trend, there is a range of possible actions that the CNSC can take, from a request for licensee action to prosecutions.

10.4.1 Ongoing oversight

Operating NPPs are subject to ongoing regulatory monitoring, screening, and inspections. Canadian CANDU plants have resident full-time CNSC inspectors on-site to provide immediate interaction with the operator and to view the station and equipment first-hand.

Aspects of continuing CNSC oversight include:

- Informal information exchange;
- Attending industry working groups as observers;
- Periodic formal audits and inspections of plant operation and management;
- Planned routine assessments and evaluations;
- Reactive inspections following events;
- Formal correspondence;
- Monitoring of progress on CANDU Safety Issues and on specific station action items;
- Conducting an annual review of the safety performance of NPPs;
- Revision to the Licence Condition Handbooks;

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- Enforcement actions;
- Orders (legal instruments);
- Licence amendments initiated by CNSC staff;
- Licence amendments or renewals initiated by licensees; and
- Commission meetings and hearings.

10.4.2 Operating licence renewal and refurbishment

There are two longer-term activities which provide an opportunity to ensure that a plant continues to meet regulatory requirements.

Operating licences are typically renewed about every five years in Canada. This requires a review, update, and resubmission of the safety case to reflect changes in plant design and to assess the plant against evolving regulatory requirements. The status of Action Items, CANDU Safety issues, station-specific items, station safety performance, operating experience, quality assurance, and other items are all reviewed at the time of renewal of an operating licence.

CANDU plants can be refurbished by replacement of the fuel channels, typically after about 25 years of operation. This action triggers a much more extensive comparison of the plant against modern standards, as described in Section 9.5.6.

10.5 Licence to Decommission and Return Site to Green Site

At the end of a nuclear power plant's useful life, it will be necessary to decommission the facility. This will require a separate licence from the Commission. Information on decommissioning plans and financial guarantees will, in practice, have been taken into account at all stages of licensing (site preparation, construction, and operation) [CNSC2008a]. Factors taken into account when evaluating an application to decommission a nuclear power plant include, but are not limited to:

- the major components and systems within the facility, which must be properly considered during decommissioning planning;
- the design features that will facilitate decommissioning activities and reduce the spread of contamination during operation;
- the expected levels of activation and contamination within the facility following the end of operation;
- an assessment of structures to ensure that they are capable of being maintained for the proposed period of storage and monitoring;
- the disposal of some of the nuclear materials and radiation devices (e.g., fresh fuel, spent fuel, heavy water, water contaminated with tritium, and other prescribed nuclear materials); and
- the quantities or volumes of wastes of all types (radioactive and hazardous) expected during decommissioning activities.

In addition, the licensees must show that they have sufficient funds to decommission the plant, provide for the long-term management of spent nuclear fuel, and provide continuing environmental monitoring and maintenance of the site for the duration of the licence.

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11 Problems

1. Tokai-Mura, 1999: Locate the following material: International Atomic Energy Agency, Report on the Preliminary Fact Finding Mission Following the Accident at the Nuclear Fuel Processing Facility in Tokaimura, Japan; IAEA report, 1999. You may also find or use other documents as well. Assess the accident in terms of regulatory effectiveness, using some of the ideas in this chapter. Propose lessons learned. Which aspects of regulatory performance were also apparent after the Fukushima accident?
2. Compare the approaches taken by the USNRC and the CNSC with respect to large Loss of Coolant Accident (LOCA) regulatory acceptance criteria.
    a. Find out what the acceptance criteria are and where they are documented. Note: not all Canadian criteria are set by the CNSC; you may have to look at CANDU safety reports.
    b. Both Canadian and U.S. practices set limits on sheath oxidation in a LOCA to prevent embrittlement on rewet by Emergency Core Cooling (ECC). Compare the two criteria and explain the rationale in each case.
    c. Find the requirements on shutdown systems for both the United States and Canada. What is the typical reliability of the shutdown system in each case? How are postulated impairments of the shutdown system handled in each case?
3. Review the U.S. General Design Criteria (GDC) in 10CFR, Appendix A to part 50. Which design criteria (if any) would be *intrinsically* difficult for CANDU to meet? If you find any such criteria, what is the safety concern that these GDCs address, and how is that concern addressed in CANDU design?
4. Locate the U.K. Safety Assessment Principles issued by HSE. Compare and contrast the requirements for safety systems with those in CNSC document RD-337. Which ideas are common? Which are different? (Compare the underlying requirement, not the exact wording).

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12 Appendix A—Examples of U.K. Safety Assessment Principles

This Appendix lists some examples of U.K. Safety Assessment Principles which illustrate the non-prescriptive technology-neutral approach. The text of the SAP is in **bold** and the commentary in normal typeface.

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"Engineering principles: design for reliability Single failure criterion EDR.4

During any normally permissible state of plant availability no single random failure, assumed to occur anywhere within the systems provided to secure a safety function, should prevent the performance of that safety function.

175 Consequential failures resulting from the assumed single failure should be considered as an integral part of the single failure. Further discussion of the single-failure criterion is given in IAEA Safety Standard NS-G-1.2"

==========

"Engineering principles: safety systems: Diversity in the detection of fault sequences ESS.7

The protection system should employ diversity in the detection of fault sequences, preferably by the use of different variables, and in the initiation of the safety system action to terminate the sequences.

342 This principle applies in particular to U.K. civil nuclear power reactor safety systems and in particular to high-integrity safety systems."

==========

"Engineering principles: reactor core Shutdown systems ERC.2

At least two diverse systems should be provided for shutting down a civil reactor.

444 Where a shutdown system is also used for the control of reactivity, a suitable and sufficient shutdown margin should be maintained at all times.

445 Reactor shutdown and subsequent hold-down should not be inhibited by mechanical failure, distortion, erosion, corrosion, etc., of plant components, or by the physical behaviour of the reactor coolant, under normal operation or design basis fault conditions."

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This last SAP was part of the basis for fitting a second shutdown mechanism (large-volume boron injection tanks) to the Sizewell-B PWR.

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13 Appendix B—Information Required in Licence Applications

(Construction and Operating Licences in Canada, as Examples)

From [GovCan2012a]

Licence to Construct

An application for a licence to construct a Class I nuclear facility shall contain the following information in addition to the information required by section 3:

(a) a description of the proposed design of the nuclear facility, including the manner in which the physical and environmental characteristics of the site are taken into account in the design;
(b) a description of the environmental baseline characteristics of the site and the surrounding area;
(c) the proposed construction program, including its schedule;
(d) a description of the structures proposed to be built as part of the nuclear facility, including their design and their design characteristics;
(e) a description of the systems and equipment proposed to be installed at the nuclear facility, including their design and their design operating conditions;
(f) a preliminary safety analysis report demonstrating the adequacy of the design of the nuclear facility;
(g) the proposed quality assurance program for the design of the nuclear facility;
(h) the proposed measures to facilitate Canada's compliance with any applicable safeguards agreement;
(i) the effects on the environment and the health and safety of persons that may result from the construction, operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects;
(j) the proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical and radiological characteristics;
(k) the proposed measures to control releases of nuclear substances and hazardous substances into the environment;
(l) the proposed program and schedule for recruiting, training and qualifying workers in respect of the operation and maintenance of the nuclear facility; and
(m) a description of any proposed full-scope training simulator for the nuclear facility.

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Licence to Operate

An application for a licence to operate a Class I nuclear facility shall contain the following information in addition to the information required by section 3:

(a) a description of the structures at the nuclear facility, including their design and their design operating conditions;
(b) a description of the systems and equipment at the nuclear facility, including their design and their design operating conditions;
(c) a final safety analysis report demonstrating the adequacy of the design of the nuclear facility;
(d) the proposed measures, policies, methods and procedures for operating and maintaining the nuclear facility;
(e) the proposed procedures for handling, storing, loading and transporting nuclear substances and hazardous substances;
(f) the proposed measures to facilitate Canada's compliance with any applicable safeguards agreement;
(g) the proposed commissioning program for the systems and equipment that will be used at the nuclear facility;
(h) the effects on the environment and the health and safety of persons that may result from the operation and decommissioning of the nuclear facility, and the measures that will be taken to prevent or mitigate those effects;
(i) the proposed location of points of release, the proposed maximum quantities and concentrations, and the anticipated volume and flow rate of releases of nuclear substances and hazardous substances into the environment, including their physical, chemical and radiological characteristics;
(j) the proposed measures to control releases of nuclear substances and hazardous substances into the environment;
(k) the proposed measures to prevent or mitigate the effects of accidental releases of nuclear substances and hazardous substances on the environment, the health and safety of persons and the maintenance of security, including measures to
(i) assist off-site authorities in planning and preparing to limit the effects of an accidental release,
(ii) notify off-site authorities of an accidental release or the imminence of an accidental release,
(iii) report information to off-site authorities during and after an accidental release,
(iv) assist off-site authorities in dealing with the effects of an accidental release, and
(v) test the implementation of the measures to prevent or mitigate the effects of an accidental release;
(l) the proposed measures to prevent acts of sabotage or attempted sabotage at the nuclear facility, including measures to alert the licensee to such acts;
(m) the proposed responsibilities of and qualification requirements and training program for workers, including the procedures for the requalification of workers; and
(n) the results that have been achieved in implementing the program for recruiting, training and qualifying workers in respect of the operation and maintenance of the nuclear facility.

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15 Glossary

ACNS	Advisory Committee on Nuclear Safety (Canada, historical)
ACRS	Advisory Committee on Reactor Safeguards (USA)
AEC	Atomic Energy Commission (USA)
AECA	Atomic Energy Control Act (Canada)
AECB	Atomic Energy Control Board (Canada)
ALARA	As Low as Reasonably Achievable
ALARP	As Low as Reasonably Practical (UK)
ASLB	Atomic Safety and Licensing Board (USA)
BDBA	Beyond Design Basis Accident
BSL	Basic Safety Limit (UK)
BSO	Basic Safety Objective (UK)
BWR	Boiling Water Reactor
CEAA	Canadian Environmental Assessment Act
CFR	Code of Federal Regulations (USA)
CNSC	Canadian Nuclear Safety Commission
C/S	Containment and Surveillance
CSA	Canadian Standards Association
DBA	Design Basis Accident
DIV	Design Information Verification
EA	Environment Agency (UK)
EA	Environmental Assessment (Canada)
ECC	Emergency Core Cooling
EIS	Environmental Impact Statement
ENSREG	European Nuclear Safety Regulators Group
EU	European Union
GAR	Global Assessment Report
GDA	Generic Design Assessment (UK)
GDC	General Design Criteria (US)
GAI	Generic Action Item

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HSE	Health and Safety Executive (UK)
IIP	Integrated Implementation Plan
IAEA	International Atomic Energy Agency
INPO	Institute of Nuclear Power Operations
INSAG	International Safety Advisory Group (IAEA)
IRRS	Integrated Regulatory Review Service (IAEA)
LBLOCA	Large Break Loss of Coolant Accident
LNT	Linear No Threshold (dose response hypothesis)
LOCA	Loss of Coolant Accident
LWR	Light Water Reactor
MBA	Material Balance Area
MDEP	Multinational Design Evaluation Programme
MPMO	Major Projects Management Office
MUF	Material Unaccounted For
NDPB	Non Departmental Public Body (UK)
NISA	Nuclear and Industrial Safety Agency (Japan)
NPP	Nuclear Power Plant
NRA	Nuclear Regulation Authority (Japan)
NRU	National Research Universal (reactor)
NRX	National Research Experimental (reactor)
NPP	Nuclear Power Plant
ONR	Office for Nuclear Regulation (UK)
PHWR	Pressurized Heavy Water Reactor
PWR	Pressurized Water Reactor
PSA	Probabilistic Safety Analysis
PSR	Periodic Safety Review
RCS	Reactor Coolant System
RIDM	Risk-Informed Decision Making
RSAC	Reactor Safety Advisory Committee (Canada, historical)
SAPs	Safety Assessment Principles (UK)
SAM	Severe Accident Management
SER	State Evaluation Report
SFAIRP	So Far As Is Reasonably Practicable (UK)

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SIR	Safeguards Implementation Report
SIS	Systems Important to Safety
SRP	Standard Review Plan (US)
SRP	Staff Review Procedure (Canada)
SSC	Systems, Structures and Components
STUK	Säteilyturvakeskus (Radiation and Nuclear Safety Authority, Finland)
Sv	Sievert, a unit of radiation dose
TAG	Technical Assessment Guide (UK)
TOR	Tolerability of Risk (UK)
TSO	Technical Support Organization
TMI	Three Mile Island
USNRC	United States Nuclear Regulatory Commission
WENRA	Western European Nuclear Regulators' Association

16 Acknowledgements

We gratefully thank the following reviewers for their hard work and excellent comments during the development of this Chapter. Their feedback has much improved it. Of course the responsibility for any errors or omissions is entirely the authors'.

Glenn Archinoff
Joe Boyadjian
Dave Newland
Terry Rogers

We also thank Diana Bouchard for expertly editing and assembling the final copy.

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Executive Summary

Public participation is presented as a central feature of nuclear regulation in Canada. This report asks whether that participation actually influences Canadian Nuclear Safety Commission licensing decisions, or whether it mainly functions as a formal step in a process whose outcomes are already shaped by CNSC staff assessments and staged regulatory approval.

By studying how the CNSC dealt with three licensing cases, at Darlington, Pickering, and Point Lepreau, we show that public concerns rarely shape the final licensing outcome. Interventions are recorded, summarized, and acknowledged in Commission decisions. However, acknowledgement is not the same as influence. In practice, the CNSC often treats intervenor concerns as issues to be managed through future oversight rather than as reasons to delay, alter, or refuse licensing approval, but the public has no formal way of intervening in such future oversight activities.

This is most evident in the Darlington and Pickering cases. In Darlington, intervenors raised serious concerns about emergency planning, incomplete safety analysis, radioactive waste, reactor design maturity, climate change, external hazards, and the continued reliance on an earlier environmental assessment. The Commission acknowledged these concerns but still granted the licence to construct. Rather than requiring the issues to be resolved before approval, the CNSC deferred them to later licensing stages, future submissions, staff review, and regulatory hold points.

The Pickering case shows the same pattern in an even sharper way. Intervenors raised concerns about operating beyond design life, probabilistic safety assessment, large release risk, cumulative site risk, emergency preparedness, and the adequacy of the evidence before the Commission. Despite these concerns, the Commission allowed continued operation. Outstanding issues were allowed to be dealt with through future reporting and CNSC staff oversight, not through a refusal to proceed or a requirement that the concerns be resolved first.

Point Lepreau is the clearest example of intervenor concerns producing a measurable effect, but even there, the influence was procedural rather than substantive. Intervenors opposed NB Power's request for a 25-year licence and CNSC staff's recommendation of a 20-year licence, arguing that long licence terms would reduce public participation and weaken accountability. The Commission granted a shorter 10-year licence and required a mid-point public proceeding. However, it still does not show a comparable influence over technical safety requirements.

The core finding of this report is that CNSC public participation is stronger as a record-building mechanism than as a decision-shaping mechanism. Civil society interventions help identify unresolved risks, expose gaps in the evidence, and create a public record of concern. However, the CNSC rarely treats those concerns as decisive. Technical and safety concerns are usually absorbed into the existing licensing framework through deferral, scope

limitation, hold points, or staff-level review. As a result, intervenors may be heard without being meaningfully considered.

This creates a serious accountability problem. Once an issue is moved from a public hearing into post-hearing CNSC staff oversight, it becomes difficult for the public to know what information was submitted, what standard was applied, whether the concern was resolved, and whether the intervenor's original warning had any effect. The process is open at the hearing stage, but much less transparent at the stage where unresolved issues are actually managed.

This report recommends that the CNSC strengthen the connection between public participation and regulatory outcomes. Key safety concerns raised during hearings should not be deferred without clear criteria, timelines, and public reporting. Records of Decision should explain whether major intervenor concerns are accepted, rejected, or deferred, and why. Hold points should identify the issues they are intended to resolve and make the basis for release publicly traceable. Public participation should also occur earlier in the licensing process, before key assumptions are already settled and when the CNSC makes decisions about lifting hold points.

Civil society interventions remain essential, but the current process limits their effect. The problem is not that intervenors are excluded. The problem is that they are included in a way that rarely changes the result. If public participation is to be meaningful, the CNSC must do more than allow intervenors to speak. It must show how their concerns are considered, weighed, and reflected in the decisions that follow.

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1. Introduction

1.1 Background

Canada's nuclear regulatory system relies on formal licensing decisions made by the Canadian Nuclear Safety Commission (CNSC), the federal authority responsible for regulating nuclear activities to protect health, safety, security, and the environment. A central feature of this regulatory framework is the use of public hearings, which allow civil society organizations, technical experts, and members of the public to participate in licensing decisions involving significant safety and environmental risks.

Civil society participation is intended to support transparency and accountability in nuclear regulation. Organizations such as the Canadian Environmental Law Association (CELA) routinely participate in CNSC hearings by reviewing technical documentation, identifying safety concerns, raising issues related to emergency preparedness, radioactive waste management, environmental protection, and long-term risk. These interventions introduce independent analysis into the regulatory record and provide an opportunity for external scrutiny of both proponent submissions and CNSC staff assessments.

Despite the presence of these participation mechanisms, the extent to which civil society interventions influence final regulatory decisions remains uncertain. Intervenor concerns are frequently acknowledged in hearing records and Commission decisions, yet such acknowledgement does not necessarily translate into measurable changes in licensing outcomes. In several cases, issues raised during hearings have been reassigned to future regulatory processes rather than resolved at the time of approval.

This report examines three major CNSC licensing cases involving Darlington, Pickering, and Point Lepreau. These cases were selected because they involve significant licensing decisions and substantial participation by civil society organizations. Together, they provide a basis for examining how intervenor concerns are recorded, interpreted, and reflected in final decisions, and whether participation functions as a mechanism of influence or primarily as a procedural requirement. Since this report draws on a large number of CNSC hearing materials, a bibliography with the full details of the decision documents, Commission Member Documents, and interventions discussed is included at the end.

1.2 Policy Problem

Despite formal mechanisms for public participation, this report demonstrates that civil society interventions are incorporated into the hearing record but do not meaningfully influence

CNSC licensing outcomes. The CNSC licensing process demonstrates strong procedural transparency. Public hearings are conducted openly, written interventions are formally recorded, and Records of Decision regularly reference issues raised by intervenors. These features suggest a regulatory system designed to incorporate public input into complex technical decision-making.

However, in multiple licensing decisions, intervenor concerns are often acknowledged without resulting in measurable changes to regulatory outcomes. Issues raised during hearings are frequently deferred to later licensing stages, treated as outside the scope of the current decision, or responded to through CNSC claims of ongoing oversight, rather than through enforceable licence conditions that would allow the public to verify whether concerns are meaningfully resolved.

This gap between formal participation and measurable influence represents a central policy concern. If participation mechanisms do not meaningfully affect decision outcomes, public hearings risk functioning primarily as procedural exercises rather than as effective tools of regulatory accountability.

1.3 Research Question

This report is guided by the following primary research question:

Do interventions by the public and civil society make a difference to CNSC licensing decisions? If so, can we identify how, at what stages, and through what types of issues interventions influence regulatory outcomes?

To address this central question, the report examines several related sub-questions:

- What issues are included in Proponent and CNSC Staff Commission Member Documents, compared to those raised by intervenors?
- How are these issues treated in the Commission's Reasons for Decision?
- Do intervenor concerns influence the imposition of regulatory hold points?
- Are certain categories of issues more likely to receive regulatory attention than others?

These questions focus on measurable patterns of decision-making rather than participation alone. The analysis, therefore, evaluates not only whether intervenors are heard, but whether their concerns affect regulatory outcomes.

1.4 Hypothesis

This report proceeds on the hypothesis that certain categories of issues are more likely than others to influence Commission decisions, but that overall intervenor influence remains constrained by structural features of the CNSC licensing framework.

Technical safety concerns, including safety analysis completeness, emergency preparedness, and radioactive waste management, are expected to be acknowledged but frequently deferred to later licensing stages. In contrast, governance and procedural concerns, such as licence duration and review requirements, may be more likely to produce observable regulatory responses.

This expectation reflects the influence of key institutional features within the CNSC framework, including staged licensing, scope limitations, and reliance on CNSC staff assessments as the primary source of technical judgment. These features shape how concerns are evaluated and often allow licensing decisions to proceed while unresolved issues are assigned to future regulatory processes.

By examining the Darlington, Pickering, and Point Lepreau cases, this report identifies recurring patterns in how intervenor concerns are treated and evaluates whether public participation functions as a mechanism of influence, a mechanism of oversight, or primarily as a mechanism of record building.

2. Background: The CNSC Licensing Process

2.1 The Canadian Nuclear Safety Commission and Its Mandate

The Canadian Nuclear Safety Commission (CNSC) is Canada's federal regulator responsible for overseeing the use of nuclear energy and materials to protect health, safety, security, and the environment. Established under the Nuclear Safety and Control Act, the CNSC regulates nuclear activities across the lifecycle of facilities, including construction, operation, and radioactive waste management. Its mandate ensures that Canada meets international obligations related to the peaceful use of nuclear technology.

In carrying out its responsibilities, the CNSC evaluates licence applications submitted by proponents such as Ontario Power Generation and NB Power. CNSC staff conduct technical assessments of these applications and present their findings to the Commission through Commission Member Documents (CMDs). These documents provide the primary technical basis used by the Commission when making licensing decisions.

2.2 The Nuclear Licensing Process in Canada

Nuclear facilities in Canada are regulated through a staged licensing process that reflects the lifecycle of a nuclear project. While the specific process may vary depending on the type of facility, major licensing decisions typically include environmental assessment, licence to construct, and licence to operate stages.

At each stage, proponents must demonstrate compliance with regulatory requirements related to safety, environmental protection, and security. CNSC staff review submitted information and provide recommendations to the Commission through CMDs. The Commission then conducts public hearings before issuing a Record of Decision (RoD).

Licensing decisions are often accompanied by conditions or hold points that require the proponent to complete additional work before proceeding to later stages. These conditions commonly relate to safety analysis, environmental monitoring, or emergency preparedness. Compliance with these requirements is primarily verified through CNSC staff review, which means that the information used to assess and release hold points may not return to the Commission through a new public hearing process. This can make it difficult for intervenors and members of the public to determine what technical or safety information was submitted, what CNSC staff found satisfactory, and the basis on which hold point release decisions were made.

A defining feature of this process is staged decision-making. Rather than requiring all uncertainties to be resolved at once, issues may be addressed incrementally across licensing phases. While this structure provides flexibility, it also allows concerns raised during hearings to be deferred to future stages rather than resolved at the time of approval.

2.3 Role of Public and Civil Society Interventions

Public participation is a formal component of CNSC licensing hearings. Individuals, community groups, and non-governmental organizations, referred to as intervenors, are invited to submit written or oral interventions. These submissions allow civil society actors to introduce additional evidence, challenge technical conclusions, and raise concerns that may not be fully addressed in proponent or staff materials.

Intervenors commonly raise issues related to emergency planning, reactor safety, radioactive waste management, and environmental impacts. Organizations such as the Canadian Environmental Law Association (CELA) play a recurring role in reviewing technical documentation and identifying areas of uncertainty or incomplete analysis.

Although the Commission is required to consider all interventions, the degree to which these submissions influence final decisions remains vague. Commission outcomes frequently rely on CNSC staff assessments as the primary basis for determining whether regulatory requirements have been met. As a result, intervenor critiques may be acknowledged but not incorporated into final decision outcomes.

2.4 Decision-Making and Public Participation Outcomes

Following public hearings, the Commission issues a Record of Decision outlining its conclusions and the rationale for granting, denying, or modifying a licence. These decisions

summarize key issues considered during the hearing process, including those raised by intervenors.

In many cases, the Commission concludes that CNSC staff assessments provide sufficient assurance of safety and environmental protection. Intervenor concerns are therefore often acknowledged but shifted into licence conditions, continued oversight, hold points, or future licensing stages rather than producing immediate changes to approval decisions. This is significant because once concerns are moved into post-hearing staff oversight, the public may have limited ability to follow whether or how those concerns are ultimately resolved.

This pattern creates a gap between procedural participation and observable influence. While intervenor concerns are recorded and referenced, the reasoning behind how those concerns affect final decisions is not always clearly articulated. As a result, participation supports transparency in the process but does not consistently produce measurable changes in regulatory outcomes.

This reliance on staff assessments reinforces the central role of internal technical authority within the CNSC framework and limits the ability of external actors to alter decision outcomes, even when concerns are thoroughly supported by evidence.

3. Scope of Review

This report reviews three CNSC licensing cases involving Darlington, Pickering, and Point Lepreau. These cases were selected because they involve major licensing decisions, substantial participation by civil society intervenors, and sufficient publicly available documentation to allow comparison across regulatory stages. Each case represents a different licensing context, allowing the report to examine how intervenor concerns were addressed across construction, operational, and licence renewal decisions.

The review focuses on formal regulatory documents associated with each case, including CNSC Staff Commission Member Documents, intervenor submissions, and Records of Decision issued by the Commission. For the Darlington case, the review also considers the role of the environmental assessment framework, particularly where questions were raised regarding the applicability of earlier environmental approvals. Together, these materials make it possible to trace how specific issues were introduced, discussed, and resolved within the regulatory process.

Across the three cases, recurring areas of issue were examined, including emergency planning, safety analysis, radioactive waste management, environmental protection, and public participation. Particular attention was given to whether these issues were acknowledged in Commission decisions, deferred to later licensing stages, or addressed through procedural mechanisms such as hold points and licence conditions.

The purpose of this review is not only to identify whether intervenor concerns appear in the regulatory record, but to assess whether those concerns produced observable effects on decision-making. By comparing how similar issues were treated across multiple cases, the report identifies recurring patterns in how public participation operates within the CNSC licensing framework and whether intervenor input results in meaningful regulatory outcomes.

4. Case Study 1: Darlington Nuclear Generating Station

4.1 Project Overview

The Darlington New Nuclear Project involved an application by Ontario Power Generation (OPG) for a licence to construct a small modular reactor (SMR) at the existing Darlington nuclear site in Ontario. The application was significant because it concerned a first-of-a-kind reactor design that had not yet been deployed in Canada or anywhere else in the world. This raised important regulatory questions regarding safety analysis, environmental assessment applicability, emergency preparedness, and whether the CNSC's staged licensing process was adequate for assessing unresolved issues associated with an unproven technology.

The CNSC conducted a public hearing process to evaluate the licence to construct applications. CNSC staff prepared technical assessments addressing reactor design, safety systems, environmental protection, and emergency planning requirements. Civil society organizations, including CELA, participated as intervenors and raised concerns regarding the adequacy of safety analysis, the readiness of emergency planning measures, radioactive waste management responsibilities, and the continued reliance on earlier environmental assessment approvals.

These issues were not limited to technical design details but reflected broader concerns about the timing of regulatory approval and the treatment of unresolved risks within the staged licensing framework. Intervenor submissions emphasized that certain safety and environmental uncertainties remained outstanding at the time of the licence to construct decision.

The Commission ultimately approved the licence to construct while acknowledging the concerns raised during the hearing process. However, rather than resolving these issues at the time of approval, several concerns were deferred to future licensing stages or addressed through planned regulatory follow-up. This decision established the foundation for the issue-specific patterns examined in the following sections.

4.2 Emergency Planning

Emergency planning was a major issue raised during the Darlington licensing process and provides a clear example of how intervenor concerns were acknowledged but not incorporated into binding regulatory requirements at the licence to construct stage. CNSC staff stated that

detailed nuclear emergency planning was not required during construction because no nuclear fuel would be present at the site. Instead, staff indicated that detailed emergency preparedness requirements, including evacuation time estimates, emergency planning zone sizing, and revisions to the Provincial Nuclear Emergency Response Plan, would be addressed at later stages of the licensing process.

The Record of Decision adopts the same staged approach. The Commission concluded that Ontario Power Generation had adequate emergency management and fire protection programs in place for construction activities and emphasized that a radiological emergency could not originate from the Darlington New Nuclear Project during this phase. Existing regional emergency infrastructure, including municipal and provincial emergency services, was identified as sufficient to support construction activities. At the same time, the Commission stated that final evacuation modelling and emergency planning zone development would be completed during future licensing applications, particularly before authorization to operate.

Intervenors challenged this staged treatment of emergency preparedness. CELA requested that the licence to construct not be issued until evacuation time estimates were updated using recent population data and the Provincial Nuclear Emergency Response Plan was revised. Northwatch and Ecojustice argued that emergency planning could not be meaningfully assessed while accident scenarios and reactor design information remained incomplete. Their broader argument was that emergency preparedness should be treated as an integral component of licensing rather than being deferred to later operational stages.

Despite these concerns, the Commission did not impose any licence conditions requiring updated evacuation modelling or completion of emergency planning revisions before construction approval. No regulatory hold points were specifically tied to emergency preparedness readiness. Instead, emergency planning requirements were deferred to future regulatory processes and later licensing stages.

This outcome reflects a pattern of procedural acknowledgement combined with deferral. Emergency planning was recognized as important and repeatedly referenced in the decision-making process, but it did not result in substantive changes to the licensing outcome. By defining emergency preparedness as phase-specific and postponing detailed evaluation to later stages, the Commission insulated the licence to construct decisions from intervenor demands for a more complete emergency planning record approval, reinforcing the role of staged licensing as a mechanism for managing unresolved risks.

4.3 Safety Analysis and Completeness of Safety

Safety analysis was also a central issue in the Darlington licensing materials and provides one of the clearest examples of the Commission’s reliance on staged licensing. CNSC staff concluded that Ontario Power Generation had met regulatory requirements for a licence to

construct while also noting that additional information would be required as the BWRX-300 safety analysis progressed. Staff described the safety case as iterative and identified several areas requiring further work, including probabilistic safety assessment uncertainty analysis, seismic and wind hazards, climate-related information, and broader hazard analysis.

The Record of Decision reflects the same approach. The Commission determined that the preliminary safety analysis contained sufficient detail to support construction activities and concluded that the proposed design could meet applicable safety goals at this stage. At the same time, the decision emphasized that the design and safety analysis would continue to evolve and that additional submissions would be required before regulatory hold points could be removed or future operating licences considered.

Intervenors directly challenged this staged approach. CELA requested that the licence to construct not be issued until the probabilistic safety assessment included full uncertainty, sensitivity, and importance analyses. Northwatch raised concerns about the maturity of the reactor design and the adequacy of accident modelling, while Ecojustice argued that phased licensing fragmented risk assessment and prevented a comprehensive evaluation of safety and environmental impacts. Additional intervenors identified unresolved accident scenarios, incomplete safety system information, and the deferral of essential technical evidence to later stages.

Despite these concerns, the Commission accepted CNSC staff's conclusion that the available information was sufficient to proceed with construction. Intervenor requests to delay licensing until the safety case was more complete were not adopted. Instead, unresolved safety questions were addressed through future submissions, staff review, and regulatory hold points. This outcome reflects procedural uptake rather than substantive change, as uncertainty was treated as an expected feature of the licensing process rather than a barrier to approval.

4.4 Waste Management and Decommissioning

Waste management provides one of the clearest examples of scope limitation through staged licensing in the Darlington record. CNSC staff defined the waste management safety and control area as covering waste-related programs during construction and preliminary decommissioning planning. Staff concluded that Ontario Power Generation had met regulatory requirements and that waste management and decommissioning plans would be updated as additional information became available. Additional hazardous waste information was required before removal of the first regulatory hold point, and licence conditions required the licensee to implement and maintain waste management and preliminary decommissioning programs.

The Record of Decision reflects the same framing. The Commission stated that no radioactive waste would be generated during the licence to construct phase and that construction waste would consist primarily of conventional materials such as excavated soil, rock, and

non-nuclear hazardous substances. Long-term radioactive waste management was explicitly deferred to future licensing stages, particularly the licence to operate. The Commission therefore concluded that waste management arrangements were adequate for the construction phase.

CELA requested that the licence not be issued until SMR-specific radioactive waste plans were provided and objected to treating long-term waste management as outside the scope of the construction decision. Northwatch argued that the absence of defined waste pathways created uncertainty because waste generation is an inevitable consequence of construction decisions. Ecojustice framed waste management as a lifecycle issue, arguing that construction approval commits the project to future waste obligations. Ole Hendrickson raised specific concerns about the preliminary decommissioning plan, including the absence of a defined decommissioning strategy and uncertainty in cost and timeline estimates.

These concerns did not alter the licensing outcome. The Commission acknowledged waste management as relevant but treated long-term radioactive waste planning as outside the scope of the licence to construct. The result reflects scope exclusion rather than substantive uptake. While near-term construction waste was addressed through existing regulatory programs, long-term waste governance was deferred to future licensing stages.

4.5 Reactor Design, Shutdown Systems, and SMR Novelty

The Darlington materials show that public intervenors repeatedly emphasized the novelty of the BWRX-300 and characterized it as a first-of-a-kind reactor design. However, the CNSC did not treat this novelty as a reason to require a more complete design before granting a licence to construct. Rather, the CNSC framed design incompleteness as a normal and manageable feature of staged licensing, effectively treating unresolved safety and design details as acceptable at the construction stage so long as they could be deferred to future submissions, hold points, and CNSC staff oversight.

Specific concerns about reactor design maturity and shutdown system independence were raised. CELA requested that the licence not be issued unless two truly independent shutdown systems were demonstrated. Northwatch and Ecojustice raised broader concerns about approving construction while key safety systems remained under development. Dr. Sunil Nijhawan argued that the design lacked several fundamental safety features, including independent shutdown systems, robust emergency core cooling capability, adequate containment volume, and reliable backup power, and recommended that construction not proceed until a more complete and independently reviewed design was available.

The Record of Decision did not adopt these requests as binding requirements. Shutdown system independence and other design concerns were treated as technical matters for continued review within the broader safety analysis framework, and no licence condition required full demonstration of design completeness before construction. This outcome reflects technical

acknowledgement without binding uptake, as design novelty was recognized but did not alter the licensing threshold.

More broadly, SMR-specific risks were acknowledged but segmented across licensing stages rather than resolved before approval. Design uncertainties were treated as preliminary matters to be addressed through future submissions and regulatory oversight, reinforcing the role of staged licensing as a mechanism for managing uncertainty rather than requiring full technical resolution before construction.

4.6 Climate Change, External Hazards, and the Environmental Assessment Context

The Darlington case raises an additional and distinctive set of concerns related to the continued applicability of the project's 2011 environmental assessment and the treatment of climate change and external hazards within that framework. Together, these issues reveal a further dimension of the deferral pattern observed across the licensing record: rather than re-examining integrated risk as conditions changed, the regulatory process converted the EA's original cumulative function into a narrower retrospective exercise focused on whether earlier conclusions remained technically bounded.

The 2011 Joint Review Panel environmental assessment was conducted without a specific reactor technology selected. OPG relied on a plant parameter envelope designed to encompass multiple possible designs, and the Panel accepted that design-specific risks would be addressed through subsequent licensing. When the BWRX-300 was selected, the regulatory question therefore shifted from evaluating the project's environmental acceptability as a whole to determining whether the new technology fell within the bounds of conclusions already reached in 2011. This shift is not merely procedural, but substantive. Where the 2011 EA treated safety, proximity, emergency planning, and waste as interlinked and cumulative concerns to be assessed together, the 2023 applicability process fragmented these issues across separate CMDs, licensing documents, and future commitments, evaluating them on an issue-by-issue basis rather than cumulatively.

The consequences of this approach are most visible in the treatment of radioactive waste. CNSC staff acknowledged that the BWRX-300 generates a higher source term for alpha-emitting radionuclides than the technologies modelled in the 2011 EA. For an intermediate-level waste pool fire scenario, the estimated dose to workers and members of the public under the BWRX-300 is approximately ten times greater than the dose estimated for the equivalent scenario in the original EA. Despite this significant deviation from EA assumptions, CNSC staff concluded that the EA's findings remained valid. The basis for this conclusion was not that the EA predictions still held, as they demonstrably did not for this scenario, but rather that the applicable regulatory dose limits continued to be met. In other words, exceedance of EA estimates was treated as acceptable so long as outcomes remained within regulatory thresholds. The environmental assessment shifted from a predictive baseline into a threshold-compliance

instrument, measuring outcomes against dose criteria rather than the accuracy of original findings.

The 2011 EA also contains commitments that were not carried forward in any visible way into the 2023 review. The Joint Review Panel explicitly recommended that, before construction, the CNSC require OPG to evaluate the cumulative effect of a common-cause severe accident involving all nuclear reactors in the site study area. This recommendation directly addressed the risk implications of building a new reactor adjacent to the existing Darlington Nuclear Generating Station. In the 2023 applicability process and subsequent licensing record, this cumulative accident scenario was not resolved. Accidents at the neighbouring Darlington NGS were screened out of the BWRX-300 hazard analysis based on low probability, and multi-unit accident scenarios were not evaluated in the manner the JRP had contemplated. The original EA's precautionary framing of proximity risk was therefore quietly set aside, absorbed into the same boundedness logic that governed the rest of the applicability review.

CELA challenged OPG's position that climate change had no nuclear safety implications, arguing that extreme weather events could affect infrastructure reliability, emergency response capacity, and evacuation feasibility in ways that design-phase assumptions might not capture. Northwatch linked climate-related risks to gaps in emergency planning. Ecojustice characterized climate change as a system-level safety concern rather than a discrete engineering variable. The Nuclear Transparency Project argued more broadly that the fragmentation of environmental review across licensing stages had left key questions about radionuclide emissions, air quality modelling, thermal impacts on Lake Ontario, and liquid effluent discharges unresolved, and that intervenors could not meaningfully assess project impacts without those studies being completed.

None of these concerns produced a substantive regulatory response. Climate risks were addressed within existing design assumptions, and no independent climate risk assessment or emergency stress-testing requirement was imposed before approval. The boundedness determination proceeded on the basis that the 2011 EA's evaluation criteria remained applicable to the BWRX-300, even where specific parameters had changed materially.

The environmental assessment’s original function as a comprehensive, integrated review of cumulative environmental and safety risk was replaced by a retrospective compliance routine that asked only whether earlier conclusions could still be technically defended. In doing so, it reproduced at the EA level the same structural feature observed throughout the licensing record: the displacement of unresolved risk from the current decision to later stages, future oversight, or ongoing regulatory management, without requiring that the underlying questions be answered before authorization to proceed.

4.7 Hold Points and the Structure of Regulatory Control

The Darlington Record of Decision includes three key regulatory hold points tied to installation of the reactor building foundation, installation of the reactor pressure vessel, and fuel-out commissioning. These hold points prevent progression to major technical milestones until additional design, safety, emergency planning, and operational requirements are accepted. However, the decision itself does not determine when these hold points will be satisfied. Their verification and release are delegated to CNSC staff through subsequent technical review.

This structure is significant for understanding the intervenor influence. Hold points demonstrate that unresolved technical and safety issues remain at the time of licensing and that the regulatory framework formally recognizes those uncertainties. However, the release of hold points occurs through staff-level review rather than through a return to the Commission for a new public decision. As a result, even when intervenor concerns overlap with the subject matter of hold points, the authority to resolve those concerns remains primarily administrative.

A recent correspondence exchange between CELA and the CNSC regarding the Darlington New Nuclear Project illustrates the difficulty of following this process after the hearing. CELA requested the licence as issued following the hearing, the Licence Conditions Handbook, more specific descriptions of the regulatory hold point preconditions that CNSC staff must be satisfied with before release, and any documentation regarding hold point releases to date. CNSC staff responded that removal of a regulatory hold point requires OPG to demonstrate that all commitments related to the hold point have been completed to the satisfaction of CNSC staff, with details documented under licence condition 15.3 of the Licence Conditions Handbook. However, the response did not clearly identify the specific technical or safety information required, what had been submitted or found satisfactory, or the basis on which hold point release decisions would be made. This illustrates how difficult it can be for members of the public and intervenors to follow unresolved issues once they move from the public hearing record into post-hearing staff oversight.

This reinforces a broader pattern observed throughout the Darlington case: civil society interventions may influence the sequencing of information and regulatory requirements, but they rarely reshape the core licensing authorization itself. Where unresolved issues are moved into hold points, the public may have limited ability to verify whether concerns raised during the hearing are later resolved in a meaningful or transparent way.

4.8 Findings from the Darlington Case

The Darlington New Nuclear Project provides strong evidence that civil society participation is formally incorporated into the CNSC process, but weak evidence that it meaningfully changes the outcome of major licensing decisions. Intervenor concerns were clearly present throughout the record. Emergency planning, probabilistic safety assessment

maturity, radioactive waste management, shutdown system independence, climate risk, and the continued applicability of the environmental assessment were all repeatedly raised by intervenors and explicitly acknowledged by CNSC staff and the Commission. On its face, this suggests a process that is open and responsive.

However, the record shows a different pattern. Acknowledgement did not translate into a meaningful constraint on the licence to construct. None of the intervenor requests resulted in denial of the licence, a new site-specific licence condition directly tied to an intervenor's request, or a regulatory hold point explicitly attributed to public intervention. Where influence is visible, it appears mainly through procedural uptake, deferral, or exclusion. In practice, intervenor concerns were recognized, but they were rarely allowed to disrupt the licensing trajectory.

Darlington also shows that this outcome is not accidental. It is built into the structure of staged licensing itself. Concerns about safety, emergency preparedness, waste, and environmental uncertainty were not rejected outright. Instead, they were absorbed into a regulatory framework that treats incompleteness as acceptable at the licence to construct stage, fragments interconnected risks across multiple future processes, and assigns responsibility for unresolved issues to later staff review. This allows the Commission to proceed with approval while postponing the moment at which difficult questions must actually be resolved.

For that reason, Darlington supports the report's central hypothesis in a particularly clear way. Public interventions do make it into the record, but they shape timing and process far more than they shape substantive outcomes. The problem is not that intervenors are ignored. The problem is that their concerns can be acknowledged, documented, and even treated as legitimate, while still being displaced to later stages where they no longer have the same capacity to constrain the original decision. In this sense, Darlington reveals a licensing system that appears participatory, but remains structurally resistant to meaningful public influence at the point of authorization.

5. Case Study 2: Pickering Nuclear Generating Station

5.1 Project Overview

The Pickering Nuclear Generating Station case provides a contrasting example of how civil society interventions interact with CNSC decision-making. Unlike Darlington, which concerns a new reactor at the licence to construct stage, Pickering involves an aging facility and a discrete regulatory decision: whether to remove a hold point — Licence Condition 16.3 — that restricted operation beyond 210,000 equivalent full-power hours, the assumed design lifetime for CANDU pressure tubes.

Ontario Power Generation sought approval to continue operating Pickering beyond this limit without refurbishment, relying on aging management programs, monitoring, and ongoing

safety assessments. The Commission had originally imposed the hold point during the 2013 licence renewal hearing, explicitly declining to delegate its removal to CNSC staff and requiring instead that the matter return to a public proceeding. Before the hold point could be removed, the Commission directed OPG to submit a revised Pickering A PSA compliant with Regulatory Standard S-294, updated PSAs for both Pickering A and B incorporating Fukushima Action Plan enhancements, and either a whole-site PSA or a methodology for developing one. These requirements reflected the Commission's own recognition that the existing safety evidence was insufficient to proceed without further scrutiny.

The May 2014 hearing, therefore, presented a specific and bounded question with immediate operational consequences. Intervenors largely argued that the hold point should remain in place until key safety and emergency planning concerns were resolved. The Commission ultimately removed the hold point, allowing continued reactor operation beyond the established design lifetime.

5.2 Reactor Aging and Operation Beyond Design Life

A central issue raised across multiple interventions was the extension of reactor operation beyond the original design lifetime. Intervenors argued that the 210,000 equivalent full-power hour limit had historically functioned as a firm safety benchmark for CANDU reactors and that exceeding it without refurbishment represented a significant departure from established regulatory practice.

The Canadian Coalition for Nuclear Responsibility emphasized that reactors reaching this limit had historically been shut down or refurbished and argued that continued operation without replacing critical components such as pressure tubes and feeder pipes increased the probability of component failure. The International Institute of Concern for Public Health similarly argued that extending operation beyond the designed limits of pressure tubes reduced safety margins and introduced additional risk.

Intervenors framed these concerns not only as technical issues but also as questions of regulatory philosophy. They argued that exceeding established design limits should trigger a precautionary approach, particularly given the potentially severe consequences of reactor failure and Pickering’s proximity to the Greater Toronto Area.

Despite these concerns, the Commission accepted CNSC staff’s conclusion that continued operation would not pose an undue risk to public health, safety, or the environment. The decision relied on evidence of inspection programs, fuel channel monitoring, aging management strategies, and ongoing operational safety assessments. This outcome demonstrates that even where intervenors challenged a core regulatory threshold such as design lifetime, the Commission relied on its judgment that ongoing management programs would be sufficient, even though historical operating limits would be exceeded.

5.3 Safety Analysis and Probabilistic Risk Assessment

Safety analysis, particularly probabilistic safety assessment, was one of the most contested issues in the 2014 Pickering hearing, and the Record of Decision reveals a more complicated picture than a simple staff endorsement.

CNSC staff confirmed that OPG had submitted updated PSAs for both Pickering A and B, incorporating Fukushima Action Plan enhancements and meeting the requirements of S-294. For Pickering B, the results were unambiguous: both core damage frequency and large release frequency met safety goal limits across all hazard categories, individually and in aggregation. For Pickering A, however, the picture was more qualified. Staff confirmed that CDF and LRF met safety goal limits when calculated individually for internal and external events, but the aggregated LRF calculated by simple summation for Pickering A was noted to be slightly above the OPG safety goal limit, primarily due to conservative assumptions in the Fire PSA. Staff also acknowledged that not all Fukushima Action Plan enhancements had yet been credited in the PSA due to a lack of validated models, and that further reductions were expected to be achieved only once those enhancements were fully incorporated. This process was not expected to conclude for approximately two more years.

Several intervenors challenged these results directly. Submissions argued that PSA results showed Pickering A exceeded both core damage frequency and large release frequency limits under fully S-294-compliant analyses, and that conservative bounding assumptions were being dismissed rather than applied. Greenpeace Canada argued that the probability of a large accidental radioactive release causing offsite damage remained unacceptably high. Intervenors also raised the absence of an independent third-party review of the PSA results, and the fact that summary reports had been published only one week before the public hearing, giving the public and intervenors insufficient time to assess their contents. The Commission acknowledged the timing concern and granted intervenors an additional week to submit supplementary comments, a procedural concession that reflected the legitimacy of the concern, even if it did not alter the substantive outcome.

Despite these acknowledged limitations, the Commission accepted CNSC staff's conclusion that the PSA evidence was sufficient to support removal of the hold point. The Commission did not require the Pickering A LRF issue to be resolved before proceeding, nor did it require the outstanding Fukushima enhancements to be credited in the PSA before authorizing continued operation. Instead, it directed OPG to submit a detailed risk improvement plan and required annual reporting on the status of improvements. These were ongoing oversight measures rather than preconditions for the decision itself.

5.4 Large Release Risk, Cumulative Risk, and External Hazards

The question of cumulative, whole-site risk was explicitly before the Commission in 2014, having been raised as a Commission-directed requirement at the 2013 licence renewal. The Commission had required OPG to develop either a whole-site PSA or a methodology for one before the hold point could be removed.

OPG responded by providing a methodology for a whole-site PSA rather than a completed assessment. CNSC staff acknowledged that the detailed review of the submitted PSAs would not be completed until June 30, 2015, more than a year after the hold point removal decision. The Commission accepted this, directing OPG to continue developing the whole-site PSA methodology and requiring annual updates on its progress, but not conditioning the hold point removal on the completion of that work.

Intervenors, particularly Greenpeace Canada, challenged the per-unit basis of risk assessment directly, arguing that evaluating reactors individually understated the total risk from the entire Pickering site and created an effective regulatory loophole that allowed the multi-reactor facility to appear safer than a proper aggregation of site-wide risk would show.

The Commission raised the loophole argument directly with CNSC staff during the hearing. Staff responded that they were not aware of any such loophole, suggested the intervenor had taken a staff comment on PSA calculation methodology out of context, and reiterated that safety goals are designed on a per-unit basis precisely to ensure that individual unit designs are as safe as possible. The Commission accepted this response without requiring further elaboration or independent verification of the claim. No additional analysis of site-wide aggregated risk was required as a condition of hold-point removal.

This exchange is revealing for two reasons. First, it shows the Commission actively defending the per-unit methodology rather than treating cumulative risk as an open regulatory question, despite having itself directed OPG to develop a whole-site PSA at the previous hearing. Second, it illustrates how intervenor arguments that directly mirror prior Commission concerns can be reframed, once staff has responded, as matters already considered and resolved. The concern did not disappear from the record. It was answered, and the answer was accepted.

The result was that the whole-site risk assessment remained incomplete at the time of the decision, the per-unit basis of safety goal evaluation was preserved, and intervenor arguments about cumulative risk were recorded but did not alter the regulatory outcome. The Commission directed continued development of the whole-site PSA methodology and required annual progress updates, prospective oversight measures that acknowledged the gap without treating it as a barrier to authorization.

5.5 Emergency Planning and Public Protection

Emergency planning was another major area of concern, particularly given Pickering's location near one of the most densely populated urban regions in Canada. Intervenors argued that evacuation planning is inadequate, that existing emergency plans are not designed for severe or multi-unit accident scenarios, and that public health protection measures are insufficient for a reactor operating beyond its design lifetime.

CELA emphasized that Pickering no longer meets modern siting expectations due to its proximity to a large population centre and argued that evacuation under severe accident conditions may not be feasible. They further argued that emergency preparedness does not adequately account for Fukushima-scale events and recommended that the hold point not be removed until emergency readiness is demonstrably improved. Additional recommendations included expanded emergency planning zones and the broader pre-distribution of potassium iodide tablets to nearby populations.

The International Institute of Concern for Public Health similarly argued that evacuation and emergency response plans are inadequate given the surrounding population density. These concerns were framed as central to the licensing decision, not as secondary operational matters, because emergency preparedness directly determines whether large-scale public protection is realistically achievable.

Despite these concerns, the Commission proceeded with the removal of the hold point. There is no indication that emergency planning deficiencies resulted in new licence conditions tied specifically to continued operation, nor that improvements in evacuation readiness were required before approval. The outcome suggests that, as in the Darlington case, emergency planning concerns were acknowledged but not treated as determinative at the decision stage.

5.6 Regulatory Standards, Evidence, and Precaution

Several intervenors argued that Ontario Power Generation had not met the regulatory requirements necessary for hold-point removal. Greenpeace Canada asserted that OPG had failed to submit a required action plan to reduce risk below safety limits and therefore had not fulfilled the Commission's conditions. Other submissions similarly argued that the evidentiary basis supporting continued operation was insufficient, noting that key safety studies were incomplete and that important analyses had not been provided at the time of the decision.

Concerns were also raised about regulatory independence, transparency, and the role of CNSC staff. Some intervenors argued that CNSC staff appeared to support the licensee's safety case rather than critically evaluate it, and called for greater independent scrutiny by the Commission itself. Others noted that certain safety studies were deferred for submission to CNSC staff without public review, raising concerns that key evidence was being assessed outside

the visible hearing process. These arguments framed regulatory transparency and evidentiary sufficiency as central to the legitimacy of the decision.

A related set of interventions focused on the broader question of precaution and justification. Several submissions argued that continued operation beyond design life was not necessary and, therefore, exposed the public to avoidable risk. These arguments emphasized that when safety limits are uncertain or contested, a precautionary approach should apply. Some intervenors framed this position as a legal obligation under the Nuclear Safety and Control Act, arguing that the CNSC has a responsibility to require risk reduction or closure where safety margins are exceeded.

Despite these challenges, the Commission accepted CNSC staff's assessment that continued operation would not pose an undue risk to public health, safety, or the environment. There is no indication that the Commission required additional independent review, strengthened evidence standards, or adopted precautionary restrictions as conditions of approval. As in the Darlington case, procedural and normative challenges to the adequacy of the evidence were acknowledged but did not result in changes to the final regulatory outcome.

5.7 Findings from the Pickering Case

The 2014 Pickering decision strengthens the patterns observed in the Darlington case while also revealing a distinctive feature of hold-point decisions: the tension between the Commission's own prior requirements and its willingness to accept partial satisfaction of those requirements as a basis for proceeding.

The Commission had explicitly refused to delegate hold-point removal to CNSC staff in 2013, requiring a public proceeding and setting out specific evidentiary conditions. By May 2014, several of those conditions had been met, but others remained outstanding. Specifically, the aggregated Pickering A LRF slightly exceeded safety targets, outstanding Fukushima enhancements had not yet been credited in the PSA, and the whole-site PSA remained a methodology rather than a completed assessment. The Commission proceeded nonetheless, directing future reporting and risk improvement planning rather than requiring resolution before authorization.

Intervenor concerns on PSA completeness, large release risk, cumulative site risk, emergency planning, and the adequacy of the evidentiary basis were acknowledged and recorded, but did not result in the hold point being maintained, new preconditions being imposed, or safety thresholds being revised. The Commission's response was to require ongoing monitoring, annual reporting, and continued work on the whole-site PSA, all of which were conditions of continued oversight rather than conditions of approval.

The pattern of influence is therefore more constrained in Pickering than even in Darlington. Whereas Darlington demonstrates deferral of intervenor concerns to future licensing stages, Pickering demonstrates something closer to conditional acceptance. The Commission moved forward despite identified gaps, relying on prospective improvements and staff oversight to address outstanding issues. In this sense, Pickering reveals the limits of intervention influence most sharply at precisely the moment when that influence might be expected to matter most, a discrete, time-sensitive decision with immediate operational consequences.

6. Case Study 3: Point Lepreau

6.1 Project Overview

The Point Lepreau Nuclear Generating Station case reinforces the broader pattern observed in the Darlington and Pickering cases. Although the central issue differed, focusing primarily on the length of the operating licence and its implications for public participation, accountability, and future regulatory oversight, the case similarly shows that civil society interventions had limited influence on the final licensing outcome.

In 2022, New Brunswick Power applied to renew the operating licence for Point Lepreau for a period of 25 years. CNSC staff supported licence renewal but recommended a shorter 20-year term. Intervenors, however, overwhelmingly opposed long licence periods and argued that extended licence terms would reduce opportunities for public engagement and weaken accountability. The Commission ultimately rejected both the 25-year request and the 20-year staff recommendation, instead granting a 10-year licence with a required mid-point public proceeding. This makes Point Lepreau the clearest example in this report of intervenor input being reflected in the final decision.

6.2 Licence length, Public Participation, and Future Developments

Licence length was the central issue in the Point Lepreau hearing. NB Power framed its 25-year request as a matter of stability, alignment with the remaining life of the facility, and consistency with international practice. CNSC staff similarly argued that effective regulatory oversight would continue regardless of licence duration.

Across submissions, long licence terms were treated not as neutral administrative choices but as decisions with direct consequences for public participation and accountability. Several groups argued that shorter licences would preserve regular hearing opportunities, allow changing conditions to be revisited, and maintain meaningful Commission oversight. The Canadian Nuclear Workers' Council, the Nuclear Transparency Project, CCNR, EOS Eco-Energy, and others all linked shorter licence terms to continued public engagement.

CELA and CRED-NB added a further dimension by arguing that long licence terms were inappropriate given future uncertainty at the site, particularly planned small modular reactor development and fuel reprocessing proposals. They argued that these future developments could not be cleanly separated from the existing licensing basis and recommended that the licence include a reopening trigger if SMR licensing moved forward. More broadly, intervenors repeatedly connected shorter licence terms to regulatory flexibility, public accountability, and the need to revisit decisions as conditions changed.

6.3 Indigenous Participation and Intergenerational Accountability

A distinctive feature of the Point Lepreau case was the explicit framing of licence length as an issue of Indigenous participation and intergenerational accountability. The Passamaquoddy Recognition Group Inc. opposed a long licence term and recommended a three-year licence, grounding that position in Wabanaki legal principles and the importance of regularly revisiting decisions affecting land and communities over time. The PRGI's submission distilled this position with striking directness: "Fourteen Thousand Years says '3 Year License'", invoking the duration of Passamaquoddy presence on the land as the frame within which a 25-year regulatory closure should be understood. From that perspective, a generation-long licence was not a neutral administrative choice but a decision to silence Indigenous voices for a period that, measured against the depth of that relationship to the land, was anything but short.

This intervention reframed licence duration as more than an administrative matter. It raised the concern that a long licence would limit the participation of future generations and foreclose opportunities to revisit regulatory decisions in a meaningful way. The Commission engaged with this directly during the hearing, asking how a 20- or 25-year licence would promote reconciliation. CNSC staff maintained that licence length did not itself determine progress toward reconciliation. The PRGI representative pushed back, noting that the group had not been part of the decision-making process that led to the request for a 25-year licence, and that such a term would represent losing the voice of a generation in the discussion.

The Commission's response acknowledged the force of this concern. Its decision explicitly stated that an ongoing dialogue involving licensees, CNSC staff, the Commission, and Indigenous Nations and communities was essential to further reconciliation and to build and maintain trust. In granting a 10-year licence with a required mid-point public proceeding, the Commission directly linked its decision to the need to preserve opportunities for Indigenous Nations and communities to be heard. This makes Point Lepreau unusual within the broader case record, as it is one of the few instances where the Commission explicitly connects an intervenor's framing, including an Indigenous governance framing, to the structure of its final decision.

6.4 Commission Response and Procedural Uptake

The Commission's decision reflects a clear, though limited, uptake of intervenor concerns. Although it accepted that NB Power maintained mature programs and adequate performance, it rejected both the 25-year request and the 20-year staff recommendation. Instead, it issued a 10-year licence aligned with the 10-year periodic safety review interval.

Most importantly, the Record of Decision explicitly linked this outcome to the concerns raised by intervenors. The Commission stated that maintaining opportunities for intervenors to voice their concerns was necessary to sustain dialogue with the public and Indigenous Nations and communities. It also required a mid-point public proceeding in 2027 to ensure additional participation during the licence term.

This is a significant departure from the patterns observed in Darlington and Pickering. Rather than deferring concerns or treating them as outside the scope of the decision, the Commission incorporated them directly into the procedural structure of the licence. Licence duration and the mid-term hearing became mechanisms for preserving oversight and participation.

6.5 Findings from the Point Lepreau Case

The Point Lepreau case provides the clearest example in this report of intervenor influence affecting a CNSC outcome. Unlike in Darlington and Pickering, where intervenor concerns were largely deferred, excluded, or acknowledged without changing the result, Point Lepreau shows direct procedural uptake. Intervenor concerns about licence length, accountability, future uncertainty, and public participation were reflected in both the shortened licence term and the requirement for a mid-point public hearing.

At the same time, the influence observed here remained limited in scope. It did not alter technical safety standards, impose new engineering requirements, or change substantive operating conditions. Instead, it affected the structure of oversight. Civil society interventions appear to be the most effective when they target governance and process rather than technical sufficiency alone.

Point Lepreau, therefore, strengthens the broader pattern emerging across the cases. Intervenors can influence CNSC decisions, but that influence is more likely to appear in procedural design than in substantive licensing constraints. In this case, the Commission responded to concerns about public participation and accountability by restructuring the licence. Even so, the change remained procedural rather than technical.

7. Cross-Case Analysis: The Role and Limits of Intervenor Influence in CNSC Decision-Making

7.1 Overview

Across the Darlington, Pickering, and Point Lepreau cases, a consistent pattern emerges regarding the role of civil society interventions, particularly those advanced by the Canadian Environmental Law Association (CELA), in CNSC decision-making. Intervenor concerns are consistently recorded and acknowledged, but their influence on final licensing outcomes is uneven and highly constrained.

The clearest cross-case finding is that intervenor influence is predominantly procedural rather than substantive. Intervenors are able to shape how issues are discussed, when they are revisited, and how regulatory processes are structured. They are far less successful, however, in altering core licensing outcomes, technical determinations, or binding safety requirements.

7.2 Acknowledgement Without Substantive Uptake

In all three cases, CELA's concerns are explicitly recognized within the Commission's reasoning. Issues raised by CELA, including emergency planning, safety analysis completeness, waste management, and public participation, appear consistently in staff documents and Records of Decision. However, this acknowledgement rarely translates into binding regulatory outcomes.

In Darlington, CELA raised concerns about the incompleteness of the probabilistic safety assessment, the adequacy of emergency planning at the licence to construct stage, radioactive waste management, and the continued applicability of the earlier environmental assessment. These concerns were explicitly referenced in the Record of Decision and treated as relevant to regulatory oversight. However, they were deferred to future licensing stages, were not tied to new licence conditions, and were not used to delay or constrain the licence to construct.

In Pickering, CELA's intervention focused heavily on emergency planning inadequacy, evacuation feasibility in a densely populated region, and the lack of preparation for severe or multi-unit accidents. These concerns were framed as fundamental public safety issues and directly linked to the decision on whether to remove the hold point. Despite this, the Commission accepted CNSC staff's assessment, removed the hold point, and did not impose new conditions tied to emergency preparedness.

Across both cases, CELA's concerns are clearly visible in the record and acknowledged as legitimate subjects of regulatory consideration, but they do not alter the core decision outcome. This establishes a consistent pattern of acknowledgement without substantive uptake.

7.3 Structural Limits on Intervenor Influence

The Darlington and Pickering cases show that the limited impact of intervenor participation is not simply a matter of individual decisions. It reflects structural features of the CNSC licensing framework itself.

One of the clearest constraints is staged licensing. In Darlington, major issues raised by CELA were repeatedly deferred. Safety analysis was directed to future submissions and hold points, emergency planning was postponed to later licensing stages, and radioactive waste management was treated as outside the scope of the construction decision. In Pickering, intervenors similarly argued that safety studies were incomplete and that the evidence base was insufficient to justify continued operation beyond the original design lifetime. Even so, the Commission accepted that further analysis and oversight could continue after the decision had already been made.

A related constraint is the narrowing of issues through scope limitation. In Darlington, radioactive waste management and long-term environmental risks were segmented into future processes rather than treated as integral to the construction decision. In Pickering, cumulative site risk, emergency planning deficiencies, and broader precautionary concerns did not become decision-driving factors. Intervenor submissions often took a lifecycle or systems-level view, linking construction to long-term waste, safety analysis to emergency preparedness, and site conditions to cumulative risk. In contrast, CNSC decision-making narrowed these concerns to the specific stage or issue under review.

A related pattern is that environmental impacts, particularly those affecting aquatic ecosystems and fisheries, appear more readily translated into immediate regulatory treatment through monitoring, mitigation, and follow-up requirements. By contrast, public safety and accident-risk concerns raised by intervenors are more frequently deferred to later licensing stages under the staged regulatory framework. This does not necessarily mean that the CNSC formally prioritizes fisheries over public safety. Rather, it suggests that environmental effects are often easier to manage through discrete regulatory tools, while broader safety concerns are absorbed into future submissions, staff review, hold points, or later licensing decisions.

The Commission’s reliance on CNSC staff as the primary source of technical authority further limits the force of intervenor challenges. In both Darlington and Pickering, intervenor critiques of safety analysis and risk assessment were considered and summarized, but not independently reassessed in a way that reopened the underlying technical conclusions. Staff determinations of adequacy functioned as the threshold for decision-making. This created an asymmetry in which intervenors could challenge the sufficiency of evidence, but the Commission ultimately deferred to staff expertise rather than revisiting core technical judgments.

Taken together, these features show that intervenor influence is constrained not only by the outcome of individual hearings but by a regulatory structure that enables deferral, narrows the scope of relevant issues, and anchors final decisions in internal technical assessments.

7.4 Where Influence Is Visible: Procedural Outcomes

The Point Lepreau case demonstrates that intervenor influence is not absent, but that it is concentrated in specific areas. In that case, CELA and other intervenors opposed long licence terms, raised concerns about reduced public participation and accountability, and linked licence length to future uncertainty, including possible SMR development at the site.

The Commission explicitly referenced intervenor concerns, rejected both the proponent's and CNSC staff's recommended licence lengths, imposed a shorter 10-year licence, and required a mid-term public hearing. This represents procedural uptake with measurable impact.

At the same time, the nature of that influence is important. It affected when and how decisions would be revisited, not what technical or safety standards would be applied. Point Lepreau, therefore, provides the clearest example in this report of meaningful influence, but it remains influence over process rather than substance.

7.5 Overall Findings

Across the three cases, intervenors had a clear but constrained impact on CNSC decision-making. Their submissions shaped the regulatory record, required the Commission and CNSC staff to respond to unresolved concerns, and influenced how issues were documented, deferred, or revisited. However, intervenors rarely changed the core licensing outcome. In Darlington, concerns about emergency planning, safety analysis, waste management, design maturity, and environmental assessment applicability were acknowledged but did not prevent the licence to construct from being granted. In Pickering, concerns about probabilistic safety assessment completeness, large release risk, cumulative site risk, operation beyond design life, and emergency preparedness did not prevent the hold point from being removed.

Point Lepreau provides the clearest exception. In that case, intervenor concerns about licence length, public participation, Indigenous engagement, future uncertainty, and regulatory accountability were reflected in the Commission's decision to issue a shorter 10-year licence and require a mid-point public proceeding. This shows that intervenors can influence CNSC outcomes, but that influence is more likely to appear in governance and procedural design than in technical safety determinations.

Overall, the cross-case evidence suggests four main findings. First, intervenor influence is consistent but constrained: concerns are regularly included in the record and acknowledged in Commission reasoning, but they do not usually alter approval outcomes. Second, the dominant

form of influence is procedural rather than substantive, appearing through deferral, reporting requirements, review mechanisms, or hearing opportunities. Third, substantive influence remains limited, particularly where intervenors challenge CNSC staff's technical conclusions about safety, risk, or regulatory adequacy. Fourth, governance issues offer the strongest avenue for impact because they allow intervenors to frame concerns around accountability, transparency, and continued public oversight rather than technical sufficiency alone.

8. Policy Implications

8.1 Overview

The Darlington, Pickering, and Point Lepreau cases show that the CNSC licensing process formally incorporates public participation, but the substantive impact of intervenor input remains limited and uneven. Intervenor concerns are consistently acknowledged in Records of Decision, yet they rarely translate into binding licence conditions, altered outcomes, or revised safety requirements. This has important implications for the transparency, effectiveness, and legitimacy of nuclear regulation in Canada.

8.2 Transparency in Nuclear Regulation

One of the clearest implications of this analysis is the gap between formal transparency and substantive transparency. Across all three cases, the CNSC process is procedurally open: intervenor submissions are placed on the record, hearings are publicly conducted, and Records of Decision reference concerns raised during the process. However, the reasoning by which those concerns influence outcomes is often unclear.

In both Darlington and Pickering, intervenor issues were acknowledged but not clearly resolved. Decisions proceeded despite identified uncertainties, and the rationale for rejecting, narrowing, or deferring intervenor concerns was not always explicitly articulated. As a result, participation is visible, but its practical impact is difficult to trace.

For organizations such as CELA, this limits the ability to assess whether interventions meaningfully shaped decisions, to understand how evidence was weighed relative to CNSC staff assessments, and to evaluate the consistency of regulatory reasoning across cases. Point Lepreau provides a partial exception, because the Commission explicitly linked intervenor concerns to licence length and the imposition of a mid-term public proceeding. Even there, however, the influence remained procedural rather than technical.

These cases suggest that while the CNSC process is open, it is not fully transparent in its decision logic, particularly in showing how intervenor evidence is incorporated, narrowed, or rejected.

8.3 Public Participation

The cases also highlight structural limits in how public participation operates within the CNSC licensing framework. Intervenors are able to raise complex technical and governance concerns, introduce evidence into the record, and appear directly before the Commission. However, their influence is constrained by recurring features of the regulatory process.

First, participation often occurs after key technical assessments have already been developed by CNSC staff. Intervenors therefore respond to pre-existing safety conclusions, defined project scopes, and staged licensing structures rather than shaping the foundational assumptions of the decision itself.

Second, staged licensing allows the Commission to defer intervenor concerns to future phases. In Darlington and Pickering, issues such as safety analysis, emergency planning, and waste management were repeatedly treated as matters for later stages, future oversight, or staff review. This reduced the immediate effect of participation even where the concerns raised were substantive and technically grounded.

Third, even when intervenor concerns are acknowledged, they are rarely tied directly to licence conditions, approval thresholds, or decision reversals. Point Lepreau shows that participation can influence procedural aspects of oversight, such as licence duration and review timing, but this remains the exception rather than the norm.

The result is that public participation often functions more as a mechanism for input and record-building than as a mechanism for direct decision influence.

8.4 Regulatory Accountability

The findings also raise significant concerns about regulatory accountability. In all three cases, the Commission relied heavily on CNSC staff assessments as the basis for decision-making. While this reflects the technical nature of nuclear regulation, it also means that staff conclusions largely define the boundaries of the decision, while intervenor challenges to those conclusions are rarely independently tested in a way that alters the outcome.

At the same time, issue deferral, scope limitation, and non-binding acknowledgement allow concerns to be recognized without being resolved. This creates two related accountability problems. The first is diffusion of responsibility. When concerns are deferred to later phases or ongoing oversight, it becomes more difficult to determine when they will be addressed, how they will be evaluated, and whether they will affect future decisions. The second is the absence of a strong feedback loop. There is no consistent requirement for the CNSC to demonstrate how intervenor concerns were incorporated, explain why specific recommendations were not adopted, or track how deferred issues are eventually resolved. Point Lepreau partially addressed this

through the requirement for a mid-term public proceeding, but that remains an exception and does not extend to technical or safety-related issues.

9. Policy Recommendations

Based on the findings of this analysis, the following recommendations are intended to strengthen public participation, improve transparency in CNSC decision-making, and enhance regulatory accountability within the existing licensing framework.

Recommendation 1

The CNSC should not grant licences where key safety concerns raised during hearings remain unresolved, particularly where those concerns relate to emergency planning, safety analysis completeness, reactor design maturity, waste management, or accident risk. Where such issues are central to the case of safety, they should be resolved before approval or attached to clear, enforceable pre-approval conditions that must be satisfied before the licence takes effect.

Recommendation 2

The CNSC should provide explicit and traceable responses to intervenor submissions in its Records of Decision by clearly identifying the major issues raised, stating whether each concern was accepted, rejected, or deferred, and explaining why, so that the relationship between public participation and the final decision can be meaningfully assessed.

Recommendation 3

The CNSC should ensure that hold points and licence conditions explicitly identify the issues they are intended to address, including where those issues were raised during public hearings. Each hold point should include measurable criteria for how and when the issue will be considered resolved, and the CNSC should make publicly available the information, analysis, and rationale used to assess and release the hold point.

Recommendation 4

The CNSC should expand public participation so that it occurs earlier and across multiple stages of the licensing process, including before key technical assumptions are finalized, and should incorporate additional review opportunities where important issues are likely to be deferred across licensing stages.

Recommendation 5

The CNSC should use licence length more strategically as a tool of accountability by avoiding excessively long licence terms where significant uncertainty, future site changes, or unresolved

governance concerns exist, and by incorporating scheduled mid-term public proceedings where continued public oversight is warranted.

These recommendations would strengthen transparency, improve the effectiveness of public participation, and make CNSC decision-making more accountable to the concerns raised during licensing hearings.

10. Conclusion

10.1 Summary of Findings

This report examined the extent to which intervenor submissions influence CNSC licensing decisions across three cases: Darlington, Pickering, and Point Lepreau.

The analysis finds that intervenor input is consistently incorporated into the record and acknowledged in Records of Decision. However, its effect on final regulatory outcomes is limited and uneven. In both Darlington and Pickering, CELA raised substantive concerns related to emergency planning, safety analysis, and waste management. While these concerns were recognized by the Commission, they were not translated into binding licence conditions and did not alter the approval decisions. Instead, they were frequently deferred to future licensing stages or treated as outside the scope of the decision under review.

Point Lepreau provides a partial exception. In that case, concerns raised by CELA and other intervenors regarding licence length, public participation, and regulatory accountability were explicitly reflected in the Commission's decision. The Commission rejected longer licence terms, issued a 10-year licence, and required a mid-term public proceeding. However, this influence remained procedural and did not extend to technical or safety-related requirements.

Overall, the findings show a consistent pattern: intervenor influence is most visible in procedural outcomes, such as licence structure and review mechanisms, but remains limited in substantive areas, including safety standards, risk thresholds, and core licensing decisions.

10.2 Limitations

This report focuses on three CNSC licensing decisions and therefore provides a targeted analysis rather than a comprehensive review of all CNSC regulatory activity. While the selected cases offer meaningful variation by covering new reactor construction, life extension, and licence renewal, they may not capture the full range of regulatory practices across different facility types or contexts.

The analysis is also limited to publicly available documents, including Records of Decision, staff materials, and intervenor submissions. It does not capture internal deliberations, informal

engagement between CNSC staff and licensees, or other decision-making dynamics that fall outside the public record.

In addition, the report centers on CELA and similar civil society intervenors. While this provides a focused perspective on public-interest participation, it does not assess the influence of other actors, such as industry stakeholders, Indigenous governments, or technical experts, whose roles may differ in both form and impact.

10.3 Implications for Nuclear Governance

The findings carry important implications for nuclear governance in Canada, particularly in relation to transparency, public participation, and regulatory accountability.

First, the gap between acknowledgement and outcome suggests that current processes do not fully translate public participation into regulatory influence. Hearings and submissions provide opportunities for engagement, but their effect on decisions, especially in technical and safety domains, remains limited.

Second, the reliance on staged licensing and issue deferral reduces the ability of intervenors to influence decisions at critical points. Concerns raised during hearings can be postponed without clear timelines or mechanisms for resolution, weakening the connection between participation and accountability.

Third, the consistent reliance on CNSC staff assessments as the basis for decision-making limits the role of external scrutiny. While technical expertise is essential in nuclear regulation, the absence of clear mechanisms to challenge or independently test staff conclusions constrains the effectiveness of intervenor participation.

At the same time, the Point Lepreau case demonstrates that change is possible within the existing framework. Where issues are framed in terms of governance, accountability, and participation, intervenor input can shape regulatory outcomes, particularly in areas where the Commission retains procedural discretion.

Taken together, these findings suggest that strengthening nuclear governance does not require reducing public participation, but making it more effective. This means improving transparency in decision-making, ensuring that intervenor concerns are clearly addressed, and creating stronger links between public input and regulatory outcomes.

For CELA, the report highlights both the value and the limits of current intervention strategies. Participation remains essential, but greater impact may depend on targeting procedural leverage points, linking technical concerns to regulatory conditions, and advocating for reforms that strengthen accountability within the licensing process.

Appendices

Appendix A - Case Study Overview Table

Facility	Decision Type	Key Issue	Final Outcome
Darlington New Nuclear Project	Licence to Construct	SMR Design and safety completeness	License granted; issues deferred
Pickering Nuclear Generating Station	Hold-point Removal	Operation beyond design life	Hold point removed
Point Lepreau Nuclear Generating Station	Licence Renewal	Licence length	10-year license issued

Appendix B - Key Documents Reviewed

Darlington New Nuclear Project (DNNP)

CNSC Hearing and Staff Documents

- **CMD 24-H3-2A** — CNSC Staff Supplemental Information
- **CMD 24-H3-B** — CNSC Staff Information
- **CMD 24-H3-1** — Application Materials
- **CMD 24-H3** — CNSC Staff Report
- **CMD 24-H3-1A** — Supplemental Application Information
- **CMD 24-H3-1B** — Additional Supporting Materials
- **CMD 24-H3-Q** — Questions from Commission Panel Members
- **CMD 24-H3.2** — Supplementary Staff Information
- **Record of Decision — Licence to Construct: OPG DNNP BWRX-300**
- **CMD 24-H2** — Related Supporting Documentation

Written Intervenor Submissions

- **CMD 24-H3-36** — Ole Hendrickson
- **CMD 24-H3-38** — Ecojustice Working Group
- **CMD 24-H3-39** — Dr. Sunil Nijhawan
- **CMD 24-H3-58** — Northwatch
- **CMD 24-H3-67** — Canadian Coalition for Nuclear Responsibility & Prolet Inc.
- **CMD 24-H3-68** — Nuclear Transparency Project
- **DNNP LTC** — Canadian Environmental Law Association Submission

- **PMD 11-P1.164** — Lake Ontario Waterkeeper Submission

Environmental Assessment Materials

- **Darlington Project Review (2011)** — Joint Review Panel Environmental Assessment
- **Joint Review Panel Environmental Assessment (2011)** — Final Report
- **2023 Submission — Comments on DNNP Environmental Assessment Applicability to BWRX-300 Technology** — Canadian Environmental Law Association

Pickering Nuclear Generating Station

CNSC Records of Decision

- **Record of Proceedings, Including Record of Decision** — May 7, 2014
- **Record of Proceedings, Including Record of Decision** — December 22, 2014
- **Record of Decision** — Application to Renew the Nuclear Power Reactor Operating Licence for the Pickering Nuclear Generating Station
- **Record of Decision, DEC 20-H100** — April 9, 2020
- **Record of Decision, DEC 22-H107** — November 24, 2022

Written Intervenor Submissions

- **CMD 14-H2.31** — Canadian Environmental Law Association
- **CMD 14-H2.53** — Canadian Coalition for Nuclear Responsibility
- **CMD 14-H2.36** — New Nuclear Free Solutions
- **CMD 14-H2.47** — Greenpeace
- **CMD 14-H2.41** — International Institute of Concern for Public Health

Proponent and Supporting Documentation

- **CD# P-CORR-00531-23980 — Ontario Power Generation Correspondence**

Additional Supporting Materials

- **Evaluating Environmental Protection and Emergency Preparedness at the Pickering Nuclear Generating Station (Ref. 2018-H-03) — Canadian Environmental Law Association**

Point Lepreau Nuclear Generating Station

CNSC Records of Decision

- **Record of Decision, DEC 22-H2 — Licence Renewal for the Point Lepreau Nuclear Generating Station**

- **Summary Record of Decision, DEC 22-H2**

Written Intervenor Submissions

- **CMD 22-H2-5 — Leap4wards**
- **CMD 22-H2-23 — EOS Eco-Energy Inc.**
- **CMD 22-H2-179 — Canadian Nuclear Workers' Council**
- **CMD 22-H2-219 — Nuclear Transparency Project**
- **CMD 22-H2-228 — Canadian Coalition for Nuclear Responsibility**
- **CMD 22-H2-244 — Passamaquoddy Recognition Group Inc.**
- **Joint Intervention — Coalition for Responsible Energy Development in New Brunswick and Canadian Environmental Law Association**

Appendix C: Darlington Environmental Assessment Applicability

2011 EA vs 2023 EA

Issue	2011 EA	2023 EA Applicability	Pattern
Reactor technology	No specific technology selected; Plant Parameter Envelope used	BWRX-300 is assessed as bounded by the old EA	Retrospective validation
Proximity	Treated as a risk amplifier and cumulative site issue	Treated as a manageable site condition	Risk normalized
Emergency planning	Public protection and evacuation were EA-level concerns	Treated as a future/ provincial/ licensing issue	Deferral
Waste	Long-term waste and used fuel are addressed as EA issues	Waste effects assessed through regulatory dose criteria	Narrowed Scope

Appendix D: Environmental and Aquatic Issue Assessments

Issue	Lake Ontario Waterkeeper Concern	Why it Matters

Lake Ontario as a drinking water source	Lake is a public health and ecological resource	Shows environmental/public health framing
Lake infilling	Up to 40 hectares of lake infilling, shoreline alteration, and habitat loss	Direct fisheries/aquatic habitat concern
Cooling water	Once-through cooling causes entrainment, impingement, thermal discharge, and contaminants	Strong evidence for the aquatic impact category
Radiological and chemical emissions	Lack of clear information on contaminants, limits, and monitoring	Supports the transparency/environmental monitoring theme
EA adequacy	Mitigation measures have not sufficiently demonstrated	Supports broader critique of environmental assessment
Need and alternatives	Alternatives not adequately assessed	Shows broader public-interest framing

Chapter 12
Nuclear Law, Oversight and Regulation: Seeking Public Dialogue and Democratic Transparency in Canada

Kerrie Blaise, Theresa McClenaghan and Richard Lindgren

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J. L. Black-Branch and D. Fleck (eds.), *Nuclear Non-Proliferation in International Law - Volume IV*, https://doi.org/10.1007/978-94-6265-267-5_12

Abstract To begin public discourse on acceptable policies and strategies surrounding Canada's continued reliance on nuclear energy and the waste legacy it generates, this chapter explains the work of the Canadian Nuclear Safety Commission (CNSC), the regulatory body which oversees Canada's nuclear industry. The authors describe the federal laws surrounding nuclear licensing and environmental approvals. They comment on current plans relating to radioactive waste disposal and emergency planning in light of the Fukushima Daiichi accident. They conclude that to strengthen the independence of the CNSC, opportunities for meaningful public participation should be developed, Indigenous engagement in CNSC decision-making processes be affirmed, and the federal government's role and responsibilities for nuclear emergency management clarified.

Keywords Environmental Protection · Human Health · Nuclear Legacy · Nuclear Oversight · Nuclear Power · Nuclear Waste · Public Dialogue · Transparent Decision-Making

12.1 Introduction

One of six Canadians live within sixty kilometres of a nuclear power plant. Even more people work or travel in areas transected by the annual transportation of approximately 1 million packages containing radioactive substances.[1] While North America witnesses a decline in operable nuclear reactors—triggered by a growing awareness of safety problems, the rising cost of operating aging reactors and competition from new forms of energy generation, such as renewables—Canada continues to pursue the refurbishment of its nuclear reactors and operations into the 2060s.[2]

Accompanying Canada's continued reliance on nuclear power, is a regulatory process which effectively expedites the approval of nuclear licences and radioactive waste repositories without meaningful engagement with the Canadian public and Indigenous peoples. In order to begin a long overdue nation-wide discussion on acceptable policies and strategies surrounding Canada's continued reliance on nuclear energy and the waste legacy it generates, it is necessary to first understand the regulatory body which oversees Canada's nuclear industry (Sect. 12.2); the laws surrounding nuclear licensing and environmental approvals (Sects. 12.3 and 12.4); current plans relating to radioactive waste disposal (Sect. 12.5); and emergency planning in light of the Fukushima Daiichi accident (Sect. 12.6).

[1] Canadian Nuclear Safety Commission, Regulatory Oversight Report on the Use of Nuclear Substances in Canada: 2016 (January 2018), http://www.nuclearsafety.gc.ca/pubs_catalogue/uploads/DNSR-Report-Use-of-Nuclear-Substances-2016-eng.pdf.

[2] Canadian Nuclear Safety Commission, CMD 18-H4, A Licence Renewal—Bruce Power Inc., Bruce Nuclear Generating Station A and B—Environmental Assessment Report (12 February 2018), p. 11.

12.2 Nuclear Regulation and Oversight in Canada

When Canada first undertook active participation in the nuclear fuel chain, it was to support the UK and US in their efforts during World War II. The first research facilities in Canada were established for that purpose. Subsequently the federal government decided to support the development of nuclear power generation, but was advised by industry that it would not participate without governmental measures to limit their exposure to liability for damages arising from accidents. The federal government then issued an Order-in-Council declaring the 'peaceful' use of nuclear energy to be for the general advantage of Canada, and provide protection of the private sector from liability. Later constitutional amendments and legislative developments further entrenched this approach, which is now embedded in the division of powers under Canada's *Constitution Act* (1982). As a result, federal regulation of nuclear power use and safety is currently undertaken by the Canadian Nuclear Safety Commission (CNSC), the successor to the former Atomic Energy Control Board (AECB).

The CNSC functions both as a quasi-judicial licensing tribunal under its enabling statute, the *Nuclear Safety and Control Act*, and as a regulator that performs monitoring, inspections and enforcement activities to ensure compliance by the nuclear sector. As an administrative tribunal, the CNSC generally operates in an inquisitorial, rather than an adversarial trial-like manner, where adjudicators act together as fact-finder and make decisions on consensus.

As a result of its inquisitorial approach, the CNSC lacks the rules of procedure necessary for hearing participants to fairly and effectively test the factual, technical and scientific evidence often presented at licencing hearings. For instance, at CNSC licencing hearings, there are no requirements for duly qualifying experts who wish to present opinion evidence, and no provisions that enable participants to cross-examine experts or other witnesses under oath. As a result, Commissioners appear to presuppose the validity and rigour of the evidence which is provided primarily by the licence applicant, but evaluated internally by CNCS technical and professional staff. Similarly, disclosure of key information can be particularly difficult for public intervenors to obtain at CNSC hearings, as licence applicants frequently withhold documents on the basis that they are 'proprietary', while the Commission may assert privilege over the requested information on the grounds that it is 'security sensitive'. Since there is no process before the Commission that allows requested documents to be disclosed to the hearing participants in strict confidence (i.e. a 'sealing order'), intervenors are routinely limited to making submissions based only on publicly available information.

The CNSC's regulatory approach also creates challenges for public disclosure and transparency of decision-making. As the majority of CNSC regulatory documents and standards are not prescriptive, licensees can 'put forward a case to demonstrate that the intent of a requirement is addressed'.[3] This negotiated approach to regulation between the CNSC and licensees results in consultations which are neither transparent nor open to broader dialogue with Canadians. This in turn limits the extent of recommendations which can be made by public intervenors appearing before the Commission at licensing hearings.

By controlling the dissemination of information and relying predominantly on its internal staff, the CNSC is able to centralize oversight of the nuclear sector in a single entity. This in turn creates the preconditions for regulatory capture, a phenomenon wherein the regulatory agency becomes beholden to the interests of the business or sector it is supposed to regulate.[4] While other federal regulators have recently undergone extensive public review in an effort to rebuild public trust, Canada's nuclear regulator has been insulated from this review.[5]

International obligations requiring the nuclear regulator be indeed, fully independent, also directly bear on the CNSC's mandate and decision-making authority. As a member of the International Atomic Energy Agency (IAEA), Canada is required to 'take appropriate steps to ensure an effective separation between the functions of the regulatory body and those of any other body …concerned with the promotion … of nuclear energy'[6] and 'ensure that the regulatory body is effectively independent in its safety related decision making'.[7] As further discussed in Sects. 12.3 and 12.4, to address the risk or apprehension of regulatory capture and ensure compliance with international obligations, Canada's nuclear laws must be modernized and the procedural safeguards for the public and intervenors enhanced.[8]

[3] Canadian Nuclear Safety Commission, Canadian Perspective on Risk-Informed Regulation (27 March 2018), http://www.nuclearsafety.gc.ca/eng/pdfs/Presentations/CNSC_Staff/2018/20180328-doug-miller-international-smr-advanced-reactor-eng.pdf.

[4] Mahar 2015, p. 9.

[5] Minister of Environment and Climate Change, Building Common Ground: A New Vision for Impact Assessment in Canada (2017), https://www.canada.ca/content/dam/themes/environment/conservation/environmental-reviews/building-common-ground/building-common-ground.pdf, pp. 50–51 [Expert Report]; Natural Resources Canada, Forward Together: Enabling Canada's Clean, Safe and Secure Energy Future (2017), https://www.nrcan.gc.ca/19667.

[6] IAEA, Convention on Nuclear Safety, Article 8. Regulatory Body.

[7] IAEA Safety Standards, *Government, Legal and Regulatory Framework for Safety*, GSR Part 1 (Rev 1), Requirement 4.

[8] Canadian Environmental Law Association, Consultation on Draft RegDoc 3.2.1 Public Information and Disclosure (2017), http://www.cela.ca/CELAcommentsonREGDOC3.2.1.

12.3 Modernizing Canada's Nuclear Laws

It has been over twenty years since Canada's *Nuclear Safety and Control Act* (NSCA)[9] was passed by Parliament to replace the previous *Atomic Energy Control Act*. However, a number of stakeholders maintain that further modernization of Canada's nuclear legislation is necessary to remedy the CNSC's widely criticized lack of institutional independence from the community it regulates,[10] and to address the issue of regulatory capture. The CNSC's predecessor, the Atomic Energy Control Board (AECB), had a legislated mandate to 'promote' nuclear power. While this mission statement was removed from the CNSC's mandate under the NSCA, there remains the public perception that the CNSC continues to be a promoter of the industry it regulates.[11]

The need for legislative reform becomes more acute in light of the findings from the Japanese Parliamentary panel's report following the Fukushima Daiichi nuclear disaster. Although the earthquake and tsunami are often typically portrayed as the cause of the accident in Canadian media, the Japanese government's investigation concluded that the nuclear accident resulted from government-industry 'collusion' and it was clearly 'manmade' due to a lax and industry-friendly nuclear regulator.[12] The report found that while the Japanese regulator and Fukushima's operator were aware of the tsunami risk, they did not adequately act on the information.

To address the human and institutional causes of the Fukushima disaster, Japan and the European Union increased legal requirements for the independence and transparency of their nuclear regulators. Similar action is required in Canada in order to modernize Canada's nuclear laws. Among other things, this requires a detailed review and revision of the CNSC's mandate, decision-making procedures and processes for public engagement. It is also necessary to review the criteria upon which the CNSC makes its decisions, ensure that the information that the CNSC relies upon is publicly available and accessible, and evaluate whether—or to what extent—participants in licencing hearings felt that their evidence and submissions were heard and respected.

As the Auditor General of Canada found in its examination of the CNSC and its management of nuclear power plants:

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[9] *Nuclear Safety and Control Act*, SC 1997, c 9.

[10] Letter by the Canadian Environmental Law Association et al to the Right Honourable Justin Trudeau (Office of the Prime Minister), 8 March 2016, http://www.cela.ca/sites/cela.ca/files/Trudeau-NuclearReview.pdf.

[11] Letter by Michael Binder (President of the CNSC) to the Honourable David Heurtel (Minister of Sustainable Development, Environment and the Fight Against Climate Change), 27 July 2015, http://www.nuclearsafety.gc.ca/eng/pdfs/letters/BAPE-letter-eng.pdf; Expert Report, *supra* note 5.

[12] The National Diet of Japan, The official report of The Fukushima Nuclear Accident Independent Investigation Commission, 2012, p. 16 [Fukushima Report].

> Overall, we found that the Canadian Nuclear Safety Commission (CNSC) had insufficient or incomplete documentation to support or explain its planning decisions. For example, it could not show how it had taken risks into account when making decisions about which inspections it would and would not carry out each year. We recognize that the CNSC's planning process must be flexible enough to respond to unforeseen events or issues. However, that does not preclude the need for a systematic, well-documented process so that the CNSC can demonstrate that its planning considers risk and that it allocates enough staff at the levels needed, commensurate with risk.[13]

In another recent report, a federal expert panel tasked with updating Canada's laws relating to environmental assessment found the following:

> The apprehension of bias or conflict of interest, whether real or not, was the single most often cited concern by participants with regard to the … CNSC. The apprehension of bias …eroded confidence in the assessment process.[14]

A legislative review of Canada's *Nuclear Safety and Control Act* is both timely and necessary to ensure an independent and robust licensing process in the public interest. This requires strengthening the independence of the Commission, improving transparency and accountability in CNSC decision-making, enhancing opportunities for meaningful public participation, and affirming the necessity for Indigenous engagement in CNSC decision-making processes. The federal government's role and responsibilities for nuclear emergency management must also be clarified, and the CNSC's oversight shifted to a Ministry without a mandate to promote nuclear power.

12.4 Environmental Assessment Law

12.4.1 Canada's Nuclear Regulator Has the Authority to Conduct Environmental Assessments

In addition to licensing and regulating the nuclear industry in Canada, the CNSC is one of only two federal authorities[15] apart from the Canadian Environmental Assessment Agency empowered to conduct environmental assessments (EAs) under the *Canadian Environmental Assessment Act, 2012* (CEAA 2012).[16] However, allowing the CNSC to conduct EAs under the current law has proven problematic in fundamental ways. For instance, CNSC-led EAs of designated

[13] Office of the Auditor General of Canada, Report 1—Inspection of Nuclear Power Plants—Canadian Nuclear Safety Commission (2016), http://www.oag-bvg.gc.ca/internet/English/att__e_41721.html.

[14] Expert Report, *supra* note 5, p. 48.

[15] The other authority that is currently empowered to conduct federal EAs is the National Energy Board.

[16] *Canadian Environmental Assessment Act, 2012,* SC 2012, c 19, s 52, s 15 [CEAA 2012].

nuclear projects have been undertaken in a manner that public interest participants have found to be narrowly focused, insufficiently robust and procedurally unfair.

More generally, the CNSC has an industry-specific mandate which does not allow it to address more broadly scoped environmental planning issues (including the rigorous consideration of alternatives) which are necessary in the EA process. For instance, hearings for the re-licensing of nuclear power plants under the NSCA —which typically occur every 5–10 years—focus largely on technical issues rather than environmental planning considerations or social acceptability (e.g. issues such as whether there is a demonstrable public need for more nuclear power, or whether there are there better, cheaper or safer ways of meeting or managing electricity demand). This is exacerbated by the Commission's view that it is not the CNSC's role to adjudicate upon the suitability of nuclear power, or the choice to use nuclear power as opposed to renewable energy sources, or the determination of what is the optimum mix of different energy sources and conservation measures. The Commission opines that these matters generally fall within the purview of the relevant provincial energy officials, and that the CNSC is limited to acting as a safety regulator. This narrow view stands in the way of proper environmental planning under federal law, as demonstrated in recent EA cases where CNSC members had the statutory duty to consider the need for, or to consider alternatives to, the proponent's proposed nuclear project, but did not adequately discharge that duty in the view of public interest intervenors.

In light of these and other fundamental concerns, the federal government established an Expert Panel in 2016 to conduct a review of the federal EA process and to recommend reforms that would restore public trust, protect the environment, and advance reconciliation with Indigenous peoples. Public interest groups, including the Canadian Environmental Law Association (CELA), welcomed this consultation by the Expert Panel which promised to 'restore robust oversight', 'introduce new, fair processes', and 'provide ways for Canadians to express their views and opportunities for experts to meaningful participation'.[17]

The 2017 report produced by the Expert Panel reflects the views it heard from numerous Canadians, and finds that the CNSC has not operated in a transparent way, nor demonstrated a commitment to public engagement and the rigorous testing of evidence, within federal EA processes. The Panel's report also cites public concerns about the perceived lack of independence and objectivity because of the close relationship between the CNSC and the nuclear industry. Because Canadians felt that public trust was lacking in the federal EA process, so too, was the social acceptance necessary to facilitate projects being assessed under CEAA 2012.

Throughout the EA reform public engagement process, CELA recommended that the federal government should repeal CEAA 2012, start with a clean slate, return to first principles, and develop an integrated package of forward-looking statutory, regulatory, policy and administrative reforms aimed at ensuring

[17] Expert Report, *supra* note 5, p. 5.

sustainability.[18] It was imperative, CELA argued, for the Government of Canada to establish a new impact assessment authority that by law is required to conduct robust reviews, hold procedurally fair proceedings, and deliver credible, transparent and evidence-based decisions.

In February 2018, the federal government tabled its proposed revised EA law entitled the *Impact Assessment Act* (IAA).[19] On the day that Bill C-69 was introduced, federal Ministers fanned out across Canada to hold press conferences praising the new legislation, and extensively using social media to claim that the IAA represented 'Better Rules'. However, a close reading of the IAA reveals that in many key aspects, the IAA is not demonstrably 'better' than CEAA 2012. To the contrary, the IAA replicates many of the same significant flaws and weaknesses found within the widely discredited CEAA 2012.

For example, Bill C-69 advances an ill-conceived model that permits CNSC members to be appointed to review panels that will conduct impact assessments of designated projects that require licencing under the NSCA, contrary to the Expert Panel's recommendation that regulators should not conduct assessments.[20] In particular, the IAA specifies that 'at least' one member from the CNSC must be appointed to the review panel when assessing a designated nuclear project. The skewed membership of review panels that could be established to assess energy projects under these provisions of the proposed IAA would essentially repeat the same contentious types of regulator-led assessments that occurred under CEAA 2012, and that resulted in a loss of public trust which prompted the above-noted commitments by the federal government to review and revise the national EA regime. Public confidence cannot be regained by creating a system that still allows regulators to lead, control or otherwise dominate assessments of projects that the regulators also happen to licence under other statutes.

12.4.2 Major Nuclear Projects Lack Environmental Assessments

A regulation under CEAA 2012 lists a small number of physical activities that currently trigger the CNSC's obligation to conduct a federal EA. While the construction of a new nuclear waste facility or the construction of a new nuclear reactor triggers an EA, projects such as reactor life extensions and refurbishments, or end of life decommissioning do not.[21] Therefore, unless a project is designated by regulation as a physical activity to which CEAA 2012 applies, then a federal EA is not required as a matter of law before the CNSC issues a licence under the NSCA.

[18] Lindgren 2017.

[19] Bill C-69, *An Act to Enact the Impact Assessment Act*, 1st Sess, 42nd Parl, 2018 [IAA].

[20] Lindgren 2018.

[21] Regulations Designating Physical Activities, SOR 2012-147.

As the proposed IAA is being debated in Parliament at the present time, the federal government has not released a draft projects list to identify which nuclear-related activities will trigger an impact assessment. However, the government is currently consulting on the criteria that should be used to designate projects under the IAA, which indicates that the proposed Act will continue the narrow project-listing approach utilized under CEAA 2012.

It is noteworthy that under the predecessor to CEAA 2012,[22] changes to the operating licences of nuclear power plants constituted an EA trigger, and therefore nuclear projects such as power plant refurbishment were automatically subject to federal EA requirements. For example, Bruce Power, the licensee which operates an 8-reactor nuclear station on the shores of Lake Huron, Ontario, is currently seeking approval from the CNSC to refurbish its reactors, allowing its operating life to be extended to 2064. Although previous refurbishment projects triggered federal EA requirements under the prior version of the law, this latest proposal is not a designated project under CEAA 2012, and would therefore be Canada's first nuclear power plant rebuild to *not* undergo a federal EA.

Even if Canada's proposed IAA was amended to either remove its project-listing approach, or to specifically include refurbishment and life extension projects within its scope, there has been no indication that any currently proposed nuclear refurbishment and life extension projects would be transitioned into the new Act.

The end stage of the nuclear power plant life cycle is similarly not subject to a federal EA nor proposed to be a designated project, requiring review under the IAA. This is particularly significant given nuclear powers plants constructed in the 1960s and 70s did not undergo environment assessments evaluating how the site would be remediated, nor review approaches to decommissioning which were cognizant of social and environmental considerations.

While the host municipality of the Pickering Nuclear Generating Station has expressly requested it be notified and engaged in the decision-making process with respect to conducting an EA for the plant's decommissioning,[23] their involvement is moot as decommissioning is so far, not a project requiring an EA either federally or provincially. The Municipality has also expressed concerns related to the socio-economic impacts of station closure and has requested they be considered in an environmental assessment. Only an EA conducted pursuant to Canada's federal environmental assessment legislation would allow authorities to review the socio-economic and cultural values and effects associated with decommissioning, as the *Nuclear Safety and Control Act* is too narrow a regulatory statute to permit this review.[24]

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[22] *Canadian Environmental Assessment Act*, SC 1992, c 37.

[23] Durham Region, Submission from the Regional Municipality of Durham regarding the application of Ontario Power Generation to renew the Power Reactor Operating Licence for the Pickering Nuclear Generating Station (7 May 2018), http://www.nuclearsafety.gc.ca/eng/the-commission/hearings/cmd/pdf/cmd18-h6/CMD18-H6-67.pdf, p. 16.

[24] See IAA, *supra* note 19, s 6(1); CEAA 2012, *supra* note 16, s 19.

If appropriate changes to Canada's proposed IAA are not implemented, then Parliament will lose an important opportunity to strengthen and improve Canada's approach to assessing nuclear projects. While the CNSC should be able to participate (like any other federal authority) in the impact assessment process, the CNSC should not lead, nor co-lead, assessments under the new regime for the very reasons that precipitated Canada's federal review of its broken EA process.[25]

12.5 Canada's (Growing) Waste Legacy

Planning for the long-term storage and disposal of radioactive waste should be a prerequisite to any proposal seeking to extend nuclear reactor operating life, refurbish infrastructure or introduce new reactor technology.[26] The worst environmental 'legacy' issues in Canada exemplify this requirement, where there are five ongoing nuclear waste cases, four of which are being evaluated in ongoing federal EA processes. Scholars have noted that the identification and construction of suitable storage and disposal sites not only poses technological challenges but lacks general social acceptance.[27] Local communities are very worried about short- and long-term impacts on adjacent waterways, drinking water quality, environmental and ecosystem health, and the potential failures of the proposed 'solutions'.

All four EA cases, discussed below, involve approaches that are either completely untested, or have failed elsewhere, and involve radioactive wastes that can remain toxic for hundreds and thousands of years. The fifth proposal involves the underground disposal of used nuclear fuel—highly toxic, long-lived radioactive waste that results from the fissioning of the CANDU fuel pellets during power production. This proposal has not yet been assessed in a federal EA process, but has been undergoing a lengthy siting process to find a geologically suitable location and a willing host community under the auspices of the Nuclear Waste Management Organization pursuant to the federal *Nuclear Fuel Waste Act*. The other four proposals deal with non-fuel radioactive waste, much of which is still highly toxic and long-lived, and all of which still needs to be managed to separate the waste from the living environment.

[25] CELA Review, *supra* note 18.

[26] Sudbury Star Staff, Mining research corporation exploring mini reactors for North, in Sudbury Star (9 April 2018), online: http://www.thesudburystar.com/2018/04/08/mining-research-corp-exploring-mini-reactors-for-north.

[27] Odendahl 2016, p. 282.

12.5.1 The Waste Bunker—Also Known as the Deep Geological Repository

The first of Canada's long-term disposal plans for low- and intermediate-level nuclear waste to undergo an EA process (which is still underway) is the Deep Geological Repository (DGR) in Kincardine, Ontario. According to the proposal, this nuclear waste, much of which is very long-lived and/or quite hazardous despite being styled 'intermediate-level' radioactive waste, would be placed in a deep rock excavation just one kilometre from Lake Huron. A lack of proper evaluation of alternatives, failure of similar technology elsewhere in the world, and irreversibility of this proposal are among the reasons that people are calling for Canada's federal environment Minister to deny the approval.

The federal decision on whether to approve the DGR is currently on hold since the Minister of the Environment and Climate Change, Catherine McKenna, has requested further information from the proponent, Ontario Power Generation, particularly about 'the potential cumulative effects of the project on physical and cultural heritage' of the Saugeen Ojibway Nation.[28] To date, Ontario Power Generation remains confident the DGR 'will have no impact on the surrounding environment'.[29]

12.5.2 Entombing Nuclear Plant Structures and Its Waste—Also Know as In Situ Decommissioning

The second and third nuclear waste disposal cases are among the first nuclear EA matters to trigger the CNSC's assessment role, as lead agency, under CEAA 2012. These two cases both propose to 'grout and abandon' radioactive wastes at the Whiteshell Reactor in Pinawa, Manitoba and the Rolphton nuclear power demonstration reactor in Chalk River, Ontario.

A 'government-owned/contractor operated' contract, unprecedented in Canada, has charged Canadian Nuclear Laboratories (CNL) with implementing solutions to nuclear legacy wastes owned by the Atomic Energy of Canada Limited (AECL). The AECL is a Crown-owned corporation which is now a shell of its former self, and serves as the intermediary corporation between its government shareholders and the contracted managers. CNL has proposed to leave everything from these two

[28] Minister of Environment and Climate Change, Letter to Ontario Power Generation (21 August 2017), http://www.ceaa-acee.gc.ca/050/documents/p17520/120087E.pdf.

[29] Ontario Power Generation, The Deep Geological Repository (2016), http://opgdgr.com/.

reactors on-site (exclusive of used fuel and heavy water which were removed from the body of both reactors) alongside the Winnipeg and Ottawa Rivers respectively, and fill the buildings and cavities with grout and cement.

The proposed *in situ* decommissioning (also known as entombment) of these sites would result in the subsurface reactor systems, components, structures and their associated hazards being permanently buried, and encased with grout. At Whiteshell, high quantities of radionuclides will remain inside the facility despite it being shut down in 1985, in part because of the 'approximately 150 documented fuel failures in the reactor between 1966 and 1983'.[30]

Despite the proposed *in situ* decommissioning of these sites, current international standards indicate that, short of an emergency scenario, *in situ* decommissioning should be limited to nuclear facilities that only contain short-lived or limited concentrations of long-lived radionuclides. The IAEA does not recognize entombment or in-situ confinement as strategies for decommissioning, noting:

> Entombment is not relevant for a facility that contains long lived isotopes because these materials are not suitable for long term surface disposal. Consequently, reprocessing facilities, fuel fabrication facilities, enrichment facilities or facilities that use or process thorium or uranium would not be appropriate for entombment. However, entombment could be a viable option for other nuclear facilities containing only short lived or limited concentrations of long lived radionuclides, i.e. in order to comply with the site release criteria.[31]

Worrying facts also emerge from the review of documents for the planned decommissioning of the Rolphton reactors in Chalk River, Ontario. For instance, the current environmental impact statement provided by the proponent notes that the buried reactor vault will become flooded after 40–60 years and the nuclides will travel underground to the Ottawa River where they will be diluted. This proposal adopts the outdated and unacceptable assumption that dilution and dispersion is an acceptable method of dealing with radionuclides (the oft-asserted, 'dilution is the solution to pollution'), and it fails to consider the cumulative effects from the project, with a near surface waste facility also proposed for the site. As discussed below, to date no study has been undertaken to review the combined environmental effects caused by the Rolphton entombment and the proposed Chalk River near-surface facility on the Ottawa River, its downstream residents and ecosystems.

[30] Canadian Nuclear Laboratories, In Situ Decommission of Whiteshell Reactor 1 Project—Decommissioning Safety Assessment Report (2017, Revision 2), para 5.3.1.1.1.

[31] IAEA, *Decommissioning Strategies for Facilities Using Radioactive Material*, SRS 50 (2007), s 3.2.2.

12.5.3 The Radioactive Landfill—Also Known as a Near Surface Disposal Facility

The fourth EA case is the 'Near Surface Disposal Facility' also proposed by CNL. Once again, the regulator— the CNSC—is conducting the federal EA process under CEAA 2012. An extremely large volume of nuclear waste from Chalk River's past activities, and from other locations in Ontario, would be placed in a 'near surface' mound with a liner and cap similar to domestic landfills, in close proximity to the Ottawa River.

The EA currently underway does not consider the two major nuclear accidents which occurred at Chalk River in the 1950s and their resulting wastes. A partial meltdown occurred in 1952 at the National Research Experimental reactor operated by the AECL. The second event occurred in 1958, involving a fuel rupture and fire in the National Research Universal reactor building. It appears likely that highly radioactive debris from these accidents still exists in some of the waste targeted for this facility, given the long long-half lives of the nuclides on-site.

As noted by many concerned groups, international guidelines do not recommend the use of near surface facilities for wastes other than those with short half-lives and low toxicity (i.e. low-level waste). While a recent change by the CNL to its proposal removes 'intermediate' level waste from the proposal, much of the waste destined for this site is still long-lived and toxic radioactive waste that does not meet the international guidelines for near surface radioactive waste facilities. As the IAEA's Specific Safety Guide for near surface disposal facilities outlines:

- Neither Intermediate Level Waste (ILW) (i.e. lasting more than a few hundred years) or High Level Waste (HLW) should be placed in these facilities;
- Near-surface disposal is an appropriate disposal option only for very low-level wastes; and
- ILW and HLW, which contain larger quantities of long-lived radionuclides, should not be stored in surface or near surface facilities.[32]

Despite this guidance, the NSDF is proposed to contain ILW. By volume, ILW will constitute 1% of total waste and given the sites total capacity of 1,380,000 m^3,[33] the resulting amount of IWL is not negligible, at 13,800 m^3.[34]

[32] IAEA, *Near Surface Disposal Facilities for Radioactive Waste*, Safety Standard, SSG-29 (2014).

[33] CNL Performance Assessment for Near Surface Disposal Facility to support the Environmental Impact Statement. Report 232-509240-ASD-001, amended Table 4-2.

[34] CEAA, Consolidated Near Surface Disposal Facility Project EIS Review Comment Tables, http://www.ceaa-acee.gc.ca/050/documents/p80122/119841E.pdf.

12.6 Emergency Planning and Preparedness

12.6.1 The Planning Basis and Accompanying Response Measures

Given Canada's current and future reliance on nuclear power, we must ask: if a large-scale nuclear accident with off-site releases of radioactive substances were to occur at any of Canada's nuclear power plants, would we be able to prevent widespread adverse health, safety and environmental consequences?

The Pickering Nuclear Generating Station in the Province of Ontario is one of the largest plants in the world.[35] In comparison to other nuclear power plants, the Pickering facility also has one of the largest populations surrounding such a plant, in its immediate vicinity. This region has also been designated a 'place to grow' by the province—meaning the population may increase from over 650,000 people currently, to 1.4 million people in the next twenty years.[36] The operator of the Pickering station recently received a licence extension to allow the plant to continue operating until 2024. While this application did not trigger a federal EA, during the licensing hearings conducted pursuant to the NSCA, members of the public, mindful of the Fukushima accident, submitted that nuclear power plants should not be re-licensed until emergency plans are in place and *proven* to be effective for a catastrophic accident.

Fundamentally, the magnitude of a nuclear accident chosen as the reference accident for emergency planning determines the consequences and risks which can be averted. The planning basis is the baseline upon which detailed off-site emergency response plans are based and therefore, it must reflect a large, off-site release, such as that which occurred at Fukushima. All emergency plans should be ready for an accidental radioactive release on par with Level 7 accident on the International Nuclear Event Scale (INES). This is not what is planned for in Canada. Instead, planners assume only smaller accident scenarios, involving the release of short-lived radionuclides. This reduces the area in which detailed off-site emergency preparedness measures are required.

Currently, Canada's nuclear response plans do not account for a Fukushima-level accident (e.g. a severe multi-reactor accident with large off-site releases of radioactive substances). Therefore, there is no detailed planning, resourcing or testing for catastrophic events that cause large releases into the atmosphere. Instead, current emergency plans are based on a scenario in which plan operators *would* likely be able to contain and control the radioactive releases in the initial hours and days.

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[35] Ontario Power Generation, Pickering Nuclear Generating Station (2018), https://www.opg.com/generating-power/nuclear/stations/pickering-nuclear/Pages/pickering-nuclear.aspx.

[36] Ministry of Municipal Affairs, Release of the Growth Plan for the Greater Golden Horseshoe, 2017 (18 May 2017), https://placestogrow.ca/index.php?option=com_content&task=blogcategory&id=4&Itemid=36.

While emergency preparedness requires cooperation between the regulator, provincial authorities and licensee, as the licensing body and regulator of an activity under federal jurisdiction, the CNSC's regulatory jurisdiction extends to verifying the sufficiency of emergency response plans. Not only does the CNSC have authority to require, review and approve emergency plans which are in the purview of its licensees, it also has authority to review emergency plans in place for off-site response. The CNSC can use its assessment of the adequacy of those plans as part of its determination as to whether a nuclear power plant may operate, and under what terms and conditions, by reason of its obligation to ensure that licensees do not cause unreasonable risk to the environment or to the health and safety of persons. As endorsed by a March 2017 ruling from the Maebashi District Court in Japan, both the government and operator of the Fukushima nuclear plant were responsible for the disaster, as a result of their failure to take preventative measures. The judges found that the major risks from the plant were foreseeable by the government, but were ignored and not acted upon.

We have learned since Fukushima that planning for emergency response and evacuation should be extended beyond current emergency planning zones.[37] This was affirmed by an independent investigation commission, which concluded that the Fukushima Daiichi nuclear power plant accident could not be 'regarded as a natural disaster. It was profoundly a manmade-disaster that could and should have been foreseen and prevented'.[38]

However, despite this global experience, current emergency measures are geographically limited to areas close to nuclear stations due to the current small-scale reference accident. For instance, the current planning zones around Ontario's three nuclear power plants include a 10 km Detailed Planning Zone (DPZ—formerly referred to as the primary zone), a newly introduced 20 km Contingency Planning Zone (CPZ)—which is meant to address the possibility for expanded evacuations in the event of a more severe accident—and a 50 km Ingestion Planning Zone (IPZ—formerly known as the secondary zone). Only in the immediate 10 km zone surrounding a plant, however, is detailed planning required for default protective actions, such as evacuation and the pre-distribution of potassium-iodide (KI) pills. Outside of this boundary, off-site emergency plans list response measures such as KI pill distribution, public awareness and monitoring as 'considerations' only. The efficacy of the emergency response 'considerations', however, crucially depends upon the level of preparedness and planning undertaken *prior* to a radiological emergency.

[37] Handl 2016, p. 332.

[38] Fukushima Report, *supra* note 12.

12.6.2 Canada's International Standing

At a minimum, in order to reflect the global experience of severe off-site accidents that have occurred in other jurisdictions, the Detailed Planning Zone in Ontario must expand its existing 10 km radius to a distance of 20 km, and likewise and extend the Contingency Planning Zone from 20 to 100 km. In order to reflect the impacts and lessons arising from the Fukushima accident, expanded emergency zones are needed.[39] This requires, for instance, that radiation monitoring programs exist out to the 100 km boundary in order to determine in the event of an accident, whether additional evacuations, sheltering or KI pill consumption is required.

Ontario has also not yet modelled projected radiation doses or exposure to vulnerable individuals (i.e. children and pregnant women) within 100 km of Ontario's reactors, nor has modelling being completed to study weather contingencies based on 'worst case' weather scenarios.[40]

Furthermore, Ontario's emergency planning zones do not align with IAEA guidance (see Table 12.1). A side-by-side comparison of Ontario's response zones and emergency measures and those proposed by the IAEA reveals the following deficiencies. First, the response measures recommended for the IAEA's Precautionary Action and Urgent Protective Action Planning Zones extend to a distance of 30 km. In Ontario, similar levels of detailed planning and response only extend to a distance of 10 km with response measures out to 20 km zone being implemented on an *ad hoc* basis. Secondly, the Ontario's Contingency Planning Zone extends to a distance of 20 km, while the IAEA in its Extended Planning Distance zone recommends a size of 100 km. Lastly, where the Ingestion Planning Zone in Ontario is based on a 50 km radius, the analogous zone in IAEA guidance, the Ingestion and Commodities Planning Distance, extends to 300 km.

12.6.3 Response Planning and Public Input

In the latter part of 2017, Ontario's emergency response plans were updated. However, the need to extend the size of planning zones remains outstanding and currently subject to technical review.

The advisory committee tasked with reviewing public comments on Ontario's proposed changes to the nuclear emergency response plans found that 'planning zones may require revision' if the planning basis was to include a multi-unit failure event.[41] The advisory committee's examination of planning zone effectiveness also

[39] Greenpeace et al. 2017.

[40] McClenaghan 2017.

[41] Office of the Fire Marshal and Emergency Management, Update on Emergency Management in Ontario and the Provincial Nuclear Emergency Response Plan (PNERP), Presentation—Commission Meeting (4 April 2018), p. 15.

Table 12.1 Emergency Planning Zones and Accompanying Response Measures per IAEA Guidance [*Source* Canadian Environmental Law Association, Supplemental Submission on Emergency Planning (2018) http://www.cela.ca/SupplementalSubmissionOnEmergencyPlanning]

Zone	Size (km)	Description	Response Measures
Precautionary Action Zone	3–5	Area within which arrangements should be made to implement precautionary urgent protective actions before or shortly after a major release with the aim of preventing or reducing the occurrence of severe deterministic effects[a]	Urgent protective actions include: isolation of a contaminated area or radioactive source; prevention of inadvertent ingestion; evacuation; Sheltering; respiratory protection and protection of skin and eyes; decontamination of individuals; prophylaxis with stable iodine; protection of the food supply and prevention of the consumption of significantly contaminated foodstuffs and water; management of the medical response; and, protection of international trade[b]
Urgent Protective Action Planning Zone	5–30	Area where preparations are made to promptly shelter in place, to perform environmental monitoring and to implement urgent protective actions on the basis of the results of monitoring within a few hours following a release[c]	
Extended Planning Distance	100	Distance around a nuclear power plant where arrangements are made to conduct early monitoring of deposition to locate hotspots with dose rates warranting (1) evacuation within a day following a release or (2) relocation within a week to a month following a release	Relocation, decontamination, replacement of food, milk and water[d]
Ingestion and Commodities Planning Distance	300	The distance around a nuclear power plant for the area within which arrangements are made, within hours of being notified by the nuclear power plant of the declaration of a General Emergency	Place grazing animals on covered feed; protect drinking water supplies that directly use rainwater (e.g. to disconnect rainwater collection pipes); restrict consumption and distribution of non-essential local produce, wildgrown products (e.g. mushrooms and game), milk from grazing animals, rainwater, animal feed; and, restrict distribution of commodities until further assessments are performed[e]

[a]IAEA Safety Standards, 'General Safety Requirements Part 7 - Preparedness and response for a Nuclear Radiological Emergency' (2015), p. 76
[b]International Atomic Energy Agency, *Arrangements for Preparedness for a Nuclear or Radiological Emergency No. GS-G-2.1* (2007) 2
[c]IAEA Safety Standards, *General Safety Requirements Part 7—Preparedness and response for a Nuclear Radiological Emergency* (2015), p. 77
[d]IAEA 2013, p. 102
[e]Ibid., p. 103

noted that models were 'confined to only a few weather patterns' and 'may not be appropriate for distances farther away' from nuclear generating stations. Consequently, as recommended by the advisory committee, technical studies, with an estimated completion date at the end of 2018 are now underway, to assess and identify changes to planning zones. In the meantime, the public expectation is that the additional technical study of off-site radioactive releases, multi-unit accidents, and 365-day meteorology monitoring (as was recommended by the advisory committee) will be released to the public for comment and result in further changes to planning zones and protection action levels. Both the City of Toronto and Durham Region, whose populations are adjacent to two of Ontario's nuclear power plants, have also requested the technical assessment be released and available for public review.[42]

Prior review of the provincial plans did not always include public input. The most recent review attempted to improve this situation with a public posting for comment and the appointment of an expert advisory committee to review the provincial proposal and all public comments. The current Provincial Nuclear Emergency Response Plan (PNERP) now calls for public review every five years. According to the updated PNERP, these public reviews are intended to 'uphold the province's commitment to transparency and accountability, and to ensure that these plans reflect current emergency response directives, legislation, lessons learned and improvements to emergency management methodologies'.[43]

The PNERP also includes a new administrative principle related to transparency, specifically, 'a policy of truth and transparency *should* be followed in providing information to the public and media prior to and during a *nuclear or radiological emergency*'.[44]

In other jurisdictions the importance of transparency in relation to the peaceful use of nuclear technology has been highlighted by civil society. For example, the European organization Nuclear Transparency Watch states:

> The Fukushima disaster has raised concerns about the risks and catastrophic potential entailed by civil nuclear activities. Nuclear electric generation, regardless of the opinion that one may have on its place in the energy mix, is a technology that requires a very high level of safety through extremely demanding conditions (financial, technical, social, political and legal).

[42] Toronto City Council, EX33.6—Emergency Management Program Update—2017, April 24 2018; Durham Region, Minutes from the Regional Council of Durham, April 11 2018, p. 13.

[43] Ministry of Community Safety & Correctional Services, Provincial Nuclear Emergency Response Plan, Master Plan 2017, https://www.emergencymanagementontario.ca/english/emcommunity/response_resources/plans/provincial_nuclear_emergency_response_plan.html, s 1.3.4.

[44] Ibid., s 1.2.12.

> These conditions can only exist if the civil society is really able to take part into the governance of nuclear activities. The vigilance of the civil society is not only requested at local and national levels but also at the European level where regulators, operator and experts cooperate more and more.[45]

Continued transparency on the topic of nuclear emergency planning is essential. Crucial to emergency planning is facilitating public awareness, building community trust in the underlying preparations, and ensuring that the potentially affected public in the vicinity of operating nuclear power plants in Canada understand how to respond quickly and appropriately in the event of an emergency. No reactor technology is immune from accident, human or otherwise. The occurrence of real world reactor accidents indicates that the likelihood of nuclear accidents is much higher than claimed in industry risk assessments.[46] Thus, in the wake of Fukushima, as well as previous nuclear accidents in Canada and world-wide, we must unfortunately heed the lesson that *any* likelihood of an off-site radiation release necessitates a detailed planned and tested emergency response.

12.7 Conclusion

Canada currently lacks the proper venue for meaningful public debate and consideration of the implications of continued reliance on nuclear power production in Canada, its environmental effects, and its legacy waste. The Fukushima disaster highlighted that nuclear regulators are vulnerable to regulatory capture. Transparency and public participation in regulatory oversight of the sector can assist in mitigating risks of regulatory capture.

Project-specific hearings conducted by the CNSC are too narrow a forum for Canada-wide discussion of larger policy issues. Canada's present and proposed EA legislation continues to promote a regulator-directed process that demonstrably results in the loss of public trust. In the face of increasing complexity and public concern, there should be a proportionate increase in opportunities to promote democratic transparency and decision-making accountability.

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DURHAM NUCLEAR AWARENESS, SLOVENIAN HOME ASSOCIATION & THE CANADIAN ENVIRONMENTAL LAW ASSOCIATION

Comments on Ontario Power Generation's Application for the Authorization to Operate Pickering Nuclear Generating Station Units 5 to 8 until 2026

Prepared by:
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Expert Review by:
M.V. Ramana, Professor and Simons Chair in Disarmament, Global and Human Security

April 26, 2024

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April 26, 2024

Senior Tribunal Officer, Secretariat
Canadian Nuclear Safety Commission
280 Slater Street, P.O. Box 1046, Station B
Ottawa, Ontario K1P 5S9

Dear Sir or Madam:

Sent by email interventions@cnsc-ccsn.gc.ca

Re: Joint Submission of Durham Nuclear Awareness, Slovenian Home Association, and the Canadian Environmental Law Association, Regarding Ontario Power Generation's application for the authorization to operate Pickering Nuclear Generating Station Units 5 to 8 until 2026 (Ref. 2024-H-05)

The Canadian Environmental Law Association ("CELA") has enclosed its comments, on behalf of Durham Nuclear Awareness, Slovenian Home Association, and CELA on the OPG's application to extend the operations of Pickering Nuclear Generating Station Units 5 to 8 until 2026.

Please find below our submission for your review.

By this letter, and pursuant to the CNSC's *Rules of Procedure*, CELA request status to participate as an intervenor in the public hearing and an opportunity to make a 30-minute oral presentation at the June 2024 hearing.

Sincerely,

CANADIAN ENVIRONMENTAL LAW ASSOCIATION

Sara Libman

Sara Libman
Legal Counsel, CELA

I. INTRODUCTION

Durham Nuclear Awareness (DNA) and Slovenian Home Association (SHA) together with the Canadian Environmental Law Association (CELA) and the expert review by Dr. M.V. Ramana,[1] submit this written report in response to the Canadian Nuclear Safety Commission's (CNSC) Participant Funding Notice to review Ontario Power Generation's (OPG) application and associated documents seeking to extend the operations of Pickering Nuclear Generating Station Units 5 to 8 until December 2026.[2]

DNA, SHA, and CELA's (herein, "the intervenors") submission is the result of a review of OPG's documents which have been made available to the public, as well as the CNSC staff Commission Member Document (CMD) prepared in response to OPG's application. In addition to reviewing the documents submitted by OPG, this submission considers the CNSC's jurisdiction pursuant to the *Nuclear Safety and Control Act* (NSCA), which requires that in making a licensing decision, the CNSC ensure the adequate protection of the environmental and human health. In meeting this objective, per section 24(4) of the NSCA, the intervenors' findings and concerns are itemized below. Our recommendations are summarized in **Appendix A.**

II. INTEREST AND EXPERTISE OF THE INTERVENORS

i. Durham Nuclear Awareness

Durham Nuclear Awareness (DNA) is a citizens' group with a longstanding interest in the Darlington Nuclear Generating Station and the Pickering Nuclear Generating Station. DNA was first organized in 1986 in the wake of the Chernobyl disaster and born out of a need for people in Durham Region to come together, learn & empower themselves.

As a volunteer group of concerned citizens, DNA dedicates themselves to raising public awareness about nuclear issues facing Durham Region, and fostering greater public involvement in the nuclear decision-making process. DNA has appeared on numerous occasions before the CNSC and has a lengthy history lobbying for critical public health and safety measures, including improved emergency planning and baseline health studies, and setting standards for tritium in drinking water. DNA continues to advocate for upgrades to nuclear emergency plans to ensure the protection of communities in the event of a nuclear accident.

[1] M.V. Ramana is the Simons Chair in Disarmament, Global and Human Security and Professor at the School of Public Policy and Global Affairs, University of British Columbia, Vancouver, Canada.
[2] Canadian Nuclear Safety Commission, *Participant Funding Notice*, online: https://www.nuclearsafety.gc.ca/eng/the-commission/participant-funding-program/opportunities/2023-pic-ext/

ii. Slovenian Home Association

Slovenian Home Association ("SHA") is a non-profit cultural organization dedicated to the preservation of Slovenian culture language, heritage and identity in Canada. Many Slovenians reside in the vicinity of the Pickering and Darlington nuclear plants and are concerned about the proposed plans to expand nuclear power generation within the region, particularly with OPG proposing novel reactor technology at the Darlington site. Much of these concerns stem from emergency planning for nuclear accidents.

SHA members are not aware of what to do in case of a nuclear alert from the Province of Ontario. Some questions posed to SHA by its members include: *Should they be prepared to evacuate or stay at home? Where is their closest evacuation center? How to protect themselves by staying at home?* Despite emergency planning being a heavy concern for its members, SHA not been made aware of any public information meetings where the details of the actions taken by the citizens, in case of a nuclear alert, were discussed. SHA would welcome an opportunity to distribute emergency preparedness instructions to its members and to organize and host a preparedness workshop on the topic of emergency preparedness.

iii. Canadian Environmental Law Association

CELA is a non-profit, public interest law organization. CELA is funded by Legal Aid Ontario as a speciality legal clinic to provide equitable access to justice to those otherwise unable to afford representation for environmental injustices. For nearly 50 years, CELA has used legal tools to advance the public interest, through advocacy and law reform, in order to increase environmental protection and safeguard communities across Canada.

CELA has been involved in number of nuclear facility licensing and regulatory matters before the CNSC including federal environmental assessments. CELA also maintains an extensive library of public legal education materials related to Canada's nuclear sector on its website.[3]

iv. Dr. M.V. Ramana

Expert review of this submission was provided by M. V. Ramana, Professor and Simons Chair in Disarmament, Global and Human Security at the School of Public Policy and Global Affairs (SPPGA), University of British Columbia. M. V. Ramana has extensive knowledge and expertise in the nuclear non-proliferation safety risks of reactor designs and accompanying adverse environmental effects. His research interests are in the broad areas of international security and

[3] Canadian Environmental Law Association, online: www.cela.ca

energy supply, with a particular focus on topics related to nuclear energy and fissile materials that can be used to make nuclear weapons. He combines technical skills and interdisciplinary methods to address policy relevant questions related to security and energy issues.

III. BACKGROUND

The Pickering Nuclear Generating Station (hereinafter, "Pickering" or "NGS") is located on the north shore of Lake Ontario, in the City of Pickering in the regional municipality of Durham, Ontario. The facility lies 32 km northeast of downtown Toronto and 21 km southwest of Oshawa. The facility is owned and operated by OPG. Pickering consists of eight nuclear reactors, Units 1-4 (Pickering A) and Units 5-8 (Pickering B). Currently, Units 1 and 4, and Units 5-8 are operational, while Units 2 and 3 have been placed in a safe storage state.[4]

In August 2018, the CNSC renewed the licence for Pickering for a period of 10 years. During the relicensing hearing, the Commission noted that should OPG intend to operate any NGS Unit beyond December 31, 2024, a decision from the Commission via a public hearing process would be required to change OPG's licencing basis.[5]

In June 2023, OPG submitted a licence amendment application requesting to operate NGS units 5-8 to December 31, 2026 (units 1 and 4 would adhere to the current licence condition of being shut down by December 31, 2024). Additionally, OPG's application is requesting the Commission's authorization for an increased operating limit for units 5-8 pressure tubes from up to 295,000 effective full power hours (EFPH) to up to 305,000 EFPH.[6]

This licence amendment application marks the third request by OPG to extend Pickering's operating licence beyond its design life. Prior to the 2018 licence renewal hearing, in 2012 OPG applied to renew Pickering's licence for a 5-year period ending in December 2020. Because 2020 was beyond the assumed design life of certain operational parts in the NGS, especially the pressure tubes, OPG was required to demonstrate that Pickering could be operated safely until the end of 2020. The CNSC renewed the licence for Pickering on August 9, 2013, valid to August 31, 2018, but imposed hold points to review additional safety information from OPG about the ability of the pressure tubes and other components to withstand longer operations. The hold- points were removed in 2014.

[4] Canadian Nuclear Safety Commission, "CMD 24-H5: A Licence Amendment – Ontario Power Generation Inc. Pickering Nuclear Generating Station," (27 February 2024), p 4 **[Staff CMD]**
[5] *Ibid*, p 5
[6] *Ibid*, p 6

The Intervenors have previously expressed concerns about the renewal and extension of operations at Pickering.[7] The Pickering NGS reactors are the oldest operating nuclear power reactors in Canada. The original Pickering Generating Station licence application to operate Unit 1 was submitted to the CNSC's predecessor, the Atomic Energy Control Board on August 14, 1970; the construction permit was issued February 24, 1966. The original application to operate Unit 5 was submitted to the then Atomic Energy Control Board on May 6, 1980; the original construction permit had been issued July 19, 1974.[8]

As Canada's oldest operating nuclear power reactor, the Pickering NGS has had a history of poor performance and created a risk to the public. It has also been plagued by substantial operating costs while producing energy that is not needed in the face of much more feasible alternative sources of power. Just as CELA submitted at the 2018 licence renewal hearing, CELA again reiterates that there are significant concerns as to whether OPG will be able make adequate provision for the protection of the environment and the health and safety of persons if it were to be granted an extension of operations at Pickering units 5-8 beyond the current end date. For this and the reasons outlined below, the Intervenors request the CNSC deny OPG's application to extend the operations of Pickering units 5-8 beyond the current expiration of December 31, 2024.

IV. PRELIMINARY MATTERS & PROCEDURAL CONCERNS

Transparency, accessibility and disclosure of documents of critical value should be a priority in licencing stages

During the research and revision processes to prepare this submission, the Intervenors encountered several barriers to ensuring that the hearing process is transparent and accessible for members of the public to adequately engage with the licencing amendment materials.

Firstly, the Intervenors raise the issue of ease of access for reviewing documents. When reading through OPG's written submission, there are numerous references made, and various reports and studies mentioned.[9] However, these references are not hyperlinked within the 135-page document. The references within the document are not hyperlinked to the References section at the end of the

[7] Canadian Environmental Law Association, "Evaluating Environmental Protection and Emergency Preparedness at the Pickering Nuclear Generating Station (Ref. 2018-H-03)" (7 May 2018), online: https://cela.ca/wp-content/uploads/2019/07/1189-CELA-Submission-to-the-CNSC-Pickering-Licence-Renewal-2018-H-03.pdf , p 10 **[CELA submission 2018]**

[8] Canadian Environmental Law Association, "Emergency Planning at Pickering NGS - Submission to the CNSC," (May 3, 2013), online: http://www.cela.ca/sites/cela.ca/files/899PickeringEmergencyPlanning.pdf

[9] Ontario Power Generation, "Written submission from Ontario Power Generation Inc." CMD 24-H5.1 (16 June 2023). *For example*, see p 25, where there are references to Pickering's Periodic Safety Review 2 (PSR2-B), as well as CNSC regulatory documents and International Atomic Energy Agency information, but no easy way to access these references. **[OPG CMD]**

document. This requires the reader to scroll to the bottom of the document to track down the referenced document. To further complicate a reader's search for references, the Refence section does not provide hyperlinks to each reference. This means that the reader must conduct a web search to locate important documents, assuming that the document is publicly available for review. Members of the public do not necessarily have the luxury of time scour a document to find a reference made within the proponent's application, and then look up the referenced document separately. When reading through the CNSC Staff's CMD by comparison, there are hyperlinked references throughout the document, which greatly assists a reader's research process. The Intervenors submit that when preparing licencing applications, proponents should be required to hyperlink references made within the document.

Secondly, the Intervenors note that there is a lack of procedural fairness associated with the volume of complex technical documents for the licence amendment versus the time and participant funding available to review the information. While the Intervenors have numerous concerns surrounding the extension of operations at units 5-8, there is simply a lack of resources for members of the public and intervenors to heavily engage with materials to flesh out concerns surrounding safety given the level of funding made available to the funded Intervenors. Comparing these resources to other major federal energy tribunals such as the National Energy Board, the CNSC funding and resulting participation process is inadequate.

Thirdly, there are documents that are not readily available for review prior to the hearing. For instance, in the Periodic Safety Review 2-B (PSR2-B), "Appendix B: Integrated Implementation Plan Resolution Action Overview" shows a list of Resolution Action targets with their completion dates. There are targets as late as 30 November 2024, and the "overall safety evaluation" target date being 31 December 2024, for "correspondence confirming that Pickering NGS safety analysis results in supporting of commercial operations of Pickering Units 5 to 8 to the end of 2026 are submitted by OPG to the CNSC"[10] The Intervenors submit the public should have a chance to see these reports and the CNSC's evaluation of these reports and comment on it before the Commission approves continued operations at Pickering. In other words, the decision timeline set out in PSR2-B undercuts any possibility of meaningful public input during this hearing. For meaningful public engagement to occur, the CNSC needs to ensure there is adequate access and resources available for the members of the public to access and digest complex materials associated with the proponent's requested amendments to Pickering's licence.

Finally, another issue in terms of transparency and fair access to crucial documents stems from the process of document retrieval. During a workshop hosted by OPG in March 2024, a number of intervenors sought clarity from OPG about the process of requesting documents that are not in the possession of the CNSC. It turned out that OPG had not thought about how members of the public

[10] Ontario Power Generation, "Pickering NGS Periodic Safety Review 2-B (PSR2-B) Integrated Implementation Plan" Appendix B, p 45 of 56

and intervenors would access certain documents within the licence application. Initially, intervenors were instructed by OPG to file a freedom of information (FOI) request to access documents from OPG. Submitting an FOI request would take too long for an individual to gain access to specific documents. This process would prohibit an individual from receiving, reading, and responding to the contents of a requested document before the hearing would occur. The Intervenors submit that an FOI request is an unacceptable process for obtaining materials for a public hearing, and should not be the process suggested to member of the public seeking additional documents.

After intervenors expressed concerns to OPG about the unfair recommendation to submit an FOI request, OPG assured that there would be a designated email address established for requesting documents from OPG. However, subsequent advice to the intervenors indicated that some documents might still require a FOI request. The Intervenors submit that proponents should be required to, at a minimum, ensure there is a contact point to quickly and easily request documents for a public hearing. The Intervenors further submit that to best achieve procedural fairness, a proponent should be required to host a public accessible database of all referenced documents (with the exception of those with security-sensitive content) to ensure members of the public have quick and easy access to documents without awaiting a response from a proponent employee. Even in case of security concerns these should be thoroughly scrutinized by the Commission to ensure that no more information is protected than absolutely necessary.

Recommendation No. 1: To increase transparency, the Intervenors submit that OPG should be required to make references easily accessible via hyperlinks within documents.

Recommendation No. 2: To ensure there is meaningful public engagement and input during public hearings such as this to extend the operations at an ageing nuclear power plant, it is essential that there is ample time provided for the public to have access to and read through essential documents and reports before a decision can be made to approve licencing amendments and renewals.

Recommendation No. 3: An FOI request is an unacceptable process for obtaining materials for a public hearing, and should not be the process suggested to member of the public seeking additional documents.

Recommendation No. 4: Proponents should be required to, at a minimum, ensure there is a contact point to quickly and easily request documents for a public hearing. Furthermore, to best achieve procedural fairness, a proponent should be required to host a public accessible database of all referenced documents (with the exception of those with security-sensitive content) to ensure members of the public have quick and easy access to documents without awaiting a response from a proponent employee.

V. ACTION REQUESTED OF THE COMMISSION

After reviewing submissions by CNSC staff and OPG, the Intervenors submit that the CNSC should deny OPG's request to extend the operations of Pickering Units 5-8 to December 2026. Under the NSCA, the Commission has an obligation to "…prevent unreasonable risk, to the environment and to the health and safety of persons, associated with that development, production, possession or use [of nuclear substances].."[11] Upon reviewing the relevant documents associated with OPG's application, the Intervenors identified several areas of concern that would contravene the objective of the CNSC, namely: inadequate emergency planning and evacuation planning measures; risks associated with Pickering's ageing parts and facilities; a history of poor performance at Pickering; and inadequate consideration of climate change impacts.

These concerns are discussed below, along with a number of recommendations and requests for the CNSC to adopt in response to OPG's request to extend the operations at this ageing nuclear power facility.

Based on the findings within this submission, in the interest of protecting the environment and ensuring public health and safety, the Intervenors submit that the Commission cannot, in good conscience, approve an extension of the operation of Pickering Units 5-8 to 2026.

Because the operation timeframe for units 5-8 should not be extended, the Intervenors further submit that the CNSC should not grant OPG's request for an increased operating limit for units 5-8 pressure tubes from up to 295,000 EFPH to up to 305,000 EFPH.

A. Emergency Planning and Evacuation Planning are inadequate

One of the primary concerns of the Intervenors collectively is ensuring there is adequate emergency planning measures in place to protect the public. This is an issue that the intervenors have vocalized in various public engagement opportunities with the CNSC. During the 2018 licence renewal application for Pickering, CELA expressed concerns surrounding emergency response planning and preparedness, urging the CNSC to exercise its stringent oversight role as to whether emergency planning and preparedness has been proven prior to exercising its discretion with regards to the Pickering NGS licence renewal.[12]

These concerns remain relevant for OPG's request to extend operations at the NGS units 5-8, and the Intervenors urge the Commission to exercise its stringent oversight role and review OPG's emergency plans for Pickering to make a determination whether the risk to the public is acceptably low per section 24(4) of the NSCA. The Intervenors submit the risk is not acceptably low. On a

[11] *Nuclear Safety and Control Act*, SC 1997, c 9 at s 9(a)(i)
[12] CELA submission 2018, p 43

full review of emergency preparedness and especially evacuation planning, the state of emergency preparedness in the Durham Region and beyond is not sufficient.

The concern described above about the risk of severe accidents is exacerbated by weaknesses in plans for how to deal with the large population in the event of a severe accident that requires evacuation of people. As of 2021, the population of the Toronto census metropolitan area (CMA) was 6.2 million, which constitutes nearly 17 percent of the entire population of Canada.[13] Between 2016 and 2021, the population of the Durham Region Census Division grew by 7.3 percent. This rapid population growth rate means that plans for emergency management can quickly become obsolete.

In the event of a severe accident with offsite releases, a significant fraction of these people could have to evacuated.

In the case of the Fukushima Daiichi accidents, areas as far away as 50 km from the site had to be evacuated due to high radiation levels.[14] The U.S. Nuclear Regulatory Commission recommended that U.S. citizens living with 80 kilometers of the Fukushima plants evacuate the area. Finally, in response to Prime Minister Naoto Kan's instructions, the Japan Atomic Energy Commission drew up a worst-case scenario for the nuclear accident, which explained that "all residents living within 170 kilometers or more of the Fukushima plant might need to be relocated, and relocation might need to be advised for those living within 250 kilometers…evacuation of the 30 million residents in the Tokyo metropolitan area could become necessary, depending upon wind direction".[15]

In 2017, CELA recommended that the province expand "the current 50 km secondary zone to 100 km from every Ontario and non-Ontario reactor, and provide for education, outreach, preparation, inventories, communication channels, contingency planning and other efforts needed to be prepared to restrict ingestion and provide alternative food and water in the case of a severe offsite accident".[16] Given the growing population of the area, CELA reiterates that recommendation.

According to OPG's application, the 2023 Pickering Evacuation Time Estimate (ETE) study determined "the time to evacuate 90% of the population from Pickering Detailed Planning Zone (DPZ) (i.e., out to 10 km) ranges from 4 hours and 45 minutes to 8 hours and 25 minutes (e.g.,

[13] https://www.durham.ca/en/regional-government/resources/Documents/Council/CIP-Reports/CIP-Reports-2022/2022-INFO-31.pdf

[14] Greenpeace, "Lessons from Fukushima", February 2012 at p. 18

[15] Funabashi, Y., and K. Kitazawa. "Fukushima in Review: A Complex Disaster, a Disastrous Response." *Bulletin of the Atomic Scientists* 68, no. 2 (2012): 9–21 at p. 10.

[16] Canadian Environmental Law Association, "Discussion Paper on Planning Basis Review and Recommendations and List of Proposed Changes to the PNERP 2009" (28 July 2017), online: https://cela.ca/provincial-nuclear-emergency-response-plan-pnerp-2009-discussion-paper/ at p 23

wintertime, with heavy snow at mid-day on a weekday)."[17] OPG's application does not discuss the even longer estimates to evacuate 100% of the DPZ: "the ETE for the 100th percentile are significantly longer than those for the 90th percentile, ranging from 4:30 to 10:15 for Regions R01 through R03 and 11:00 to 17:00 for Region R04. This is the result of significant congestion within the study area."[18] In the event of a severe accident, taking up to 17 hours to evacuate 100% of just the DPZ is deeply concerning, as this timing does not capture the impact of a larger zone radius (i.e., our proposed 100 km radius).

The Pickering ETE study notes:

> Current Provincial Nuclear Emergency Response Plans indicate pre-planned protective actions are to be developed for the Detailed Planning Zone Outer Ring (DPZ Outer Ring) and contingency planning and arrangements are to be made for the Contingency Planning Zone (CPZ). As such, this report focuses on the DPZ Outer Ring, but provides information, including data, analyses and results, for the CPZ for contingency planning purposes.[19]

With the current radius limitations, the evacuation times are likely downplaying the potential public confusion and even chaos, and increased evacuation times needed for a larger area in the event of a severe offsite accident. Given the lengthy evacuation times needed to mitigate the health impacts on all the surrounding population of a severe accident at this ageing nuclear plant, the CNSC should insist on the expansion of the planning zones around Pickering.

Another area of concern arising from OPG's application is the failure to carefully sufficiently consider potential accidents with large offsite releases. Hazards such as malfeasance or air accidents, for instance, are not considered by OPG, nor are unexpected chains of events and unexpected consequences of human error. When assessing internal and external hazards that may result in large offsite releases, certain hazards are have been screened out because OPG has deemed the likelihood of occurrence as low, even though risks such as those mentioned above are not outside the realm of possibility of occurring. When discussing the probabilistic safety assessment (PSA), OPG states:

> PSAs are conducted separately for internal and external types of hazards, in particular, they are for internal events, internal fires, internal floods, seismic hazards, and high wind hazards. Many other hazards are also examined and addressed as part of the PSA hazard

[17] OPG CMD at p 36.
[18] Ontario Power Generation, "Pickering Nuclear Generating Station Evacuation Time Estimate" (24 March 2023), online: https://www.opg.com/documents/pickering-ngs-development-of-evacuation-time-estimates/ p ES-2.
[19] *Ibid*, p xiii

> screening process (some hazards – meteorites, for example – are deemed to be of such low likelihood that they were screened out, and not developed into PSA models).[20]

CELA has previously expressed concerns to the CNSC about OPG's lack of consideration for severe accidents, and planning for worst-case scenarios due to low likelihood of occurrence.[21] While meteorite strikes might not be a significant risk, screening out hazards from a PSA on the basis of the OPG's evaluation of the likelihood of the hazard raises concerns that OPG is not prepared for worst case scenario accidents. Many scholars have explained why risk assessments are often unreliable because they cannot account for "unknown unknowns" and estimates of very low probabilities of accidents are not credible.[22] No nuclear plant is immune to the possibility of a major accident, even in case of reactors featuring multiple safety systems. The case of the multiple reactor meltdowns at the Fukushima-Daiichi nuclear power plant involved the failure of many safety systems due to a single root cause – the earthquake, which was also the cause of the tsunami.

The failure was also institutional. The Tokyo Electric Power Company that owned and operated the Fukushima reactors and the nuclear safety agency in Japan both convinced themselves that the reactors were safe. CELA is concerned that safety assessments of the ageing Pickering nuclear plant might be suffering from similar blind spots. As CELA has emphasised in previous submissions before the CNSC, simply because the likelihood of an event occurring is very low, it does not guarantee that a severe nuclear reactor accident will never happen.

The Intervenors submit that the safety assessment for Pickering is inadequate because it does not plan for severe accident scenarios and the CNSC cannot come to the determination that the risk associated with extending units 5-8 would not pose a risk to human health and the environment.

Recommendation No. 5: The province should expand the current 50 km secondary zone to 100 km from every Ontario and non-Ontario reactor, and provide for education, outreach, preparation, inventories, communication channels, contingency planning and other efforts needed to be

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[20] OPG CMD at p 29, *emphasis added*.

[21] *See for instance*, CELA, "Comments on the applicability of the Darlington New Nuclear Project's environmental assessment and plant parameter envelope to Ontario Power Generation's selected BWRX-300 reactor technology" (17 November 2023), online: https://api.cnsc-ccsn.gc.ca/dms/digital-medias/CMD24-H2-8.pdf/object?subscription-key=3ff0910c6c54489abc34bc5b7d773be0 p15

[22] Raju, Suvrat. "Estimating the Frequency of Nuclear Accidents." *Science & Global Security* 24, no. 1 (2016): 37–62; Ramana, M. V. "Beyond Our Imagination: Fukushima and the Problem of Assessing Risk." *Bulletin of the Atomic Scientists*, April 19, 2011. https://thebulletin.org/2011/04/beyond-our-imagination-fukushima-and-the-problem-of-assessing-risk/; Downer, John. "Disowning Fukushima: Managing the Credibility of Nuclear Reliability Assessment in the Wake of Disaster." *Regulation & Governance* 8, no. 3 (2014): 287–309. https://doi.org/10.1111/rego.12029; Downer, John, and M. V. Ramana. "Empires Built on Sand: On the Fundamental Implausibility of Reactor Safety Assessments and the Implications for Nuclear Regulation." *Regulation & Governance* 15, no. 4 (2021): 1304–25. https://doi.org/10.1111/rego.12300.

prepared to restrict ingestion and provide alternative food and water in the case of a severe offsite accident.

Recommendation No. 6: CNSC should declare that the accident analysis prepared for Pickering is inadequate and it cannot come to a determination that the risk associated with extending units 5-8 would not pose a risk to human health and the environment.

B. Ageing Parts and Facilities at the Pickering Nuclear Generating Station pose a cause for concern

CELA emphasizes that the extension request is for a nuclear reactor with old and ageing parts and equipment, while the size of the surrounding population has continued to increase, with more population planned for the area. The request is also being made when OPG has not completed all of the necessary outstanding safety studies relevant to this extension request. We note that Pickering NGS Periodic Safety Review 2-B (April 2023) includes no less than thirteen (13) unresolved issues, many of which have great safety significance. Since they have not been resolved, there is an insufficient basis to authorize continued operations past December 2024. Among the critical components that are at risk of aging related problems are fuel channels and steam generators.

Fuel Channels and their degradation pose major challenges. Each of these fuel channels is in a different location and operating at a different individual power level. Therefore, their degradation will not occur at the same rate; modelling these different components and how they might behave in the event of an accident is very complex.[23]

These concerns should be considered in view of the insights from the Fukushima accident and dangers of climate change related events.

When they melted down, the first three reactor units at Fukushima Daiichi had been operating for 41 years, 38 years and 37 years respectively.[24] Pickering units 5 through 8 have been operating for 42 years, 41 years, 40 years, and 39 years.[25] Old nuclear plants are particularly susceptible to accidents, the likelihood of which can be described by something called the bathtub curve.[26] The failure rate is initially high due to manufacturing problems and operator errors associated with new technology. Then curving like a tub, the failure rate declines with experience and rises again as

[23] Nijhawan, Sunil. "Severe Accident Related Vulnerabilities, Potential Design Enhancements and Opportunities for International Cooperation in Risk Reduction in Pressurized Heavy Water Reactors." American Society of Mechanical Engineers Digital Collection, 2015. https://doi.org/10.1115/NUCLRF2015-49574.
[24] https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=JP
[25] https://pris.iaea.org/PRIS/CountryStatistics/CountryDetails.aspx?current=CA
[26] Lochbaum, David. "The Bathtub Curve, Nuclear Safety, and Run-to-Failure." *The Equation* (blog), November 17, 2015. https://blog.ucsusa.org/dlochbaum/the-bathtub-curve-nuclear-safety-and-run-to-failure/.

aging related wear and tear starts increasing. In other words, the dangers of continuing operations are high and increasing.

The problem of ageing is likely to be even more severe in the case of the Pickering units because they were intended to be shut down. The scale of the challenge might be appreciated from looking at the case of the Diablo Canyon nuclear plant in California in the USA. The two reactor units were to be shut down in 2024 and 2025 but due to political reasons, there are now plans to extend operations at this reactor.[27] This has led to an evaluation of its safety.

In July 2022, the Chair of the Diablo Canyon Independent Safety Committee explained why this case was very difficult: "so many different programs and projects and so on have been put in place over the last half a dozen years predicated on that closure in 2024-25 and each one of those would have to be evaluated and some of them are okay and some of them won't be and some are going to be a real stretch and some are going to cost money and some of them aren't going to be able to be done maybe".[28] This is the kind of challenge that we are confronting at Pickering too.

One should note that in February 2011, just a month before the devastating accident at the Fukushima Daiichi nuclear plant, Japan's Nuclear and Industrial Safety Agency, approved a 10-year extension for the oldest of the six reactors at the power station. This was done despite warnings about its safety.[29] This decision, in part, resulted in the disastrous nuclear accident in 2011, posing immense consequences to people's health, the environment, and the economy.

The Intervenors submit that with so many uncertainties surrounding the ageing equipment of Pickering, allowing the NGS to continue operating beyond 2024 would irresponsibly place the environment and the public at enhanced and unsupportable risk. Due to the ageing parts and equipment at Pickering, along with the numerous unresolved issues within the Periodic Safety Review 2-B, the risk for severe incidents is too great to grant OPG's application.

Recommendation No. 7: The Commission should deny the application for a further extension of the aging units since allowing the NGS to continue operating beyond 2024 would irresponsibly place the environment and the public at risk.

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[27] Sara Nelson and M. V. Ramana. "Managing Decline: Devaluation and Just Transition at Diablo Canyon Nuclear Power Plant." *Environment and Planning A: Economy and Space* 55, no. 8 (2023): 1951–69. https://doi.org/DOI: 10.1177/0308518X231167865.

[28] *June 22, 2022: Diablo Canyon Independent Safety Committee--Diablo Relicensing Concerns Discussed*, 2022. https://www.youtube.com/watch?v=WjTq-ybnaSY.

[29] Tabuchi, Hiroko, Norimitsu Onishi, and Ken Belson. "Japan Extended Reactor's Life, despite Warning." *New York Times*, March 22, 2011.

C. Poor Performance at the Pickering Nuclear Generating Station

Pickering's performance data as reported by the International Atomic Energy Agency's ("IAEA") Power Reactor Information System Database does not suggest that these reactors are performing well.[30] The load factors for the last two years reported on the PRIS database for Pickering-5 are below its average lifetime load factor, with the 2022 load factor being a paltry 52.8 percent. Likewise, Pickering-7 and Pickering-8 fell below their lifetime averages in 2021 (with load factors of only 62.8 percent and 54.4 percent respectively).

According to the IAEA's Operating Experience with Nuclear Power Stations in Member Stations document for 2023, Pickering-5 was not functioning for 3826 hours (44 percent of the time), which includes 57 hours because of environmental conditions such as lack of cooling water due to dry weather, cooling water temperature limits, flood, storm, lightning; Pickering-6 for 915 hours, including 94 hours because of environmental conditions; Pickering-7 for 790 hours; and Pickering-8 for 971 hours.[31] In other words, two of the units were not contributing energy for extended periods of time, and two were shut down for 2-4 days due to causes that might well have to do with climate change.

In addition to the findings by the IAEA, the CNSC Staff CMD itself highlights several concerns indicating poor performance at Pickering.

Firstly, in the application of the "as reasonably low as achievable" (ALARA) principle—which is a requirement under the *Radiation Protection Regulations,*[32] through a desktop inspection in December 2022, CNSC staff identified non-compliant findings of low and negligible safety significance:

> During the inspection, CNSC staff reviewed OPG's 5-year ALARA plan and found that OPG did not have the governance support documents in place to drive the creation and content of this plan. As a result, CNSC staff requested that OPG develop governance support documents to formalize the requirements for creating and maintaining a 5-year ALARA plan and create an implementation plan for corrective actions. CNSC staff will continue to monitor and assess OPG's implementation of these corrective actions and its 5-year ALARA plan at the Pickering NGS.[33]

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[30] IAEA, "Power Reactor Information System (PRIS)", Canadian plant details, online: https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=CA

[31] IAEA, "Operating Experience with Nuclear Power Stations in Member States 2023 Edition" (2023), online: https://www-pub.iaea.org/MTCD/Publications/PDF/OPEX_2023_web.pdf.

[32] Radiation Protection Regulations, under the *Nuclear Safety and Control Act* (SOR/2000-203)

[33] CNSC Staff CMD, p 49.

While the non-compliance with the ALARA principle's application was deemed to be of "low" and "negligible" safety significance by CNSC staff, the failure to have a properly developed ALARA plan in place is concerning, as these plans are intended to ensure that contamination levels and radiation doses received by individuals are monitored, controlled and maintained as low as reasonably achievable. By failing to implement such important safety plans, the Intervenors are concerned that allowing Pickering to continue operations beyond 2024, poses a risk that important health and safety measures and plans will fall to the wayside. As Pickering continues to age and reach the end of its operational lifespan, there are concerns surrounding worker complacency resulting in further non-compliances. The Intervenors submit that the CNSC should not discount the non-compliance with important safety measures.

Yet another non-compliance highlighted within the CNSC staff CMD concerned the maintenance of fire emergency response equipment and the utilization of firefighting equipment and tools.[34] While OPG has implemented corrective action plans to address these non-compliances, it is concerning that fire emergency equipment within a nuclear power plant was not being properly maintained.

CNSC staff have noted that in previous years (2021 and 2022), Pickering failed to meet CNSC staff standards for the Security SCA. Due to an increase in regulatory scrutiny, the CNSC staff note that they are satisfied that OPG is "adequately addressing the identified issues," and "overall, CNSC staff conclude that OPG's programs within the security SCA are adequate for continued commercial operations until the end of 2026."[35] This "enhanced regulatory scrutiny" involved the issuance of an Administrative Monetary Penalty (AMP) to OPG in 2023 as a result of a failure to comply with a licence condition in relation to its security program at the Pickering and Darlington Nuclear Generating Stations. The AMP was issued to promote compliance and deter recurrence."[36] The issuance of an AMP by the CNSC is uncommon, so the requirement to issue a monetary penalty to encourage OPG to maintain compliance at not one but two nuclear power plants does not exude confidence in the ability to maintain proper safety measures at Pickering for two more years beyond the current licence.

While most instances of non-compliance at Pickering have been deemed by CNSC staff to be low risk, the frequency of these occurrences in Pickering's later stages of its operation suggest that safety measures are not being treated as a top priority in day-to-day operations at Pickering. The Intervenors submit that the Commission needs to consider these instances of non-compliance, and balance their frequency of occurrence with the risk of accidents impacting the environment and the public.

[34] *Ibid* at p 64.
[35] *Ibid* at p 69.
[36] *Ibid* at p 69-70.

Recommendation No. 8: Given the poor performance of Pickering's reactors, the CNSC should deem units 5-8 as having reached the end of their operational phase, and deny a further extension of their operations.

Recommendation No. 9: The Commission should consider these instances of non-compliance, their frequency of occurrence, the age of the plant, and the consequences of a severe accident with large offsite releases that would impact the environment and the public.

D. The Pickering Nuclear Generating Station is not prepared to adapt to and mitigate the impacts of Climate Change

The impacts of climate change on nuclear reactors includes various climate-related hazards such as heat waves, floods, droughts, storms, lightning events and wildfires which can pose challenges to nuclear safety, necessitating enhanced safety assessments and adaptive management strategies.[37] Rising temperatures can threaten nuclear reactors, as they rely on large external sources of water for cooling. In the case of Canadian reactors, this water is usually from the Great Lakes, or the Bay of Fundy in the case of Point Lepreau. Heightened water stress, either from reduced availability or competing uses, is a concern since nuclear reactors require significant amounts of water for cooling purposes.[38]

Higher water temperatures can reduce the thermal efficiency of nuclear reactors. Studies of climate-driven alterations in energy sectors in Ontario, Canada, show anticipated temperature increases and consequent loss of thermal efficiency in the output from reactors will result in reduced generation of electricity of somewhere between 1.5 and 2.5 percent by the 2050s, and between 2.5 and 4.0 percent by the 2080s.[39] While the current proposed extension is for a much shorter time, the trend is clear.

A second source of reduced output is having to shut down nuclear reactors for shorter or longer periods of time because of extreme weather conditions, including high water temperatures.[40]

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[37] Linnerud, K., Mideksa, T. K., & Eskeland, G. S. (2011). The impact of climate change on nuclear power supply. *The Energy Journal*, 32(1); Kromp-Kolb, Helga, Nikolaus Muellner, Duck Kim, Jaroslav Holy, Sanna Syri, Jean-Yves Caneill, Gernot Thuma, et al. "Climate Change: Assessment of the Vulnerability of Nuclear Power Plants and Approaches for Their Adaptation." Paris: Nuclear Energy Agency, OECD, 2021.

[38] Linnerud, K. et al.

[39] Wang, S., Zhu, J., Huang, G., Baetz, B., Cheng, G., Zeng, X., & Wang, X. (2020). Assessment of climate change impacts on energy capacity planning in Ontario, Canada using high-resolution regional climate model. *Journal of Cleaner Production*, 274, 123026.

[40] *See*: Ahmad, Ali. "Increase in Frequency of Nuclear Power Outages Due to Changing Climate." *Nature Energy* 6, no. 7 (July 2021): 755–62. https://doi.org/10.1038/s41560-021-00849-y; *see also* Ahmad, Ali, Andrei Covatariu, and M. V. Ramana. "A Stormy Future? Financial Impact of Climate Change-Related Disruptions on Nuclear Power Plant Owners." *Utilities Policy* 81 (April 1, 2023): 101484. https://doi.org/10.1016/j.jup.2022.101484.

Reactors have to cease generating power because the required cooling cannot be ensured or because of other operational challenges.

We may already be witnessing such a climate driven impact as increasing temperatures lead to accelerating growth of algae and jellyfish, which can block input of water.[41] Climate impacts have already happened repeatedly at Pickering, most recently in 2018 when heavy rain and warmer temperatures when units 5, 6, 7, and 8 had to be shut down after algae clogged the cooling water intakes.[42] This has happened in earlier years too including in 2003, when OPG had to shut down Unit 7, and in 2005, when Units 5, 6, and 8 were shut down because of a large incursion.[43] Those shut downs prompted OPG to install a diversion net by the water intake and improving its operating procedures, at considerable cost.[44] But as the subsequent forced shut down of four units in 2018 showed, these changes have not eliminated the problem. As climate change intensifies, such shut downs will become more common and result in revenue being lost.

Within OPG's application, the only climate change mitigation discussion surrounds algae accumulation.[45] There is no discussion of how OPG plans to address and prevent other impacts of climate change from shutting down the Pickering reactors. For example, extreme weather events such as floods, heatwaves, wildfires, lightning events or extreme heat events are not discussed within the "Equipment Reliability and Climate Change" section of OPG's application. The Intervenors are concerned that Pickering is not currently equipped to address the impacts of climate change, and granting an additional 2 years of operation time for units 5-8 will not improve OPG's (lack of) mitigation measures. The Intervenors submit that without robust climate change mitigation strategies in place, it would be irresponsible for the Commission to extend the operation period for units 5-8 beyond 2024.

In April 2024, the U.S. Government Accountability Office (GAO) released a report titled "Nuclear Power Plants: NRC Should Take Actions to Fully Consider the Potential Effects of Climate Change." The GAO notes that "climate change is likely to exacerbate natural hazards—such as floods and drought. The risks to nuclear power plants from such hazards include damage to systems

[41] Lin, H., Zhang, S., Cao, R., Yu, S., Bai, W., Zhang, R., ... & Zhang, X. (2023). A review on the risk, prevention and control of cooling water intake blockage in coastal nuclear power plants. *Nuclear Engineering and Technology*.
[42] Dalton, David. "Four Pickering Units Safely Shut Down After Algae Clogs Cooling Water Intakes." *NucNet*, February 22, 2018. https://www.nucnet.org/news/four-pickering-units-safely-shut-down-after-algae-clogs-cooling-water-intakes.
[43] Hamilton, Tyler. "Algae Prompt Reactor Shutdown." *Toronto Star*, August 10, 2007. https://www.thestar.com/business/algae-prompt-reactor-shutdown/article_efafc9ca-d686-5e22-ab91-5300e2f5e1b0.html.
[44] Kopytko, Natalie. "Spineless Attacks on Nuclear Power Plants Could Increase." *Bulletin of the Atomic Scientists*, February 19, 2015. https://thebulletin.org/2015/02/spineless-attacks-on-nuclear-power-plants-could-increase/.
[45] OPG CMD at p 94-95.

and equipment that ensure safe operation."[46] The GAO noted that the Nuclear Regulatory Commission (NRC) does not fully consider potential increases in risk from climate change, and therefore recommended the NRC assess whether its existing processes adequately address climate risks and develop and implement a plan to address any gaps identified.[47] The Intervenors recommend that the CNSC conduct an assessment and mapping of vulnerability of Canadian nuclear power plans to climate change hazards, similar to the one recommended in the GAO report.

Recommendation No. 10: Without robust climate change mitigation strategies in place, it would be irresponsible for the Commission to extend the operation period for units 5-8 beyond 2024.

Recommendation No. 11: The CNSC should conduct an assessment and mapping of vulnerability of Canadian nuclear power plans to climate change hazards, similar to those shown in the GAO report.

VI. CONCLUSION

For the foregoing reasons provided in this submission, DNA, SHA, and CELA submit it would be contrary to the responsibility of the Commission to protect the environment and ensuring the health and safety of persons if it were to allow Pickering Units 5-8 to operate beyond 2024, and recommend the CNSC issue an order:

(1) Granting Durham Nuclear Awareness, Slovenian Home Association, and the Canadian Environmental Law Association the status of intervenor;

(2) Granting Durham Nuclear Awareness, Slovenian Home Association, and the Canadian Environmental Law Association the opportunity to make an oral presentation at the January 2024 public hearing;

(3) Denying OPG's request to amend the PROL of Pickering Nuclear Generating Station to operate units 5-8 to December 31, 2026; and

(4) Denying OPG's request to amend the PROL to increase the pressure tube operating limit to 305,000 EFPH.

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[46] U.S. Government Accountability Office, "Nuclear Power Plants: NRC Should Take Actions to Fully Consider the Potential Effects of Climate Change." GA-24-106326 (2 April 2024), online: https://www.gao.gov/products/gao-24-106326?utm_campaign=usgao_email&utm_content=topic_naturalresources&utm_medium=email&utm_source=govdelivery

[47] *Ibid.*

Sincerely,

On behalf of

CANADIAN ENVIRONMENTAL LAW ASSOCIATION
DURHAM NUCLEAR AWARENESS
SLOVENIAN HOMEOWNERS ASSOCIATION

Sara Libman

Sara Libman
Legal Counsel

APPENDIX A - SUMMARY OF RECOMMENDATIONS

Recommendation No. 1: To increase transparency, the Intervenors submit that OPG should be required to make references easily accessible via hyperlinks within documents.

Recommendation No. 2: To ensure there is meaningful public engagement and input during public hearings such as this to extend the operations at an ageing nuclear power plant, it is essential that there is ample time provided for the public to have access to and read through essential documents and reports before a decision can be made to approve licencing amendments and renewals.

Recommendation No. 3: An FOI request is an unacceptable process for obtaining materials for a public hearing, and should not be the process suggested to member of the public seeking additional documents.

Recommendation No. 4: Proponents should be required to, at a minimum, ensure there is a contact point to quickly and easily request documents for a public hearing. Furthermore, to best achieve procedural fairness, a proponent should be required to host a public accessible database of all referenced documents (with the exception of those with security-sensitive content) to ensure members of the public have quick and easy access to documents without awaiting a response from a proponent employee.

Recommendation No. 5: The province should expand the current 50 km secondary zone to 100 km from every Ontario and non-Ontario reactor, and provide for education, outreach, preparation, inventories, communication channels, contingency planning and other efforts needed to be prepared to restrict ingestion and provide alternative food and water in the case of a severe offsite accident.

Recommendation No. 6: CNSC should declare that the accident analysis prepared for Pickering is inadequate and it cannot come to a determination that the risk associated with extending units 5-8 would not pose a risk to human health and the environment.

Recommendation No. 7: The Commission should deny the application for a further extension of the aging units since allowing the NGS to continue operating beyond 2024 would irresponsibly place the environment and the public at risk.

Recommendation No. 8: Given the poor performance of Pickering's reactors, the CNSC should deem units 5-8 as having reached the end of their operational phase, and deny a further extension of their operations.

Recommendation No. 9: The Commission should consider these instances of non-compliance, their frequency of occurrence, the age of the plant, and the consequences of a severe accident with large offsite releases that would impact the environment and the public.

Recommendation No. 10: Without robust climate change mitigation strategies in place, it would be irresponsible for the Commission to extend the operation period for units 5-8 beyond 2024.

Recommendation No. 11: The CNSC should conduct an assessment and mapping of vulnerability of Canadian nuclear power plans to climate change hazards, similar to those shown in the GAO report.

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CELEBRATING 50 Years 1970 - 2020 OF ENVIRONMENTAL PROTECTION

SUBMISSION BY THE CANADIAN ENVIRONMENTAL LAW ASSOCIATION TO THE CANADIAN NUCLEAR SAFETY COMMISSION REGARDING THE REGULATORY OVERSIGHT REPORT FOR CANADIAN NUCLEAR POWER GENERATING SITES: 2022 AND THE MID-TERM UPDATE OF LICENSED ACTIVITIES FOR THE PICKERING NUCLEAR GENERATING STATION

October 29, 2023

Prepared by
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Expert Review by
M.V. Ramana
Dr. Farrukh A. Chishtie

I. INTRODUCTION

This submission is filed in response to the Canadian Nuclear Safety Commission's ("CNSC") Notice of Participation at a Commission Meeting and Participant Funding[1] in respect of the *Regulatory Oversight Report for Canadian Nuclear Power Generating Sites: 2022* (herein "ROR")[2] and the mid-term update on the Pickering Nuclear Generating Station[3]. A public meeting with respect to this matter is scheduled for December 13-14, 2023.

The Canadian Environmental Law Association ("CELA") has prepared this written submission with the support of the expert review of M.V. Ramana and Dr. Farrukh A. Chishtie, with the intention of presenting oral submissions during the December 13-14, 2023 Commission meeting.

CELA has reviewed the CMDs for the December 13-14, 2023 meeting associated with this ROR. While this submission is commenting on the whole ROR, CELA has chosen to focus on the discussions surrounding the Pickering Nuclear Generating Station ("Pickering"), since the mid-

[1] CNSC, Notice of Participation at a Commission Meeting and Participant Funding, online: https://www.cnsc-ccsn.gc.ca/eng/the-commission/participant-funding-program/opportunities/npgs-pmt.cfm.
[2] CNSC, Regulatory Oversight Report for Canadian Nuclear Power Generating Sites for 2022, CMD 23-M36 [**ROR 2022**].
[3] OPG, Mid-Term Update of Licensed Activities for the Pickering Nuclear Generating Station, CMD 23-M36.1 [**Mid-term Review**].

term update for this nuclear power generating station ("NPGS") will be a focus at the upcoming Commission meeting.

CELA is a non-profit, public interest law organization. For over 50 years, CELA has used legal tools to advance the public interest, through advocacy and law reform, in order to increase environmental protection and safeguard communities across Canada. CELA is funded by Legal Aid Ontario as a specialty legal clinic, to provide equitable access to justice to those otherwise unable to afford representation.

CELA has engaged in detailed research and advocacy related to public safety and environmental protection by seeking improvements to nuclear emergency preparedness. We have also appeared before the CNSC on a number of licensing matters, as well as the federal environmental assessment proceedings for multiple NPGS and proposed projects. CELA also has an extensive library of materials related to Canada's nuclear sector which is publicly available on our website.[4]

II. PICKERING MID-TERM REVIEW REPORT

In 2018, the CNSC held a public hearing for Ontario Power Generation's ("OPG") application to renew the Nuclear Power Reactor Operating Licence for the Pickering Nuclear Generating Station. The CNSC decided to renew Pickering's licence for a 10-year term, allowing Pickering to operate until August 31, 2028.[5] One of the conditions of the licence renewal required that, "around the mid-point of the 10-year licence period and no later than 2023, OPG shall present to the Commission a comprehensive mid-term update on its licenced activities at the PNGS."[6]

Upon reviewing OPG's Mid-Term Review, CELA is disappointed in the absence of comprehensive information presented within the Mid-Term Review. Much of the content within the review essentially amounts to a PR campaign for Pickering and OPG.[7] The Mid-Term Review fails to provide a substantive discussion of issues that have arisen at Pickering over the last 5-years.

Due to the timing of the publication of the Mid-Term Review, CNSC Staff were unable to provide commentary on the review within the ROR. Instead, Appendix H: "Pickering Nuclear Generating

[4] Canadian Environmental Law Association, online: www.cela.ca

[5] CNSC, Record of Decision for Application to Renew the Nuclear Power Reactor Operating Licence for the Pickering Nuclear Generating Station [**Record of Decision**].

[6] CNSC, Record of Decision at page 138, para 673, *emphasis added.*

[7] *For instance,* page 6 of the Mid-Term Review discusses OPG's Commitment to Excellence, noting that "Pickering NGS was recognized with Business Excellence Awards", and on page 8 of the Mid-Term Review, OPG comments on being named "one of Canada's Best Diversity Employers."

Station Mid-Term Update" provides "CNSC staff's regulatory oversight context on topics covered in OPG's mid-term update that were of public interest during the 2018 relicensing hearing."[8]

As Appendix H reveals, Pickering has received a "below expectations" rating for complying with the Security safety and control area ("SCA") two years in a row (2021 and 2022). Furthermore, "CNSC staff have increased regulatory oversight of the PNGS security program and issued an Administrative Monetary Penalty (AMP) to OPG in June 2023 to deter future non-compliances."[9] This AMP totaling $21,790[10] is not mentioned in within the Mid-Term Review, nor is the "below expectations" rating that Pickering has repeatedly received for the Security SCA. AMPs are not often issued for NPGS, and therefore this notable penalty should have been highlighted within the Mid-Term Review, along with a note about how OPG has assured CNSC that such problems will not recur.

In the Mid-Term Review, OPG has claimed that "Pickering NGS is exhibiting its strongest performance ever, including achieving its highest yearly production output."[11] However, the plant's performance data as reported by the International Atomic Energy Agency's ("IAEA") Power Reactor Information System Database suggests otherwise.[12] In each of the operating reactors at Pickering, there were at least two years when the reactor's performance fell below its average lifetime load factor. The only exception was Pickering-1 whose lifetime load factor is a paltry 66 percent. But even for Pickering-1, the 2020 load factor was just 49.6 percent and the 2022 load factor was 66.2 percent, barely over the lifetime average. In the case of the other Pickering reactors, Pickering-4 fell below its lifetime average in 2018 and 2020, Pickering-5 fell below its lifetime average in 2019, 2021, and 2022, Pickering-6 fell below its lifetime average in 2018 and 2020, Pickering-7 fell below its lifetime average in 2019 and 2021, and Pickering-8 fell below its lifetime average in 2018 and 2021. Since Pickering is not a particularly well-performing nuclear power plant, this poor record should have been clearly noted.

According to the IAEA's Operating Experience with Nuclear Power Stations in Member Stations document for 2023, Pickering-1 was not functioning for 2714 hours in the year 2022, or roughly 30 percent of the time; Pickering-4 for 740 hours; Pickering-5 for 3826 hours (44 percent of the time), which includes 57 hours because of environmental conditions such as lack of cooling water due to dry weather, cooling water temperature limits, flood, storm, lightning; Pickering-6 for 915 hours, including 94 hours because of environmental conditions; Pickering-7 for 790 hours; and

[8] CNSC, ROR 2022 at page 162.
[9] CNSC, ROR 2022 at page 163.
[10] CNSC, Regulatory Action—Ontario Power Generation (31 May 2023), online: https://www.cnsc-ccsn.gc.ca/eng/acts-and-regulations/regulatory-action/ontario-power-generation.cfm
[11] OPG, Mid-Term Review at page 1.
[12] IAEA, "Power Reactor Information System (PRIS)", Canadian plant details, online: https://pris.iaea.org/pris/CountryStatistics/CountryDetails.aspx?current=CA

Pickering-8 for 971 hours.[13] In other words, two of the units were not contributing energy for extended periods of time, and two were shut down for 2-4 days due to causes that might well have to do with climate change.

CELA **submits** that the Mid-Term Review should not just be a document boasting about a licensee's accomplishments during a licence period. It should address shortfalls and provide clarity on what is being done to alleviate and prevent these shortfalls from happening in the remainder of the licence term.

Climate change discussions within the Mid-Term Review are rather minimal, with no discussion of how OPG has been and plans to mitigate the impacts of climate change on Pickering. The Mid-Term Review focuses largely on the climate mitigation element of nuclear power production, highlighting reduction of greenhouse gases by such means. While there is mention of hazards like fires, floods, high winds and seismic events with related preparation as follows:

> External events such as fires, floods, high winds, and seismic events have a specific PSA for both Pickering NGS A and B which are utilized for ad-hoc assessments that require specific risk-informed insights. Pickering has successfully updated all models to utilize the latest EPRI Instantaneous Risk Monitor software (Phoenix) with features like remote risk quantification proved to be very beneficial during COVID for off-site support staff. Pickering has a multi-hazard risk monitor which is a CANDU first-of-a-kind that allows insights for all available internal/external monitors in one screen.[14]

For those mentioned hazards, the Mid-Term Review does not discuss the connection to climate change and related adaptation. A major concern in this regard is the record-breaking Canadian wildfire season in 2023 which also affected Ontario, such impacts can be anticipated as conceivable threats to operations of nuclear reactors. Moreover, climate related events such as droughts, algae growth and related actions are also entirely missing. This report therefore misses climate related impacts and associated adaptation actions.

Our submission provides a more in-depth discussion on climate change in Section III, as the issue of climate change impacts and mitigation measures extend beyond the Mid-Term Review of Pickering, and are applicable to the regulatory oversight of all NPGS.

Another topic that would benefit from greater discussion within the Mid-Term Review is the distribution of potassium iodide pills ("KI Pills"). According to the 2018 Record of Decision:

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[13] IAEA, "Operating Experience with Nuclear Power Stations in Member States 2023 Edition" (2023), online: https://www-pub.iaea.org/MTCD/Publications/PDF/OPEX_2023_web.pdf.

[14] OPG, Mid-Term Review at page 15.

> The Commission noted the suggestion in the intervention from E. Munro that OPG use modern means of communication to inform and encourage the public beyond the DPZ to order KI pills. The Commission requests that OPG consider this suggestion in future KI distribution and emergency communications planning.[15]

Within the Mid-Term Review, OPG notes that:

> KI public awareness campaigns are held regularly (three times per year). The campaigns are focused on the 10 km DPZ but they are also extended into the 50 km IPZ, through various media (e.g., news releases, print advertisements, social media, and digital display boards). Durham Region has produced videos to raise general awareness about KI, one of which focused on the availability of KI within the 50 km IPZ.[16]

It is unknown as to whether the public awareness campaigns are actually effective means of communication to inform and encourage the public beyond the DPZ to order KI pills. The Mid-Term Review does not shed light on how frequently OPG reviews engagement metrics, and whether the public awareness campaigns result in an influx of KI pills being ordered outside of the DPZ. CELA **recommends** that the Mid-Term Review of Pickering (and other NPGS) take the time to discuss and clarify the effectiveness of communications with the public surrounding KI pill distribution. Future reports should record how many KI pills are available in the 50 km IPZ along with an estimate of the current population within the zone.

In a similar vein to KI pill awareness, the Record of Decision for renewing the Pickering licence also notes concerns surrounding public awareness surrounding nuclear preparedness:

> The Commission notes the concerns submitted by CELA, DNA and individuals that nuclear emergency preparedness awareness beyond the DPZ was low. The Commission also notes that survey results submitted as part of interventions substantiated these concerns. The Commission expects OPG to review its means of communication with the public in regard to nuclear emergency preparedness awareness outside the DPZ during the renewed licence period.[17]

The Mid-Term Review does not shed much light on how OPG has reviewed its means of communication with the public outside the DPZ during the renewed licence period. For instance, the report states:

[15] CNSC, Record of Decision at page 109.
[16] OPG, Mid-Term Review at page 21.
[17] CNSC, Record of Decision at page 112.

> The primary focus area for the engagement activities, in addition to the public at large, includes three municipalities proximate to the Pickering NGS site including the host community (City of Pickering) and adjacent communities within 10 km of the project (the City of Toronto, the Town of Ajax, and the Town of Whitby). The 10 km radius is consistent with the Pickering NGS DPZ for nuclear emergency planning purposes, an area where residents are most familiar with nuclear plant operations and regularly receive information station and operational updates.[18]

Based on OPG's focus on the 10 km radius, there is a lack of focus on the communities beyond the 10-km radius who are likely less familiar with nuclear plant operations and the actions they would have to take in the event of an emergency. As the population within the GTA has grown in the last 5 years and will continue to grow over the next 5 years, CELA **submits** that it is crucial for OPG to provide more information about how it has reviewed its means of communication with the public in regard to nuclear emergency preparedness awareness outside the DPZ during the renewed licence period.

Recommendations

1. The Mid-Term Review should not just be a document boasting about a licensee's accomplishments during a licence period. It should address shortfalls and provide clarity on what is being done to alleviate and prevent these shortfalls from happening in the remainder of the licence term.

2. When the CNSC has required mid-term reviews for licensees, the CNSC should provide required topics for discussion, including climate change impacts and mitigation.

3. OPG should provide more information on how effective their communication methods have been to improve awareness about KI pills beyond the DPZ and information on the availability of KI pills in these areas.

4. OPG must provide more information about how it has reviewed its means of communication with the public in regard to nuclear emergency preparedness awareness outside the DPZ during the renewed licence period.

[18] OPG, Mid-Term Review at page 32.

III. GENERAL FINDINGS WITHIN THE REGULATORY OVERSIGHT REPORT

When reviewing the 2022 ROR, CELA has chosen to focus the discussion on Pickering, while also touching upon a number of general comments and concerns stemming from reviewing the entirety of the report. CELA notes that a more in-depth discussion of the Darlington NPGS will be provided within CELA's upcoming submissions for the Darlington New Nuclear Project. And, subject to funding approval by the CNSC, a more in-depth discussion surrounding the extension of Pickering's Units 5-8 will be explored by CELA in future submissions.

A. Operating Performance

The ROR gives data about the numbers of unplanned transients but it does not offer any data on the length of the resulting outages.[19] That piece of information will help contextualize the severity of the unplanned transient. This information would have been particularly useful to understand the performance trends in the Point Lepreau nuclear station, which is underperforming relative to global trends.

The discussion about operating performance of the Pickering units does not mention the length of the periods when Pickering-1 and Pickering-5 were not operating, nor about the challenges posed by environmental conditions.[20]

Recommendation

5. Additional data surrounding the length of reported outages is essential for understanding and contextualizing the severity of unplanned transients. This information should be provided at the upcoming Commission meeting.

B. Fitness for Service

Table 6 on page 19 of the ROR highlights the safety system test performance for 2022.[21] The table indicated that for Pickering, all 13,283 planned safety system tests were completed in 2022. CELA notes that Pickering has seen a positive trend over the last five years in terms of completed safety system test performance, and is relieved to see that the 5 missed tests at Pickering in 2018 has not been a recurring issue since then.[22]

[19] CNSC, ROR 2022 at page 16, Table 5: Number of unplanned transients.
[20] CNSC, ROR 2022 at page 72.
[21] CNSC, ROR 2022 at page 19, Table 6: Safety systems test performance for 2022.
[22] CNSC, ROR 2022 at page 20, Figure 5: Trend of safety system performance for NPPs and industry.

Given the age of Pickering and its systems, having all planned safety system tests performed on an annual basis is crucial, and CELA hopes to see this 100% completion trend continue for the remainder of Pickering's operations.

C. Radiation Protection

On page 26 of the ROR, there are two reports of radiation doses in the 20-50 mSv range.[23] While the dose limit for nuclear workers is indeed 50 mSv per year, there is also a limit of 100 mSv over five years. If the same individual is obtaining over 20 mSv doses in successive years, they can breach the 100 mSv limit in a few years.

It is unclear if the individuals in 2018 who received radiation doses in the 20-50 mSv range (in 2018 there were 6 reported events in this range) are the same individuals who received the doses in 2022. For example, at Bruce A and B, maximum individual effective doses over the last five years have ranged between 16.69 and 22.19 mSv.[24] CELA **requests** further information on how long-term, repeated high radiation doses are being monitored and prevented at each licenced site.

Recommendations

6. Additional information is requested regarding the measures being implemented at each licenced site to monitor and prevent long-term, repeated high radiation doses being administered to individuals.

D. Environmental Performance

The environmental performance of reactors is described in terms of the assessed dose to the public.[25] For example, the dose from the Point Lepreau site is mentioned as 0.0011mSv. This piece of information does not shed any light on what radionuclides were released, how much of each radionuclide (in Bq/year) was released, and how these releases compare with corresponding figures from earlier years. At a minimum, the releases of important radionuclides that are known to be particular dangerous to human health—for example, Tritium and tritiated water—should be released on an annual basis. Without such detailed information, it is not possible to gauge how well a reactor is performing.

CELA **submits** that the publication of annual radionuclide loadings to the environment from nuclear facilities on the CNSC Open Government Portal does not provide clarity for the data

[23] CNSC ROR 2022 at page 26, Figure 10:Trend of distribution of annual effective doses received by all monitored persons at Canadian NPPs.
[24] CNSC, ROR 2022 at page 25, Figure 9: Trend of maximum individual effective doses.
[25] CNSC, ROR 2022 at page 29, Table 9: Trend of estimated dose to the public form Canadian nuclear power generating sites (mSv).

presented within the ROR, as the radionuclide loadings on the portal are presented in NPRI format, whereas the data in Table 9 is provided in mSv.

Recommendation

7. The ROR should provide detailed information (in Bq) on an annual basis regarding the releases of important radionuclides that are known to be particular dangerous to human health—for example, Tritium and tritiated water. Without such detailed information, it is not possible to gauge how well a reactor is performing.

E. Climate Change

The impacts of climate change on nuclear reactors includes various climate-related hazards such as heat waves, floods, droughts, storms, lightning events and wildfires which can pose challenges to nuclear safety, necessitating enhanced safety assessments and adaptive management strategies.[26] Rising temperatures can threaten nuclear reactors, as they often rely on large external sources of water for cooling. In the case of Canadian reactors, this water is usually from the Great Lakes, or the Bay of Fundy in the case of Point Lepreau. Higher water temperatures can reduce the thermal efficiency of nuclear reactors and even lead to temporary shutdowns when the required cooling cannot be ensured. Heightened water stress, either from reduced availability or competing uses, is a concern since nuclear reactors require significant amounts of water for cooling purposes.[27] The IAEA highlights the need to reassess nuclear installations' safety against external hazards due to climate change.[28] These hazards include meteorological events like lightning and tropical cyclones, hydrological events such as floods and storm surges, and fire-related hazards. Continuous site hazard assessment and timely management of plant reactions to extreme events are essential for ensuring nuclear safety. Besides these hazards, increasing temperatures leads to growth in algae species, which can water intake create blockages and shutdowns.[29]

A recent study by Wang et al utilizes a Regional Climate Model to analyze the climate-driven alterations in energy sectors in Ontario, Canada, with a resolution of 25 km × 25 km.[30] The study provides reliable projections of climatic variable changes, assessing their impacts on cooling degree days, heating degree days, and energy availability. Rising temperatures, influenced by

[26] Linnerud, K., Mideksa, T. K., & Eskeland, G. S. (2011). The impact of climate change on nuclear power supply. *The Energy Journal*, 32(1).
[27] Linnerud, K. et al.
[28] International Atomic Energy Agency (2023, 12 October). *Impact of Climate Related Hazards on the Safety of Nuclear Installations: Experts Discuss*, online: https://www.iaea.org/newscenter/news/impact-of-climate-related-hazards-on-the-safety-of-nuclear-installations-experts-discuss
[29] Lin, H., Zhang, S., Cao, R., Yu, S., Bai, W., Zhang, R., ... & Zhang, X. (2023). A review on the risk, prevention and control of cooling water intake blockage in coastal nuclear power plants. *Nuclear Engineering and Technology* (in-print)
[30] Wang, S., Zhu, J., Huang, G., Baetz, B., Cheng, G., Zeng, X., & Wang, X. (2020). Assessment of climate change impacts on energy capacity planning in Ontario, Canada using high-resolution regional climate model. *Journal of Cleaner Production*, 274, 123026.

Ontario's high latitudes, significantly affect the region's residential and commercial energy demand, showcasing a temperature-sensitive energy demand pattern based on seasonal variability. According to their study, anticipated temperature increases will result in nuclear power production reductions of 1.5% and 2.5% under RCP4.5 and RCP8.5 scenarios respectively for the 2050s, and 2.5% and 4.0% reductions respectively for the 2080s. Since this is a relatively coarse regional model-based analysis and not specific to particular reactors, to gather an accurate extent of climate change impacts on nuclear reactors in Canada requires more localized studies and analyses. Further, this study only looks at the loss of thermal efficiency. A second source of reduced output is having to shut down nuclear reactors for shorter or longer periods of time because of extreme weather conditions, including high water temperatures.[31] These concerns mean that it is vital to carry out comprehensive climate vulnerability assessments of nuclear reactors across Canada.

For the Pickering reactor section of the ROR, there is mention of a requirement for recent data for floods in the context climate change:

> CNSC staff has finished reviewing the PSA Hazard Screening Analysis and identified an issue regarding the external flood PSA's use of old precipitation data, when more recent data should be used. In response, CNSC staff communicated their expectations to OPG that they incorporate more up-to-date climate change precipitation data to account for the potential impact of climate change.[32]

While this covers only one type of impact, there are others such as heat waves, droughts etc., which require further accounting for to fully assess the risk from climate risks. Moreover, there is no further mention how such flood risks are planned to be ascertained. Fires are also mentioned in the Physical Design section.[33] However, there is no mention or consideration of climate impacts such as wildfires, lightning events or extreme heat events.

Overall, there is a significant gap in addressing the increased risks to nuclear plant operations and safety as a result of the impacts of climate change and the development of adaptive strategies.

Recommendations

8. With latest regional and local scale models assess present and future risk of relevant climate impacts on all nuclear reactor facilities across Canada. This includes risks from algal growth.

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[31] *See*: Ahmad, Ali. "Increase in Frequency of Nuclear Power Outages Due to Changing Climate." *Nature Energy* 6, no. 7 (July 2021): 755–62. https://doi.org/10.1038/s41560-021-00849-y; *see also* Ahmad, Ali, Andrei Covatariu, and M. V. Ramana. "A Stormy Future? Financial Impact of Climate Change-Related Disruptions on Nuclear Power Plant Owners." *Utilities Policy* 81 (April 1, 2023): 101484. https://doi.org/10.1016/j.jup.2022.101484.

[32] CNSC, ROR 2022 at page 74.

[33] CNSC, ROR 2022 at page 74.

9. Conduct a climate vulnerability assessment which ascertains adaptive capacity of present reactors and with participation from various partners, develop implementable adaptive strategies.

10. Have dedicated sections in Regulatory Oversight reports for specific reactors and respective climate change risks and related adaptation strategies along with an overview of these power plants.

F. Human Performance Management

When reviewing the human performance management commentary on Pickering, the ROR notes there were 8 non-compliant findings within the SCA, with three being of negligible safety significance, and five were of low safety significance.[34] With two of the low safety significance findings being associated with discrepancies in non-certified training programs related to Fire Protection training program, and the three other low safety significance findings being related to discrepancies in documentation and training material for OPG's Emergency Response Organization Training Program, there are concerns that arise surrounding complacency and staff competency at Pickering.

Pickering is an aging facility with its operational lifespan coming to an end, and with its limited lifespan, it calls into question whether Pickering is maintaining staff competency and adequate staffing numbers. With job security coming into question as Pickering approaches various phases of winding down, there is the risk of employees not taking safety training and safety practices seriously. CELA **requests** additional information on the employment trends at Pickering, and whether these instances of human performance management non-compliance are being an increasingly notable trend with every passing year.

Recommendation

11. Additional information is requested related to the employment trends at Pickering, and whether these instances of human performance management non-compliance are being an increasingly notable trend with every passing year.

[34] CNSC, ROR 2022 at page 72.

G. Physical Design

According to the ROR's discussion of CNSC staff's review of the 2021 Annual Fuel Performance Report, "the residency time for defected bundles in the core is high relative to the industry average, due to the design limitations."[35] This is doubly concerning because of the age of the Pickering reactors and because the discovery of "elevated hydrogen equivalent concentration near the rolled joints" at two of the units at Bruce.[36] The latter problem might well afflict the Pickering reactors too.

CNSC staff have recommended that OPG take steps to detect and remove these defects from the core in a timely manner.[37] CELA **requests** that OPG provide an update on what steps are being done, as well as a timeline for removing these defects, be provided at the upcoming Commission meeting.

Recommendation

12. OPG should provide an update on what steps are being done, as well as a timeline for removing these defects from core bundles, be provided at the upcoming Commission meeting.

H. Emergency Management and Fire Protection

Following an inspection of the Fire Protection Program at Pickering, CNSC staff found three non-compliances of low safety significance: "the non-compliances pertained to compressed gas storage, minimization of impairment durations of fire protection systems, and some inconsistencies between pre-fire plans and the conditions of the plant."[38] According to the ROR, the CNSC staff reviewed and agreed with the proposed corrective plant by OPG. However, there is no discussion or explanation of this corrective plan. To improve transparency with the public, CELA **requests** that the corrective plan be provided, or at a minimum, provide a high level summary of the plan so that the public can understand the risks surrounding these non-compliance instances. This request becomes even more relevant when the recent fire event at Pickering is taken into consideration.

The ROR briefly discusses a fire on February 22, 2022 that was traced to poor vendor quality of the motor rewind complete in 2014.[39] However, the report does not specify whether the reactor

[35] CNSC, ROR 2022 at page 75.
[36] CNSC, ROR 2022 at page 75.
[37] CNSC, ROR 2022 at page 75.
[38] CNSC, ROR 2022 at page 78.
[39] CNSC, ROR 2022 at page 78.

was shut down due to the fire, as one would expect from a cautious operator. Indeed, one would expect that the reactor should be shut until the underlying reason for the fire was fixed. CELA therefore **requests** additional information regarding this fire event be provided at the upcoming Commission meeting.

Furthermore, the ROR notes that CNSC staff found that offsite emergency centers where not tested over a minimum of a five-year period, deeming OPG as non-compliant with a requirement in REGDOC-2.10.1, *Nuclear Emergency Preparedness and Response.*[40] The findings of non-compliance were raised in the Emergency Preparedness and Fire Protection SCA, and the Human Performance SCA. These instances of non-compliance once again call into question whether staff at Pickering are taking their roles in nuclear safety seriously as the lifespan of Pickering (and their job positions) shortens over time. The ROR notes that "OPG was requested to provide updates on planned drills for two offsite centers and to implement corrective action plans to align with REGDOC-2.10.1 and correct inconsistencies within its internal procedures."[41] CELA **requests** that this update be discussed at the upcoming Commission meeting.

Recommendations

13. The corrective plan to address inconsistencies between the pre-fire plans and the conditions of the Pickering plant should be provided, or at a minimum, provide a high level summary of the plan so that the public can understand the risks surrounding these non-compliance instances.

14. Additional information surrounding the pump fire on February 22, 2022 should be provided at the upcoming Commission meeting.

15. OPG should provide an update on planned drills for two offsite centers and to implement corrective action plans to align with REGDOC-2.10.1 and correct inconsistencies within its internal procedures at the upcoming Commission meeting.

I. Security

The ROR mentions that that there were four non-compliant findings regarding security of the Pickering plant as well as five non-compliant findings in the cyber-security domain.[42] While the report dismisses these findings as having "low safety significance", in fact, vulnerabilities can open up the reactors to cyberattacks, which in turn can create a further pathway for accidents. Even

[40] CNSC, ROR 2022 at page 78.
[41] CNSC, ROR 2022 at page 78.
[42] CNSC, ROR 2022 at page 79.

if attacks themselves might not cause, say, the meltdown of the core, by potentially disabling safety systems or causing other problems, such as loss of electric power at the plant, these attacks could set the stage for a meltdown if it is combined with some other challenge to the plant's safety systems, for example a severe storm or an earthquake.[43]

The issue of compliance with the Security SCA at Pickering is not a novel one; during the licence renewal hearing for Pickering in 2018: "… the Commission notes the decrease in OPG's rating in this [Security] SCA during the previous licence period and encourages that OPG implement measures to again achieve fully satisfactory rating in this SCA."[44]

It is noted that based on a declining trend in compliance performance, CNSC staff are conducting enhanced regulatory oversight of the security SCA, and that staff findings will be detailed in a supplemental CMD as the details are confidential.[45] Due to confidentiality surrounding security breaches, little detail on these issues is available to the public. The fact that security oversight is non-transparent to the public; and that it engages international and domestic security risks elevates this issue to one of a high import for the commission. CELA **recommends** that the Commission members stringently and thoroughly delve into the causes and significance of this area of the supplemental report; and demand challenging and detailed, specific steps to show improvement and report back to the Commissioners themselves.

Recommendation

16. The Commission members must stringently and thoroughly delve into the causes and significance of this area of the supplemental report surrounding security non-compliance at Pickering; and demand challenging and detailed, specific steps to show improvement and report back to the Commissioners themselves.

J. KI Pill Distribution

The 2022 ROR briefly mentions potassium iodide ("KI" pill) distribution in the Appendices, mainly within the mid-term update for Pickering in Appendix H. For instance, the ROR states:

> In October 2022, EMO was given the responsibility to complete the Phase II objectives of the working group through the revision of the PNERP anticipated for 2023. EMO and the Ontario MOH drafted a discussion paper concerning the Phase II objectives, including the development of a strategy for emergency distribution of KI pills and examination of the

[43] M. V. Ramana and Lauren J. Borja. "The Computer Infection of Kudankulam and Its Implications." *The India Forum*, January 7, 2020. https://www.theindiaforum.in/article/computer-infection-kudankulam-and-its-implications.
[44] CNSC, Record of Decision at page 120.
[45] CNSC, ROR 2022 at page 80.

feasibility of KI pill pre-distribution to all schools in the IPZ. This discussion paper was shared with the KIPWG in December 2022 for comments. The discussion paper is intended to inform the upcoming revision to the PNERP. The revised PNERP is expected to undergo a public review period, which will also provide an opportunity for the KIPWG to discuss the revision and provide comments to EMO. Additional information on the KIPWG is available on the CNSC.[46]

According to the CNSC website, the 2023 update to the PNERP will undergo a 60-day public review, led by EMO, however, it is unclear when this review will take place.[47] CELA reiterates that adequate distribution of KI pills is an important element of emergency preparedness for all NPGS, and **submits** that a discussion of KI distribution requirements and any updates based on meetings of the CNSC-led KI Pill Working Group would fall well within the scope of this ROR.

CELA remains an active member of the advisory group to the KI Pill Working Group and **submits** that distribution of KI pills is currently inadequate. While operators and regulators have spent years working on understanding the current framework for storing and distributing potassium iodide, the critical work has not begun as committed to in the last Pickering hearing to further distribute KI pills to residents living beyond 10 km. This measure is especially critical for vulnerable populations, such as children.

CELA continues to **recommend** expanding the delivery of KI pills to a pre-distribution area of 50 km, rather than the current 10 km pre-distribution area. CELA further **recommends** that KI pill distribution requirements and updates from the KI Pill Working Group be discussed at the upcoming Commission Meeting and integrated into this ROR.

Recommendations

17. The CNSC should consider expanding the delivery of KI pills to a pre-distribution area of 50 km, rather than the current 10 km pre-distribution area.

18. KI distribution requirements and updates from the KI Pill Working Group be discussed at the upcoming Commission Meeting and integrated into this ROR.

[46] CNSC, ROR 2022 at pages 166-167.
[47] CNSC, Potassium iodide (KI) Pill Working Group, online: https://nuclearsafety.gc.ca/eng/resources/emergency-management-and-safety/potassium-iodide-pill-working-group.cfm#3

IV. CONCLUSION

We respectfully provide these comments to assist the Commission in its review of the *Pickering Mid-Term Review Report* and the *Regulatory Oversight Report for Canadian Nuclear Power Generating Sites: 2022*.

Sincerely,

CANADIAN ENVIRONMENTAL LAW ASSOCIATION

Sara Libman

Sara Libman, Legal Counsel

Appendix 1
Summary of Recommendations

1. The Mid-Term Review should not just be a document boasting about a licensee's accomplishments during a licence period. It should address shortfalls and provide clarity on what is being done to alleviate and prevent these shortfalls from happening in the remainder of the licence term.

2. When the CNSC has required mid-term reviews for licensees, the CNSC should provide required topics for discussion, including climate change impacts and mitigation.

3. OPG should provide more information on how effective their communication methods have been to improve awareness about KI pills beyond the DPZ and information on the availability of KI pills in these areas.

4. OPG must provide more information about how it has reviewed its means of communication with the public in regard to nuclear emergency preparedness awareness outside the DPZ during the renewed licence period.

5. Additional data surrounding the length of reported outages is essential for understanding and contextualizing the severity of unplanned transients. This information should be provided at the upcoming Commission meeting.

6. Additional information is requested regarding the measures being implemented at each licenced site to monitor and prevent long-term, repeated high radiation doses being administered to individuals.

7. The ROR should provide detailed information (in Bq) on an annual basis regarding the releases of important radionuclides that are known to be particular dangerous to human health—for example, Tritium and tritiated water. Without such detailed information, it is not possible to gauge how well a reactor is performing.

8. With latest regional and local scale models assess present and future risk of relevant climate impacts on all nuclear reactor facilities across Canada. This includes risks from algal growth.

9. Conduct a climate vulnerability assessment which ascertains adaptive capacity of present reactors and with participation from various partners, develop implementable adaptive strategies.

10. Have dedicated sections in Regulatory Oversight reports for specific reactors and respective climate change risks and related adaptation strategies along with an overview of these power plants.

11. Additional information is requested related to the employment trends at Pickering, and whether these instances of human performance management non-compliance are being an increasingly notable trend with every passing year.

12. OPG should provide an update on what steps are being done, as well as a timeline for removing these defects from core bundles, be provided at the upcoming Commission meeting.

13. The corrective plan to address inconsistencies between the pre-fire plans and the conditions of the Pickering plant should be provided, or at a minimum, provide a high level summary of the plan so that the public can understand the risks surrounding these non-compliance instances.

14. Additional information surrounding the pump fire on February 22, 2022 should be provided at the upcoming Commission meeting.

15. OPG should provide an update on planned drills for two offsite centers and to implement corrective action plans to align with REGDOC-2.10.1 and correct inconsistencies within its internal procedures at the upcoming Commission meeting.

16. The Commission members must stringently and thoroughly delve into the causes and significance of this area of the supplemental report surrounding security non-compliance at Pickering; and demand challenging and detailed, specific steps to show improvement and report back to the Commissioners themselves.

17. The CNSC should consider expanding the delivery of KI pills to a pre-distribution area of 50 km, rather than the current 10 km pre-distribution area.

18. KI distribution requirements and updates from the KI Pill Working Group be discussed at the upcoming Commission Meeting and integrated into this ROR.

Pickering Nuclear Generating Station REGDOC-3.1.1 Reporting Q1-2025

Report Number	*Event Title*
P-2024-14152	Degradation: Impairment of Unit 5 Boiler Emergency Cooling System, No Operability Impact
P-2025-00374	Matter of Regulatory Interest: Fish Entrainment Study Late Start Date
P-2025-01460	Minor Fire Incident at Unit 4, No Operability Impact
P-2025-01597	Exceedance of Monthly Gross Beta/Gamma Environmental Action Level for Sewage Emissions, No Safety or Environmental Impact
P-2025-01602	Counterfeit, Fraudulent or Suspect Items: Lifting Eye Nuts Received, Unexpected Identification Markings and Appearance, Safely Resolved
P-2025-03822	Unplanned Change in Reactor Power: Unit 7 Reactor Setback following Turbine Trip

Pickering Nuclear Generating Station REGDOC-3.1.1 Reporting
Q2-2025

Report Number	*Event Title*

None

Pickering Nuclear Generating Station REGDOC-3.1.1 Reporting
Q3-2025

Report Number	*Event Title*
P-2025-09164	Environmental Release of Demineralized Water, No Environmental Impact
P-2025-09642	Reactor Trip: Unit 6 Tripped Via Shutdown System 1 and Shutdown System 2
P-2025-10527	Flame Observed Inside the Unit 6 Emergency Low Pressure Service Water Pump Motor Casing, Safely Resolved
P-2025-12466	Safeguards: Temporary Loss of Power to Safeguards Equipment

Pickering Nuclear Generating Station REGDOC-3.1.1 Reporting
Q4-2025

Report Number	*Event Title*
P-2025-13276	Radiation Hazard Boundaries and Signs Not Positioned in Accordance with the Regulations
P-2025-13984	System Degradation: System Does Not Meet Design Intent of Unit 7 Auxiliary Boiler Feedwater System, No Operability Impact
P-2025-14160	Hours of Work Exceedances - Workers Had Less Than 8 Hours Free From Work Between Shifts, No Fitness For Duty Impact
P-2025-14509	Emergency Response Team Mobilized To Assist Crane Worker, Safely Resolved
P-2025-15021	Environmental Release of Fire Retardant Fluid, No Environmental Impact Observed
P-2025-15759	Environmental Release of Fire Retardant Foam, No Environmental Impact Observed
P-2025-16404	Unplanned Change in Reactor Power: Unit 7 Power Unexpected Decrease to 86%
P-2025-16784	Radiation Hazard Sign Not Posted in Accordance with the Regulations

ONTARIO POWER GENERATION

Pickering Nuclear Generating Station REGDOC-3.1.1 Reporting
Q1-2026

Report Number	*Event Title*
P-2025-08487	Radiological Incident: Unit 8 Loss of D2O Due To West Fuelling Machine Separator Failure, No Safety or Environmental Impact
P-2026-01058	Pickering Nuclear 24 Hour Daily Average Effluent Temperature Exceeded the Environmental Compliance Approval Limit, No Environmental Impact
P-2026-01122	Unplanned Change in Reactor Power: Unit 8 Reactor Setback to 90% Due to Loss of Reheat Return Flow, no Operability Impact
P-2026-01275	Unplanned Change in Reactor Power: Unit 6 Reactor Power Unexpectedly Raised by DCCX, no Operability Impact
P-2026-01539	Emergency Response Team Mobilized To Retrieve Crane Worker, no Injury to Worker and Safely Resolved
P-2026-02403	Discovery Issue Resolution Process Initiated for Legacy Unit 2 & 3 Bulkhead Structural Analysis, no Operability Impact
P-2026-02653	Licence Non-Compliance: 2025 CSA N287.7 PRD/VB Missed Inspection Scope, no Operability Impact
P-2026-03070	Hours of Work Exceedances, no Fitness for Duty Impact
P-2026-03515	Reactor Trip: Unit 8 Unexpected SDS2 Trip Following Planned SDS1 Trip, no Operability Impact

Pickering Nuclear Generating Station REGDOC-3.1.1 Reporting
Q2-2026

Report Number	*Event Title*
P-2026-04297	Hours of Work Exceedances, no Fitness for Duty Impact
P-2026-04942	Safeguards: Temporary Loss of Power to Safeguards Equipment, no Nuclear or Public Safety Impact
P-2026-06054	Radiation Hazard Sign Not Posted in Accordance with the Regulations, no Exposure Levels or Dose Limits Exceeded
P-2026-06400	Unplanned Change in Reactor Power: Unit 7 Reactor Power above 100% on Digital Control Computer (DCC) Y, no Operability Impact
P-2026-07875	Radiation Hazard Sign Not Posted in Accordance with the Regulations, no Exposure Levels or Dose Limits Exceeded

Canadian Environmental Law Association
EQUITY. JUSTICE. HEALTH.

October 2, 2023

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Re: Regulatory Oversight Report CNL 2022 Reference # 2023-M-30

The Canadian Environmental Law Association writes to provide brief comments as a written only request for Intervention in the above noted matter, being the 2022 Regulatory Oversight Report of Chalk River Nuclear Laboratories.

CELA has reviewed the CMDs for the November 1 meeting dealing with this ROR. After briefly noting the results of inspections and reportable events reviewed in the Appendices, CELA provides comments below in respect of the CNSC Staff CMD and its conclusions, as well as areas CELA recommends for focus by the Commission.

Table D-1 lists of CNSC led inspections at CRL. CELA draws the Commissions attention to the fact that every single inspection revealed non-compliant areas except baseline radiation at Whiteshell and the inspection at Port Granby. The security inspection results are not available.

Table E-2 lists the reportable events at CRL in 2022. There were 39 reportable events this past year at the CRL set of facilities. These were in the Emergency Management and Fire Protection areas of concern. One significant event included the finding of Radium 226 stored in a CRL building. Another involved loss of power. Some involved impacts or loss of fire detection, monitoring, and alarms. One event was a fire on top of an operating furnace at a CRL Building (a digital thermocouple caught fire). Another fire occurred at a barrel heater. Yet another released contaminated groundwater at 15 Bq/L to the ground, when a re-circulation pump and heat tracing simultaneously failed. In a different event, severe rainfall resulted in overflowing and discharge of water from containers holding radiologically contaminated waste. Another event reported access by an employee without proper authorization and oversight. Another event occurred with loss of power that resulted in the NRU reactor Exhaust Air Filtration System Fans to shut down unexpectedly. Another event disclosed unposted radiation fields in an accessible area outside of a storage building. Two events occurred during IAEA safeguards and non-proliferation inspections. One rendered a measurement tool inoperable. The other disclosed CNL staff misunderstanding the permitted amount of enriched uranium that could be imported. In event 34, the seal on the top of a flask containing a pressure tube was damaged. Another was

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the result of exceeding the amount of permissible fissionable materials in an area. Three events with no details in the appendix involved unauthorized site access; details reported to be classified confidential. Similarly, a significant reportable event involving security equipment failure is also classified confidential with no details provided.

At Port Hope in 2022, Table E-3 lists events including security, fire protection, worker injury, equipment performance and several other matters.

At Port Granby in 2022, Table E-5 lists events including damage to a leachate pipe, security, and package classification among others.

No reportable events occurred or were reported for Douglas Point, Gentilly 1, or the Nuclear Power Demonstration Waste Facilities in 2022.

COMMENT 1: CELA is of the view that there is an inappropriate weighting of the Safety Control Areas such that CNSC staff ought to be viewing them from the perspective of those that indicate warning signs and red flags in terms of the adequacy of CNL management and conduct of the necessary controls on all of the facilities. For example, several of the significant event reports – almost all of them in fact – occurred in areas dealing with fire and electrical as well as emergency readiness. These are fundamental "basics" for nuclear facilities and their inattention should result in much higher weighting to such red flags and therefore downgrading of the rating for the SCAs. The current rating system appears to wait for actual releases before reporting the SCAs as unacceptable. By the time there is such a problem it is too late and public / environment has been impacted.

COMMENT 2: CNSC staff should refrain from use of jargonistic acronyms in the report even after an initial definition. For example, the repeated use of the acronym NNCs for non-compliant areas buries the import of the text such as at p. 17 of the CNSC CMD, in discussing non-compliant areas of management systems.

COMMENT 3: Many of the issues discussed in the CNSC CMD are of high significance and have to do with competence of CNL to track and manage significant types of waste. These should not be down-played especially by use of phrases such as no "imminent" safety significance. The matters reviewed in the CMD are critical issues in terms of confidence in the ability of the operator to handle this waste now and into the future and given the very lengthy time frames implicated by these types of waste these deficiencies are very serious considerations.

COMMENT 4: A similar concern arises in respect of the area of non-compliance regarding change control at Port Hope. Competent change management is critical for long term knowledge management; assurance of safety to the public; future return of staff to do other work or to conduct follow up and monitoring tasks. CNSC staff appear to be giving CNL lots of leeway on this issue, even though it was found to be unacceptable to CNSC staff.

COMMENT 5: In the area of human performance management, it is concerning that CNL was not reporting to CNSC in areas of non-compliance in safety-significant positions for worker fatigue and related matters. Again, both the worker fatigue itself, and the non-reporting, are red

flags. These are the types of issues that have been found to be causal precedents to major nuclear and industrial accidents. These types of occurrences are reflective of failure of both oversight by management and of a diminishing safety culture; they also demonstrate a lack of respect for importance of reporting to the regulator.

COMMENT 6: Regarding the Chalk River power outage and initiation of CRL emergency operations centre, CELA recommends that Commission members query as to whether the public was notified of this event and what the communications consisted of. Transparent public notification at the time of serious events is essential to public trust.

COMMENT 7: At p. 23 of the CNSC staff CMD, section 4.4.1 Chalk River labs is a description of a non-compliance regarding the overall inventory of U-235 (a fissionable material) in building 429. This issue is of very high importance vis a vis safety and security; it was reported that CNL was not tracking overall inventory of U-235 in the building. Again, this constitutes another very serious red flag in terms of the management competence and performance of safeguarding the site and its materials. This is especially concerning in a context of the needed long-term management of decommissioned facilities and the necessity for the regulator and the public to have trust in the capacity of the operator to handle high level waste and fissionable materials across a range of activities.

COMMENT 8: Section 4.6.1 of the CNSC Staff CMD describes fitness for service of the Chalk River labs. This section of the staff CMD contains additional red flags in terms of operator competence. There were expired HEPA filters; calibration inadequacies; a lack of backup power in a room with a radiation monitoring system; and 7 non-compliant areas pertaining to aging management. These are critically important for a facility engaged in decommissioning nuclear systems; managing the federal government waste liabilities and legacies; and as an entity seeking long term waste approvals.

COMMENT 9: Reference 28 is CMD 23-M25 Event Initial Report, Safety stand-down at Canadian Nuclear Laboratories' Whiteshell Site following the discovery of non-compliances in the fire protection program. Page 34 of the CNSC CMD further describes the self-assessment and report by CNL that disclosed serious deficiencies in fire and emergency training, and in equipment inspection, lack of testing and maintenance and use of incomplete or expired personal protective equipment. The result was that CNL reported to the CNSC and CNL itself placed Whiteshell Labs into a safe shutdown state where only essential compliance and maintenance work could be conducted. This affected the years 2020, 2021, 2022. As noted, it was based on a self-inspection.

The full event report, reference 28, presented to the commission on June 29, 2023 also identified that there was unavailability of firewater pressure and flow; unavailability of fire hydrants; non-maintenance of fixed suppression systems (sprinklers); and non-testing of same; and non-testing of emergency lighting in buildings. (p. 34 of the staff CMD)

CELA notes that not all compliance activities related to this stand-down are expected before the end of March 2024. CNSC staff are reviewing CNL's submissions on this matter, including a multi-phase re-start plan. Further updates will be provided by CNSC staff at the November 1,

2023 meeting; as well as preparation of a lessons learned report to focus on the CNSC regulatory oversight of Whiteshell Labs from 2019 to 2022.
Table E-3 setting out other reportable Events at Whiteshell Labs in 2022, included the receipt by a worker of an electrical shock, as well as events with emergency management and fire protection significance.

CELA RECOMMENDS that the CNSC Commission members delve into better understanding the CNSC regulatory oversight role, or lack thereof, whereby this situation at Whiteshell managed to get to the state that it did on the very matters that are most critically important to the public and the environment. It bespeaks a lack of on-site inspections, or lack of inspector competence, or lack of inspection rigour, if they did occur. It also undermines the credibility of the current oversight system.

CELA also notes that the statement that the overall conclusion by CNSC staff that CNL continues to implement and maintain effective emergency management fire protection programs at CNL sites in accordance with regulatory requirements, apart from WL, is inappropriate, especially when combined with the events and areas of non-compliance at CRNLs which also engaged fire and emergency response issues.

COMMENT 10: At p. 40 of the CMD, CNSC staff concluded that CRL security was assessed below expectations in 2022. They indicated that there was no immediate risk to security and CNSC staff will continue to conduct oversight in this area. Little detail is available to the public. The fact that security oversight is non-transparent to the public; and that it engages international and domestic security risks elevates this issue to one of a high import for the commission. CELA recommends that the Commission members stringently and thoroughly delve into the causes and significance of this area of the report; and demand challenging and detailed, specific steps to show improvement and report back to the Commissioners themselves.

COMMENT 11: In the overall ratings, Security continued to be rated Below Expectations for Chalk River Labs; and Emergency Management and Fire protection Below Expectations for the Whiteshell Labs. Note that the rating for this safety and control area at Whiteshell was rated by CNSC staff as satisfactory for the years 2018 through 2021, but as discussed in the staff report, these must not have been satisfactory since the CNL self-assessment discovered that these serious issues that led to the stand-down had been in existence in 2020, 2021, and 2022.

COMMENT 12: Appendix M provides a Summary Table of the Status of Issues Concerns and Requests from Indigenous Intervenors in the 2021 CNL ROR. CELA commends inclusion of this table. CELA also appreciates the direct outreach by the CNSC staff regarding the thematic issues raised by CELA over prior years. CELA recommends that further specificity of the issues raised by Intervenors, and how they have been answered or responded to should be provided in RORs annually in a disposition chart or table of action taken and underway.

We trust these comments on the CMD 2022 for Chalk River Laboratories managed facilities are of assistance to the Commissioners, staff and CNL.

Yours very truly,
CANADIAN ENVIRONMENTAL LAW ASSOCIATION

Theresa McClenaghan
Executive Director & Counsel

United States Government Accountability Office

Report to Congressional Requesters

April 2024

NUCLEAR POWER PLANTS

NRC Should Take Actions to Fully Consider the Potential Effects of Climate Change

GAO-24-106326

GAO Highlights

Highlights of GAO-24-106326, a report to congressional requesters

April 2024

NUCLEAR POWER PLANTS

NRC Should Take Actions to Fully Consider the Potential Effects of Climate Change

Why GAO Did This Study

NRC licenses and regulates the use of nuclear energy to provide reasonable assurance of adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. Like all energy infrastructure, nuclear power plants can be affected by disruptions from natural hazards, some of which are likely to be exacerbated by climate change. Most commercial nuclear plants in the United States were built in the 1960s and 1970s, and weather patterns and climate-related risks to these plants have changed since their construction.

GAO was asked to review the climate resilience of energy infrastructure. This report examines (1) how climate change is expected to affect nuclear power plants and (2) NRC actions to address risks to nuclear power plants from climate change. GAO analyzed available federal data and reviewed regulations, agency documents, and relevant literature. GAO interviewed officials from federal agencies, including NRC, the Department of Energy, and the National Oceanic and Atmospheric Administration, and knowledgeable stakeholders from industry, academia, and nongovernmental organizations. GAO also conducted site visits to two plants.

What GAO Recommends

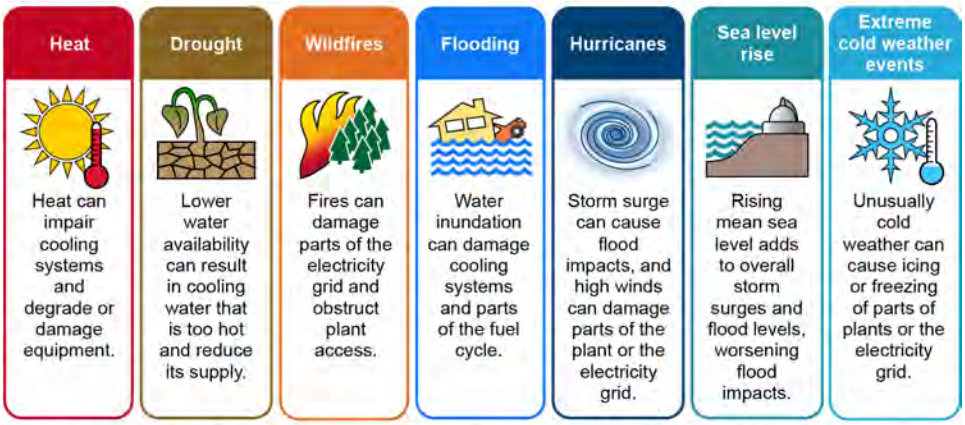
GAO is making three recommendations, including that NRC assess whether its existing processes adequately address climate risks and develop and implement a plan to address any gaps identified. NRC said the recommendations are consistent with actions that are either underway or under development.

View GAO-24-106326. For more information, contact Frank Rusco at (202) 512-3841 or ruscof@gao.gov.

What GAO Found

Climate change is expected to exacerbate natural hazards—including heat, drought, wildfires, flooding, hurricanes, and sea level rise. In addition, climate change may affect extreme cold weather events. Risks to nuclear power plants from these hazards include loss of offsite power, damage to systems and equipment, and diminished cooling capacity, potentially resulting in reduced operations or plant shutdowns.

Examples of Natural Hazards that May Pose Risks to Nuclear Power Plants



Sources: Nuclear Regulatory Commission documents; summary of literature; GAO (icons). | GAO-24-106326

The Nuclear Regulatory Commission (NRC) addresses risks to the safety of nuclear power plants, including risks from natural hazards, in its licensing and oversight processes. Following the tsunami that led to the 2011 accident at Japan's Fukushima Dai-ichi nuclear power plant, NRC took additional actions to address risks from natural hazards. These include requiring safety margins in reactor designs, measures to prevent radioactive releases should a natural hazard event exceed what a plant was designed to withstand, and maintenance of backup equipment related to safety functions.

However, NRC's actions to address risks from natural hazards do not fully consider potential climate change effects. For example, NRC primarily uses historical data in its licensing and oversight processes rather than climate projections data. NRC officials GAO interviewed said they believe their current processes provide an adequate margin of safety to address climate risks. However, NRC has not conducted an assessment to demonstrate that this is the case. Assessing its processes to determine whether they adequately address the potential for increased risks from climate change would help ensure NRC fully considers risks to existing and proposed plants. Specifically, identifying any gaps in its processes and developing a plan to address them, including by using climate projections data, would help ensure that NRC adopts a more comprehensive approach for assessing risks and is better able to fulfill its mission to protect public health and safety.

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Abbreviations

FLEX	Diverse and Flexible Coping Strategies
NCA	National Climate Assessment
NOAA	National Oceanic and Atmospheric Administration
NRC	Nuclear Regulatory Commission
POANHI	Process for the Ongoing Assessment of Natural Hazard Information
SAFER	Strategic Alliance for FLEX Emergency Response

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U.S. GOVERNMENT ACCOUNTABILITY OFFICE
441 G St. N.W.
Washington, DC 20548

April 2, 2024

The Honorable Joe Manchin III
Chairman
Committee on Energy and Natural Resources
United States Senate

The Honorable Tom Carper
Chairman
Committee on Environment and Public Works
United States Senate

Since 1990, nuclear energy has accounted for about 20 percent of the electricity generated in the United States. In 2022, nuclear energy provided nearly half of our nation's carbon-free electricity, making it the largest domestic source of carbon-free energy. Nuclear power plants emit no carbon dioxide during operations and, unlike many sources of renewable energy, typically operate around the clock, producing on average above 90 percent of their generating capacity.

However, nuclear power plants can be affected by natural hazards—including heat, drought, wildfires, flooding, hurricanes, sea level rise, and extreme cold weather events—some of which are expected to be exacerbated by climate change, with effects varying by region. Most commercial nuclear power plants in the United States were licensed and built in the 1960s and 1970s, and the risks to plants' safety and operations from natural hazards have changed since their construction.

The Nuclear Regulatory Commission (NRC) is responsible for regulating the civilian use of radioactive materials to promote the nation's common defense and security, provide reasonable assurance of adequate protection of public health and safety, and protect the environment. As electricity demand in the United States is expected to continue to grow over the coming decades, Congress and others are turning to nuclear power as one means of meeting the increased demand while reducing carbon emissions. For example, in recent years, Congress has provided incentives for the continued operation of existing nuclear power plants

and for the construction of new plants, which, if licensed, could operate into the next century.[1]

You asked us to review the climate resilience of energy infrastructure. This report focuses on nuclear power plants' resilience to climate change and examines (1) how climate change is expected to affect nuclear power plants and (2) what actions NRC has taken to address the risks to nuclear power plants from climate change.

To address both objectives, we interviewed officials from NRC headquarters and its four regional offices, NRC resident inspectors, and officials from the Department of Energy—including the Office of Nuclear Energy and the Idaho National Laboratory—the Federal Energy Regulatory Commission, the Federal Emergency Management Agency, the National Oceanic and Atmospheric Administration (NOAA), and the U.S. Forest Service. In addition, we interviewed a nongeneralizable sample of nine stakeholders knowledgeable about nuclear power plant operations and safety, climate change, and resilience measures. We also visited two selected nuclear power plants—Palo Verde Nuclear Generating Station in Buckeye, Arizona, and Turkey Point Nuclear Generating Station in Homestead, Florida—and interviewed plant staff and NRC resident inspectors at these plants. We selected these plants because of their exposure to a variety of natural hazards that may be exacerbated by climate change and regional diversity. Findings from selected stakeholder interviews and site visits are not generalizable to all stakeholders and sites.

To examine how climate change is expected to affect nuclear power plants, we conducted a literature review of articles and reports related to the effects of climate change on nuclear power plants. On the basis of this method, we identified and used 36 articles to support the findings in our report. We also reviewed the fourth and fifth U.S. Global Change

[1]NRC has efforts underway to support the licensing of advanced nuclear reactors—nuclear fission reactors that may offer significant improvements over the most recent generation of nuclear fission reactors and may involve first-of-a-kind designs—which, according to NRC officials, contribute to climate resilience by supporting an alternative to fossil-fuel-based power plants. For more information on NRC's licensing of advanced nuclear reactors, see GAO, *Nuclear Power: NRC Needs to Take Additional Actions to Prepare to License Advanced Reactors,* GAO-23-105997 (Washington, D.C.: July 27, 2023).

Research Program's National Climate Assessments (NCA),[2] federal data on natural hazards, and prior GAO reports.

Additionally, we identified and obtained national-level data sets from relevant federal agencies for six of the seven natural hazards identified by the NCA and our literature review as likely to be exacerbated by climate change: extreme heat, extreme cold, wildfires, flooding, storm surge from hurricanes, and sea level rise.[3] For heat, cold, and sea level rise, we used data that are based on climate scenarios. For heat and cold, we analyzed the projected exposure of nuclear power plants to those hazards.[4] For wildfires, hurricane storm surge, and flooding, we used data that are based on current and past conditions.[5] We assessed the reliability of the data sources used and found the data to be sufficiently reliable for the purposes of our reporting objectives. For more detailed information on our scope and methodology, and the steps we took to assess the reliability of the data used in this report, see appendix I. For more detail on data sources used in this report, see appendix II.

In addition, we obtained NRC data on the location of all 54 operating U.S. nuclear power plants as well as on the 21 shutdown nuclear power plants

[2]U.S. Global Change Research Program, *Fifth National Climate Assessment*, (Washington, D.C.: 2023); U.S. Global Change Research Program, *Impacts, Risks, and Adaptation in the United States: Fourth National Climate Assessment*, vol. II (Washington, D.C.: 2018).

[3]To identify and select national-level data sets, we used information from the NCA. The fifth NCA was released on November 14, 2023, after we had obtained and analyzed the hazard data sets. We reviewed relevant sections from the fifth NCA and did not identify major differences in the predicted or projected trends for the selected natural hazards. We did not analyze drought data because we were unable to identify national-level geospatial data that was both relevant to nuclear power plants and sufficiently reliable for our purposes.

[4]To analyze projected exposure to heat and cold hazards, we used data from the fourth NCA on the projected exposure to maximum and minimum temperatures by the midcentury (i.e., 2036–2065). We selected data using the projected change by the midcentury time frame under both a low- and high-emission scenario to show the range of potential projected change to selected natural hazards. The midcentury time frame was selected because it captures potential hazard effects during the period in which most U.S. nuclear power plants are likely to remain operational.

[5]To analyze exposure to floods, we used 2023 data from the Federal Emergency Management Agency that categorize flood exposure as a high, moderate, minimal, other, or unknown flood hazard. To analyze exposure to hurricane storm surge, we used NOAA data on storm surge exposure from Category 1 hurricanes (the lowest possible category) and Category 4 or 5 hurricanes (the highest possible categories) to show a range of potential climate change effects. To analyze exposure to wildfires, we used 2023 data from the U.S. Forest Service on wildfire hazard potential.

that have spent nuclear fuel stored onsite in spent fuel pools or in dry cask storage.[6] We analyzed these data using mapping software to identify nuclear power plants located in areas that may be affected by selected natural hazards. We determined that the data were sufficiently reliable for the purposes of our reporting objectives.

To examine NRC's actions to address risks to nuclear power plants from climate change, we reviewed relevant laws and regulations; agency guidance and documents, including NRC's 2022–2026 Strategic Plan; NRC office instructions; and the NRC inspection manual on adverse weather protection.[7] We also reviewed GAO's *Standards for Internal Control in the Federal Government*.[8]

We conducted this performance audit from November 2022 to April 2024 in accordance with generally accepted government auditing standards. Those standards require that we plan and perform the audit to obtain sufficient, appropriate evidence to provide a reasonable basis for our findings and conclusions based on our audit objectives. We believe that the evidence obtained provides a reasonable basis for our findings and conclusions based on our audit objectives.

[6]When a company decides to shut down a nuclear power plant permanently, the facility must be decommissioned by safely removing it from service and reducing residual radioactivity to a level that permits release of the property and termination of the operating license. NRC regulates the decommissioning a nuclear power plant and any spent nuclear fuel that will remain on site. See 10 C.F.R. pt. 20, subpt. E; 10 C.F.R. §§ 50.75, 50.82, 51.53, 51.95. For the purposes of this report, we use the term "shutdown" to refer to plants at various stages of decommissioning, including those in the process of decommissioning and those already decommissioned, with spent nuclear fuel stored onsite. Spent nuclear fuel is the fuel that has been removed from commercial nuclear power reactors after it has been used to produce electricity. Spent nuclear fuel is initially stored immersed in pools of water designed to cool and isolate it from the environment. Water circulates in the pools to remove the heat generated from the radioactive decay. Industry practice has been to store the spent nuclear fuel in these pools for at least 5 years or until the fuel has cooled enough to be transferred to dry cask storage. Dry cask storage consists of a steel canister that holds the fuel assemblies, protected by an outer cask made of steel and concrete designed to cool the fuel and provide shielding from its radiation. We also obtained data on the location of the two Strategic Alliance for FLEX Emergency Response (SAFER) centers that maintain emergency equipment that can be provided to plants as a backup to the plants' onsite primary backup equipment.

[7]NRC, *Strategic Plan Fiscal Years 2022-2026,* NUREG-1614, Vol. 8 (Washington, D.C.: April 2022). See also, NRC, *Inspection Manual: Adverse Weather Protection,* Inspection Procedure 71111, Attachment 01 (Washington, D.C.: Jan. 1, 2018).

[8]GAO, *Standards for Internal Control in the Federal Government*, GAO-14-704G (Washington, D.C.: Sept. 10, 2014).

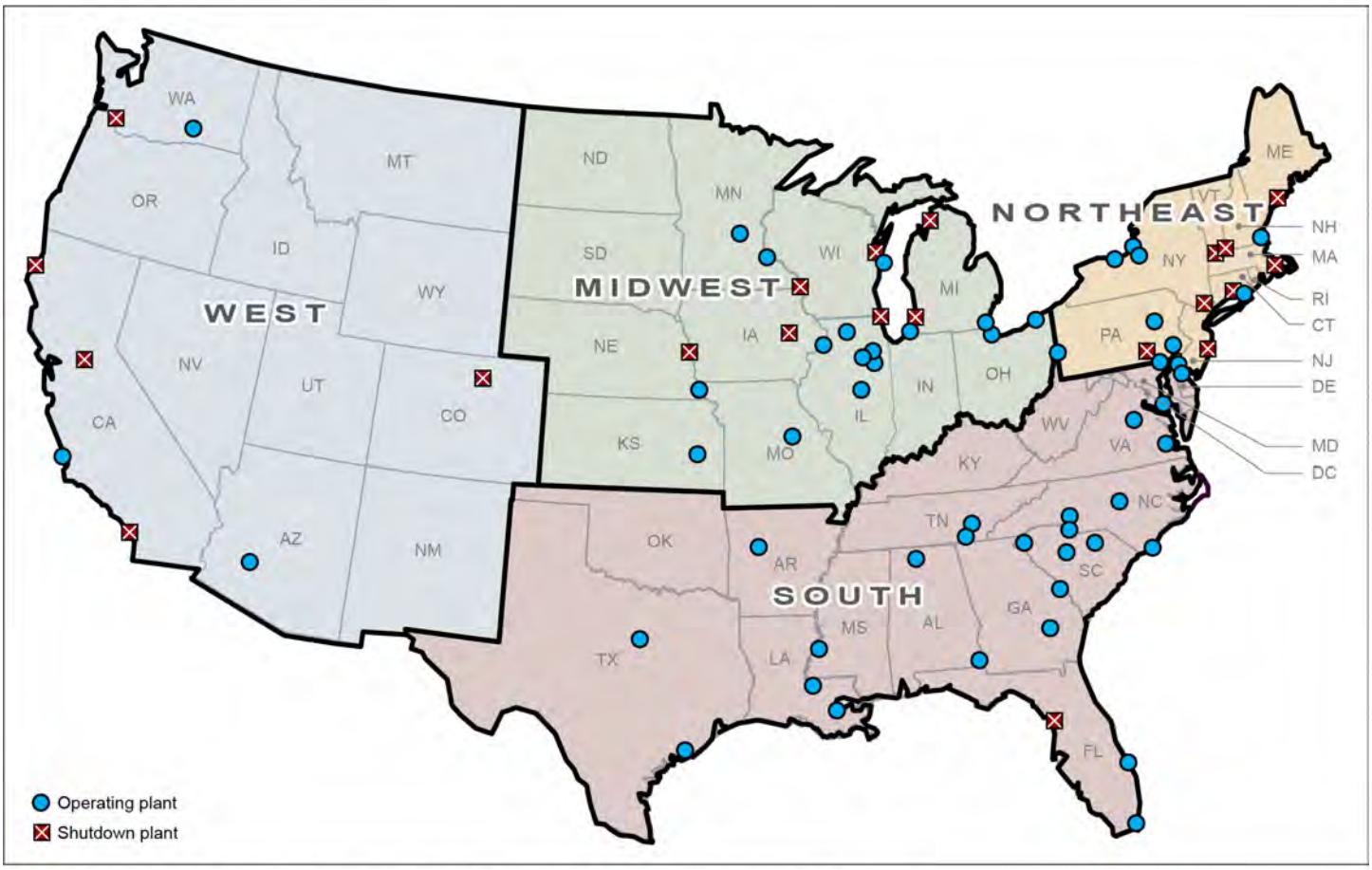
Background

The Nuclear Power Industry and U.S. Plant Operations

Private companies own nearly all nuclear power plants in the United States. As of August 2023, the United States had 93 operating commercial nuclear reactors with an average age of about 42 years old, according to the U.S. Energy Information Administration. These reactors are located at 54 nuclear power plants in 28 states.[9] In addition, as of July 2023, there were 21 shutdown plants that have spent nuclear fuel stored onsite in spent fuel pools or in dry casks. See figure 1 for the locations and regions of operating and shutdown nuclear power plants by U.S. Census region.

[9]According to the U.S. Energy Information Administration, nuclear reactors are machines that contain and control nuclear chain reactions while releasing heat at a controlled rate. A nuclear power plant uses the heat that a nuclear reactor produces to turn water into steam, which then drives turbine generators that generate electricity.

Figure 1: Map of Operating and Shutdown Nuclear Power Plants by U.S. Census Region



Sources: GAO analysis of U.S. Census Bureau and Nuclear Regulatory Commission data; U.S. Census Bureau (map). | GAO-24-106326

Note: This map includes 75 U.S. nuclear power plants—54 operating plants and 21 shutdown plants with spent nuclear fuel onsite.

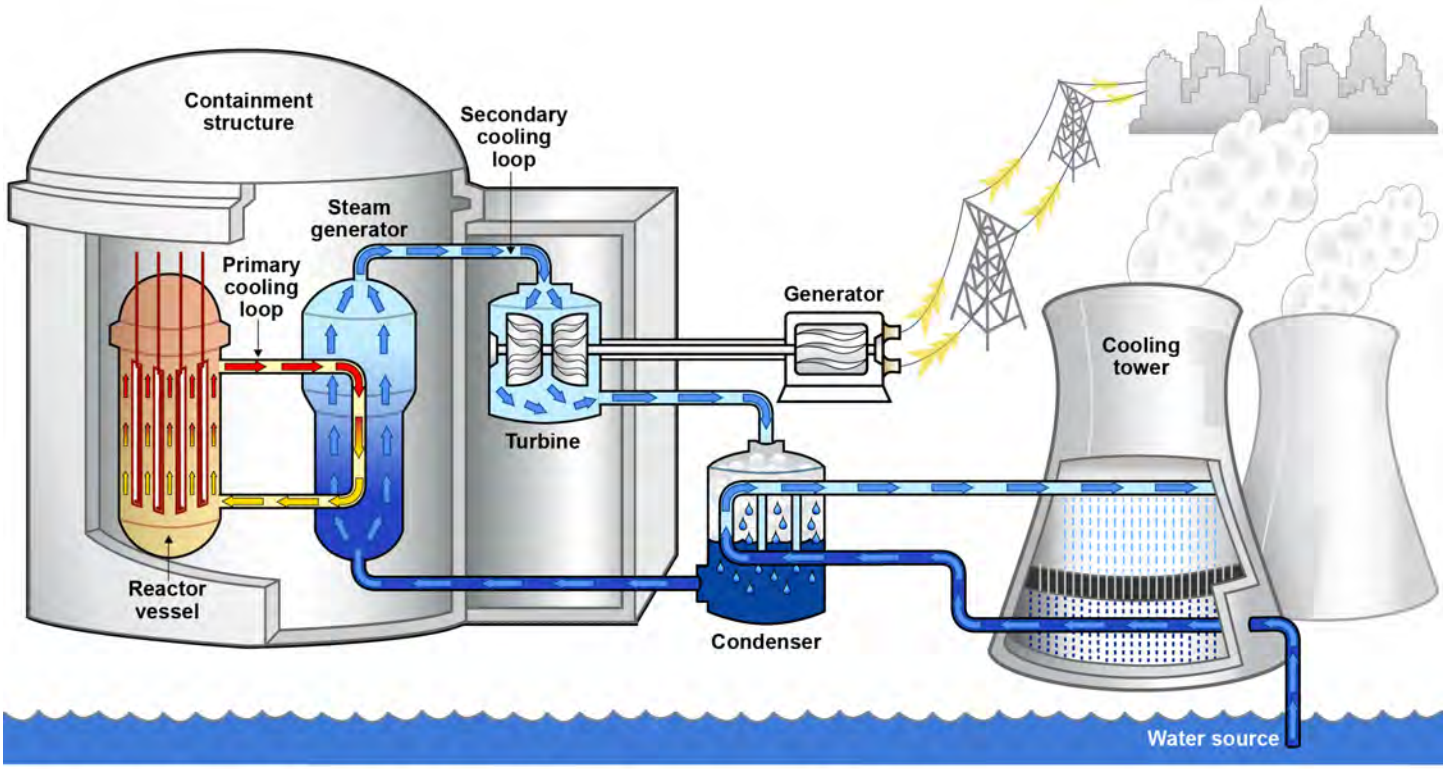
Nuclear reactors rely on technologies to initiate and control chain reactions that produce heat through a physical process called fission—whereby atoms are split to release energy. All commercial nuclear power reactors in the United States use uranium as fuel and are light water reactors, which means they use water as both a coolant and moderator to

serve critical safety and operations functions.[10] Nuclear power plants use water during normal operations to absorb the heat that is left over after making electricity and to cool the equipment and buildings used in generating that electricity. In the event of an accident, nuclear power plants also need water to remove the heat produced by the reactor core, even when it is temporarily shut down. Water is also used to cool spent fuel once it is removed from the reactor core. Because light water reactors rely on water for key safety and operational functions, nuclear power plants are typically located next to lakes, rivers, or oceans.

There are two types of light water reactors in the United States—pressurized water reactors and boiling water reactors. Pressurized water reactors, the predominant type of light water reactor in the United States, use steam generators to transfer the heat created by fission from the primary coolant loop to the secondary coolant loop, creating steam in the secondary loop that spins a turbine and generates electricity. Boiling water reactors, which constitute a third of the operating reactors in the United States, do not use steam generators or have secondary loops. Instead, the steam is generated directly inside the reactor vessel. See figure 2 for an overview of a nuclear power plant's components for a pressurized water reactor.

[10]The commercial nuclear power reactors currently operational in the United States are known as "light water reactors," meaning reactors that use ordinary water to cool and moderate the reactor, as opposed to heavy water, which contains deuterium, an isotope of hydrogen.

Figure 2: Nuclear Power Plant Components and Operations for a Pressurized Water Reactor



Sources: GAO illustration and analysis of Department of Energy and Nuclear Regulatory Commission documentation. | GAO-24-106326

Note: This illustration depicts a pressurized water reactor, the predominant reactor type in the United States. Boiling water reactors, which constitute a third of the operating reactors in the United States, do not use steam generators or have secondary loops. Boiling water reactors boil water directly inside the reactor vessel to produce steam.

To operate the cooling pumps and other systems that manage the water that reactors rely on for key safety and operational functions, nuclear plants need a reliable source of power. Nuclear power plants typically rely on the electricity grid to which the plant is connected for offsite power.[11]

[11]As we reported in 2021, climate change is expected to affect every aspect of the electricity grid—from generation, transmission, and distribution, to demand for electricity. We found that power outages can have significant cascading effects on critical sectors and electric service disruptions can significantly affect the reliability of other parts of the energy sector. These losses are of special concern because outages caused by climate effects can be widespread and affect large geographic areas all at once, according to the Department of Energy. GAO, *Electricity Grid Resilience: Climate Change Is Expected to Have Far-Reaching Effects and DOE and FERC Should Take Actions,* GAO-21-346 (Washington, D.C.: March 5, 2021).

However, if a plant loses access to offsite power, it must rely on backup power sources, such as diesel generators, to power cooling pumps. The loss of power and ability to pump cooling water can have a significant adverse impact on a plant's ability to safely shut down and maintain safe shutdown conditions. This could result in damage to a reactor's core and potentially release radiological material into the environment.

NRC's Role

NRC is an independent federal agency, headed by five commissioners, responsible for permitting the construction and licensing of commercial nuclear power reactors and regulating and overseeing their security and safe operation.[12] NRC can issue a license to operate a nuclear power reactor for up to 40 years and can renew a license for up to 20 additional years. A renewed license may be subsequently renewed for up to another 20 years, allowing a reactor to operate for up to a total of 80 years. As of December 2023, NRC had issued subsequent license renewals for six reactors at three nuclear power plants in the United States.[13] Spent nuclear fuel may remain onsite long after a plant shuts down.[14]

As part of NRC's process for issuing construction permits and licenses for nuclear power plants, agency staff conduct safety and environmental reviews. As part of the safety review, NRC reviews a plant's design to ensure it meets the technical specifications required for the safe operation of the plant. Specifically, NRC's reactor design criteria require that important safety systems, structures, and components are designed to withstand the effects of natural hazards, including climate-related hazards

[12]NRC's mission is to regulate the civilian use of radioactive materials, to provide reasonable assurance of adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. As such, any new requirements that the agency imposes on commercial nuclear plants must meet this standard, according to NRC officials.

[13]NRC issued subsequent license renewals to Turkey Point Units 3 and 4 in December 2019; Peach Bottom Units 2 and 3 in March 2020; and Surry Units 1 and 2 in May 2021.

[14]The United States does not have a consolidated storage facility or repository where plants can send their spent fuel during operations or after a plant shuts down. GAO, *Commercial Spent Nuclear Fuel: Congressional Action Needed to Break Impasse and Develop a Permanent Disposal Solution,* GAO-21-603 (Washington, D.C.: Sept. 23, 2021).

such as hurricanes and floods, without losing the ability to perform their safety functions.[15]

License applicants are responsible for ensuring their plants are protected against natural hazards by assessing the hazards that may affect their plants and designing the plants to withstand those hazards. NRC is responsible for reviewing plant and reactor designs and comparing the design limits for natural hazards with those found in applicants' hazard assessments, which consider the characteristics of the plant's geographic location. Once a nuclear power plant is licensed and operational, NRC conducts regular inspections of the plant's systems and ensures that the licensee is operating in accordance with its license. If a plant experiences external conditions that exceed the limiting conditions for operation, the licensee is required to either shut the reactor down, take remedial actions as permitted in its license, or request a license amendment or enforcement discretion from NRC to continue operations.[16]

NRC also regulates the decommissioning of nuclear power plants, which means safely removing nuclear power plants from service by reducing residual radioactivity to a level that permits the release of the property and termination of the license.[17]

NRC's Regulatory Approach

NRC uses conservatism, safety margins, and defense-in-depth to implement regulatory requirements for the design, construction, maintenance, operation, and decommissioning of nuclear power plants to prevent and mitigate accidents that could release radiation or hazardous

[15]10 C.F.R. Part 50, Appendix A, General Design for Nuclear Power Plants, Criterion 2—*Design Bases for Protection Against Natural Phenomena*. According to an NRC document, all currently operating reactors were licensed to meet the intent of the General Design Criteria, which include General Design Criterion 2. See also 10 C.F.R. §§ 50.34, 52.79 (detailing safety analysis and design requirements for a license application).

[16]Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the plant. 10 C.F.R. § 50.36(c)(2)(i). If the limiting conditions are exceeded by an extreme weather event, licensees can request the following from NRC: a temporary enforcement discretion for a brief period to allow them to continue operating despite the exceedance; a temporary license amendment to revise the limiting conditions for a specified period (e.g., 1–3 months); or a permanent license amendment to change the technical specifications.

[17]The NRC ensures that safety requirements are being met throughout the decommissioning process by reviewing decommissioning or license termination plans, conducting inspections, monitoring to ensure that radioactive contamination is reduced or stabilized, and issuing permits for spent nuclear fuel that will remain on site after license termination. See 10 C.F.R. pt. 20, subpt. E; 10 C.F.R. §§ 50.75, 50.82, 51.53, 51.95.

materials. According to agency documents and NRC officials we interviewed, the approach can be described as follows:

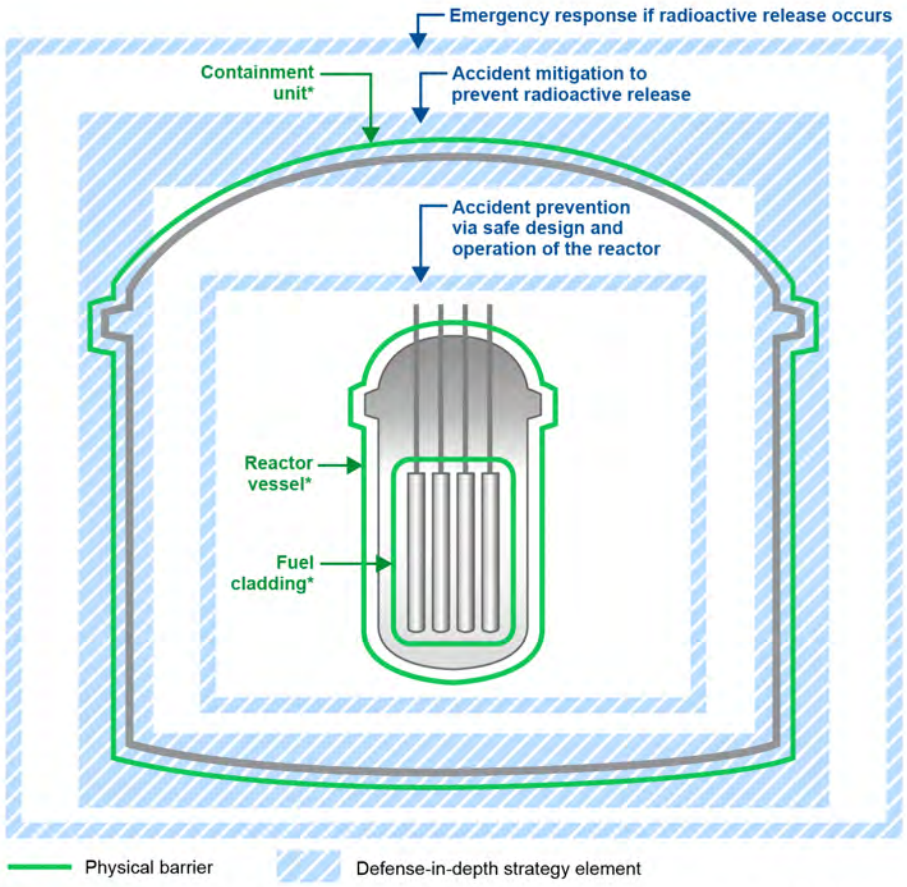
- Conservatism, for example, includes the consideration of the most severe natural phenomena that have been historically reported for a nuclear power plant site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.[18]
- Safety margins are the extra capacity factored into the design of a structure, system, or component so that it can cope with conditions beyond what is expected as a way to compensate for uncertainty.[19]
- Defense-in-depth includes multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures (see fig. 3).[20]

[18]10 C.F.R. Part 50, Appendix A, General Design Criterion 2, *Design Bases for Protection Against Natural Phenomena*. See also, NRC, *Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision-Making*, NUREG-1855 (Washington, D.C.: March 2017).

[19]NRC, *Glossary of Risk-Related Terms in Support of Risk-Informed Decision-Making,* NUREG-2122 (Washington, D.C.: Nov. 2013).

[20]For more information on defense-in-depth, see NRC, *Historical Review and Observations of Defense-in-Depth*, NUREG/KM-0009 (Washington, D.C.: April 2016).

Figure 3: Measures Consistent with the Nuclear Regulatory Commission's (NRC) Defense-in-Depth Approach



*Containment unit – airtight enclosure around the nuclear reactor to confine radiation that otherwise might be released to the atmosphere in the event of an accident

*Reactor vessel – forms a reactor coolant pressure boundary that provides a barrier against the release of radioactivity generated within the reactor

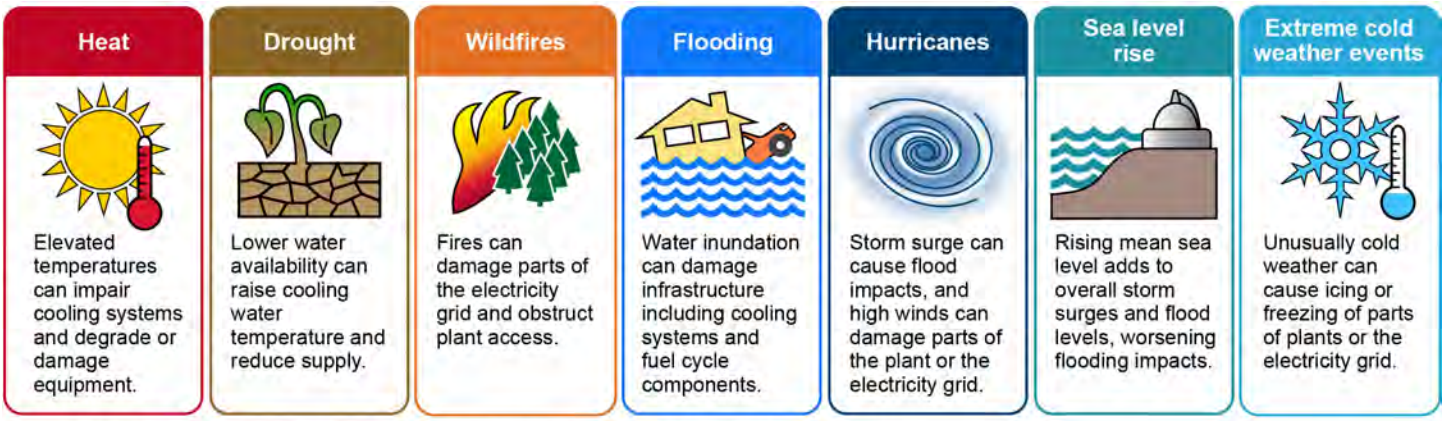
*Fuel cladding – the thin-walled metal tube that forms the outer jacket of a nuclear fuel rod

Source: GAO and NRC documentation. | GAO-24-106326

Climate Change Is Expected to Exacerbate Natural Hazards That Pose Risks to Nuclear Power Plants

Climate change is expected to exacerbate natural hazards—including heat, drought, wildfires, flooding, hurricanes, and sea level rise. In addition, climate change may affect extreme cold weather events.[21] These natural hazards pose risks to nuclear power plants (see fig. 4).

Figure 4: Examples of Natural Hazards that May Pose Risks to Nuclear Power Plants



Sources: Nuclear Regulatory Commission documents; summary of literature; GAO (icons). | GAO-24-106326

Note: The potential risks to nuclear power plants from these hazards include a loss of offsite power, diminished cooling capacity, flood damage, and reduced operations or temporary plant shutdowns. The loss of offsite power is a complete loss of electrical power from the grid to a nuclear power plant. The loss can decrease a plant's ability to maintain safe shutdown conditions. Diminished cooling capacity refers to any impact which reduces a plant's ability to cool reactor or fuel cycle components and can result in a temporary plant shutdown.

[21]According to the NCA, climate change has driven increases in the frequency and severity of some extreme weather events. For example, climate change caused Hurricane Harvey's rainfall to be an estimated 15 and 20 percent heavier than it would have been without human-caused warming. However, researchers disagree about some climate impacts. For example, whereas emerging research suggests that the frequency of cold-weather events and heavy snowfall may be increasing because of warming Arctic temperatures, there is some disagreement in the research community regarding this projection.

Heat, Drought, and Wildfires Pose Risks to Nuclear Power Plants, and Climate Change Is Expected to Exacerbate These Hazards, Particularly in the South and Southwest

According to our analysis of NCA and U.S. Forest Service data, all 75 operating and shutdown U.S. nuclear power plants are located in areas where climate change is expected to exacerbate heat, drought, wildfires, or all three.

Heat and drought. Heat and drought pose risks to nuclear power plants because they can affect the water used for cooling. Specifically, higher-than-usual ambient air temperatures may increase the temperature of water used for cooling. Drought can also reduce the supply of cooling water. If a plant has an insufficient supply of cooling water or its cooling water approaches or exceeds the maximum allowable temperature for cooling certain reactor components, a licensee may need to temporarily limit or stop operations to ensure plant safety. Higher temperatures in the bodies of water into which nuclear power plants discharge cooling water may also require a plant to limit or temporarily stop operations to comply with laws designed to protect aquatic ecosystems and wildlife.[22] In addition, high temperatures can also degrade the performance or cause failure of pumps and other equipment, reduce the lifetime of plant components, and reduce the overall efficiency of power plants. Warmer temperatures may also increase levels of certain algae or other biological material which can block cooling water systems and lead to reduced production or a temporary plant shutdown.

[22]Some plants that discharge cooling water into rivers or lakes are subject to environmental requirements. These requirements could force a power plant to shut down or reduce power generation. For example, in 2007, 2010, and 2011, the Tennessee Valley Authority had to reduce power output from its Browns Ferry Nuclear Power Plant in Alabama because river temperatures were too high to receive discharge water from the plant without posing ecological risks.

Heat and Drought at Turkey Point Nuclear Generating Station

According to the Nuclear Regulatory Commission (NRC) and Turkey Point Nuclear Generating Station officials, in 2014, extended drought conditions and high algae content caused the cooling water for the Turkey Point Generating Station to exceed its maximum allowable temperature in its license. NRC approved the licensee's requests to not enforce the temperature requirement for the plant's cooling water for a limited period. Later, NRC granted the licensee a permanent license amendment that raised the maximum allowable cooling water temperature for the plant from 100 degrees to 104 degrees Fahrenheit.

High temperatures and drought conditions at Turkey Point Nuclear Generating Station potentially created risks to local drinking water sources when decreased water levels and increased evaporation rates led to higher salinity in the cooling canals. Higher salinity levels made the water denser, causing it to sink below the canals that contain it. This could have led to intrusion of higher salinity water into the areas of the Biscayne Aquifer, a source of drinking water for the Miami-Dade area.

To mitigate these risks, the licensee constructed a series of wells to decrease the water salinity in the cooling canals.

Well used to adjust salinity in the Turkey Point Nuclear Generating Station's cooling canals

Sources: Interviews with plant personnel at the Turkey Point Nuclear Generating Station, and review of NRC documents; GAO (photo). | GAO-24-106326

All operating and shutdown nuclear power plants are located in areas where climate change is projected to increase measures of heat, including daily and average maximum temperature, according to our analysis of NCA and NRC data. The effects of climate change on maximum temperatures are projected to be most severe in the South, where one-third of the plants are located.[23] The plants in the South are projected to experience an annual average of from 21 to 31 days with higher maximum temperatures than historical high temperatures. In addition, according to the NCA, climate change is expected to increase drought intensity in some regions, specifically in the Southwest, where two operating and four shutdown nuclear power plants are located.

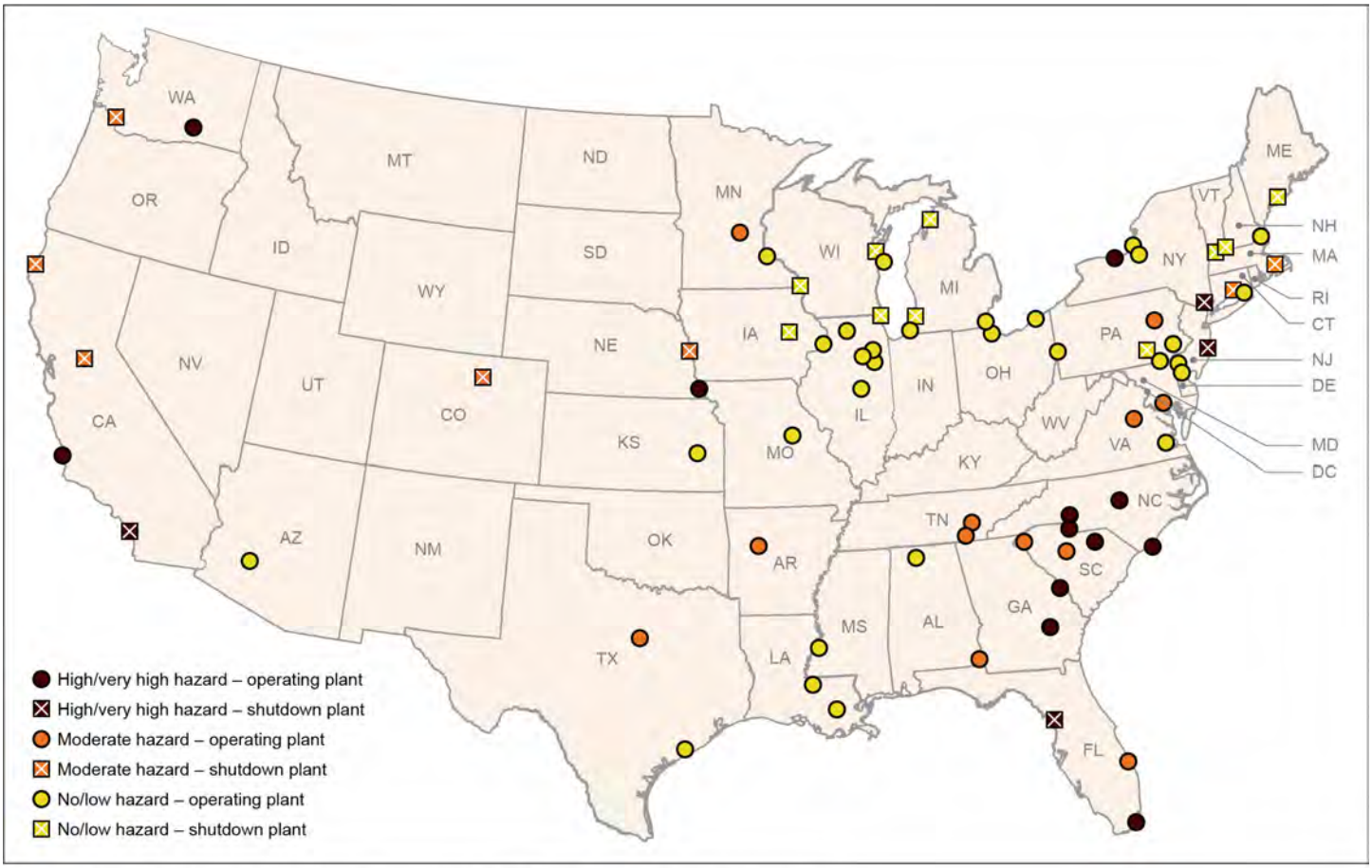
[23]Of the 25 plants in the South, 24 are operational and one is shutdown.

Wildfire. According to the NCA, increased heat and drought contribute to increases in wildfire frequency, and climate change has contributed to unprecedented wildfire events in the Southwest. The NCA projects increased heatwaves, drought risk, and more frequent and larger wildfires. Wildfires pose several risks to nuclear power plants, including increasing the potential for onsite fires that could damage plant infrastructure, damaging transmission lines that deliver electricity to plants, and causing a loss of power that could require plants to shut down. Wildfires and the smoke they produce could also hinder or prevent nuclear power plant personnel and supplies from getting to a plant.

According to our analysis of U.S. Forest Service and NRC data, about 20 percent of nuclear power plants (16 of 75) are located in areas with a high or very high potential for wildfire.[24] More specifically, more than one-third of nuclear power plants in the South (nine of 25) and West (three of eight) are located in areas with a high or very high potential for wildfire (see fig. 5).

[24]The U.S. Forest Service maps wildfire hazard potential based on landscape conditions and other observations. These maps include an index of wildfire hazard potential for the United States, based on, among other factors, annual burn probabilities and the potential intensity of large fires. The wildfire potential index is a relative ranking. The U.S. Forest Service categorizes the wildfire hazard potential index into five classes: very low, low, moderate, high, and very high. The U.S. Forest Service designates as “high” those areas with wildfire hazard potential index from the 85th to the 95th percentile, and as “very high” those areas above the 95th percentile. For this analysis, we combined the high and very high wildfire hazard potential categories; we did not identify the number of facilities in each of these categories separately. Of the 16 plants with high or very high potential for wildfire, 12 are operating and four are shutdown.

Figure 5: Nuclear Power Plants Located in Areas with Exposure to No/Low, Moderate, and High/Very High Wildfire Hazard Potential



Sources: U.S. Forest Service and Nuclear Regulatory Commission data; U.S. Census Bureau (map). | GAO-24-106326

Note: To determine if a plant is located in an area with wildfire hazard potential, we identified overlap between a 0.5-mile radius around nuclear power plant coordinates provided by the Nuclear Regulatory Commission and wildfire hazard potential data. Overlap indicates that a facility is located in an area that may be affected by the selected hazard. We used the U.S. Forest Service Wildfire Hazard Potential Map to show exposure to wildfire hazard potential. The U.S. Forest Service categorizes the wildfire hazard potential index into five classes of very low, low, moderate, high, and very high. We analyzed the moderate, high, and very high wildfire potential layers, and combined results for the high/very high layers. No/low refers to plants that are not located in an area with wildfire potential of moderate, high, or very high, based on the U.S. Forest Service Wildfire Hazard Potential Map. See appendix I for more details on our data analysis. We previously reported that the primary intended use of the wildfire hazard potential map is to identify priority areas for hazardous fuels treatments from a broad, national- to regional-scale perspective. This analysis does not account for any protective measures plants may have taken to mitigate the risk of selected natural hazards.

Appendix III provides additional details of exposure to heat and wildfire hazard potential in areas where nuclear power plants are located.

Flooding, Hurricanes, and Sea Level Rise Pose Risks to Nuclear Plants, and Climate Change Is Expected to Exacerbate These Hazards, Particularly in Coastal Regions

According to our analysis of NOAA and NRC data, about 63 percent of nuclear power plants (47 of 75) are located in areas with exposure to either Category 4 or 5 hurricane storm surge or high flood hazard, and nine are located on a coastline, where NOAA projects a range of sea level increases.[25] In addition, 20 percent of nuclear power plants (15 of 75) are located in areas with exposure to both Category 4 or Category 5 hurricane storm surge and high flood hazard. The NCA predicts that climate change will exacerbate all three hazards.

[25]To identify coastal plant locations, we used nuclear power plant coordinates from NRC and added a 0.5-mile radius around NRC's plant coordinates as a proxy for an average size nuclear power plant. Coastal plants were those with a radius that intersected with or beyond the coastline.

Flood Protection

To mitigate the impacts of flooding, licensees have implemented various measures, including the elevation of spent fuel pools and use of flood barriers.

Flood barrier protecting part of the Turkey Point Nuclear Generating Station

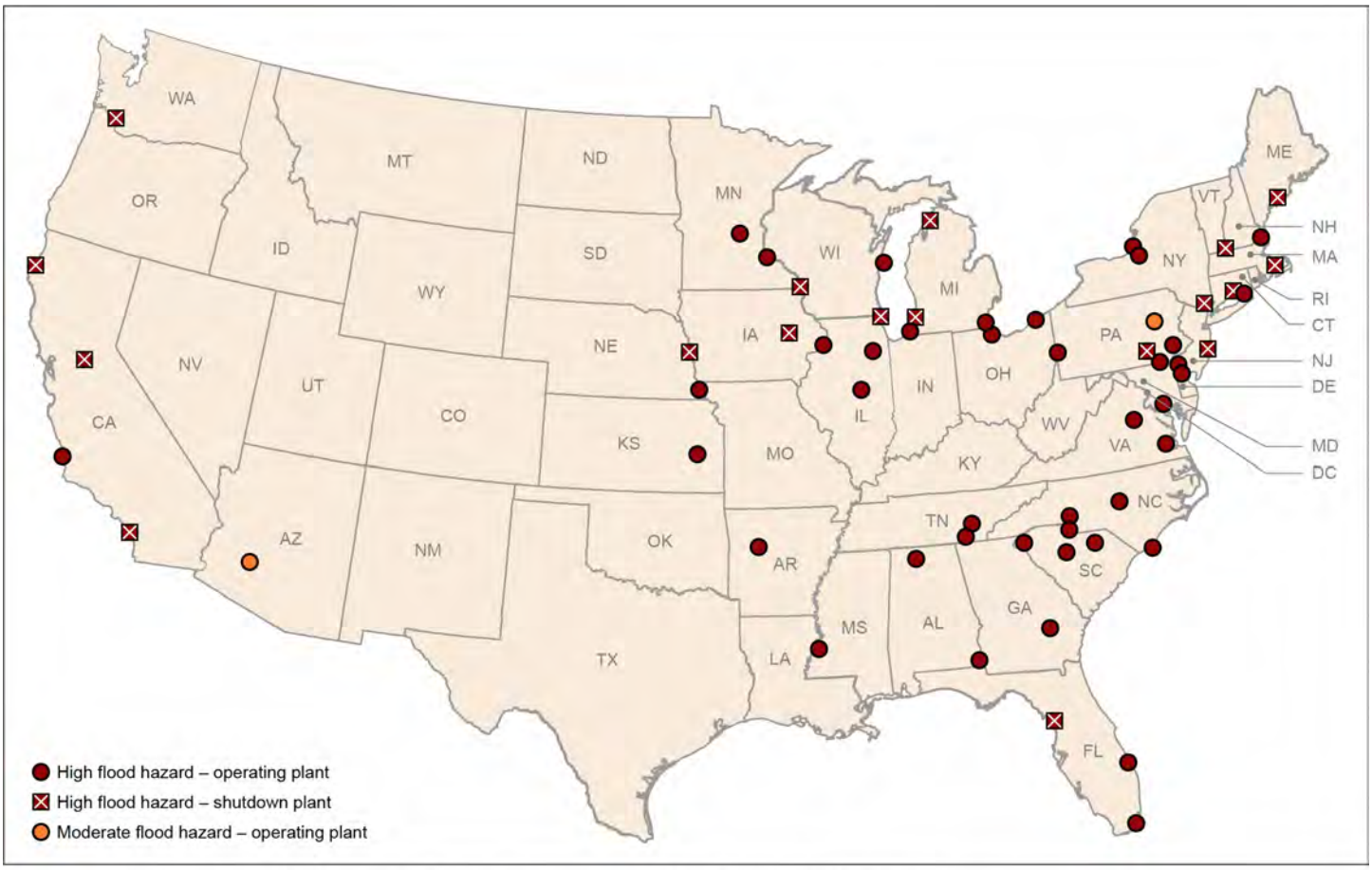
Sources: GAO site visit and interviews with plant personnel at the Turkey Point Nuclear Generating Station; GAO (photo). | GAO-24-106326

Flooding. Flooding could pose risks to nuclear power plants by, among other things, diminishing a plant's cooling capacity. Flooded roads could prevent personnel, equipment, and supplies from reaching a plant. Flooding could also cause damage to buildings, equipment, and electrical systems that could require a plant to curtail operations or shut down. In addition, flood waters could interfere with heat removal from spent fuel pools by blocking ventilation ports with water. Prolonged exposure to salt water from coastal flooding could also degrade or corrode a cask's exterior, potentially posing risks to the environment and human health.

Our analysis of Federal Emergency Management Agency data found that 60 of the 75 nuclear power plants in the United States are located in areas with high flood hazard and two are in areas with moderate flood hazard.[26] Just over one-third of the plants (21 of 60) located in areas with high flood hazard are in the South (see fig. 6). According to the NCA, heavy rainfall and flooding are expected to become more frequent and severe across the United States. The NCA predicts that climate change will continue to exacerbate hurricane storm surge, rainfall, and flood events in U.S. coastal areas.

[26]We analyzed Federal Emergency Management Agency data from 2023. For our analysis, high flood hazard corresponds to areas in 100-year floodplains (areas with a 1 percent or higher annual chance of flooding), moderate flood hazard corresponds to areas in 500-year floodplains (areas with a 0.2 percent or higher annual chance of flooding), and no/low corresponds to areas with minimal, unknown, or other flood hazards, including areas with reduced risk because of levees as well as areas with flood hazard based on future conditions, such as the future implementation of land-use plans. Of the 60 plants located in areas with high flood hazard, 42 are operating and 18 are shutdown. Both of the plants located in areas with moderate flood hazard are operating.

Figure 6: Nuclear Power Plants Located in Areas with High and Moderate Flood Hazard



Sources: Federal Emergency Management Agency and Nuclear Regulatory Commission data; U.S. Census Bureau (map). | GAO-24-106326

Note: To determine if a plant is located in an area with exposure to moderate or high flood hazard, we identified overlap between a 0.5-mile radius around nuclear power plant coordinates provided by the Nuclear Regulatory Commission and the flood hazard data. Overlap indicates that a facility is located in an area that may be affected by the selected hazard. See appendix I for more details on our data analysis. To show exposure to flooding, we use the Federal Emergency Management Agency's National Flood Hazard Layer, which estimates several levels of flood hazard, including high flood hazard (areas with a 1 percent or higher annual chance of flooding), and moderate flood hazard (areas with a 0.2 percent or higher annual chance of flooding). This analysis does not account for any protective measures plants may have taken to mitigate the risk of selected natural hazards.

Hurricanes. High winds from hurricanes can generate projectiles capable of damaging parts of nuclear power plants and electricity transmission lines that provide nuclear power plants with power. In addition, storm surge from hurricanes can cause flooding, which could diminish a plant's cooling capacity and damage buildings, equipment, and electrical

systems. About 23 percent of nuclear power plants (17 of 75) are located in areas that may be inundated by storm surge from Category 4 or Category 5 hurricanes,[27] according to our analysis of NOAA and NRC data.[28] All 17 of these plants are in the East and South, and the six plants with exposure to Category 5 hurricanes are located in the South (see fig. 7).[29] According to the NCA, climate change is expected to heighten hurricane storm surges, wind speeds, and rainfall rates.[30]

[27]Of the 17 plants located in areas that may be inundated by storm surge from Category 4 or 5 hurricanes, 11 are operating and six are shut down. For the West Coast of the United States, storm surge data were only available for Southern California.

[28]Our analysis of NOAA storm surge data is based on a model that estimates the maximum extent of storm surge at high tide. NOAA provides estimates of hurricane storm surge using a model called Sea, Lake, and Overland Surges from Hurricanes. This model includes hypothetical hurricanes under different storm conditions, such as landfall location, trajectory, and forward speed. Hurricanes reaching Category 3 and higher are considered major hurricanes because of the potential for significant loss of life and damage. In our analysis, we used the maximum extent of storm surge from Category 1 hurricanes (the lowest possible category) and Category 5 hurricanes (the highest possible category) to show a range of potential climate change effects. Category 4 hurricanes carry sustained winds of 130–156 miles per hour. Category 5 hurricanes have sustained winds exceeding 156 miles per hour.

[29]Storm surge impacts to nuclear power plants would depend on several factors, including a plant's elevation and protective measures.

[30]Climate change leads to warmer ocean surface temperatures. This, in turn, makes hurricanes more powerful because the temperature increase causes more water to evaporate from the ocean. Evaporation adds moisture to the air, and warmer air temperatures can hold more water vapor. The increased moisture in the air leads to more intense rainfall. In a hurricane, spiraling winds draw moist air toward the center, fueling the thunderstorms that surround it.

Figure 7: Nuclear Power Plants Located in Areas with Exposure to Storm Surges from Category 4 and Category 5 Hurricanes

Sources: National Oceanic and Atmospheric Administration and Nuclear Regulatory Commission data; U.S. Census Bureau (map). | GAO-24-106326

Notes: To determine if a plant exists in an area with exposure to hurricane storm surge, we identified overlap between a 0.5-mile radius around nuclear power plant coordinates provided by the Nuclear Regulatory Commission and storm surge data. Overlap indicates that a facility is located in an area that may be affected by the selected hazard. See appendix I for more details on our data analysis. To show exposure to hurricane storm surge, we use the National Oceanic and Atmospheric Administration's Sea, Lake, and Overland Surges from Hurricanes Model, which estimates storm surge heights resulting from the various categories of hurricanes. This analysis does not account for any protective measures plants may have taken to mitigate the risk of selected natural hazards.

Sea level rise. Sea level rise could affect nuclear power plants by contributing to greater storm surges and flooding. According to NOAA officials, a rise in sea level can increase corrosion from saltwater intrusion and lead to chronic long-term erosion of coastal cliffs, where some plants

are located.[31] According to a NOAA report, over the next 30 years sea levels will continue to rise as climate change warms glaciers and ice sheets, causing additional water mass to enter the ocean.[32] The rise in sea level is expected to increase coastal flooding by contributing to higher tides and storm surges that reach further inland, potentially affecting coastal nuclear power plants.

Our analysis of NOAA and NRC data indicates that about half of nuclear power plants (37 of 75) are located in a coastal region, and nine of these are located on the coastline.[33] Projected sea level rise in 2050 varies by coastal region, from 0.5 feet in the Northwest to 1.9 feet in the Western Gulf (see fig. 8). In addition, sea level rise may increase saltwater intrusion into the coastal rivers or groundwater aquifers that some nuclear power plants use for service or potable water.[34]

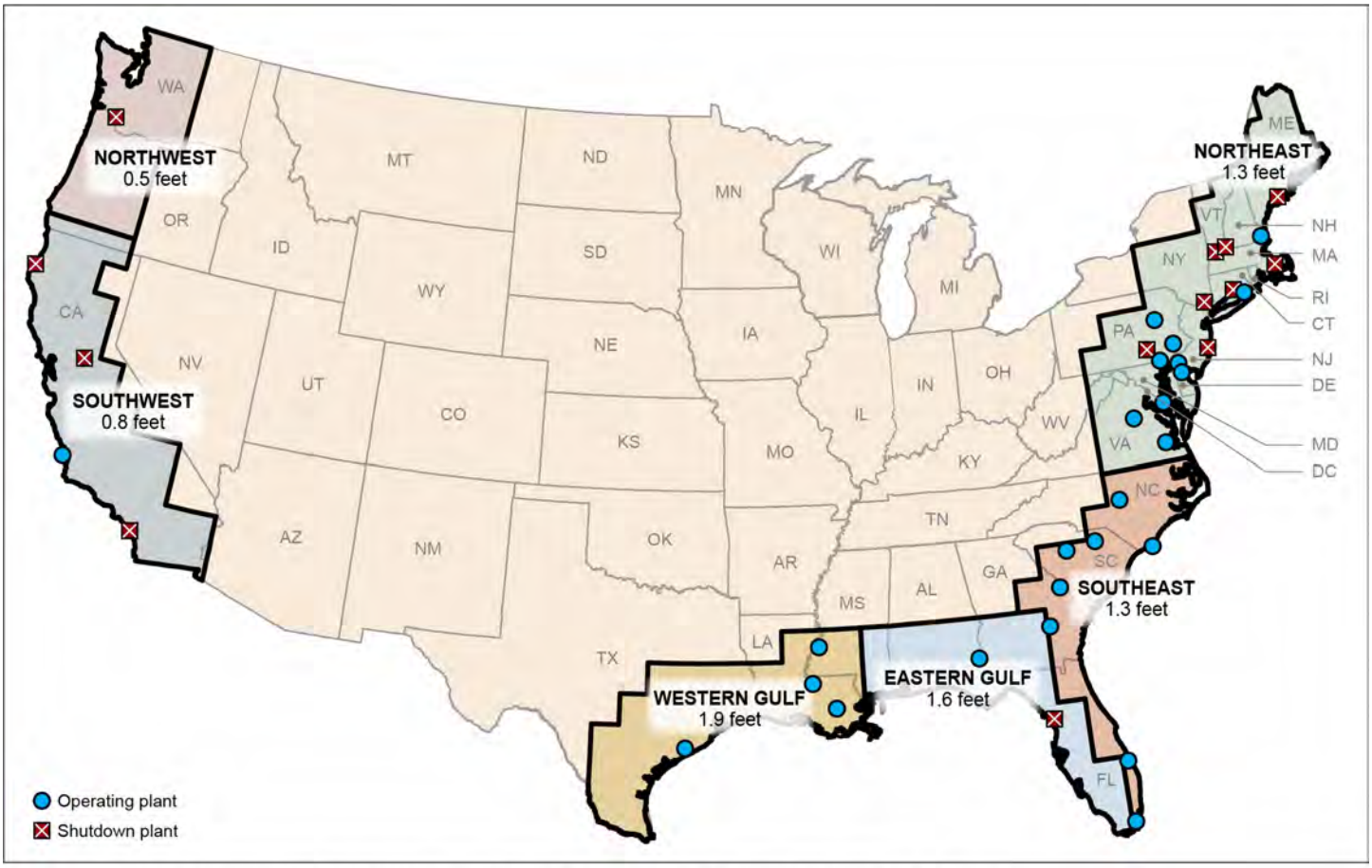
[31]NOAA officials said that Turkey Point Nuclear Generating Station is an example of a plant where, if unaddressed, sea level rise could lead to saltwater intrusion into the plant's cooling canals. Officials also said that Southern California is an example of an area where cliffs consist of unconsolidated rock, a type of loose rock composition that is particularly vulnerable to long-term erosion from sea level rise.

[32]W. V. Sweet, B. D. Hamlington, R. E. Kopp, C. P. Weaver, P. L. Barnard, D. Bekaert, W. Brooks, M. Craghan, G. Dusek, T. Frederikse, G. Garner, A. S. Genz, J. P. Krasting, E. Larour, D. Marcy, J. J. Marra, J. Obeysekera, M. Osler, M. Pendleton, D. Roman, L. Schmied, W. Veatch, K. D. White, and C. Zuzak, *2022: Global and Regional Sea Level Rise Scenarios for the United States: Updated Mean Projections and Extreme Water Level Probabilities Along U.S. Coastlines*, NOAA Technical Report NOS 01 (Silver Spring, MD: Feb. 2022).

[33]Of the 37 nuclear power plants located in a coastal region, 24 are operating and 13 are shut down. Of the nine nuclear power plants located on the coastline, seven are operating and two are shut down. To determine which nuclear power plants are located on the coastline, we identified plants whose coordinates intersect with a coastline. NRC provided coordinate data, and we used a 0.5-mile radius as a proxy for plant size in our analysis.

[34]According to one U.S. Environmental Protection Agency source, sea level rise may increase river levels and the risk of saltwater intrusion into rivers and coastal groundwater aquifers, especially during dry periods. According to NRC's Generic Environmental Impact Statement for License Renewal of Nuclear Plants, saltwater intrusion into groundwater aquifers can degrade the quality of groundwater used for potable and service water at nuclear power plants. See NUREG-1437, Vol. 1, Revision 1.

Figure 8: Nuclear Power Plants in the National Oceanic and Atmospheric Administration (NOAA) Coastal Regions and Projected Sea Level Rise in 2050



Sources: National Oceanic and Atmospheric Administration and Nuclear Regulatory Commission data; U.S. Census Bureau (map). | GAO-24-106326

Note: The regional sea level rise values for 2050 are regional observation-based extrapolations from an interagency report covering sea level rise scenarios. These extrapolations use observed changes in sea level rise and other factors to estimate the trajectory of sea level rise in the near term. Sea-level rise primarily affects coastlines but may also affect the salinity and level of coastal rivers and groundwater aquifers. This map includes all nuclear power plants that are located in NOAA coastal regions. The analysis does not account for site-specific plant elevation or protective measures plants may have taken to mitigate the risk of selected natural hazards.

Appendix III provides additional details of our analysis of exposure to flooding, hurricane storm surges, and sea level rise in areas where nuclear power plants are located.

Extreme Cold Weather Events Pose Risks to Nuclear Power Plants, and Climate Change May Affect These Events in Certain Regions

Cold temperatures can diminish cooling capacity and lead to the loss of offsite power, posing risks to nuclear power plants. Specifically, extreme cold conditions may create ice that could block a plant's cooling water intake system, potentially reducing the supply of cooling water to safety-related systems and components. In addition, frozen precipitation can cause icing of power lines and lead to full or partial loss of off-site power, potentially forcing a plant to rely on backup diesel that may be vulnerable to extremely cold air temperatures.

Extreme Cold at South Texas Project Nuclear Power Plant

On February 15, 2021, the South Texas Project experienced an automatic reactor shutdown when a 5-foot section of uninsulated water line froze, causing the failure of a feed water pump. The Nuclear Regulatory Commission (NRC) found that the facility shut down safely, but the licensee failed to implement a required Freezing Weather Plan to insulate the line. According to one NRC official, a cold weather event nearly rendered another plant's diesel generators inoperable when the air intake temperature dipped to -50 degrees Fahrenheit.

South Texas Project, reactor units 1 and 2

Sources: GAO analysis of NRC documents; U.S. NRC Blog (photo). | GAO-24-106326

Climate change may affect extreme cold weather events.[35] While the NCA found that climate change is expected to cause an overall increase in average temperatures, a 2021 study funded in part by NOAA found that Arctic warming caused by climate change may cause extremely cold air from the Arctic to stretch into the United States.[36] The study links climate change to extreme cold events, such as the record cold temperatures in Texas in 2021. Our analysis of NCA climate projections data and NRC location data found that the average operating nuclear power plant will

[35]As noted previously, according to the NCA, there is disagreement among researchers about some climate impacts. For example, whereas emerging research suggests that the frequency of cold weather events and heavy snowfall may be increasing because of warming Arctic temperatures, there is some disagreement in the research community regarding this projection.

[36]J. Cohen, L. Agel, M. Barlow, C. I. Garfinkel, and I. White, *Linking Arctic Variability and Change with Extreme Winter Weather in the United States*, Science, Volume 373, Issue 6559 (Washington, D.C.: 2021) 1116-1121.

experience from 17 to 22 fewer frost days annually.[37] However, certain regions may also see an increase in extreme cold weather events.

Cold Protection

Licensees have insulated water lines and added cold-weather insulation for turbines to protect against freezing water in pipes and damage to other plant equipment.

Example of insulation at an industrial facility

Sources: Interviews with plant personnel at the Turkey Point Nuclear Generating Station; rootstocks/stock.adobe.com (photo). | GAO-24-106326

Following 2021's Winter Storm Uri, the Federal Energy Regulatory Commission approved a new standard, effective October 2024, that will require certain owners of certain electricity generating units, including nuclear power reactors, to implement freeze protection measures to operate for at least 12 continuous hours at the unit's recorded extreme cold weather temperature.[38]

Appendix III provides additional details of our analysis of exposure to cold weather events in areas where nuclear power plants are located.

[37]Climate projections are used to show a range of future outcomes, and are limited by uncertainties in emissions, natural variability, and scientific models. To show a range of possible outcomes, we used climate projections for a low-emission scenario (17 days) and a high-emission scenario (21 days). Climate projections rely on a variety of assumptions about the future. These limitations are further discussed in appendix II.

[38]In 2023, the Federal Energy Regulatory Commission (FERC) approved Emergency Operation Standard 012-01, also known as the Extreme Cold Weather Preparedness and Operations standard. Effective October 1, 2024, the standard addresses the effects of operating in extreme cold weather by ensuring owners and operators of generating units like nuclear power reactors develop and implement plan(s) to mitigate the reliability impacts of extreme cold. FERC defines extreme cold weather as the temperature equal to the lowest 0.2 percentile of the hourly temperatures measured in December, January, and February. The standard exempts certain generating units, including nuclear power reactors, which have an extreme cold weather temperature exceeding 32 degrees Fahrenheit or operate only in a backup or non-winter capacity.

NRC's Actions to Address Risks to Nuclear Power Plants from Natural Hazards Do Not Fully Consider the Potential Effects of Climate Change

NRC's processes for licensing and overseeing nuclear power plants include actions to address risks from natural hazards. However, NRC's actions do not fully consider the potential effects of climate change.

NRC's Oversight of Nuclear Power Plants Includes Actions to Address Risks from Natural Hazards

NRC's existing processes are designed to address risks to the safety of nuclear power plants, including risks from natural hazards. For example:

- **Defense-in-depth.** A nuclear power plant must be designed and built to withstand phenomena or events such as earthquakes, tornadoes, hurricanes, and floods without the loss of the structures, systems, or components necessary to ensure public health and safety. According to NRC, NRC's defense-in-depth approach focuses on protecting plants against risks such as those related to events that exceed a plant's design basis, including flooding from intense precipitation or hurricanes.[39] As such, NRC's defense-in-depth approach includes verifying that plants have multiple physical barriers and equipment backups to ensure plant safety if plant structures and equipment are damaged due to such severe weather events or if a power outage threatens a plant's ability to continue cooling the reactor.

[39]The design basis for a plant includes the specific functions to be performed by the structures, systems, or components that could be compromised by an adverse weather event that exceeds what the plant was designed to withstand, such as the maximum flood elevation or maximum temperature limit allowed for a plant to continue operating. 10 C.F.R. Part 50, Appendix A, General Design for Nuclear Power Plants, Criterion 2—Design Bases for Protection Against Natural Phenomena.

Defense-in-Depth at Duane Arnold Energy Center

In 2020, the Iowa "Derecho Windstorm" brought heavy rains and winds up to 130 miles per hour to the Duane Arnold Energy Center. The storm resulted in the loss of offsite power, which caused an emergency shutdown of the reactor. Winds from the storm also damaged two cooling towers and buildings housing the reactor, turbine, and equipment.

However, according to a Nuclear Regulatory Commission (NRC) document, the plant's safety margins and use of a defense-in-depth approach mitigated the effects of storm damage. Specifically, the plant had multiple backup generators and pumps as well as physical barriers to protect the plant.

During the storm, the plant lost offsite power, and the cooling pump for the spent fuel pool turned off. Before the outage, two emergency diesel generators started automatically due to grid-related storm impacts. Staff immediately started a second cooling pump. This action prevented the water in the spent fuel pool from boiling and potentially exposing the fuel rods. According to NRC, the winds also damaged the reactor's containment unit, but it remained functional and would have prevented a release of radiological material in the event of damage to the reactor core.

Duane Arnold Energy Center

Sources: Nuclear Regulatory Commission; AsNuke (photo), https://creativecommons.org/licenses/by-sa/4.0/deed.en. No changes were made to this photo. | GAO-24-106326

- **Licensing**. During the licensing process, NRC assesses a plant's risks from natural hazards as part of its safety evaluation. In doing so, NRC reviews reactor and plant designs and compares the design limits for natural hazards with the site's expected exposure to natural hazards on the basis of the licensee's hazard assessments. According to NRC officials, NRC also conducts a confirmatory analysis of the licensee's hazard assessments, which if deemed insufficient, must be revised by the licensee.
- **Inspections.** NRC resident inspectors use inspection manual procedures to inspect licensees' preparations for addressing adverse weather events and extreme temperatures.[40] As part of their inspections, NRC resident inspectors verify that selected systems and components will function when affected by adverse weather. NRC officials explained that an inspection includes observing licensees repair and run pieces of equipment, conducting emergency drills, and verifying that licensees are taking appropriate actions in response to severe weather conditions. Inspectors from NRC regional offices may also conduct plant inspections after adverse weather events, such as floods or hurricanes.
- **Probabilistic risk assessments.** NRC uses probabilistic risk assessments in its licensing and inspection processes to analyze various risks, including safety risks posed by natural hazards.[41] These assessments are a systematic method for assessing what can go wrong, its likelihood, and its potential consequences to provide insights into the strengths and weaknesses of the design and operation of a nuclear power reactor. Probabilistic risk assessments are used to estimate the risk of reactor core damage, radioactive material release, and related consequences to the public and environment based on the as-built, as-operated plant.
- **Operating experience program.** NRC's operating experience program collects and evaluates information from various regulatory oversight activities and inspection findings and shares information about plants' operating experiences with NRC staff. In addition, according to NRC officials, NRC has a research office that analyzes long-term trends, such as the loss of offsite power due to severe

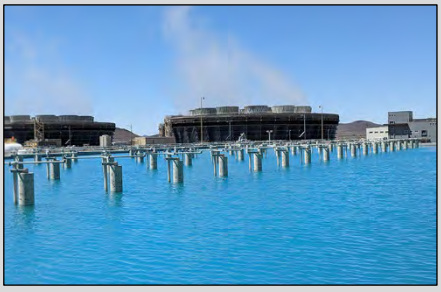
[40]Inspections by resident inspectors at the plant level are called baseline inspections, and different types of baseline inspections occur either daily, quarterly or annually.

[41]Applicants for certain licenses for new reactors must submit a description of the plant-specific probabilistic risk assessment and its results to NRC as a part of their application.

NRC Inspectors Address Heat Risks at the Palo Verde Nuclear Generating Station

To prepare for extreme summer heat, Nuclear Regulatory Commission (NRC) resident inspectors at the Palo Verde Nuclear Generating Station in Arizona inspect systems likely to be affected by high temperatures, such as diesel generators and spray ponds, both of which are used to cool the reactor. The spray ponds contain a 26-day supply of water to ensure that plants have adequate cooling capacity to safely shut down.

Because high temperatures can cause water held in the spray ponds to evaporate, the plant relies on its reservoirs and deep wells as backup sources of water.



Palo Verde Nuclear Generating Station spray pond

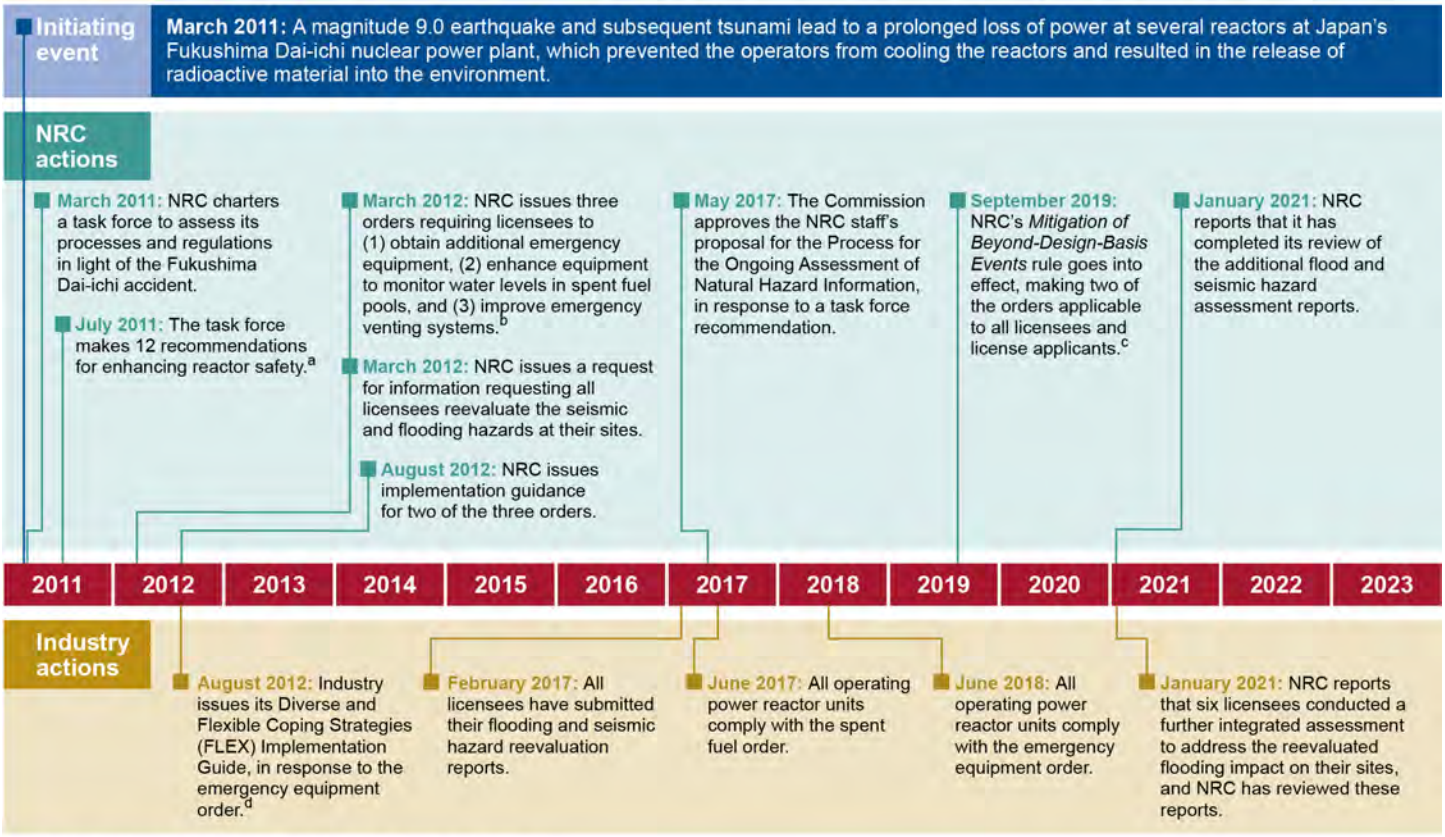
Sources: NRC; GAO (photo). | GAO-24-106326

weather, to identify lessons learned that could be applied to the oversight of other plants.

Following the 2011 accident at Japan's Fukushima Dai-ichi nuclear power plant, NRC and industry took several actions to further address risks to nuclear power plants from natural hazards.[42] Some of these actions were taken in response to recommendations from a task force NRC established to assess its regulatory approach (see fig. 9).

[42]On March 11, 2011, a 9.0-magnitude earthquake and subsequent tsunami devastated northeast Japan and led to the most extensive release of radioactive material at a nuclear power plant since the 1986 Chernobyl disaster. The Fukushima Dai-ichi nuclear power plant suffered extensive damage when a 45-foot-high tsunami wave exceeded the plant's seawall and flooded the site, causing a prolonged loss of electrical power at several of its reactors. As a result of the loss of power, plant operators were unable to keep three of the reactors cool, which led to fuel melting, hydrogen explosions, and the release of radioactive material into the environment. The disaster displaced tens of thousands of residents and contaminated the surrounding area. Nuclear-power-generating countries worldwide have since taken actions to prepare for an event like this, which far exceeded the Fukushima Dai-ichi plant's design basis.

Figure 9: Timeline of Selected Nuclear Regulatory Commission (NRC) and Industry Actions after the Fukushima Dai-ichi Accident in 2011



Source: GAO review of NRC and industry documents. | GAO-24-106326

[a]NRC prioritized the recommendations in three tiers: (1) recommendations NRC should implement without unnecessary delay; (2) recommendations that could not be initiated in the near term due, in part, to resource or critical skill set limitations; and (3) recommendations that required further study by NRC to determine if regulatory action was necessary, among other factors.

[b]NRC, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Order EA-12-049 (Washington, D.C.: Mar. 12, 2012); Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation, Order EA-12-051 (Washington, D.C.: Mar. 12, 2012); and Order Modifying Licenses with Regard to Reliable Hardened Containment Vents, Order EA-12-050 (Washington, D.C.: Mar. 12, 2012).

[c]NRC, Mitigation of Beyond-Design-Basis Events, 84 Fed. Reg. 39,684 (Aug. 9, 2019). Orders EA-12-049 and EA-12-051 are applicable to all licensees and construction permit holders. Order EA-12-050 applies to licensees with boiling water reactors that feature certain containments that require proper venting to ensure safety.

[d]Nuclear Energy Institute, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, NEI 12-06 (August 2012) and NRC, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Order EA-12-049 (Washington, D.C.: Mar. 12, 2012).

The actions NRC took in response to the task force recommendations include the following:

- **Required licensees to assess flooding risks.** NRC required all licensees to assess updated flood hazard risk information and reevaluate and upgrade, as necessary, their plants' flood protection of structures, systems, and components. On the basis of these assessments, NRC did not identify the need to require any plant modifications or revise plant safety procedures.[43]
- **Created a process for ongoing hazard assessments.** In May 2017, the Commission approved the Process for the Ongoing Assessment of Natural Hazard Information (POANHI) to determine the need for site-specific assessments, additional research, or regulatory action.[44] POANHI involves collecting and maintaining hazard information in the

[43]NRC required all nuclear power plant licensees to conduct on-site inspections of safety-related systems to verify that plant features that protect against flooding are available, functional, and properly maintained. All licensees conducted flood reevaluations for their plants, and licensees at six plants conducted further integrated assessments, which are requested by NRC if the plant's design for a potential flood is exceeded by the reevaluation's estimates of potential maximum elevation of flood waters. These assessments evaluate the plant response to flooding hazards and the effectiveness of existing systems and procedures to mitigate risks from flooding. In addition, NRC required licensees to identify and address plant-specific vulnerabilities related to flooding and verify the adequacy of monitoring and maintenance for protection features in the interim period until longer term actions were completed to reevaluate design-basis flooding hazards. Also, following the accident at the Fukushima Dai-ichi plant, NRC issued a temporary instruction directing its inspection staff to independently assess the adequacy of actions taken by licensees. NRC also required licensees to assess seismic hazard risks. Seismic hazard risks are not included in the scope of this report.

[44]Preceding the approval of POANHI, NRC conducted a 2013 Probabilistic Flood Hazard Assessment workshop following the accident at the Fukushima Dai-ichi nuclear power plant, in which participants from federal agencies and other organizations shared information about probabilistic assessment of extreme rainfall, flood-induced dam and levee failures, tsunami flooding, river flooding, extreme storm surge, and combined-events flooding. NRC continues to host Probabilistic Flood Hazard workshops nearly annually, and these workshops often share research results with the public. For example, NRC contracted with the Pacific Northwest National Laboratory to publish four national and regional reports on the potential impacts of climate change, which as of 2022 have not yet led to additional NRC guidance for probabilistic flood hazard assessment. These reports are publicly available at https://www.osti.gov/biblio/1259942, https://www.osti.gov/biblio/1593340, https://www.osti.gov/biblio/1524249, and https://www.osti.gov/biblio/1605280.

Natural Hazards Information Digest[45]—a database that supports POANHI—and reviewing and assessing the hazard information to determine whether a hazard has a potentially significant impact on plant safety.[46] To ensure that NRC is aware of new hazard information from a variety of sources for inclusion in this database, NRC regularly interacts with internal and external stakeholders, including other federal agencies, academia, industry, regulators from other countries, and other technical and scientific organizations, according to NRC officials. If a POANHI assessment of new hazard information identifies a potentially significant effect on plant safety, NRC refers the issue to the appropriate regulatory program, at which point the program office determines how to proceed. POANHI leverages and is integrated into other existing processes, such as the operating experience program, for the assessment of new information and the determination of whether a change is needed to a particular plant's licensing basis. According to NRC officials, NRC has not taken any regulatory actions as a result of POANHI.[47]

- **Required enhanced safety and emergency equipment**. In 2012, NRC ordered all licensees and nuclear power plant construction permit holders to ensure that a plant's key safety functions could be

[45]NRC incorporates new hazard information, such as records of site-specific or regional extreme weather events, into its Natural Hazards Information Digest, which NRC began using in 2019. This database is NRC's repository for information on natural hazard-related events at or near nuclear power plants. The database captures documentation provided by licensees in response to site hazard reevaluations and plant inspections as well as historical site-specific events and information about natural hazards that could affect plants. In addition to informing POANHI, the database also supports NRC efforts to (1) respond to emergent events associated with natural hazards by providing relevant information, (2) engage with stakeholders, (3) evaluate natural hazard-related inspection findings to determine their safety significance, (4) implement natural hazards research plans, and (5) update regulatory and staff guidance.

[46]According to NRC policy, the significance assessment determines whether the new information indicates that the hazard could adversely affect the capability of a plant's structures, systems, and components to perform their intended safety functions. To make this determination, NRC staff either conduct a quantitative assessment that compares the new information with risk insights from past hazard analyses to assess the impacts of plant response or conduct a qualitative assessment that considers the likelihood of the event, identifies vulnerabilities and actions to address them, and adheres to defense-in-depth principles, among other factors.

[47]NRC is reviewing new seismic information from a 2018 report to assess updated seismic hazards at the nuclear power plants located in the region addressed by the report. See Pacific Earthquake Engineering Research Center, *Central and Eastern North America Ground-Motion Characterization: NGA-East Final Report*, (Berkeley, CA: December 2018). After reviewing this report, NRC determined that 13 nuclear power plants located in the central and eastern United States needed further assessment. Based on assessments conducted as of March 2024, NRC determined that no regulatory action was needed.

maintained during a natural disaster that exceeds a plant's design basis. In response, the nuclear industry developed and implemented the *Diverse and Flexible Coping Strategies (FLEX) Implementation Guide,* which NRC has endorsed as one method to comply with the 2012 order.[48] FLEX is a strategy that uses controls, procedures, and backup equipment to ensure that the key safety functions related to cooling a reactor's core and spent fuel, as well as containment to prevent accidental releases of radiation, are maintained if a disaster occurs at a plant. According to NRC officials, as part of this strategy, all plants have backup equipment on site. In addition, the nuclear power industry operates two Strategic Alliance for FLEX Emergency Response (SAFER) centers that maintain emergency equipment that can be provided to plants as a backup to plants' primary backup equipment onsite.[49] See figure 10 for examples of FLEX and SAFER equipment.

[48]Nuclear Energy Institute, *Diverse and Flexible Coping Strategies (FLEX) Implementation Guide*, NEI 12-06, August 2012.

[49]The SAFER centers are located in Phoenix, Arizona, and Memphis, Tennessee. The SAFER centers' staff comprises staff from a private company that has contractual agreements to manage and deploy offsite equipment with every nuclear licensee in the United States as part of FLEX. The SAFER centers maintain generic equipment useful for multiple plants, including various types of generators and pumps, and site-specific equipment unique to certain plants. NRC determined there is reasonable assurance that equipment at the SAFER centers can be deployed to any plant in the United States within 24 hours, as specified by licensees' SAFER response plans. To date, no SAFER response plan has been activated. According to our analysis of federal hazard data, SAFER centers are in areas with no exposure to sea level rise or hurricane storm surge, and either low (Memphis) or high (Phoenix) flood hazard. The centers are located in areas projected to see an increase in daily temperature from 3.6 to 4.9 degrees, and the Phoenix SAFER center is in an area with high or very high wildfire risk.

Figure 10: Examples of the Diverse and Flexible Coping Strategies (FLEX) and Strategic Alliance for FLEX Emergency Response (SAFER) Center Equipment

Top: Water purification equipment, located at the National SAFER Response Center in Phoenix, Arizona, is designed to facilitate the use of higher quality water for the reactor coolant system's makeup. Bottom left: Arizona's Palo Verde Generating Station high-pressure reactor coolant system injection pump is used to support the FLEX strategy and receives borated water from an onsite storage tank to inject into the reactor coolant system to maintain reactor coolant system inventory and nuclear reactivity. Bottom right: Florida's Turkey Point Nuclear Generating Station steam generator FLEX well pump is a piece of backup equipment used to refill a storage tank from an artesian well for the purpose of adding water to the steam generator.

Sources: GAO analysis of Nuclear Regulatory Commission documents and interview with licensee staff; GAO (photos). | GAO-24-106326

NRC's Actions Do Not Fully Consider the Potential Effects of Climate Change

NRC's actions to address risks to nuclear power plants from natural hazards in its licensing, license renewal, and inspection processes do not fully consider the potential increased risks from natural hazards that may be exacerbated by climate change.

- **Licensing.** NRC does not use climate projections data to identify and assess risk as part of the safety reviews or probabilistic risk

assessment reviews it conducts during the initial licensing process.[50] Rather, NRC uses historical data to extrapolate the future risks of natural hazards that may occur during the lifetime of a nuclear power plant.[51] Extrapolating historical data into the future assumes that existing climatological trends will continue.[52] According to NRC officials, NRC uses historical data in conjunction with other information to establish a conservative licensing basis, and many of the natural hazards considered during licensing target annual exceedance probabilities such that an event is unlikely to occur during the lifetime of the plant. In such a case, NRC expects the event to occur only once in 10,000 to 10 million years, depending on the hazard. NRC officials we interviewed told us that they review regional climate projections information for some hazards but do not incorporate site-specific climate projections data, which include hazard assessments, design bases, or determining the adequate safety margin for plants. For example, NRC officials said they review the projected average increase in temperature that applies to a multi-state region according to the NCA designation and compare that with the maximum temperature limits for a particular plant in that region. The officials said that they do not use data on the projected temperature increase to inform licensing decisions at the plant site itself.

- **License renewals.** Following an initial 40-year licensing period, NRC does not reevaluate natural hazard risks, including climate-related

[50]The NCA defines a "climate projection" as the simulated response of the climate system to a scenario of future emissions or concentrations of greenhouse gases and aerosols, generally derived using climate models. Projections data could be based on a range of possible future scenarios for particular time frames, such as the projected temperature of a specific location in the year 2050, as identified by models that consider climate systems' physical, chemical, and biological properties and their interactions. According to NRC officials we interviewed, probabilistic risk assessments use current estimates of the probability of external events, and neither licensees' nor NRC's assessments incorporate climate projections data, despite their role in assessing the likelihood of future events. NRC officials said that while it is both technical and feasible to update these models with the latest information reflecting their current state of knowledge, using climate projections data would increase uncertainty in the results of the probabilistic risk assessments, and no historical trends have emerged to suggest the need to adjust these.

[51]For example, NRC regulations for evaluating sites for initial licensing require NRC to consider the seismology, meteorology, geology, and hydrology of the site and to estimate the "maximum probable flood" using historical data, among other factors. 10 C.F.R. § 100.20(c). These regulations do not preclude NRC from using climate projections data.

[52]As noted previously, according to the NCA, climate change is altering the characteristics of many extreme weather events. Specifically, some of these events have already become more frequent, intense, widespread, or of longer duration, and many are expected to continue to worsen.

risks, to update the safety reviews required for the license renewal process. NRC's license renewal process focuses on evaluating and managing the effects of aging on the extended operations of nuclear power plants and considers the original licensing basis in that context.[53] As of January 2024, NRC had issued license renewals to 49 of the 54 operating nuclear power plants, meaning most plants are operating on the basis of assessments of natural hazard risk that are over 40 years old.

- **Inspections.** During regular inspections, NRC resident inspectors are responsible for focusing on the immediate day-to-day safety of plants rather than on the potential long-term safety risks. Inspections do not include an assessment of future climate projections data. In addition, while NRC sometimes conducts additional inspections using outside teams—including staff from NRC regional offices—to address recent events or emerging issues related to safety, these inspections also do not focus on long-term safety risks.

NRC officials we interviewed told us that while their regulatory processes—including licensing, license renewals, and inspections—do not use climate projections data to assess climate risks, they believe conservatism, safety margins, and defense-in-depth provide an adequate margin of safety to address climate risks to the safety of nuclear power plants.[54] However, NRC has not conducted an assessment to demonstrate that this is the case.

Moreover, NRC actions taken to address risks to nuclear power plants from natural hazards post-Fukushima did not fully consider the effects of climate change. Specifically, NRC required licensees to assess flooding risk and enhance safety and emergency equipment, but NRC did not require licensees to use climate projections data to assess future flooding

[53]Licensees are not required to reevaluate their plant's design basis pertaining to natural hazards as part of the license renewal process.

[54]According to NRC officials, NRC uses the NCA, which includes climate projections, in the environmental reviews it conducts during licensing and license renewals to assess the expected effects of nuclear power plants on the environment. For example, NRC addresses the greenhouse gas emissions associated with the life cycle of the plant as well as the potential effects of climate change on the environment in these reviews.

risks as part of these assessments or in the FLEX equipment needs assessments.[55]

NRC also created POANHI—its process for ongoing hazard assessments following Fukushima—which, according to NRC officials, NRC relies on to identify and assess changes in natural hazard risks, including those driven by climate change. However, POANHI has several limitations as a mechanism for comprehensively identifying and assessing climate risks. Specifically:

- While POANHI was designed to assess all natural hazards, NRC has not used POANHI to assess potential changes to all natural hazards, nor has NRC comprehensively reviewed natural hazards on a regular basis to determine whether available information indicates the need for a POANHI assessment. NRC officials told us that while POANHI requires continuous evaluation of new information on natural hazards, NRC conducts POANHI assessments for one hazard at a time, and the agency does not have a schedule for reviewing natural hazards beyond the assessment of seismic hazards currently underway. As such, POANHI is used to react to new hazard information or events when NRC staff become aware of them.
- NRC has not documented the new hazard information it reviews as part of POANHI or the way it incorporates climate projections data to determine whether to initiate a POANHI assessment, require additional plant-specific assessments, conduct an overall hazard reevaluation, or take regulatory action.
- NRC has not implemented POANHI and the Natural Hazards Information Digest at all levels of the agency. For example, several regional branch chiefs and resident inspectors we interviewed were unaware of POANHI and this information database. An official from one NRC regional office said that if the database were shared more broadly, it would benefit resident inspectors, who could access and use information on weather-related events and inspection findings to inform probabilistic risk assessments. According to NRC officials, the

[55]According to NRC officials, the plant-specific mitigation strategy relied on information each licensee had previously been required to provide as part of reevaluations of external events for comparison against the current licensing bases and FLEX equipment reflect the most severe external events that could occur based on known available meteorological, geological, and geographical data. According to NRC officials, the external hazards needing to be considered were both extreme and rare in nature which resulted in the regulatory approach of using flexible, diverse strategies to maintain or restore core cooling, containment, and spent fuel pool cooling.

agency is conducting internal outreach to increase NRC staff's knowledge of POANHI.

NRC's Fiscal Year 2022–2026 Strategic Plan calls for ensuring that licensees have measures to address the potential for increased risks from climate change. The strategic plan also promotes risk-informed decision-making to support NRC's strategic objective of providing quality licensing and oversight of nuclear facilities. Moreover, *GAO's Standards for Internal Control in the Federal Government* state that management should identify, analyze, and respond to risks related to achieving defined objectives.[56] These standards also call for agency management to use quality information to achieve their objectives.

Assessing its current processes would help NRC to determine whether they adequately address the potential for increased risks to nuclear power plants from climate change. Specifically, identifying gaps in its processes and developing a plan to address them, including by using climate projections data or augmenting POANHI, would help ensure that NRC adopts a comprehensive approach for assessing risks and fulfills its mission to protect public health and safety.

NRC officials told us that they use historical data in licensing and oversight processes rather than climate projections data, in part because regulations require NRC to use available historical data to assess the safety of the reactor site and design and they believe these data are reliable and sufficient for developing an adequate margin of safety for plants.[57] According to NRC officials, using site-specific climate projections

[56]GAO-14-704G. Risk assessment is the identification and analysis of risks related to achieving defined objectives to form a basis for developing responses to these risks. Our prior work has shown that assessing risks includes assessing both the likelihood of an event occurring and the effect the event would have. Agency leaders and subject matter experts should assess each risk by assigning the likelihood of the event's occurrence and the potential effect if the event occurs. GAO, *Enterprise Risk Management: Selected Agencies' Experiences Illustrate Good Practices in Managing Risk*, GAO-17-63 (Washington, D.C.: Dec. 1, 2016).

[57]See, e.g., 10 C.F.R. § 100.20(c) (requiring NRC to consider for initial licensing of new reactors the seismology, meteorology, geology, and hydrology of the site and to estimate the "maximum probable flood" using historical data). See also 10 C.F.R. § 60.2 (defining "design bases" to include using severe natural events estimates based on historical and physical data); and 10 C.F.R. Part 50, Appendix A, General Design for Nuclear Power Plants, Criterion 2—*Design Bases for Protection Against Natural Phenomena* (requiring the design bases for the reactor's safety structures, systems, and components to consider the "most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated," among other factors).

data for extreme hazard levels in nuclear power plant design and safety reviews is challenging because of the uncertainty associated with applying these data to specific sites. However, NRC regulations do not preclude NRC from using climate projections data, and new sources of reliable projected climate data are available to NRC. In 2023, the White House Office of Science and Technology Policy issued guidance to federal agencies on selecting and using climate data to assess risks and their potential impacts.[58] This guide provides information on climate models and projections to help federal agencies understand exposure to current and future climate-related hazards and their potential impacts. Without incorporating the best available information into its licensing and oversight processes, it is unclear whether the safety margins for nuclear power plants established during the licensing period—in most cases over 40 years ago—are adequate to address the risks that climate change poses to plants.

Conclusions

Commercial nuclear power plants in the United States were licensed and built an average of 42 years ago, and weather patterns and climate-related risks to their safety and operations have changed since their construction. Climate change is expected to exacerbate natural hazards—such as heat, drought, wildfires, flooding, hurricanes, sea level rise, and extreme cold weather events—that can affect nuclear power plant safety and operations in various ways. Some of these effects are already occurring, and many are expected to continue to worsen.

However, NRC does not use climate projections data to identify and assess risk as part of the safety reviews it conducts or the probabilistic risk assessments it reviews during the initial licensing process. NRC has also not fully developed POANHI, which the agency relies on to identify and assess changes in natural hazard risks, including climate change.

NRC has the opportunity to consider climate risks more fully and, in doing so, to better fulfill its mission to protect public health and safety.

[58]Office of Science and Technology Policy, *Selecting Climate Information to Use in Climate Risk and Impact Assessments: Guide for Federal Agency Climate Adaptation Planners*, (Washington, D.C.: March 2023). Although climate projections data and guidance are available to federal agencies, we previously recommended that the federal government, through the Executive Office of the President, make authoritative climate data and information accessible and assist in translating that information for decision makers. GAO, *High-Risk Series: Efforts Made to Achieve Progress Need to Be Maintained and Expanded to Fully Address All Areas,* GAO-23-106203 (Washington, D.C.: April 20, 2023) and *Climate Information: A National System Could Help Federal, State, Local, and Private Sector Decision Makers Use Climate Information,* GAO-16-37 (Washington, D.C.: Nov. 23, 2015).

Specifically, assessing whether its licensing and oversight processes adequately consider climate risks to nuclear power plants and developing and implementing a plan to address any gaps identified would help the agency do so. As NRC makes licensing, license renewal, and oversight decisions, adopting an approach that incorporates the best available information on climate risks and ways that those risks may affect nuclear plants, would provide greater assurance that licensees have adequate measures to address risks from climate change.

Recommendations for Executive Action

We are making the following three recommendations to NRC:

The Chair of the NRC should direct NRC staff to assess whether its licensing and oversight processes adequately address the potential for increased risks to nuclear power plants from climate change. (Recommendation 1)

The Chair of the NRC should direct NRC staff to develop, finalize, and implement a plan to address any gaps identified in its assessment of existing processes. (Recommendation 2)

The Chair of the NRC should direct NRC staff to develop and finalize guidance on incorporating climate projections data into relevant processes, including what sources of climate projections data to use and when and how to use climate projections data. (Recommendation 3)

Agency Comments and Our Evaluation

We provided a draft of this report for review and comment to NRC.

In its written comments, reproduced in appendix IV, NRC stated that the three recommendations are consistent with actions that are either underway or under development. In addition, NRC stated that the layers of conservatism and defense-in-depth incorporated into NRC's processes provide reasonable assurance regarding any plausible natural hazard and combinations at a site for the licensed operational lifetime of the reactor, including those that could result from climate change. As we noted in our report, NRC has not conducted an assessment to demonstrate that the safety margins for nuclear power plants established during the licensing period are adequate to address the risks that climate change poses to plants. According to the NCA, many of the climate conditions and impacts experienced in the United States today are unprecedented for thousands of years. Across all regions of the United States, extremes, including heat, drought, flooding, wildfire, and hurricanes, are becoming more frequent and/or severe, with a cascade of effects in every part of the country. We continue to believe that NRC cannot fully consider potential

climate change effects on plants without using the best available information—including climate projections data—in its licensing and oversight processes.

NRC also provided technical comments, which we incorporated, as appropriate.

We are sending copies of this report to the appropriate congressional committees, the Chair of the NRC, and other interested parties. In addition, the report is available at no charge on the GAO website at https://www.gao.gov.

If you or your staff have any questions about this report, please contact me at (202) 512-3841 or ruscof@gao.gov. Contact points for our Offices of Congressional Relations and Public Affairs may be found on the last page of this report. GAO staff who made key contributions to this report are listed in appendix V.

Frank Rusco
Director, Natural Resources and Environment

Appendix I: Objectives, Scope, and Methodology

This report examines (1) how climate change is expected to affect nuclear power plants and (2) actions the Nuclear Regulatory Commission (NRC) has taken to address the risks to nuclear power plants from climate change.

To address both objectives, we interviewed officials from NRC headquarters, all four NRC regional offices, and two nuclear power plants. We also interviewed officials from the Department of Energy—including the Office of Nuclear Energy and the Idaho National Laboratory—the Department of Homeland Security, the Federal Energy Regulatory Commission, and the National Oceanic and Atmospheric Administration (NOAA). We also interviewed nine stakeholders knowledgeable about nuclear power plant safety and operations, climate change, and resilience measures. These included stakeholders from three industry groups, four nongovernmental organizations, and two academic institutions. We identified stakeholders using snowball sampling.[1] Views from selected stakeholders cannot be generalized to all stakeholders.

We conducted site visits to two nuclear power plants—Palo Verde Nuclear Generating Station in Buckeye, Arizona, and Turkey Point Nuclear Generating Station in Homestead, Florida. We toured the power plants and interviewed plant staff and NRC resident inspectors at each site. To answer both objectives, we chose these sites for in-person visits based on factors including exposure to distinct natural hazards, regional diversity, reactor type, licensee size, and agency resources. Findings from selected site visits are not generalizable to all sites.

To address how climate change is expected to affect nuclear power plants, we reviewed prior GAO reports and sources of climate change information (including the fourth and fifth National Climate Assessments (NCA)), completed a literature review, and conducted data analysis.[2] To conduct the literature review of articles and reports related to the effects of selected hazards and climate change on nuclear power plants, we searched a variety of scholarly, trade, and news databases, such as Ei

[1]In snowball sampling, the methodology begins with an initial list of contacts and asks each person interviewed to refer the interviewer to additional cognizant persons. The group of referred contacts (or “snowball”) grows larger and then narrows as a group of individuals are identified frequently.

[2]Few supporting sources distinguish between the impact of selected natural hazards on operating versus shutdown nuclear power plants. As a result, we most often do not make this distinction.

Encompass LIT, Geobase, Inspec, the National Technical Information Service, ProQuest Environmental Science Professional, and Scopus using relevant keywords (e.g., “nuclear power,” “climate change,” “risk,” and “extreme weather”) for articles and other documents published since 2012. The results yielded 107 potentially relevant articles and other documents published from January 2012 through January 2023. To determine which articles were relevant to our scope, one analyst reviewed the articles’ abstracts and determined whether the articles were in scope using professional judgment based on their knowledge of the engagement’s scope. A second analyst reviewed the first analyst’s determinations, and the two came to a consensus on which articles were in scope. Using this method, we selected 56 articles and other documents for further review. Reviewing them for relevance, we ultimately identified and used 36 articles to support findings in our report.

To conduct our data analysis, we identified national-level data sets from relevant federal agencies for six of the seven natural hazards identified by the NCA, and our review of literature, as likely to be exacerbated by climate change in the United States. The six hazards are heat, cold, wildfires, flooding, storm surge from hurricanes, and sea level rise.[3] For heat and cold, we used climate projections data that incorporate emission scenarios to project future exposure to those hazards.[4] For wildfires, flooding, and hurricane storm surge, we used climate data that show current conditions based on past conditions (which do not incorporate climate projections).[5] For sea level rise, we used data for coastal regions and sea level rise projections from an interagency report covering sea level rise scenarios to identify coastal nuclear power plants and projected

[3]To identify the best available federal-level hazard data, we relied on interviews with agency officials and prior GAO reports. We did not analyze drought data because we were unable to identify national-level geospatial data that was both relevant to nuclear power plants and sufficiently reliable for our purposes.

[4]To analyze projected exposure to heat and cold hazards, we used NCA data on the projected exposure to maximum and minimum temperatures in the midcentury (i.e., 2036–2065), using both a low and high emission scenario for projected climate change.

[5]To analyze flood exposure, we used 2023 data from the Federal Emergency Management Agency that categorizes flood exposure into high, moderate, minimal or other, and unknown flood hazard categories. To analyze exposure to hurricane storm surge, we used NOAA data on storm surge exposure from Categories 1, 4, and 5 hurricanes. To analyze exposure to wildfires, we used 2023 data from the U.S. Forest Service on wildfire hazard potential.

sea level rise in their respective regions.[6] In this report, we refer to these hazards collectively as selected natural hazards that may be exacerbated by climate change.

For our national-level data from federal agencies, we used data we determined to be the most appropriate to represent selected natural hazards.[7] Data sources for each of the hazards we analyzed are, as follows:

- **Heat and cold.** To analyze projected exposure to heat and cold, we used data from the fourth NCA on the projected exposure to maximum temperatures in the midcentury (i.e., 2036–2065).[8]
- **Wildfire.** To analyze exposure to wildfire hazard potential, we used 2023 data from the U.S. Forest Service's Wildfire Hazard Potential Map. For reporting purposes, we grouped wildfire hazard potential into the following three categories: no/low, moderate, and high/very high.[9]
- **Flooding.** To analyze exposure to flood hazards, we used 2023 data from the Federal Emergency Management Agency's National Flood Hazard Layer. For reporting purposes, we grouped flood hazard

[6]W. V. Sweet, B. D. Hamlington, R. E. Kopp, C. P. Weaver, P. L. Barnard, D. Bekaert, W. Brooks, M. Craghan, G. Dusek, T. Frederikse, G. Garner, A. S. Genz, J. P. Krasting, E. Larour, D. Marcy, J. J. Marra, J. Obeysekera, M. Osler, M. Pendleton, D. Roman, L. Schmied, W. Veatch, K. D. White, and C. Zuzak, *Global and Regional Sea Level Rise Scenarios for the United States: Updated Mean Projections and Extreme Water Level Probabilities Along U.S. Coastlines*, NOAA Technical Report NOS 01 (Silver Spring, MD: February 2022).

[7]Data sources were chosen based on use in prior GAO reports, review of the NCA, and interviews with federal agencies responsible for collecting and reporting on data related to the selected natural hazards.

[8]The fifth NCA was released on November 14, 2023, after we had obtained and analyzed the hazard data sets from the fourth NCA. We reviewed relevant sections from the fifth NCA and did not identify major differences in the predicted or projected trends for the selected natural hazards.

[9]We combined layers of "high" and "very high" wildfire hazard potentials, which correspond to areas at the 85th percentile or greater for wildfire hazard potential. The no/low category includes plants that are in areas that are not covered by the "moderate," "high," or "very high" wildfire potential layers.

zones into the following three categories: no/low, moderate, and high.[10]

- **Hurricane storm surge.** To analyze exposure to various levels of hurricane storm surge, we used data from NOAA's Sea, Lake, and Overland Surges from Hurricanes model. We used a range of categories from the data, including no exposure to hurricanes, and Categories 1, 4, and 5 hurricanes.[11]
- **Sea level rise.** To analyze potential exposure to sea level rise in 2050, we used data from an interagency report covering sea level rise scenarios to illustrate regional climate projections for sea level rise in coastal regions. The data include two types of estimates–observation-based extrapolations and regionalized global mean sea level scenarios. NOAA officials recommended using these projections for our analysis of sea level rise data.

To identify nuclear power plant locations, we used nuclear power plant location data from NRC.[12] We used a 0.5-mile radius around the plant coordinates provided by NRC as a proxy for approximate plant size. We based the size of the radius on approximations we made for an average U.S. nuclear power plant.[13]

See appendix II for further discussion of these data sources.

[10]No/low corresponds to areas with minimal, unknown, or other flood hazards, including areas with reduced risk because of levees as well as areas with flood hazard based on future conditions, such as the future implementation of land-use plans. Moderate flood hazard zones correspond to a 500-year floodplain, which indicates between 0.2 percent and 1 percent annual chance of flooding. High flood hazard zones correspond to a 100-year floodplain, which indicates a 1 percent or higher annual chance of flooding.

[11]In our analysis, we used data on storm surge from Category 1 hurricanes (the lowest possible category) and for Categories 4 and 5 hurricanes (the highest possible categories) to show a range of climate change effects.

[12]In March of 2023, we obtained NRC nuclear power plant coordinates for all 54 operating nuclear power plants. In July 2023, we obtained NRC nuclear power plant coordinates for the 21 nuclear power plants that have shut down and have spent nuclear fuel stored onsite in spent fuel pools or in dry cask storage. NRC provided coordinates, including a latitude and longitude value for each plant. In addition, NRC's location data file contained other identifying plant information including operating status, license number, and reactor type.

[13]We requested average plant size from NRC, but NRC was unable to provide these data. Instead, we approximated the size of a typical nuclear power plant using DOE documentation on nuclear power plants.

Using hazard and nuclear power plant location data, we analyzed natural hazard exposure in the areas around nuclear power plants. In our analysis, we included operating plants and plants at various stages of decommissioning, including those in the process of decommissioning and those already shut down, with spent nuclear fuel stored onsite. We did not include experimental or test reactors in our analysis.

For certain hazards, we analyzed exposure to a range of intensities. For example, we analyzed nuclear power plant exposure to storm surge from the weakest (Category 1) and strongest (Category 5) hurricanes, as modeled by NOAA.

To analyze whether nuclear plants are located in areas that may be affected by heat, cold, wildfire, flooding, and hurricane storm surge, we used MapInfo mapping software to determine whether the nuclear power plant locations were located in areas with exposure to the selected hazards. Exposure indicates that a facility is located in an area that may be affected by a selected hazard. If the plant overlapped with multiple hazard layers, the layer representing the highest level of exposure was reported. For example, in our report, we coded a plant whose locations showed exposure to both layers for Category 1 and Category 5 storm surge data as having exposure to Category 5 storm surge.

We assessed the reliability of the fourth NCA climate projections data we used to analyze heat and cold exposure by (1) interviewing NOAA officials knowledgeable about the data and (2) reviewing existing information about the data and system that produced them.

To assess the reliability of the Federal Emergency Management Agency's National Flood Hazard Layer, NOAA's data on Sea, Lake, and Overland Surges from Hurricanes, and the U.S. Forest Service's Wildfire Hazard Potential data, we reviewed prior GAO data reliability assessments for reports using the same data.[14] Then, through interviews and email correspondence with NOAA, the Federal Emergency Management Agency, and U.S. Forest Service officials, we ensured that these data remained appropriate and reliable, considering any subsequent updates or changes made to the data.

[14]GAO, *Chemical Accident Prevention: EPA Should Ensure Regulated Facilities Consider Risks from Climate Change*, GAO-22-104494 (Washington, D.C.: Feb. 28, 2022) and GAO, *Superfund: EPA Should Take Additional Actions to Manage Risks from Climate Change,* GAO-20-73 (Washington, D.C.: Oct. 18, 2019).

To assess the reliability and appropriate use of sea level rise data for use in our analysis, we reviewed regional sea level rise data in an interagency report covering sea level rise scenarios and interviewed NOAA officials knowledgeable about sea level rise data.

To assess the reliability of NRC's data on nuclear power plant locations, we communicated with NRC staff about data accuracy and conducted limited data testing.[15] As a result of the steps described above, we found the data from the NCA, the Federal Emergency Management Agency, NOAA, the U.S. Forest Service, and NRC to be sufficiently reliable for the purpose of our reporting objectives.

To examine NRC's actions to address risks to nuclear power plants from climate change, we conducted interviews and reviewed relevant agency documents. We interviewed officials from NRC headquarters, all four NRC regional offices, and two nuclear power plants. During our two nuclear power plant site visits, we interviewed plant operator staff as well as NRC's resident inspectors to assess whether NRC processes to mitigate the risks of natural hazards and extreme weather at those plants adequately consider climate change risks. We also observed an NRC safety evaluation review to understand the extent to which NRC incorporates considerations of climate change risks when determining whether and under what conditions to license a nuclear power plant.[16] We reviewed relevant documents consisting of the following: relevant laws and regulations, agency documents (including guidance on probabilistic risk assessments and NRC's 2022–2026 Strategic Plan), two NRC office instructions, NRC's inspection manual on adverse weather protection, and other documents.[17] We compared NRC's actions against requirements to identify any relevant gaps. We also reviewed GAO's

[15]Specifically, we inputted a selection of NRC's location data into mapping software to ensure NRC's latitude and longitude location data for nuclear power plants correctly corresponded to plant names and identifying information provided by NRC. Also, we compared the plant operating status of selected plants in NRC's dataset with public information to ensure the operating status of plants matched.

[16]This safety evaluation review was for Turkey Point's Units 6 and 7, which were granted an operating license under 10 C.F.R. Part 52 but have not been built, according to NRC officials.

[17]NRC, *Strategic Plan Fiscal Years 2022–2026,* NUREG-1614, Vol. 8 (Washington, D.C.: April 2022). See also, NRC, *Inspection Manual: Adverse Weather Protection,* Inspection Procedure 71111, Attachment 01 (Washington, D.C.: Jan. 1, 2018).

Standards for Internal Control in the Federal Government and compared NRC's actions against those standards.[18]

We conducted this performance audit from November 2022 to April 2024 in accordance with generally accepted government auditing standards. Those standards require that we plan and perform the audit to obtain sufficient, appropriate evidence to provide a reasonable basis for our findings and conclusions based on our audit objectives. We believe the evidence obtained provides a reasonable basis for our findings and conclusions based on our audit objectives.

[18]GAO, *Standards for Internal Control in the Federal Government*, GAO-14-704G (Washington, D.C.: Sept. 10, 2014).

Appendix II: Available Federal Data on Heat, Cold, Wildfires, Flooding, Storm Surge, and Sea Level Rise

This appendix provides information on data sources we used to analyze potential exposure of nuclear power plants to selected natural hazards—including heat, cold, wildfires, flooding, storm surge from hurricanes, and sea level rise. We include information, when available, on the data source name, description, purpose, update frequency, and limitations.

National Climate Assessment Heat and Cold Climate Projections Data

The U.S. Global Change Research Program posts climate projections data on its website so that authors and other users can access their data.[1] The variables we used for heat and cold are part of a suite of variables intended to provide users insights into the effects of climate change on different variables under multiple emission scenarios.[2]

We analyzed and reported on the following heat or cold variables from the fourth National Climate Assessment (NCA):[3]

- projected change in maximum daily temperature;

[1]NOAA's Technical Support Unit and the Scripps Institute of Oceanography were involved in creating these data. Together, these stakeholders contributed to creating 100 variables derived through statistical downscaling—a process used to take climate model data, which are typically at a low resolution, and produce more detailed data relevant to a specific location or region. Climate projections have limitations that include uncertainties in emissions, natural variability, and differences in scientific models. "Emission uncertainty" refers to a climate projection's reliance on emission scenarios reliant on assumptions about future emissions, changes in population, energy use, and technology. "Natural variability" refers to unpredictable climate events like volcanic eruptions. Scientific models refer to the way processes are understood and incorporated. For example, any change in the scientific understanding of cloud properties and ocean circulation can affect projections of future climate. In this report, we refer to these data as NCA data.

[2]Climate projections are based on emissions scenarios. These scenarios are produced using a range of future assumptions about underlying socioeconomic conditions, such as population and global gross domestic product projections. The climate projections data from the fourth NCA enable users to analyze projected exposure to temperature, precipitation, and other related variables by using a range of emission scenarios and time periods. Four scenarios are available, including historical climate (averages based on the 1976–2005 climate), lower (averages based on NCA assumptions for intermediate-low sea level rise, lower population, and lower development land use), higher (averages based on intermediate sea level rise, higher population, and higher development land use), and the upper bound (averages based on extreme sea level rise, higher population, and higher development land use). All four scenarios base their future projections on historical climate data for 1976–2005. These scenarios are available for three time periods, which include the early 21st Century (2016–2045), mid-21st Century (2036–2065), and late 21st Century (2070–2099).

[3]The fifth NCA became available in November 2023, after we had obtained and analyzed heat and cold data from the fourth NCA.

- projected change in the annual days with a maximum temperature greater than the 99th percentile;
- projected change in the annual number of days with a maximum temperature greater than 115 degrees Fahrenheit;
- projected change in annual highest maximum temperature averaged over a 5-day period; and
- projected change in the annual number of days with a maximum temperature lower than the 1st percentile.[4]

U.S. Forest Service Wildfire Hazard Potential Data

The U.S. Forest Service maps wildfire hazard potential based on landscape conditions and other observations. We previously reported that the primary intended use of the wildfire hazard potential map is to identify priority areas for hazardous fuels treatments from a broad, national- to regional-scale perspective. The data do not explicitly show wildfire threat or risk.[5]

The U.S. Forest Service maps an index of wildfire hazard potential for the contiguous United States based on, among other factors, annual burn probabilities and potential intensity of large fires. The U.S. Forest Service categorizes the wildfire hazard potential index into five classes: very low, low, moderate, high, and very high. The U.S. Forest Service designates as “high” those areas with wildfire hazard potential index from the 85th to the 95th percentiles, and as “very high” those areas above the 95th percentile. The U.S. Forest Service also categorizes some areas as non-burnable (including agricultural lands, developed lands, and water).

As we previously reported, according to the U.S. Forest Service, areas with higher levels of wildfire hazard potential have fuels that are more likely to burn with high intensity under certain weather conditions. However, areas with moderate, low, and very low wildfire hazard potential

[4]We selected temperature data using the projected change by the midcentury time frame under both a low and high emission scenario to show the range of potential projected change to selected natural hazards. The midcentury time frame was selected because it captures potential hazard effects during the period in which nuclear power plants are likely to remain operational. Other available variables include the average daily temperature and maximum 1- or 5-day precipitation. In this report, we refer to these data as NCA data.

[5]The objective of the wildfire hazard potential map is to depict the relative potential for wildfire that would be difficult for suppression resources to contain. The U.S. Forest Service’s Wildfire Hazard Potential map is available at https://doi.org/10.2737/RDS-2015-0047-4.

may still experience wildfires, particularly near areas with higher wildfire hazard potential.

We used 2023 wildfire hazard potential data. These data incorporated methodological changes to the fire simulation modeling to better represent probabilistic components of wildfire hazard for the fuel and climate conditions as they exist today, according to U.S. Forest Service officials we interviewed. For our analysis, we combined the high and very high wildfire hazard potential categories; we did not identify the number of facilities in each of these categories separately.

Federal Emergency Management Agency Flood Hazard Data

The Federal Emergency Management Agency's National Flood Hazard Layer provides data on the most current coastal and riverine flooding hazard data.[6] Among other uses, the flood hazard data are used for flood insurance ratings and floodplain management. The National Flood Hazard Layer identifies areas with the highest risk of flooding, with a 1 percent or higher annual chance of flooding.[7] In some locations, the National Flood Hazard Layer also identifies areas with a 0.2 percent or higher annual chance of flooding, which the Federal Emergency Management Agency considers moderate flood hazards, and other flood hazards.[8] The National Flood Hazard Layer also identifies areas with minimal flood hazards, including those with less than 0.2 percent annual chance of flooding, and unknown flood hazards, including areas the Federal Emergency Management Agency has not assessed for flood hazards.

[6]Riverine flooding is flooding related to or caused by a river, stream, or tributary overflowing its banks because of excessive rainfall, snowmelt, or ice. The Federal Emergency Management Agency provides a tool for viewing, downloading, and printing flood maps for specific locations. The Federal Emergency Management Agency's flood hazard maps are available at https://www.fema.gov/flood-maps/national-flood-hazard-layer. Federal law requires the Federal Emergency Management Agency to assess the need to revise and update the nation's flood maps once every 5 years or more often as the Administrator determines necessary. 42 U.S.C. § 4101(e).

[7]These areas are known as Special Flood Hazard Areas. Under federal law, in communities that participate in the National Flood Insurance Program, homeowners are required to purchase flood insurance for properties located in Special Flood Hazard Areas that are secured by mortgages from federally regulated lenders. 42 U.S.C. § 4012a(b)(1).

[8]Other flood hazards include areas with reduced risk because of levees, as well as areas with flood hazard based on future conditions, for example, if land use plans were implemented.

National Oceanic and Atmospheric Administration Storm Surge Hazard Data

The National Oceanic and Atmospheric Administration (NOAA) provides estimates of hurricane storm surge using a model called Sea, Lake, and Overland Surges from Hurricanes.[9] Estimates for storm surge are available for coastal areas in the eastern United States from Texas to Maine as well as in Hawaii, Puerto Rico, and the U.S. Virgin Islands. As of November 2023, storm surge data for coastal areas in the western United States were only available for Southern California.

The model accounts for specific shorelines by incorporating bay and river configurations, water depths, bridges, roads, levees, and other physical features. It estimates the maximum extent of storm surge at high tide by modeling hypothetical hurricanes under different storm conditions, such as landfall location, storm trajectory, and forward speed.

NOAA models storm surge for Category 1 through Category 5 hurricanes for the Atlantic coast south of the North Carolina-Virginia border, the Gulf of Mexico, Puerto Rico, and the U.S. Virgin Islands; and Category 1 through Category 4 hurricanes for the Atlantic coast north of the North Carolina-Virginia border and Hawaii.[10] As we previously reported, the model is to be used for educational purposes and to increase awareness of storm surge hazards at a city or community level. According to NOAA's website, the agency updates the model for portions of the shoreline each year to account for, among other changes, new data and the addition of flood protection devices, such as levees. The model does not account for future conditions such as erosion, subsidence (i.e., the sinking of an area of land), construction, or sea level rise.

2022 Interagency Sea Level Rise Technical Report Sea Level Rise Data

The 2022 Interagency Sea Level Rise Technical Report provides observation-based extrapolations and model-based global mean sea level scenarios as two distinct estimates of future sea level rise. Observation-based extrapolations use observed changes in sea level rise to estimate

[9]According to NOAA, "storm surge" is an abnormal rise of water generated by a storm, over and above the predicted tides. Storm surge is produced by water being pushed toward the shore by the force of the storm's winds. NOAA's storm surge hazard maps are available at https://www.nhc.noaa.gov/nationalsurge/.

[10]We previously reported that NOAA does not estimate storm surge for Category 5 hurricanes in areas where such hurricanes have not historically made landfall, such as areas north of the North Carolina-Virginia border.

the trajectory of sea level rise.[11] Model-based-global mean sea level scenarios use emission scenarios to estimate future sea level rise. The 2022 Interagency Sea Level Rise Technical Report provides both types of estimates for sea level rise in 2050 (relative to a baseline of the year 2000) for eight coastal regions of the United States. Formed by analyzing aggregated tide gauge data, the regional boundary data that NOAA provided our team include the Northeast (Maine to Virginia), the Southeast (North Carolina to the east coast of Florida), the Eastern Gulf (west coast of Florida to Mississippi), the Western Gulf (Louisiana to Texas), the Southwest (California), the Northwest (Oregon to Washington), the Hawaiian Islands, and the Caribbean.

The 2022 Interagency Sea Level Rise Technical Report providing the sea level rise estimates and coastal regions is intended to help inform federal agencies, Tribes, state and local governments, and stakeholders in coastal communities about current and future sea level rise.[12] The two primary limitations that the report discusses for the sea level rise estimates we use include process uncertainty and emission uncertainty. Process uncertainty refers to uncertainty about the impact of emissions on ice sheet loss, ocean expansion, and local ocean dynamics. Emission uncertainty refers to the uncertain amount of greenhouse gas emissions that will enter the atmosphere, trap heat, and affect temperature and sea level rise.

[11]The observation-based extrapolations are intended to serve as a comparison with the model-based-global mean sea level scenarios.

[12]W. V. Sweet, B. D. Hamlington, R. E. Kopp, C. P. Weaver, P. L. Barnard, D. Bekaert, W. Brooks, M. Craghan, G. Dusek, T. Frederikse, G. Garner, A. S. Genz, J. P. Krasting, E. Larour, D. Marcy, J. J. Marra, J. Obeysekera, M. Osler, M. Pendleton, D. Roman, L. Schmied, W. Veatch, K. D. White, and C. Zuzak, *2022: Global and Regional Sea Level Rise Scenarios for the United States: Updated Mean Projections and Extreme Water Level Probabilities Along U.S. Coastlines*, NOAA Technical Report NOS 01 (Silver Spring, MD: February 2022).

Appendix III: Nuclear Power Plant Exposure to Selected Natural Hazards

Table 1 shows the exposure of areas around operating nuclear power plant locations to six current or projected natural hazards: flooding, hurricane storm surge, wildfire, sea level rise, heat, and cold. Data for flooding, hurricane storm surge, and wildfire are current and based on historical observation data. Data for sea level rise and heat and cold temperature variables are climate projections data, which incorporate emission scenarios. For more information about the data sources used, see appendix II.

Table 1: Potential Exposure to Current and Future Hazards at Operating Nuclear Power Plants

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios, (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios (days/ year)[j]
Browns Ferry	AL	High	No exposure	None/ low	N/A	3.74, 4.61°F	21.07, 29.35 days	0.01, 0.05 days	5.05, 6.32°F	-1.86, -2.18 days
Joseph M. Farley	AL	High	No exposure	Moderate	1.0, 1.7 ft.	3.53, 4.33°F	19.79, 28.87 days	0.01, 0.02 days	4.33, 5.44°F	-1.88, -2.27 days
Arkansas Nuclear One	AR	High	No exposure	Moderate	N/A	3.90, 4.85°F	14.15, 20.63 days	0.09, 0.17 days	4.94, 6.10°F	-2.01, -2.25 days
Palo Verde	AZ	Moderate	No exposure	None/ low	N/A	3.64, 4.73°F	16.66, 24.67 days	15.09, 22.64 days	3.82, 4.85°F	-2.34, -2.72 days
SAFER Phoenix[k]	AZ	High	No exposure	High/ very high	N/A	3.59, 4.66°F	19.21, 28.16 days	12.86, 19.86 days	4.31, 5.32°F	-2.09, -2.52 days
Diablo Canyon	CA	High	No exposure	High/ very high	0.5, 1.2 ft.	2.59, 3.27°F	5.14, 7.10 days	0, 0 days	3.56, 4.35°F	-2.49, -2.82 days
Millstone	CT	High	Category 4	None/ low	1.2, 1.8 ft.	3.38, 4.32°F	8.28, 12.78 days	0, 0 days	3.52, 4.76°F	-2.37, -2.77 days
St. Lucie	FL	High	Category 5	Moderate	0.9, 1.6 ft.	3.05, 3.99°F	37.60, 60.07 days	0, 0 days	3.52, 4.60°F	-1.52, -2.05 days
Turkey Point	FL	High	Category 5	High/ very high	0.9, 1.6 ft.	2.91, 3.75°F	54.28, 79.80 days	0, 0 days	3.11, 3.94°F	-1.99, -2.40 days
Edwin I. Hatch	GA	High	No exposure	High/ very high	0.9, 1.6 ft.	3.39, 4.22°F	20.62, 30.21 days	0, 0.01 days	4.38, 5.56°F	-1.78, -2.25 days

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios, (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios (days/ year)[j]
Vogtle	GA	No/low	No exposure	High/ very high	0.9, 1.6 ft.	3.54, 4.38°F	16.23, 22.88 days	0.01, 0.01 days	4.30, 5.41°F	-2.02, -2.41 days
Braidwood	IL	No/low	No exposure	None/ low	N/A	4.26, 5.35°F	16.66, 24.74 days	0.01, 0.01 days	5.06, 6.57°F	-2.48, -2.76 days
Byron	IL	No/low	No exposure	None/ low	N/A	4.37, 5.46°F	15.92, 24.51 days	0, 0 days	5.26, 6.88°F	-2.61, -2.84 days
Clinton	IL	High	No exposure	None/ low	N/A	4.35, 5.35°F	19.12, 26.98 days	0.05, 0.03 days	5.50, 6.87°F	-2.41, -2.74 days
Dresden	IL	High	No exposure	None/ low	N/A	4.27, 5.38°F	17.00, 25.26 days	0.01, 0.01 days	5.08, 6.61°F	-2.52, -2.80 days
LaSalle	IL	No/low	No exposure	None/ low	N/A	4.28, 5.40°F	16.67, 24.88 days	0.02, 0.01 days	5.21, 6.79°F	-2.47, -2.77 days
Quad Cities	IL	High	No exposure	None/ low	N/A	4.20, 5.22°F	17.12, 25.92 days	0.01, 0 days	5.20, 6.73°F	-2.45, -2.70 days
Wolf Creek	KS	High	No exposure	None/ low	N/A	3.91, 4.91°F	11.36, 16.85 days	0.15, 0.22 days	4.92, 6.16°F	-1.87, -2.26 days
River Bend	LA	No/low	No exposure	None/ low	1.6, 2.3 ft.	3.09, 3.91°F	29.25, 41.28 days	0, 0 days	3.85, 4.88°F	-1.33, -1.70 days
Waterford	LA	No/low	Category 5	None/ low	1.6, 2.3 ft.	2.88, 3.59°F	22.43, 33.67 days	0, 0 days	3.14, 4.04°F	-1.70, -2.04 days
Calvert Cliffs	MD	High	Category 4	Moderate	1.2, 1.8 ft.	3.62, 4.58°F	16.21, 24.27 days	0, 0 days	4.75, 6.22°F	-2.32, -2.61 days

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios, (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios (days/ year)[j]
Donald C. Cook	MI	High	No exposure	None/ low	N/A	4.25, 5.43°F	16.75, 25.52 days	0, 0 days	5.23, 6.85°F	-2.67, -2.97 days
Fermi	MI	High	No exposure	None/ low	N/A	4.18, 5.29°F	15.12, 23.04 days	0, 0 days	5.29, 7.01°F	-2.84, -3.09 days
Monticello	MN	High	No exposure	Moderate	N/A	4.36, 5.41°F	14.72, 22.40 days	0, 0 days	5.27, 6.87°F	-2.72, -2.96 days
Prairie Island	MN	High	No exposure	None/ low	N/A	4.44, 5.51°F	14.28, 22.42 days	0, 0 days	4.86, 6.45°F	-2.78, -3.00 days
Callaway	MO	No/low	No exposure	None/ low	N/A	4.34, 5.33°F	15.36, 23.26 days	0.15, 0.22 days	5.71, 7.11°F	-2.31, -2.54 days
Grand Gulf	MS	High	No exposure	None/ low	1.6, 2.3 ft.	3.72, 4.59°F	24.81, 34.84 days	0, 0.01 days	4.30, 5.46°F	-1.93, -2.18 days
Brunswick	NC	High	Category 5	High/ very high	0.9, 1.6 ft.	2.67, 3.39°F	12.00, 18.35 days	0, 0 days	3.02, 3.91°F	-1.94, -2.28 days
McGuire	NC	High	No exposure	High/ very high	N/A	3.89, 4.82°F	17.41, 24.70 days	0, 0.02 days	4.64, 5.91°F	-2.07, -2.36 days
Shearon Harris	NC	High	No exposure	High/ very high	0.9, 1.6 ft.	3.79, 4.73°F	17.66, 25.38 days	0, 0.01 days	4.47, 5.72°F	-2.05, -2.37 days
Cooper	NE	High	No exposure	High/ very high	N/A	4.32, 5.32°F	12.21, 18.53 days	0.05, 0.09 days	5.09, 6.55°F	-2.26, -2.46 days
Seabrook	NH	High	Category 4	None/ low	1.2, 1.8 ft.	3.72, 4.77°F	9.24, 13.56 days	0, 0 days	3.89, 5.18°F	-2.72, -3.07 days

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios, (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios (days/ year)[j]
Hope Creek	NJ	High	Category 4	None/ low	1.2, 1.8 ft.	3.74, 4.79°F	12.70, 19.86 days	0, 0 days	4.24, 5.71°F	-2.63, -2.95 days
Salem	NJ	High	Category 4	None/ low	1.2, 1.8 ft.	3.74, 4.79°F	12.70, 19.86 days	0, 0 days	4.24, 5.71°F	-2.63, -2.95 days
James A. FitzPatrick	NY	High	No exposure	None/ low	N/A	4.06, 5.16°F	14.59, 21.08 days	0, 0 days	4.87, 6.44°F	-2.65, -3.06 days
Nine Mile Point	NY	High	No exposure	None/ low	N/A	4.06, 5.16°F	14.59, 21.08 days	0, 0 days	4.87, 6.44°F	-2.65, -3.06 days
R. E. Ginna	NY	No/low	No exposure	High/ very high	N/A	4.39, 5.53°F	14.43, 20.97 days	0, 0.01 days	4.98, 6.72°F	-2.84, -3.15 days
Davis-Besse	OH	High	No exposure	None/ low	N/A	4.06, 5.08°F	15.18, 22.48 days	0, 0 days	4.93, 6.46°F	-2.71, -2.98 days
Perry	OH	High	No exposure	None/ low	N/A	4.27, 5.41°F	15.04, 23.08 days	0, 0 days	4.82, 6.35°F	-2.82, -3.07 days
Beaver Valley	PA	High	No exposure	None/ low	N/A	3.76, 4.79°F	16.38, 24.97 days	0, 0 days	4.84, 6.54°F	-2.55, -2.87 days
Limerick	PA	High	No exposure	None/ low	1.2, 1.8 ft.	3.88, 4.91°F	13.22, 20.69 days	0, 0 days	4.96, 6.66°F	-2.57, -2.87 days
Peach Bottom	PA	High	No exposure	None/ low	1.2, 1.8 ft.	4.06, 5.08°F	16.27, 24.13 days	0, 0 days	5.20, 6.79°F	-2.56, -2.89 days
Susquehanna	PA	Moderate	No exposure	Moderate	1.2, 1.8 ft.	4.23, 5.27°F	15.05, 21.82 days	0, 0.02 days	5.58, 7.13°F	-2.69, -3.00 days

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios, (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios (days/ year)[j]
Catawba	SC	High	No exposure	High/ very high	N/A	4.00, 4.96°F	18.27, 25.53 days	0, 0.03 days	4.73, 6.02°F	-2.05, -2.30 days
H. B. Robinson	SC	High	No exposure	High/ very high	0.9, 1.6 ft.	3.57, 4.44°F	15.03, 21.81 days	0, 0.01 days	4.39, 5.59°F	-1.94, -2.29 days
Oconee	SC	High	No exposure	Moderate	N/A	3.66, 4.54°F	19.32, 26.96 days	0, 0.03 days	4.86, 6.22°F	-1.56, -1.89 days
Virgil C. Summer	SC	High	No exposure	Moderate	0.9, 1.6 ft.	3.55, 4.50°F	15.91, 22.53 days	0.01, 0.05 days	4.50, 5.72°F	-1.71, -2.12 days
SAFER Memphis[k]	TN	No/low	No exposure	None/ low	N/A	3.99, 4.85°F	21.63, 31.04 days	0.01, 0.04 days	5.09, 6.41°F	-1.98, -2.24 days
Sequoyah	TN	High	No exposure	Moderate	N/A	3.70, 4.55°F	19.20, 26.92 days	0, 0.01 days	4.99, 6.30°F	-1.68, -2.04 days
Watts Bar	TN	High	No exposure	Moderate	N/A	3.70, 4.59°F	18.75, 26.76 days	0, 0.01 days	4.78, 6.04°F	-1.90, -2.22 days
Comanche Peak	TX	High	No exposure	Moderate	N/A	3.69, 4.67°F	14.75, 21.52 days	0.18, 0.62 days	4.07, 5.42°F	-1.89, -2.15 days
South Texas Project	TX	No/low	Category 5	None/ low	1.6, 2.3 ft.	2.93, 3.74°F	22.77, 34.60 days	0, 0 days	3.16, 3.99°F	-1.62, -1.90 days
North Anna	VA	High	No exposure	Moderate	1.2, 1.8 ft.	3.82, 4.83°F	18.28, 26.64 days	0.01, 0.02 days	4.95, 6.43°F	-2.20, -2.55 days
Surry	VA	High	Category 4	None/ low	1.2, 1.8 ft.	3.47, 4.40°F	13.85, 21.08 days	0, 0 days	3.96, 5.17°F	-2.36, -2.68 days

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios, (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios (days/ year)[j]
Columbia	WA	No/low	No exposure	High/ very high	N/A	3.90, 4.87°F	9.56, 14.05 days	0.01, 0.06 days	4.15, 5.29°F	-1.64, -1.88 days
Point Beach	WI	High	No exposure	None/ low	N/A	3.91, 4.93°F	10.01, 15.74 days	0, 0 days	4.58, 6.13°F	-2.48, -2.78 days

Source: GAO analysis of data from the fourth National Climate Assessment (NCA), U.S. Forest Service, National Oceanic and Atmospheric Administration (NOAA), the 2022 Interagency Sea Level Rise Technical Report, the Federal Emergency Management Agency, and the Nuclear Regulatory Commission (NRC). I GAO-24-106326

[a]To identify plant locations, we used nuclear power plant coordinates from NRC and added a one-half-mile radius around NRC's plant coordinates to approximate the size of a nuclear power plant. To analyze whether nuclear plants are located in areas that may be affected by heat, cold, wildfire, hurricane storm surge, and flooding, we used MapInfo mapping software to determine whether the nuclear power plant locations are located in areas with exposure to the natural hazards. Exposure indicates that a facility is located in an area that may be affected by the selected hazard. If the plant overlapped with multiple hazard layers, the layer representing the highest level of exposure was reported.

[b]To analyze exposure to flood hazards, we used 2023 data from Federal Emergency Management Agency's National Flood Hazard Layer. We grouped flood hazard zones into three categories: no/low, moderate, and high. "No/low" refers to areas with minimal, unknown, or other flood hazards, including areas with reduced risk because of levees as well as areas with flood hazard based on future conditions, such as the future implementation of land-use plans. "Moderate" corresponds to a 500-year floodplain, which indicates between 0.2 percent and 1 percent annual chance of flooding. "High" corresponds to a 100-year floodplain, which indicates a 1 percent or higher annual chance of flooding.

[c]To analyze exposure to various levels of hurricane storm surge, we used data from NOAA's Sea, Lake, and Overland Surges from Hurricanes model. We used a range of categories from the data, including no exposure to hurricanes, and Category 1, and 4, and 5 hurricanes.

[d]To analyze exposure to wildfire hazard potential, we used 2023 data from the U.S. Forest Service's Wildfire Hazard Potential Map. "None/low" refers to plants in areas that are not covered by the "moderate," "high," or "very high" wildfire potential layers. "Moderate" refers to plants in areas with moderate wildfire hazard potential. "High/very high" refers to plants in areas with high or very high wildfire hazard potential.

[e]To analyze potential exposure to sea level rise in 2050, we used data from an interagency report covering sea level rise scenarios to illustrate climate projections for sea level rise in coastal regions, under both a low and high scenario in the regions.

[f]"Projected change in daily max. temp." refers to the projected change in daily maximum temperature by the midcentury (i.e., 2036-2065) using both a low- and high emission scenario for projected climate change from the fourth NCA. Values are measured in degrees Fahrenheit.

[g]"Projected change in max. temp. exceeding historical highs" refers to the change in the annual number of days with a maximum temperature greater than the 99th percentile by the midcentury (i.e., 2036-2065), using both a low and high emission scenario for projected climate change from the fourth NCA. This

variable measures the annual number of days when the highest temperature of the day exceeds the hottest (99th percentile of) historical (1976-2005) high temperatures. Values are measured in number of days per year.

[h]"Projected change in max. temp. over 115°F" refers to the projected change in annual number of days with a maximum temperature over 115°F by the midcentury (i.e., 2036-2065) using both a low and high emission scenario for projected climate change from the fourth NCA. Values are measured in number of days per year.

[l]"Projected change in 5-day max. temp." refers to the projected change in highest maximum temperature averaged over a 5-day period by the midcentury (i.e., 2036-2065) using both a low and a high emission scenario for projected climate change from the fourth NCA. Values are measured in degrees Fahrenheit.

[j]"Projected change in min. temp. below historical lows" refers to projected change in the annual number of days with a maximum temperature lower than the 1st percentile by the midcentury (i.e., 2036-2065), using both a low and high emission scenario for projected climate change from the fourth NCA. This variable measures the annual number of days when the highest temperature of the day is lower than the coldest (1st percentile of) historical (1976-2005) high temperatures. A negative value indicates that there will be fewer days when the daily highest temperature falls below the 1st percentile. Values are measured in number of days per year.

[k]The nuclear power industry operates two Strategic Alliance for FLEX Emergency Response (SAFER) centers that maintain emergency equipment that can be provided to plants as a backup to a plant's onsite primary backup equipment.

Table 2 shows the exposure of areas around shutdown nuclear power plant locations to six current or projected natural hazards: flooding, hurricane storm surge, wildfire, sea level rise, heat, and cold.[1] Data for flooding, hurricane storm surge, and wildfire are current data and based on historical observation data. Data for sea level rise and heat and cold temperature variables are climate projections data, which incorporate emission scenarios. For more information about the data sources used, see appendix II.

[1]We included plants at various stages of decommissioning, including those in the process of decommissioning and those already decommissioned, with spent nuclear fuel stored onsite. We refer to these as shutdown plants because they are no longer operational.

Table 2: Potential Exposure to Current and Future Hazards at Shutdown Nuclear Power Plants

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/ year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios(days/ year)[j]
Humboldt Bay	CA	High	No exposure	Moderate	0.5, 1.2 ft.	2.65, 3.43°F	8.70, 14.32 days	0, 0 days	3.22, 4.20°F	-2.36, -2.68 days
Rancho Seco	CA	High	No exposure	Moderate	0.5, 1.2 ft.	3.35, 4.26°F	10.78, 15.07 days	0.12, 0.26 days	4.29, 5.35°F	-2.28, -2.65 days
San Onofre	CA	High	Category 1	High/ very high	0.5, 1.2 ft.	2.52, 3.34°F	6.59, 9.92 days	0, 0 days	3.10, 3.93°F	-2.66, -3.01 days
Fort Saint Vrain	CO	No/low	No exposure	Moderate	N/A	4.43, 5.55°F	20.00, 27.98 days	0, 0 days	5.07, 6.48°F	-1.43, -1.83 days
Haddam Neck	CT	High	Category 4	Moderate	1.2, 1.8 ft.	3.64, 4.69°F	9.65, 15.07 days	0, 0 days	3.86, 5.24°F	-2.42, -2.80 days
Crystal River	FL	High	Category 5	High/ very high	1.0, 1.7 ft.	3.01, 3.87°F	34.35, 52.84 days	0, 0 days	3.24, 4.22°F	-1.76, -2.22 days
Duane Arnold	IA	High	No exposure	None/ low	N/A	4.50, 5.55°F	17.40, 26.03 days	0.01, 0.02 days	5.47, 7.04°F	-2.60, -2.80 days
Zion	IL	High	No exposure	None/ low	N/A	4.18, 5.25°F	13.44, 20.26 days	0.01, 0.01 days	5.20, 6.80°F	-2.41, -2.65 days
Pilgrim	MA	High	Category 4	Moderate	1.2, 1.8 ft.	3.23, 4.12°F	7.35, 10.76 days	0, 0 days	3.53, 4.60°F	-2.06, -2.44 days
Yankee Rowe	MA	No/low	No exposure	None/ low	1.2, 1.8 ft.	4.24, 5.52°F	11.92, 18.63 days	0, 0 days	4.82, 6.74°F	-2.73, -3.07 days

Plant[a]	State	Flood hazard level[b]	Hurricane storm surge level[c]	Wildfire potential level[d]	Projected regional sea level rise in 2050, low and high emission scenarios (feet)[e]	Projected change in max. daily temp., low and high emission scenarios (°Fahrenheit)[f]	Projected change in max. temp. exceeding historical highs, low and high emission scenarios (days/ year)[g]	Projected change in max. temp. over 115°F, low and high emission scenarios (days/ year)[h]	Projected change in 5-day max. temp., low and high emission scenarios (°Fahrenheit)[i]	Projected change in max. temp. below historical lows, low and high emission scenarios(days/ year)[j]
Maine Yankee	ME	High	Category 4	None/ low	1.2, 1.8 ft.	3.74, 4.85°F	9.87, 14.50 days	0, 0 days	4.09, 5.41°F	-2.58, -2.9 days
Big Rock Point	MI	High	No exposure	None/ low	N/A	4.22, 5.39°F	11.06, 17.29 days	0, 0 days	4.13, 5.71°F	-3.02, -3.28 days
Palisades	MI	High	No exposure	None/ low	N/A	3.84, 4.92°F	13.75, 21.03 days	0, 0 days	4.35, 5.78°F	-2.61, -2.99 days
Fort Calhoun	NE	High	No exposure	Moderate	N/A	4.26, 5.32°F	13.67, 20.69 days	0.01, 0.01 days	4.78, 6.15°F	-2.32, -2.58 days
Oyster Creek	NJ	High	Category 4	High/ very high	1.2, 1.8 ft.	3.60, 4.54°F	9.99, 14.93 days	0, 0 days	4.33, 5.65°F	-2.51, -2.81 days
Indian Point	NY	High	Category 4	High/ very high	1.2, 1.8 ft.	3.78, 4.84°F	12.33, 18.84 days	0, 0 days	4.78, 6.39°F	-2.63, -3.00 days
Trojan	OR	High	No exposure	Moderate	0.3, 1 ft.	3.44, 4.37°F	6.64, 10.03 days	0, 0 days	4.19, 5.32°F	-1.96, -2.22 days
Three Mile Island	PA	High	No exposure	None/ low	1.2, 1.8 ft.	4.14, 5.21°F	14.66, 21.37 days	0.01, 0.05 days	5.68, 7.31°F	-2.78, -3.05 days
Vermont Yankee	VT	High	No exposure	None/ low	1.2, 1.8 ft.	3.77, 4.89°F	13.48, 21.03 days	0, 0 days	4.80, 6.64°F	-2.48, -2.84 days
Kewaunee	WI	No/low	No exposure	None/ low	N/A	3.92, 4.96°F	9.50, 14.96 days	0, 0 days	4.38, 5.92°F	-2.59, -2.88 days
Lacrosse	WI	High	No exposure	None/ low	N/A	4.31, 5.38°F	17.34, 26.24 days	0, 0 days	5.30, 6.96°F	-2.58, -2.79 days

Source: GAO analysis of data from the fourth National Climate Assessment (NCA), U.S. Forest Service, National Oceanic and Atmospheric Administration (NOAA), the 2022 Interagency Sea Level Rise Technical Report, the Federal Emergency Management Agency, and the Nuclear Regulatory Commission (NRC). I GAO-24-106326

[a]To identify plant locations, we used nuclear power plant coordinates from NRC and added a one-half-mile radius around NRC's plant coordinates to approximate the size of a nuclear power plant. To analyze whether nuclear plants are located in areas that may be affected by heat, cold, wildfire, hurricane storm surge, and flooding, we used MapInfo mapping software to determine whether the nuclear power plant locations are located in areas with exposure to the natural hazards. Exposure indicates that a facility is located in an area that may be affected by the selected hazard. If the plant overlapped with multiple hazard layers, the layer representing the highest level of exposure was reported.

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[d]To analyze exposure to wildfire hazard potential, we used 2023 data from the U.S. Forest Service's Wildfire Hazard Potential Map. "None/low" refers to plants in areas that are not covered by the "moderate," "high," or "very high" wildfire potential layers. "Moderate" refers to plants in areas with moderate wildfire hazard potential. "High/very high" refers to plants in areas with high or very high wildfire hazard potential.

[e]To analyze potential exposure to sea level rise in 2050, we used data from an interagency report covering sea level rise scenarios to illustrate climate projections for sea level rise in coastal regions, under both a low and high scenario in the regions.

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Appendix IV: Comments from the Nuclear Regulatory Commission

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

March 7, 2024

Frank Rusco, Director
Natural Resources and Environment
U.S. Government Accountability Office
441 G Street, NW
Washington, DC 20548

SUBJECT: U.S. NUCLEAR REGULATORY COMMISSION COMMENTS ON DRAFT GOVERNMENT ACCOUNTABILITY OFFICE REPORT GAO-24-106326, "NUCLEAR POWER PLANTS: NRC SHOULD TAKE ACTIONS TO FULLY CONSIDER THE POTENTIAL EFFECTS OF CLIMATE CHANGE"

Dear Mr. Rusco:

Thank you for the opportunity to review and comment on the U.S. Government Accountability Office (GAO) draft report GAO-24-106326, "Nuclear Power Plants: NRC Should Take Actions to Fully Consider the Potential Effects of Climate Change," which the U.S. Nuclear Regulatory Commission (NRC) received on February 6, 2024.

The NRC appreciates the constructive engagement we have had with the GAO throughout the audit and the GAO's thoroughness in seeking to understand the NRC's regulatory framework related to external hazards, particularly given the complexity of the detailed processes and programs that the NRC uses to support its safety mission.

We are in general agreement with the findings. The three recommendations in the draft report are very broad but are consistent with actions that are either underway or under development. The NRC considers this report to be a fair characterization of its regulatory structure, processes, and strengths. However, considering the description of the conservatisms, safety margins, and defense-in-depth policies described in the report, the NRC does not agree with the conclusion that the agency does not address the impacts of climate change. In effect, the layers of conservatism and defense in depth incorporated into NRC's processes provide reasonable assurance regarding any plausible natural hazard and combinations at a site for the licensed operational lifetime of the reactor, including those that could result from climate change. Specifically, the processes, tools, methods, models, data, and additional margins provide reasonable assurance of the ability to withstand or mitigate projected changes in natural hazards.

It is also important to note that the NRC's mission is focused on nuclear safety; as such, we cannot impose requirements that would increase energy resilience or require consideration of potential future climate impacts without a sufficient nuclear safety justification. This is an important distinction since the predominant concerns expressed often in the draft report are that a licensee's operations may be disrupted more frequently by climate change and that the NRC should more fully consider future climate projections. The NRC is focused on nuclear safety and any potential increase in the risk of a radiological release, consistent with the NRC's mandate.

F. Rusco - 2 -

We note there was no focus during the audit and no mention in the report of the NRC's proactive approach to safety reviews to support the licensing of new reactors. This is a significant contribution by the NRC to climate resilience; specifically, the NRC is licensing safe new energy sources that are an alternative to fossil-fuel-based plants. As noted in the enclosure, we would recommend a brief reference to the NRC's contributions to climate resilience via the proactive planning and implementation of new and advanced reactor licensing and license renewals.

If you have any questions concerning the staff's comments, please contact John Jolicoeur. Mr. Jolicoeur can be reached at (301) 415-1642 or by email to John.Jolicoeur@nrc.gov.

Sincerely,

Signed by Furstenau, Raymond on 03/07/24

Ray V. Furstenau
Acting Executive Director for Operations

Appendix V: GAO Contact and Staff Acknowledgments

GAO Contact

Frank Rusco, at (202) 512-3841 or ruscof@gao.gov

Staff Acknowledgments

In addition to the contact named above, Janice Ceperich (Assistant Director), Marissa Dondoe (Analyst-in-Charge), Bethany Benitez, Colleen Candrl, Breanne Cave, Lilia Chaidez, John Delicath, Cindy Gilbert, Claire McLellan, John Mingus, Katrina Pekar-Carpenter, Dan C. Royer, Wesley Sholtes, John Tanis, Joseph Dean Thompson, Linda Tsang, and Kristen Watts made significant contributions to this report.

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EQUITY. JUSTICE. HEALTH.

Jacqueline Wilson <jacqueline@cela.ca>

CELA-OPG: Follow Up to PNGS Workshop

Aditi Bhardwaj <aditi.bhardwaj@opg.com> Wed, Aug 5, 2026 at 3:27 PM
To: Caleb Broeker <lawstudent2@cela.ca>
Cc: Jacqueline Wilson <jacqueline@cela.ca>, Hannah <jinijiwai@gmail.com>, Theresa McClenaghan <theresa@cela.ca>, "Analiese St. Aubin" <analiese.staubin@opg.com>

Good afternoon Caleb,

Thank you for your email.

The information related to PSR3 is referenced throughout the OPG Commission Member Document. Is there a specific section or topic you are looking for?

OPG plans to upload the document at the conclusion of the PSR project, after the Integrated Implementation Plan has been accepted by the CNSC.

Thank you,

Aditi

Aditi Bhardwaj

289-387-2110

From: Caleb Broeker <lawstudent2@cela.ca>
Sent: Wednesday, August 5, 2026 2:50 PM
To: Aditi Bhardwaj <aditi.bhardwaj@opg.com>
Cc: Jacqueline Wilson <jacqueline@cela.ca>; Hannah <jinijiwai@gmail.com>; Theresa McClenaghan <theresa@cela.ca>; Analiese St. Aubin <analiese.staubin@opg.com>
Subject: Re: CELA-OPG: Follow Up to PNGS Workshop

*** Exercise caution. This is an EXTERNAL email. DO NOT open attachments or click links from unknown senders or unexpected email. Please report suspicious emails using “Report Phishing button”. ***

Good afternoon Ms. Bhardwaj,

Can you point out where the Pickering PSR3 Pickering Nuclear Generating Station 251 Power Reactor

Operating Licence Renewal Basis Document, P-REP-03680-00054 Pickering NGS Periodic Safety Review 3 (PSR3) Basis Document (accepted by CNSC staff in May 2024) is?

We weren't able to locate it within the documents linked to this email.

Thank you very much for your attention to this.

Best regards,

Caleb Broeker

CELA Summer Law Student

613-408-8288

On Tue, Aug 4, 2026 at 1:30 PM Aditi Bhardwaj <aditi.bhardwaj@opg.com> wrote:

> Good afternoon folks, hope you are doing well.
>
> The action that was taken the last meeting with CELA on 21 July 2026 was to provide the website information and detailed information about the Safety Improvement Opportunities and how it would improve the overall safety (in terms of Control, Cool and Contain) of the Pickering Nuclear Generating station once refurbishment is complete.
>
> Pickering Nuclear Licencing Documents are currently available on opg.com (Home - OPG Archive).
>
> OPG continues to meet regulatory safety requirements and is leveraging the Pickering Refurbishment project as an opportunity to further improve Nuclear Safety in several specific areas. Design and operational changes that can be implemented at the station during refurbishment to improve and enhance safety are referred to as Safety Improvement Opportunities (SIOs). SIOs are identified through a number of methods including reviews of the station PSAs, system reliability models, and through the Periodic Safety Review (PSR).
>
> SIOs improve the Probabilistic safety assessment (PSA) results by providing an additional layer of defense-in-depth, which ultimately result in a reduction in Severe Core Damage Frequency (SCDF) and Large Release Frequency (LRF) in the station PSAs. The selected SIOs are progressing through appropriate design phases to ensure practical solutions are able to be implemented to achieve the expected benefits.
>
> The following SIO modifications are included as core refurbishment scope and demonstrate a strong commitment to Nuclear Safety and the continued safe operation of Pickering post refurbishment:
>
> Improving COOL function:
>
> - **Firewater makeup System**: An alternate beyond design basis emergency water supply will be established for the Pickering NGS units 5 to 8 by connecting the diesel firewater pump system to existing emergency service water infrastructure. This will provide a redundant supply of cooling water to the heat transport system and

moderator. The new pumps are planned to be located in the Pickering B screenhouse, which is common to all four Pickering B units. Pump capacity will be designed to be sufficient to supply decay heat sink cooling for all four Pickering B units.

- **Enhanced Emergency Water System (EWS) supply to Boilers, Heat Transport System, and Moderator:** The emergency water supply to the boilers, heat transport system, and moderator will be enhanced by providing new flow paths. This SIO will eliminate single point of vulnerabilities by providing a redundant flowpath to the existing system and thereby will improve the overall reliability of the existing Emergency Water system at Pickering.
- **Alternate Water Supply to the Moderator:** The Emergency Water Storage Tank (EWST) is currently supplied by the Pickering 014 units' Service Water system and serves as an interim heat sink for the Pickering NGS units 5 to 8 Moderator during beyond design basis accidents. Following separation of Pickering 014, to maintain a diverse and independent supply post-refurbishment, an alternate water source to the moderator will be provided

Improving CONTAIN function:

- **Containment Filtered Venting System (CFVS)** - a new system to mitigate Beyond Design Basis Accidents (BDBA). It relieves containment overpressure, providing a filtered relief pathway to enhance containment integrity in Severe Accidents. CFVS vents steam, hydrogen and other non-condensable gases to relieve pressure, prevent failure of the containment envelope and thereby minimize any uncontrolled release.
- **Calandria Vault Overpressure Protection** – The implementation of this SIO provides increased safety margin for Severe Accidents. This SIO will install a large relief capacity on the End Shield / Calandria Vault to improve the margin to calandria vault overpressure resulting from rapidly progressing accidents.

External Hazard (High Wind) Protection:

- **Strengthening U7 Control Equipment Room Exterior Wall:** This SIO will provide additional resistance to safety related equipment failures from hazards such as windborne missiles (e.g. reinforcement of the metal clad portion of the Unit 7 Control Equipment Room exterior wall). Based on existing plant layout and design, Units 5/6/8 are already sufficiently protected.

These SIOs when implemented during the refurbishment project will result in a decrease in the risk estimates for the Pickering B PSA and will make Pickering safer.

Significant modifications are also being undertaken to improve long-term plant reliability and operational margins. Despite these not being driven specifically by Nuclear Safety enhancements (therefore **not "SIOs**"), they will contribute to an overall improvement in Plant Safety.

- **Replacement of all Pressure Tubes, Calandria Tubes and Feeders** - Including all affected instrumentation and control equipment, improving reliability. Through the use of OPEX, lessons learned and fabrication improvements, the new feeders will be more resistant to flow assisted corrosion (FAC) and cracking degradation.
- **Replacement of all 48 Boilers** - With unit defueled, dewatered, and vacuum dried, 12 boilers per unit will be replaced by hoisting through reactor building dome at East and West end banks. This will reset the effects of aging related issues as well as chemical/operational degradation mechanisms.
- **Replacement of Standby Generators** - Improves reliability for backup Class III power, a system important to safety.
- **Replacement of Digital Control Computers** - The new equipment will have better reliability associated with improved, modern technology.
- **Fuel Handling Equipment improvements** - Scope items to extend equipment life and improve reliability include but are not limited to: New closure plug replacements to reduce risk of sticking during fueling activities; Fueling Machine and carriage refurbishment: Fueling Machine head and carriage to be overhauled including internal shaft support and magazine bearings; Bridge ball screw replacement; and inspection and repair of elevator and conveyor.
- **Fire Protection Equipment improvements** - upgrades planned for several components such as valves, sprinkler systems, and fire hydrants improving reliability and supports compliance with CSA N293 and the Fire Hazard Assessment.

Thank you and should you have any questions, please feel free to reach out directly.

Aditi Bhardwaj

Senior Manager, Strategic Projects

Nuclear Regulatory Affairs

289-387-2110

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SAFETY SERIES No. 75-INSAG-6

SAFETY REPORTS

safety series

INSAG-6
Probabilistic Safety Assessment

A REPORT BY THE
INTERNATIONAL NUCLEAR SAFETY ADVISORY GROUP

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1992

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A new hierarchical categorization scheme has been introduced, according to which the publications in the IAEA Safety Series are grouped as follows:

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PROBABILISTIC SAFETY ASSESSMENT
INSAG-6

A report by the International Nuclear Safety Advisory Group

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

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Printed by the IAEA in Austria
July 1992

SAFETY SERIES No. 75-INSAG-6

PROBABILISTIC SAFETY ASSESSMENT

INSAG-6

A report by the International Nuclear Safety Advisory Group

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 1992

The International Nuclear Safety Advisory Group (INSAG) is an advisory group to the Director General of the International Atomic Energy Agency, whose main functions are:

(1) To provide a forum for the exchange of information on generic nuclear safety issues of international significance;
(2) To identify important current nuclear safety issues and to draw conclusions on the basis of the results of nuclear safety activities within the IAEA and of other information;
(3) To give advice on nuclear safety issues in which an exchange of information and/or additional efforts may be required;
(4) To formulate, where possible, commonly shared safety concepts.

THIS SAFETY SERIES IS ALSO PUBLISHED IN FRENCH, RUSSIAN AND SPANISH

VIC Library Cataloguing in Publication Data

Probabilistic safety assessment : INSAG-6 : a report by the International Nuclear Safety Advisory Group. — Vienna : International Atomic Energy Agency, 1992.
p. ; 24 cm. — (Safety series, ISSN 0074-1892 ; 75-INSAG-6)
STI/PUB/916
ISBN 92-0-102492-4
Includes bibliographical references.

1. Nuclear power plants—Risk assessment. 2. Nuclear reactors—Safety measures. 3. Nuclear power plants—Reliability. I. International Atomic Energy Agency. II. International Nuclear Safety Advisory Group. III. Series.

VICL 92-00022

FOREWORD
by the Director General

Probabilistic safety assessment (PSA) has contributed significantly to the understanding of how best to ensure the safety of nuclear power plants. By means of PSA, a nuclear power plant, including its safety systems and installations, can be analysed in its entirety. Such an analysis can yield insights into plant processes and mechanisms and possible interactions between plant systems, both for existing plants with operating histories and for plants still in the design stage.

The rapid development and increased use of PSA and its methods in recent years have been accompanied by some exaggerated claims of its capabilities and of the applicability of the results of PSA in safety analysis, plant design, and the regulation and control of operating practices. Concern over the misapplication of PSA and its methods and the misinterpretation of the results of PSAs led INSAG to prepare the present report, Probabilistic Safety Assessment, on its merits and limitations.

I am pleased to have received this report and am happy to release it to a wider audience.

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1. INTRODUCTION

At nuclear power plants, extensive safety precautions are aimed at reducing the possibility of accidents with serious consequences. The necessary level of safety is demonstrated and monitored by analysis and by compilation of information on operating experience.

Most safety analyses in the past did not explicitly consider probabilities. Compliance with safety requirements was checked by deterministic analyses that used pessimistic assumptions in order to ensure that the results of assessments were 'on the safe side'.

Operating experience with proven reactor designs shows that events of direct safety relevance have been rendered more and more scarce by the safety precautions that have been taken. However, the lack of events of safety significance in an individual case does not preclude the existence of underlying safety deficiencies. Thus, there is a gap between the bulk of operating experience and events of safety significance which may be bridged only by theoretical analysis.

Probabilistic safety assessment (PSA) is a systematic approach to performing that analysis. Its use is closely related to the use made of operating experience in general.

The rapid development of the method has resulted in its extensive use, possibly at times beyond the point of satisfactory application. This report therefore reviews the general bases of probabilistic safety assessment, emphasizing its merits and limits as well as the general lines of the future development of that methodology and its application.

In this report, some of the discussion of PSA, its methods and its results may seem very negative. If so, this is because of INSAG's conviction that PSA, if properly conducted, is an important tool in achieving safety, and INSAG believes it is essential to highlight the areas where most benefit can come from improvement of the methodology.

2. CAPABILITIES AND LIMITATIONS OF THE METHODOLOGY OF PSA

2.1. HISTORICAL BACKGROUND

The first large scale applications of probabilistic methodology to nuclear safety were the risk analyses performed in the 1970s successively in the United States of America, the United Kingdom, the Federal Republic of Germany and other countries. In particular the analysis in the USA [1] and the German investigation [2, 3] were aimed at the calculation of individual and population risks from the operation of nuclear power plants, and comparison with other natural and industrial risks. Earlier applications of statistical methods, for instance in the aircraft industry, were limited to reliability analyses.

Originally it was believed that these methods could be used for generating probabilistic acceptance criteria for the development of nuclear power. At some stage comparisons with general risks in life, and with goals for risks to individuals and populations, were proposed in the USA and discussed in several other countries. However, the use of quantitative safety goals in regulatory requirements turned out to be difficult. The main reasons were the dependence of the results of PSA on the scope and methodology, and on subjective elements, as well as a lack of plant specific data suitable for obtaining sufficiently accurate results.

On the other hand, the first risk analyses provided important insights into strengths and weaknesses of design and operation of the plants under investigation. For instance, WASH-1400 [1] and the German study [2, 3] demonstrated the significance of small break loss of coolant accidents (LOCAs) for pressurized water reactors (PWRs). The investigations also pointed to possible ways to improve the plants' safety. PSAs have subsequently been carried out for many existing plants and for new designs, and they have continued to confirm the benefit of PSA in identifying plant weaknesses to be remedied. In addition, attempts to develop safety goals and technical acceptance criteria have continued.

The 'state of the art' is described in some detail by the IAEA Safety Series report The Role of Probabilistic Safety Assessment and Probabilistic Safety Criteria in Nuclear Power Plant Safety [4] and by the NUREG-1420 peer review [5] of NUREG-1150 [6].

2.2. GENERAL ASPECTS OF PSA

In practice, PSA aims at:

— identifying and delineating the combinations of events that may lead to a severe accident;

— assessing the expected probability of occurrence for each combination;
— evaluating the consequences.

In order to perform these tasks, PSA methodology integrates information about plant design, operating practices, operating history, component reliability, human behaviour, accident phenomena, and (in its widest application) potential environmental and health effects. The approach aims at achieving completeness in identifying possible mishaps, deficiencies and plant vulnerabilities, and providing a balanced picture of the safety significance of a broad spectrum of issues, including the uncertainties of the numerical results.

Customarily, three levels of PSA are distinguished, depending on the scope of the analysis:

— A Level 1 PSA provides an assessment of plant design and operation, focusing on sequences that could lead to core damage. It can provide major insights into design strengths and weaknesses and into ways of preventing core damage that would be a precursor to a large release of radioactive material.
— In addition to the analyses performed in a Level 1 PSA, a Level 2 PSA also addresses the phenomenon of a core damage accident, the response of the containment to the expected loads, and the transport of radioactive material from the damaged core to the environment. Such analyses provide information about the probabilities of accidental radioactive releases (source terms). The analyses show the relative importance of events in terms of the primary safety concerns arising from the possibility of off-site releases, and they allow the identification of measures for mitigation of the consequences of accidents.
— In addition to the aspects analysed within a Level 2 PSA, a full scope or Level 3 PSA also analyses the dispersion of radionuclides in the surrounding environment and potential environmental and health effects.

At each level, PSA provides the probabilities (frequencies) of adverse consequences and information on the dependence of these values on specific technical features (risk profiles). These findings depend to some degree on subjective elements in the various steps of the analysis.

2.3. SOURCES OF DATA

The general approach to PSA is to define a set of initiators that challenge the safety of the plant and then to derive the probability of sequences of events leading to unacceptable consequences beyond the design basis, using a detailed plant model, initiator probabilities and component reliabilities. Data required relate to reliabilities of components, systems and operator actions. One approach is to use databases that summarize all pertinent data. Some of the data will be from plants with different

operating conditions and from components with different detailed designs (generic data). An alternative and preferable approach is to use information on direct operating experience from the particular plant to estimate initiator probabilities and system failure probabilities (plant specific data). In practice, a combination of the two approaches is used.

2.4. TYPE OF ANALYSIS

Ideally, two types of PSA may be distinguished.

- The first type of analysis, termed a posteriori analysis, refers to existing plants with operating histories. It is based on information from past operating experience, normally plant specific data. For instance, each event identified from historical records may be analysed to determine the conditional probability of its progression to an accident, and these conditional probabilities are combined to provide an estimate of core damage probability. Thus, a posteriori analyses are based on the totality of operating experience having safety significance, and they are also used in the continuous search for means of improving safety. Generic data are used where plant specific data are lacking.
- The second type of analysis, termed a priori analysis, relates to a plant with no operating history (a new project on paper or a new plant in its initial stage). Here a prediction is made for a period in the future, and generic databases or models provide the basic information for the probabilistic study.

2.5. ASSESSMENT OF RESULTS

There are two basically different ways in which PSA results can be evaluated in practice.

- One possibility is to draw technical conclusions from the relative importance of the contributions to risk of different accident sequences that could be initiated by failures in plant equipment or modes of operation. These conclusions may include evaluation of the effects of improvements. Such 'relative conclusions' are of prime importance for improving safety, as they indicate the appropriateness of and the priorities for further steps to be taken.
- The other possibility is to draw conclusions about the absolute risks from a facility and about the tolerability of those risks, taking into account the associated uncertainties. Such a course requires a Level 3 PSA. In principle, this can be done both for the technical risks (probability of a severe core damage accident, probabilities of unacceptable source terms) and for the health risks that are related to such accidents.

Regarding the quality of the final judgement, the following general points can be made. Firstly, a PSA may be a combination of the a posteriori and a priori approaches in that both generic and plant specific experience is used in the database. In that case, the uncertainties will be reduced according to the extent of the a posteriori component. It is particularly important to take this into account when trying to compare the results of different PSAs. Secondly, and by definition, relative conclusions are more reliable than absolute statements. In addition, biases and uncertainties in results increase substantially from Level 1 to Level 3 assessments. It is essential that the associated range of uncertainty be quoted in conjunction with any absolute statement, and it must be accepted that the range will be so wide as to require caution in the application of the results.

2.6. MERITS OF PSA APPLICATIONS

The success of PSA in identifying vulnerabilities and possible improvements in plant safety is often closely related to the process of performing such investigations. That process opens up new opportunities to detect safety deficiencies, and it stimulates people involved to improve plant safety through careful rethinking of the entire range of possible malfunctions. That independent 'orthogonal' look at the technical system is among the most important merits of PSA methodology. It should be noted that such benefit requires a questioning attitude on the part of the analyst, all the more so because PSA lacks a strictly standardized methodology.

Altogether, PSA has proven beneficial in areas such as the following:

- evaluation of the completeness and balance of the design for safety (including the identification of risk relevant plant conditions and of possible 'cliff edge effects' with regard to accidents and their consequences);
- evaluation of modifications of plant design and procedures from a safety viewpoint;
- evaluation of management and operational strategies;
- assistance to plant management in establishing effective maintenance and testing (including 'living probabilistic safety assessments');
- training of staff;
- assistance in decision making on backfits;
- evaluation of accident management strategies;
- guidance for improving technical specifications;
- systematic monitoring of the safety level, for instance by evaluation of precursor events for their safety significance;
- establishment of priorities for future research by identification of those areas where knowledge is most incomplete, e.g. by evaluation of PSA uncertainties;
- assistance in the development of new reactor concepts.

Most of these benefits are achieved by Level 1 (core damage probability) and Level 2 (source term probability) a posteriori analyses. The success of such investigations makes them most valuable for systematic safety reviews of operating nuclear power plants.

2.7. PRECAUTIONS CURRENTLY NECESSARY IN THE USE OF PSA

Irrespective of its benefits, the use of PSA is subject to limitations. They are due to its dependence on deterministic design, to uncertainties concerning data and models, to difficulties in properly treating some issues, and to the resulting subjective views of the team performing a PSA. Such limitations must be taken into account when using PSA for decision making.

It should also be mentioned that there is as yet no recognized standardized PSA methodology. This lack makes it difficult to compare numerical results of different PSAs.

2.7.1. Dependence of accuracy on plant design

The accuracy of a PSA depends in a fundamental way on the design of the plant. A design with several independent lines of defence in relation to a particular fault permits derivation of data from past experience, since the reliability requirements for individual components or systems can be moderate and failure rates are thus observable. Where lines of defence are fewer or not fully independent, the reliability requirements must be much more stringent, and data from past experience will be scarce and highly uncertain.

Common cause failure and system interactions may play an important role. If the design based separation between systems and trains is adequately implemented, many causes for dependences, interactions and common cause failures are eliminated, and systems and trains act nearly independently of each other. This facilitates the modelling and lessens the vulnerability of the plant, systems and components to these failures. However, if such provisions are not made in the design, the vulnerability to common cause failure and complex system interactions might be large and difficult to assess.

Plant design also affects the accuracy of a PSA through the measures taken to reduce dependence on satisfactory operator performance, since human reliability is inherently difficult to quantify.

2.7.2. Uncertainty

Most PSAs include only internal initiators. They exclude external events such as earthquakes or flooding due to storms or tsunamis. INSAG does not refer to these sources of uncertainty at this point.

Much of the uncertainty in data can be estimated. However, when failures in components of nuclear plants lead to actions to prevent their recurrence, the influence of such corrections on reliability might be difficult to evaluate.

The limited capability to model with reasonable accuracy some relevant issues (e.g. phenomena of accident progression including core melt, human behaviour, health effects of low level radiation) also contributes to the uncertainty of PSA results. Such uncertainties are sometimes estimated by expert opinion.

Both types of uncertainties can be propagated through the steps of the analysis, leading to probability distributions of the calculated results. According to the presently available Level 1 PSAs of proven reactor designs, the uncertainties (90% confidence interval) in predictions of core damage probability cover a range of roughly one order of magnitude. In Level 2 analyses, the uncertainties are much larger, because of the difficulties in modelling containment loads and containment failure mechanisms associated with severe accidents. Available results show that the uncertainty range of Level 2 results can extend over several orders of magnitude. In Level 3 analyses, somewhat more uncertainty is added by the inclusion of atmospheric dispersion and of low-dose-response relationships in the evaluation of the significance of relatively few health effects in large populations. Limitations of this kind must be recognized in the application of PSA.

2.7.3. Limitation of scope

PSA is limited to certain classes of issues. Issues may be neglected because they are believed unimportant. (Such issues may begin to contribute when the importance of other issues is reduced by improvements in technology.) Issues may also be neglected because they cannot be readily included in the structure of PSA models. For instance, the scope of accident sequence analysis is generally limited to component failures of specific types and to operator failures to correctly perform described actions. The consideration of initiating events is also limited in scope. As just stated, most PSAs do not include external events as initiators. This point is discussed further in Section 2.7.7.

In the following, specific limitations of scope are described in more detail.

2.7.4. Human factors

It is difficult to model human behaviour. Thus the influence of the human factor is among the issues most difficult to quantify in a PSA. Designs whose safety largely depends on human intervention are particularly sensitive in this regard.

For proven designs, the influence of human error has been substantially reduced by automation and an improved man–machine interface. However, as the overall reliability of plant technology is constantly being improved, human error, even if reduced in absolute terms, remains an important contributor to risk.

A particular problem is the assessment of the type and probability of operator errors of commission. Those errors would be intentional acts by operators outside of procedures, which could occur as a result of vagueness of procedures, misleading instrumentation or simply errors on the part of the operators. Including such errors in a PSA is extremely difficult, because the number of possible actions to be considered is almost unlimited. Some kinds of such errors, i.e. misdiagnosis with related erroneous operator actions, are dealt with in some recent PSAs. On the whole, however, the capability to model human factors is still limited, and the results have large uncertainties. Even if such actions can be identified, quantification is difficult. Greater progress has been made in studying human response with reactor simulators; however, there may also be a systematic bias in results obtained this way, because the stress that would be present under actual conditions is absent.

2.7.5. Common cause failures

Common cause failures of redundant safety systems are observed extremely rarely, largely owing to the high quality of nuclear power plant equipment. Nevertheless, common cause failures may have a major influence on system failure rate in a highly redundant system.

There are several models in use to assess the probability of common cause failures. Most of them use existing information about single failure events in order to formulate conjectures about multiple failure probabilities. Sometimes, additional information about specific common cause events (such as double or triple failure events) is evaluated. However, there exists a basic difficulty in extrapolating these data to multiple failure probabilities, as there is in most cases no clear causal dependence between the observations and the multiple failure modes that are to be analysed. In particular for highly redundant systems (e.g. safety relief systems in boiling water reactors (BWRs)), it is therefore extremely difficult to reliably assess the failure probabilities.

This difficulty in accurately predicting common cause failure probabilities is believed to result, in many cases, in overestimation of common cause contributions. Nevertheless, because of the large uncertainty in the prediction of such effects, the potential for and consequences of common cause failure need to be addressed, and more emphasis on diversity in systems seems justified.

2.7.6. Low probability events

In general, event families of very low probability are neglected because of judgement of their unimportance. In this regard, it is common practice (and seems to be without significant influence on the outcome of an analysis) to discount event families that occur with probabilities that are two or three orders of magnitude lower than the overall probability of the postulated adverse outcome. As most PSAs of

proven reactor designs predict core damage probabilities of between 10^{-5} and 10^{-4} per reactor-year, a cut-off probability of 10^{-7} per reactor-year seems appropriate in this regard. If a total core damage probability is found to be significantly below 10^{-5} per reactor-year, issues that would be outside the scope of PSAs on the aforementioned basis may become significant if they are numerous.

The end results of a PSA represent degrees of belief that certain specified detrimental effects will occur in a given time frame. That degree of belief is described by probabilities which may be based on two different types of information:

- The first type of information is empirical data and their stochastic variability. The corresponding input into a PSA can be interpreted as the probabilities of occurrence of certain phenomena observed.
- Information of the second type is judgement about relevant issues such as phenomena, failure modes and related probabilities. Such judgements may be based upon scientific background and engineering experience, but are not directly supported by empirical data such as probabilities of occurrence.

It is a basic aim of PSA to derive the results to the largest possible extent from empirical data. As explained earlier, that is accomplished by a breakdown of the possible failure mechanisms into the product of various independent submechanisms, which permits deducing even very low accident probabilities from limited operating experience. The need to use such a method is a direct result of the success of defence in depth in nuclear plant safety. However, the share of subjective elements in results increases as the probabilities representing the end results decrease. Below a certain level of probability, the empirical component of the results will be negligible, and the PSA results will essentially be a judgement that the outcome in mind is impossible. Experience with existing PSAs of proven designs indicates that frequencies below 10^{-7} per reactor-year for event families are well below that level.

For families of events that would threaten the containment and could therefore lead directly to a large release, it might be necessary to consider even lower probabilities. However, it is useful to consider such events in a separate analysis from that for events for which the containment provides a further barrier (see Section 4).

2.7.7. Plant internal and plant external hazards

Typical hazards such as seismic events, fires and flooding (external, internal), which might compromise several safety systems simultaneously, are treated in some recent PSAs and have frequently given significant contributions to severe accident probabilities. They are clearly distinct in nature from initiators that would be associated with a localized part of the plant. They are rather predecessors of accident initiators than initiators themselves.

The contribution of those hazards to risk is site and plant specific. Relatively large contributions have sometimes been found for older plants, due to vulnerable

or non-existent physical protection and separation. In new designs, qualification of components to withstand harsh environments and adequate physical separation significantly diminish the vulnerability to hazards.

The methods presently used for assessing hazards are characterized by large uncertainties. Confidence intervals extend over several orders of magnitude even for Level 1 analyses. Such uncertainties make it meaningless to combine these results with those of events due to localized initiators.

2.7.8. Interaction between designers and operators and the PSA team

Effective use of PSA depends on the premise that the evaluation is as close as possible to reality. That requires complete information about possible failure modes and the use of best estimate data. It generally requires much more effort and better knowledge than using design data from the licensing process. It also requires keeping data up to date when modifications are made to the plants.

It is evident that the establishment and maintenance of such an information base and database can only be achieved by close co-operation between the team performing the PSA, the operator and the designer of the plant. For instance, PSA teams need free access to all relevant information from periodic tests, maintenance and repair. Walk-through of the plant by members of the PSA team is important so that they can become familiar with ongoing developments and changes in the plants and can receive dependable information about other relevant aspects of operation. Co-operation with the plant designer must support understanding of realistic system response under accident conditions and, in particular, beyond design limits. On the other hand, involvement of the plant operator and designer in PSAs furthers safety consciousness and a risk attentive attitude on the part of the staff and can thus essentially contribute to safety culture. Thus the flow of information and the benefit are in both directions: one direction improves the PSA; the other contributes to safer design and operation.

3. FURTHER DEVELOPMENT OF METHODOLOGY

The precautions currently necessary in the use of PSA are related to the availability and quality of data and to the appropriateness of models as well as to the completeness and the quality of analyses. Improvements seem possible in all fields. The following aspects are of particular importance in this regard.

3.1. PLANT DATA

Many PSAs still use a large amount of generic data for the probabilities of accident initiators and for component reliabilities. Improved collection of plant specific data and a more widespread use of methods such as the Bayes approach to combine plant specific observations with generic data should become common practice. In addition, the analysis of subtle statistical and probabilistic correlations still needs further development. Such correlations may be related to hidden deficiencies in components, for instance due to errors in manufacture, installation, inspection or maintenance.

3.2. ACCIDENT PROGRESSION MODELS

The events occurring if an accident were to progress to severe core damage and beyond would be physically and chemically complex. To date, no validated mechanistic computer codes are available to consistently model the relevant phenomena with satisfactory model accuracy and predictive capability. This holds in particular for phenomena expected to occur in the containment after vessel breach. Effort to develop adequate models should be sustained. In addition, design improvements would assist in reducing the large uncertainties of Level 2 analyses, such as the conditional probabilities of containment failure and release modes.

3.3. CONDITIONS WHILE THE PLANT IS SHUT DOWN

Until recently, virtually all PSAs for power reactors were performed for full power conditions. This was considered to be a conservative approach, based on having the maximum amount of energy available in the core, which maximizes the system response requirements and minimizes the time available to prevent damage. However, during shutdown major maintenance activities related to safety systems might be performed; the redundancy usually required at full power may not exist; emergency operating procedures may be limited; the state of the plant may be less clear owing to the many activities under way; the integrity of the primary cooling

system and that of the containment might be compromised; and the operating and maintenance staff may be expected to be less attentive. The human factor, especially in errors of commission, may be very important under those conditions. Therefore, shutdown conditions could make significant contributions to the core damage probability. Inclusion of such states in PSAs will require substantial effort, as has been seen in PSAs recently conducted in France, because within each different plant operating mode different technical specifications apply, reliability data might be different and scarce, and therefore separate analyses are required.

3.4. HUMAN FACTORS

The difficulty in modelling human behaviour is due to the lack of good data, the shortcomings in describing human actions by models and the dependence on circumstances.

Thus, it is important to improve the collection of human factor data from operating experience. Such a collection should include data on maintenance and repair. Increased use of full scale simulators would be helpful in strengthening the base of the information about human response during potential accidents, though the reservation mentioned earlier in this connection remains important.

Reduction of uncertainties from human factors will follow the development of designs that make safety less dependent on human action and the modification of existing designs to such an end, and the accumulation of greater operating experience that includes the effect of human errors.

3.5. COMMON CAUSE FAILURES

It has been stated before that data on multiple failures are extremely scarce. Therefore, significant improvement in the quantification of probabilities of common cause failure events depends on both investigating all available data on phenomena with a potential for resulting in a common cause failure, and developing common cause models emphasizing the causal dependence of common cause failure probabilities on such phenomena.

3.6. CONSIDERATION OF TIME DEPENDENCE

To date, only a few time dependent phenomena are modelled in PSA studies. Exceptions are time dependent success criteria in long term accident sequences in French PSAs. The effects of several time dependent phenomena such as ageing of components, and time dependent unavailabilities (test intervals, latent failures,

repair) and time dependences of accident sequences (time dependent success criteria, time dependent operator actions, time dependent physical phenomena) are usually treated by averaging in PSAs.

For some time dependent phenomena, such as dependence of recovery actions on time available and varying minimum success requirements, more sophisticated models should be developed. These issues, however, are not the most urgent problems in PSA, and they may be regarded more as mid-term priorities.

3.7. QUALITY ASSURANCE

Quality assurance of PSA is important, as PSA will be more widely used in the future. Careful peer review is an essential part of the review process in each case, in order to ensure that PSAs correspond to the 'state of the art'.

4. GUIDELINES FOR PRESENTATION AND INTERPRETATION OF PSA RESULTS

PSA's complex methodology is difficult for non-specialists to understand. Findings can easily be misunderstood. Therefore, presentation of results should be made as 'transparent' as possible, and should include presentation of uncertainties, scope and coverage, omitted phenomena, the peer review process, and guidance on the use of the results.

As explained in Section 2.7.6, PSA results generally lose their meaning below a certain level of probability, especially if they are related to single event families. It has also been said before that this level depends on both deterministic design and data from operating experience, and the lower level for proven designs is currently assumed to be in the range of 10^{-7} per reactor-year. As stated earlier, inclusion of events above this threshold leads to confidence in overall accuracy above 10^{-5} per reactor-year. Results related to probabilities below that threshold should generally be excluded from presentation.

Estimates of uncertainties can be among the important findings of probabilistic assessments. Therefore, it seems reasonable to depict the dependence of uncertainties on technical issues (uncertainty profiles) and the sensitivities of results to imperfectly understood phenomena.

The assessment of a risk contribution from a single event with consequential failure of all levels of defence will normally rely on a small probability of occurrence of just one phenomenon or of a group of closely related phenomena. It has been explained earlier (Sections 2.7.1 and 2.7.2) that in such a case, a calculated probability of resultant severe off-site consequences is highly uncertain and may even be, at least below a certain level of probability, rather speculative[1]. It is evident that such events should be excluded by design. If they are not, they need to be addressed in a PSA. However, such results should be presented separately from other results because of their different character and the eventual need for specific actions.

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[1] The Chernobyl plant, as it was before the accident in 1986, may serve as example. That design permitted the triggering of an autocatalytic excursion with destruction of all barriers by malfunctioning of a single system, i.e. the reactivity control system. Thus, if a probabilistic assessment had been made, the calculated probability of severe consequences would have depended almost exclusively on terms such as a common cause failure of the control system or a human error.

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5. HOW SHOULD PSA BE USED IN THE FUTURE?

5.1. USE FOR TECHNICAL CONCLUSIONS

The success of Level 1 PSA in a posteriori analyses being taken into account, an expanded use in this field promises further benefits. Increased use of plant specific analyses would be helpful. Further, PSA could be applied more intensely to systematic monitoring of safety. Plant personnel should be involved in those activities in order to promote the thought processes of PSA among the staff of nuclear facilities. Owing to the fact that Level 1 results can be translated relatively easily into technical findings, many PSAs are restricted to that level. Nevertheless, the integrity of the containment is highly relevant to the fundamental objective of nuclear safety, i.e. the protection of the public against radiation hazards from accidents in nuclear facilities. Therefore, PSA results should include at least the presentation of major conclusions with regard to the containment, especially if these depend significantly on specific event sequences and phenomena.

Some standardization might be required to ensure that decision making has an equivalent basis for different plants. That requires PSA guidelines[2] to define methodology standardized according to the state of the art. However, PSA methodology has not yet been perfected, and a questioning attitude when conducting a PSA is a significant contributor to safety. Therefore, such guidelines are to be adapted from time to time to the evolution of PSA, which should be advanced by non-standardized 'exploratory investigations' in depth.

5.2. USE IN CONNECTION WITH SAFETY CRITERIA

Various deterministic criteria are used for NPP design in different countries. International co-operation in the nuclear industry and the need to explain transborder safety levels clearly to the public require a more consistent international approach in that area. Probabilistic assessments could provide an essential contribution in making deterministic criteria more consistent, more 'transparent' and more comparable. However, care should be taken to maintain the distinction between a priori and a posteriori assessments.

5.3. USE FOR CONCLUSIONS ABOUT TOLERABILITY

As noted in Section 2.5, PSA results can be used either to determine the significance of different contributors to accidents or to judge the risks from a nuclear

[2] See, for example, Ref. [7] and references contained therein.

power plant. In the latter regard, it is widely accepted that decisions concerning the tolerability of risk should be based on three principles:

— There exist levels of risk from technology to individuals or society that should not be tolerated irrespective of the technology's benefits. Such levels are often referred to as tolerability limits.
— At risks lower than that level, safety cannot be absolute, and the knowledge of how to improve it is never complete. Responsible action includes continued striving for risk reduction, provided that the effort to achieve these improvements is not unreasonably high.
— Well below the tolerability limit, risks are so low that they should be regarded as negligible in order to avoid unnecessary deployment of resources which diverts attention from substantial safety issues which could lead to larger risks of other types. That corresponding low level is sometimes called a de minimis limit.

The implementation of these principles requires safety goals based on appropriate definitions of risks, which ensure that a comparison of the actual levels of risks with the goals is practicable, meaningful and 'transparent'. Goals can be set as technical safety objectives, for example on core melt probability. The corresponding criteria can be assessed using a Level 1 PSA for a core melt criterion, or a Level 2 PSA for a source term criterion. They can be related to design characteristics and potential improvements. However, to be able to proceed to comparisons with other risks, including those from other industries, goals must be set in terms of overall risks for the individuals or the society, which are called high level risk objectives[3]. In that case, a Level 3 PSA is needed, with all the associated uncertainties identified earlier.

Although this framework is likely to receive general support, achievement of an international agreement on the two values delineating the regions, tolerability limit and de minimis limit, is likely to be difficult for at least three reasons:

— it demands agreement on explicit societal risk criteria;
— it has to take into account the uncertainty margins of the PSA results;
— when comparison with other risks is sought, systematic flaws in evaluations might make the comparisons meaningless.

That holds in particular for Level 3 and a priori analyses, where the development of appropriate methods to overcome the aforementioned difficulties is an important task which will take quite some time.

On the other hand, technical objectives, such as core damage probability or probability of source terms of given sizes, are in use. The related limit values could

[3] It is understood that other factors such as land contamination must be taken into account in determining objectives for societal risks.

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and should be strengthened by appropriate consideration of the problems of uncertainties in PSA results and of comparability between different PSAs. The use of confidence levels (e.g. 95% percentiles) is recommended in this regard. It also seems reasonable to establish quantitative requirements for issues difficult to assess, such as common cause failures and human errors. The definition of minimum contributions of those factors to the risk level would help to eliminate meaningless evaluations from comparisons.

6. CONCLUSIONS

Nuclear power plants are generally designed to deterministic requirements. Nevertheless, many probabilistic assessments have now been carried out on existing plants and proposed designs. Numerous lessons have been learned from these applications, and there is now a considerable and continuously increasing knowledge base on the capabilities and the limits of PSA.

Thus it can be said that PSA has matured to a point where there is now a broad basis of understanding. There is agreement that it constitutes a valuable additional basis for safety assessment and for decisions on safety improvements. In particular, the methodology has proven to be a most valuable tool for identification of plant weaknesses.

The use of PSA is therefore strongly encouraged. Plant operators and plant operating personnel should be involved in these activities. However, the methodological limitations and the range of uncertainties involved must be taken into account in the interpretation and use of results.

With appropriate caution, PSA has applicability in safety analysis, plant design, control of operating practices and licensing. Another important area where past applications have been found useful, and where future activities are strongly encouraged, is the use of PSA for evaluation of operating experience (e.g. precursor analyses).

It is essential that uncertainties should be emphasized in displaying results. The presentation of point estimates should be restricted as far as possible, and uncertainties should be reflected even in that case. It must also be recognized that PSA results change their character and lose their meaning below a certain level of probability. In special cases, if new phenomena are involved, a separate presentation of low probability results might be warranted. In general, however, results should not be displayed if probabilities are below a level in the range of 10^{-7} per reactor-year.

Particular attention to the possibility of biases and uncertainties due to subjective views of PSA teams, human factor and common cause contributions, dependence of accuracy on deterministic design, and uncertainties related to modelling containment loads and off-site consequences is required when interpreting PSA results. It must also be recognized that the uncertainties of results escalate greatly for a PSA beyond Level 1. Furthermore, the extent to which external events have been included among the initiators must be borne in mind.

The use of PSA for the establishment of acceptance criteria has turned out to be difficult. That is due to both the difficulty in completely assessing all biases and uncertainties in numerical terms and to the dependence of acceptance safety levels on a number of factors, such as expected lifetime, perceived risk, availability of data, ability to carry out realistic calculations and the methods used in the original design.

Nevertheless, it is necessary to establish assessment criteria in order to identify to what extent resources should be spent in continuing to reduce risks. In this regard,

fundamental risk criteria (health effects) are most readily comparable and in spite of the problems attached to higher uncertainties they provide the opportunity to base such decisions on a footing comparable with that applied in other areas of life. However, intermediate technical surrogates are often more useful as their probabilities can be calculated more precisely with PSA. There is an emerging international consensus on target probabilities of core damage and large accidental release.

There have been differences in the methodologies used in PSAs, and there is a definite need for improvement in methods and data. In particular, for future reactors for which utilities and regulators may wish to compare an international range of designs there is a clear need to agree on consistent methodology and assessment criteria in order to ensure that comparisons are valid. Such standardized procedures should be subject to periodic review as PSA methodology is not sufficiently mature for its present status to be frozen.

Important areas requiring further effort in methodology include human factors, common cause failures, hazards analysis, and low probability cut-off levels. Efforts in data collection should be strengthened in order to improve modelling in these areas and to extend the use of plant specific PSA. Attention should also be given to extended application of PSA in areas such as identification of relevant research issues, and training of plant personnel and regulatory staff. A further important issue is improving communication on PSA and risk among nuclear engineers and scientists, scientists in other fields, and the general public. All of these activities need international co-operation and co-ordination.

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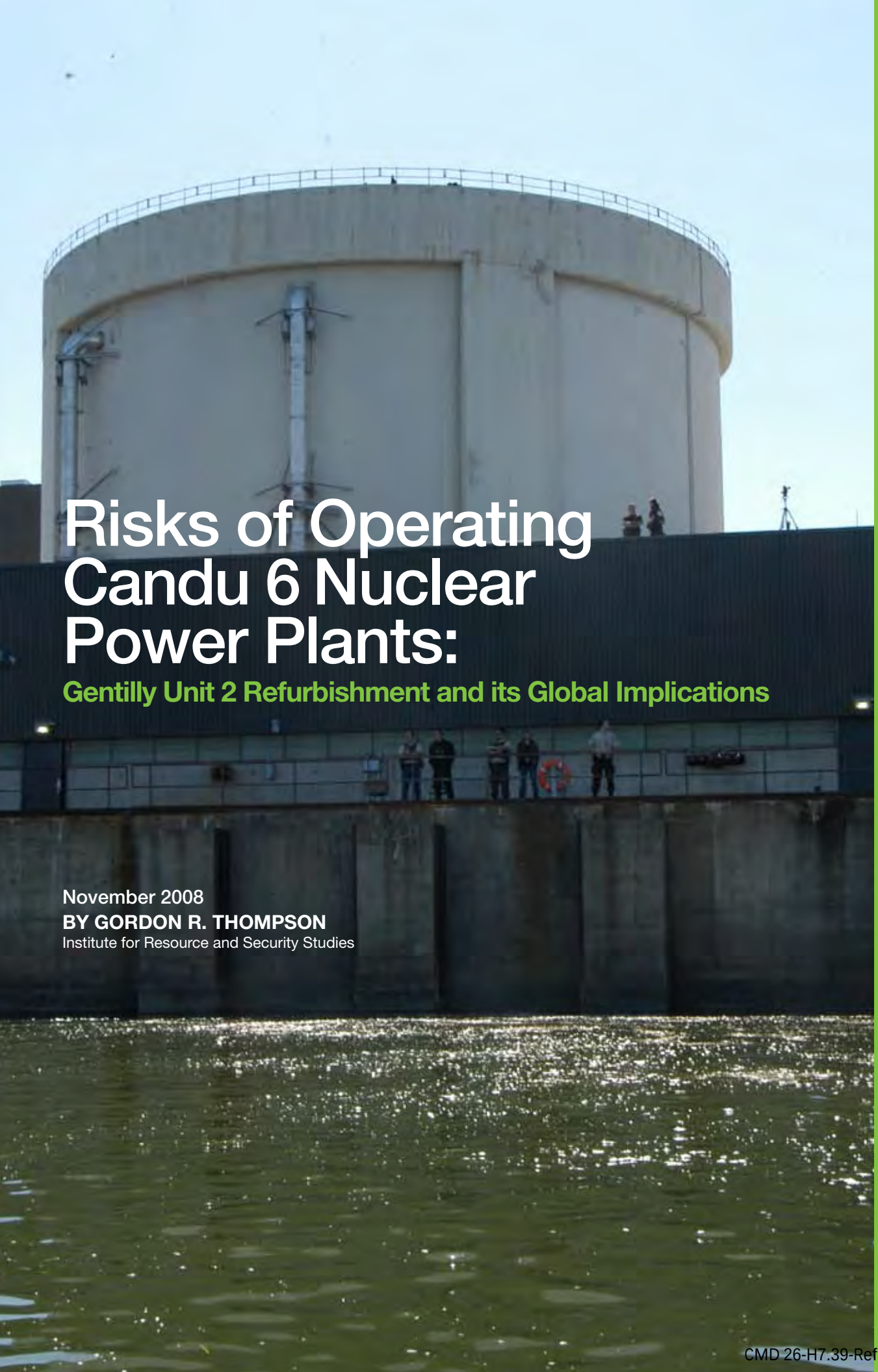
ISBN 92-0-102492-4



Risks of Operating Candu 6 Nuclear Power Plants:

Gentilly Unit 2 Refurbishment and its Global Implications

November 2008

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ABSTRACT

Operation of any nuclear power plant creates risks. CANDU 6 plants pose additional risks arising from their use of natural uranium as fuel and heavy water as moderator. A CANDU 6 reactor could experience a violent power excursion, potentially leading to containment failure and a release of radioactive material to the environment. Spent fuel discharged from a CANDU 6 could be diverted and used to produce plutonium for nuclear weapons. Those risks are examined here with special attention to Hydro-Quebec's plan for refurbishment and continued operation of the Gentilly 2 plant. That action would lead to continued radiological risk in Quebec and could promote sales of CANDU 6 plants in other countries, thereby contributing to an enhanced risk of nuclear-weapon proliferation. Hydro-Quebec's plan also faces regulatory risks. Safety issues could increase the cost of refurbishing Gentilly 2, weakening an already marginal economic case for refurbishment. This report proposes an approach for systematic, public assessment of the risks associated with Gentilly 2.

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ACKNOWLEDGEMENTS

This report was prepared by IRSS under the sponsorship of Greenpeace Canada. Shawn-Patrick Stensil assisted the author by obtaining information that was used during preparation of the report. The author, Gordon R. Thompson, is solely responsible for the content of the English-language version of this report. If the report or parts of it are translated into other languages, the author is not responsible for errors of translation.

Executive Summary

Hydro-Quebec plans to refurbish the Gentilly Unit 2 nuclear power plant to extend its operating life to about 2040. The plan involves three categories of risk that have not been properly assessed. One is the risk of an unplanned release of radioactive material, through accident or malevolence. The second is the risk that spent nuclear fuel will be diverted and used to produce plutonium for nuclear weapons. The third is the risk of regulatory actions that would increase the costs of refurbishing Gentilly 2. This report discusses all three categories of risk, and proposes an approach for systematic, public assessments of the risks associated with Gentilly 2. Those assessments could provide important information to citizens in Quebec, across Canada, and around the world.

BACKGROUND

Gentilly 2 is the only nuclear power plant operating in Quebec, providing about 3 percent of the province's electricity. It was commissioned in 1983, and was built to a Canadian design known as the CANDU 6. Hydro-Quebec plans to refurbish Gentilly 2 at a currently estimated cost of about Can$1.9 billion. The economic case for refurbishment and continued operation is weak, according to Hydro-Quebec. Operation of the plant, after its refurbishment, must be approved by the Canadian Nuclear Safety Commission (CNSC).

Atomic Energy of Canada Limited (AECL), a Crown corporation, has built two CANDU 6 plants in Canada – Gentilly 2, and the Point Lepreau plant in New Brunswick. In addition, nine CANDU 6 plants supplied by AECL are operating in Argentina, China, Romania and South Korea. AECL hopes to build new CANDU 6 plants around the world, and is pursuing opportunities in Argentina, Jordan, Romania, Turkey and elsewhere.

RISKS ASSOCIATED WITH THE CANDU 6 DESIGN

The concept of "risk" encompasses the probability and magnitude of an adverse impact on humans and the environment. Operation of any nuclear power plant creates risks. Plants of the CANDU 6 design pose additional risks that arise from basic features of the design, especially the use of natural uranium as fuel and heavy water as moderator and coolant. Those features create additional risks in two respects. First, a CANDU 6 reactor could experience a violent power excursion, potentially leading to containment failure and a release of radioactive material to the environment. Second, at a CANDU 6 plant it is comparatively easy to divert spent fuel in order to produce plutonium for nuclear weapons.

Continued operation of Gentilly 2 would involve risks specific to that particular plant, the Gentilly site, Hydro-Quebec's management, and Canada's socio-political climate. Risks associated with other CANDU 6 plants would differ in various respects. Nevertheless, the design features of the CANDU 6 plant are a common thread that links the risks associated with every CANDU 6 plant. Canada, as the home of the CANDU 6 design, has a responsibility to assess the influence of that design on the risks associated with CANDU 6 plants worldwide. The Director General of the International Atomic Energy Agency (IAEA) stated in October 2008: "Suppliers of nuclear technology owe a duty of care to the recipients and to the world at large." Hydro-Quebec's plan to refurbish Gentilly 2 provides an opportunity for Canada to carry out that duty in regard to the CANDU 6 design.

THE RISK OF AN UNPLANNED RELEASE OF RADIOACTIVE MATERIAL

In the context of an unplanned, radioactive release, a CANDU 6 plant has many characteristics in common with other nuclear power plants now operating worldwide. Almost all of those plants are in the "Generation II" category, and most (80 percent) are light-water reactors (LWRs) that are moderated and cooled by light water. Plants constructed during the next few decades would be in the Generation III category.

Any of the nuclear power plants now operating could experience an unplanned release of radioactive material as a result of an accident or a malevolent act. There are plant-specific aspects of the potential for such a release, but also broad similarities. For example, each Generation II plant has a comparatively modest capability to resist attack by a well-informed, well-resourced, sub-national group.

CANDU 6 plants are, in many respects, representative of Generation II plants. There is one respect, however, in which CANDU plants, including the CANDU 6, differ significantly from most Generation II plants. CANDU plants use natural (un-enriched) uranium as fuel, and heavy water as moderator and coolant. As a result, a CANDU reactor has a positive void coefficient of reactivity. Thus, if the flow of cooling water to its core were interrupted and shutdown systems were ineffective, the reactor would experience a violent power excursion, challenging the integrity of the containment structure. Such an event occurred at Chernobyl Unit 4 in 1986, with a large release of radioactive material to the atmosphere. Chernobyl 4 was an RBMK plant, not a CANDU, but it also had a positive void coefficient of reactivity.

AECL is currently offering a new version of the CANDU design concept, known as the ACR-1000. No ACR-1000 plant has been built to date. Design changes incorporated in the ACR-1000 include the use of light water as primary coolant, and low-enriched uranium fuel. As a result of those changes, AECL expects that the void coefficient of reactivity for the ACR-1000 would be slightly negative. That design feature, if achieved by AECL, might allow the ACR-1000 to be licensed in countries that would not accept a positive void coefficient. There are indications that the CNSC would refuse to license a new Canadian plant with a positive void coefficient. Concern about that licensing issue may have influenced the Ontario government to exclude the CANDU 6 from the list of designs for which it has invited bids for construction of new nuclear power plants in Ontario. The list now consists of the ACR-1000, the EPR design offered by AREVA, and the AP1000 design offered by Westinghouse. The latter two designs are LWRs.

In 2000, the IAEA established international standards for the design of new nuclear power plants. Those standards are, in many respects, the “lowest common denominator” of standards set by national regulators. Nevertheless, the IAEA recommends that plants have “inherently safe” behavior. The CANDU 6 design does not meet that standard, because it has a positive void coefficient of reactivity. Moreover, national regulators, including the CNSC, are beginning to require that new nuclear power plants have some capability to resist malevolent acts. The CANDU 6 design has limited capability in that regard. Thus, new CANDU 6 plants will become increasingly difficult to license as safety and security standards rise around the world. The IAEA is encouraging a trend toward standards that are more stringent and more uniform across national regulators.

THE RISK OF DIVERSION OF SPENT FUEL

AECL hopes to sell the CANDU 6 to a number of countries. Presumably, those countries would see advantages to the CANDU 6 that would offset risk issues such as a positive void coefficient and vulnerability to malevolent acts. It appears that the Turkish government sees such advantages. In soliciting bids for construction of new nuclear power plants in Turkey, the Turkish government has stated that it will consider the construction of CANDU-type plants only if they are fueled by natural uranium. The ACR-1000 is excluded by that requirement, but the CANDU 6 is allowed.

One reason for a government to favor a plant design that uses natural-uranium fuel would be the lack of need to purchase uranium-enrichment services for that plant. Moreover, if uranium could be mined within the country, the nuclear fuel cycle could become entirely indigenous. A government might choose that arrangement from the perspective of economics and/or energy security. There is also another consideration that a government would be unlikely to discuss in public. Deployment of an indigenous nuclear fuel cycle, featuring reactors that employ on-line refueling, would provide the country with a reserve capability to produce plutonium sufficient for a substantial arsenal of nuclear weapons. The country’s government could draw upon that capability at some future date, depending on the government’s assessment of the net benefit of establishing a nuclear arsenal.

The CANDU 6 design employs natural-uranium fuel and on-line refueling. Thus, CANDU 6 could be a preferred plant choice for a government that contemplates the possibility of deploying a nuclear arsenal. Canadians must, therefore, consider the risk that AECL’s marketing of the CANDU 6 could contribute to the proliferation of nuclear weapons, albeit inadvertently. In contemplating that risk, it should be noted that growth in the number of nuclear-weapon states could increase the probability of nuclear war, in part by expanding the number of decision centers. Canada has experience in inadvertently contributing to nuclear-weapon proliferation, having supplied the CIRUS research reactor to India in the 1950s, with the condition that the reactor be used only for peaceful purposes. In fact, India produced plutonium in CIRUS for its 1974 test of a nuclear weapon, and for subsequent nuclear weapons.

LINKAGE BETWEEN RISKS AT GENTILLY 2 AND RISKS AT OTHER CANDU 6 PLANTS

The risk of a violent power excursion at a CANDU 6 plant, and the risk of diversion of spent fuel, both derive from the plant's use of natural uranium as fuel and heavy water as coolant and moderator. The risk of a violent power excursion exists at Gentilly 2 and all CANDU 6 plants, with some local differences. By contrast, the risk of diversion of spent fuel is highly country-specific. At present, there is little prospect of Canada using plutonium from Gentilly 2 in nuclear weapons. The same cannot be said for every country where CANDU 6 plants exist or might be built. AECL would undoubtedly use the refurbishment of Gentilly 2 as an asset in its marketing of the CANDU 6. Thus, in weighing the costs and benefits of refurbishing Gentilly 2, citizens of Quebec and other parts of Canada are obliged to consider not only the risk of an unplanned release, but also the risk of contributing to nuclear-weapon proliferation.

REGULATORY RISK ASSOCIATED WITH REFURBISHMENT OF GENTILLY 2

The CNSC criteria for approving license extensions for Canadian CANDUs are vague. It is difficult to determine the stringency with which the CNSC will apply those criteria to license extensions, and the extent to which all plants seeking license extensions will be treated equally. That uncertainty reflects a current tension within the CNSC between its traditional regulatory approach, which has ad hoc and incestuous qualities, and a more modern and professional approach. If the professional approach gains influence, then any licensee seeking a CANDU license extension will be obliged to conduct lengthy and expensive studies on shutdown-system effectiveness and other matters. The licensee could be required to implement safety-enhancing measures, which could be costly.

In view of these considerations, Hydro-Quebec faces significant regulatory uncertainty regarding the extension of the Gentilly 2 operating license. If the CNSC takes a uniform, professional approach to all license extensions, Hydro-Quebec will be obliged to make substantial expenditures on safety studies that could reveal needs for costly modifications of the plant. Also, delays pursuant to CNSC requirements could arise during the refurbishment of Gentilly 2. Hydro-Quebec has already stated that the economic case for refurbishment and life extension of Gentilly 2 is weak. Accounting for regulatory uncertainty could further weaken that case. Additional weakening could come from consideration of the risk of onsite economic impacts from fuel-damage events.

Hydro-Quebec announced its plan to refurbish Gentilly 2 without waiting to complete studies on safety issues and the potential to improve safety through plant modifications. Presumably, that action reflects Hydro-Quebec's judgment that the regulatory risk is low. In reaching that judgment, Hydro-Quebec was probably encouraged by the example of the Point Lepreau plant. Refurbishment of that plant, which is similar to Gentilly 2, is proceeding. However, ongoing studies on shutdown-system effectiveness and other matters related to CANDU safety could convince the CNSC that safety-enhancing measures – such as the use of low-enriched uranium fuel – should be required. The Point Lepreau plant might escape that requirement because the licensing process has moved further in that case. Taken together, the factors discussed here indicate that Hydro-Quebec's judgment about regulatory risk could be faulty in two respects. First, a trend to professionalism in the CNSC could lead to higher safety standards. Second, ongoing studies could reveal, within the next few years, needs for safety-enhancing measures at existing CANDU plants.

Recommendations

Each of the three categories of risk discussed in this report deserves a thorough assessment before Hydro-Quebec proceeds with its plan to refurbish Gentilly 2. Those assessments should be published, with limited exceptions for sensitive information. Openness and transparency are essential if the findings are to be credible. The assessments should be conducted soon, before substantial expenditures are made on refurbishment of Gentilly 2. Thorough assessments could show that refurbishment is neither cost-effective nor prudent.

The CNSC should require Hydro-Quebec to perform a full-scope probabilistic risk assessment that examines unrestrained reactivity excursions and other fuel-damage scenarios at Gentilly 2. A complementary study should assess the risks of unplanned releases caused by malevolent acts. Both studies should be available for independent review. Hydro-Quebec should be required to identify and characterize a range of risk-reducing options, including the use of low-enriched uranium fuel. Descriptions of the options and their effects on risk should be published.

The government of Canada should direct its relevant agencies, including the CNSC, to assess the risk that international marketing of the CANDU 6 will contribute to the risks of nuclear-weapon proliferation and nuclear war. That assessment should be published.

Hydro-Quebec should support the recommendations set forth above. In addition, Hydro-Quebec should independently assess the regulatory risk associated with refurbishment of Gentilly 2, and the risk of onsite economic impacts from fuel-damage events. Those assessments should inform a review of the costs and benefits of refurbishing Gentilly 2. The risk assessments and the cost-benefit review should be published.

Legislators in Quebec and across Canada should call for open assessments of risks, as described above. If the CNSC, the Canadian government and Hydro-Quebec do not perform thorough assessments, legislators should consider sponsoring alternative actions such as the conduct of independent hearings and studies.

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1. Introduction

Hydro-Quebec plans to refurbish the Gentilly Unit 2 nuclear power plant to extend its operating life to about 2040.[1] The Gentilly 2 plant was commissioned in 1983, and was built to a Canadian design known as the CANDU 6. This report discusses three categories of risk that are highly relevant to Hydro-Quebec's refurbishment plan. None of these risks has been properly assessed. An approach for systematic, public assessments of all three categories of risk is proposed here. Those assessments could provide important information to citizens in Quebec, across Canada, and around the world.

The concept of "risk" encompasses the probability and magnitude of an adverse impact on humans and the environment.[2] The three categories of risk that are discussed here involve potential adverse impacts in Quebec, other Canadian provinces, and other countries. One category is the risk of an unplanned release of radioactive material from a CANDU 6 plant, through accident or malevolence. The second is the risk that spent nuclear fuel will be diverted from a CANDU 6 plant and used to produce plutonium for nuclear weapons. The third is the risk of regulatory actions that would increase the costs of the refurbishment of Gentilly 2.

BACKGROUND

Gentilly 2 is the only nuclear power plant operating in Quebec, providing about 3 percent of the province's electricity. Hydro-Quebec plans to refurbish Gentilly 2 at an estimated cost of about Can$1.9 billion.[3] The economic case for refurbishment and continued operation is weak, according to Hydro-Quebec.[4] Refurbishment of CANDU plants at the Pickering and Bruce sites in Ontario has been substantially more expensive than expected.[5] Post-refurbishment operation of the Gentilly 2 plant must be approved by the Canadian Nuclear Safety Commission (CNSC).

The CANDU design concept for a nuclear power plant was developed in Canada, beginning in the 1950s. Most CANDU plants now operating are in two groups. One group consists of the CANDU 6 plants, including Gentilly 2. The other group consists of commercial CANDU plants built in Ontario.[6] The Ontario plants share the same basic reactor technology as the CANDU 6, but their support and safety systems (e.g., containment) are significantly different. India has a number of nuclear power plants of indigenous manufacture, based on a CANDU design. They are not classified as CANDUs.

Atomic Energy of Canada Limited (AECL), a Crown corporation, has built two CANDU 6 plants in Canada – Gentilly 2, and the Point Lepreau plant in New Brunswick. In addition, nine CANDU 6 plants supplied by AECL are operating in Argentina, China, Romania and South Korea.[7] AECL hopes to build new CANDU 6 plants around the world, and is pursuing opportunities in Argentina, Jordan, Romania, Turkey and elsewhere. At the same time, AECL is developing a new version of the CANDU known as the ACR-1000, and hopes to sell this version in Canada and elsewhere.

Like most nuclear power plants now operating worldwide, the CANDU plants now operating are in the "Generation II" category. That designation distinguishes the present plants from the first generation of nuclear power plants, most of which have been closed. The great majority (80 percent) of the Generation II plants employ light-water reactors (LWRs) that are moderated and cooled by light water. Plants constructed during the next few decades would be in the Generation III category. The CANDU 6 design is from Generation II. An ACR-1000 plant, if one is built, would be from Generation III.

1 Throughout this report, the term "nuclear power plant" means a nuclear fission reactor and its associated equipment, including equipment to produce electricity.
2 The term "risk" is sometimes used to refer to the arithmetic product of: (i) a quantitative indicator of adverse impact; and (ii) the quantitative probability that the impact will occur. In this report, the term is used in a more general sense, to encompass a range of qualitative and quantitative information about the potential for an adverse outcome.
3 Hydro-Quebec, 2008.
4 Pageau, 2004, pages 4 and 5 of attached document.
5 WNA, 2008, Canada data.
6 Two of the twenty commercial CANDUs built in Ontario have been closed, and two are now undergoing refurbishment.
7 See Table 2-3.

SUSTAINABILITY AS A GUIDING PERSPECTIVE

This report examines three categories of risk associated with one type of nuclear power plant, focusing on a particular site in Quebec. Issues raised here should be viewed in a broader context. There is much contemporary discussion about human-induced climate change and its connection with energy systems. That discussion is part of a broader challenge – ensuring that our civilization is sustainable. In setting out guidelines for an environmental assessment of a proposed new nuclear power plant at the Bruce site in Ontario, the Canadian Environmental Assessment Agency (CEAA) acknowledged the importance of sustainability. CEAA approached sustainability through the concept of sustainable development, defined as follows:[8] "Sustainable development seeks to meet the needs of present generations without compromising the ability of future generations to meet their own needs." That definition was first articulated by the World Commission on Environment and Development in 1987.

Sustainability of engineered systems, such as nuclear power plants, is a large subject.[9] Indeed, developing, refining and applying the principles of sustainability are likely to be major preoccupations of humanity throughout the 21st century. There is no generally accepted, overall framework of sustainability principles, and no prospect of such a framework emerging soon. Yet, there is consensus among governments and international agencies that any new, large, long-lived engineered system should be designed according to sustainability principles. A refurbished Gentilly 2 would be, in effect, a new system of that kind. Hydro-Quebec expects that Gentilly 2 would operate until about 2040 if refurbished. Therefore, the merits of the refurbishment should be evaluated according to the sustainability needs and standards of the mid 21st century, to the extent that these can be predicted now. This author has discussed sustainability issues related to nuclear power plants, in another report.[10]

SCENARIOS FOR FUTURE USE OF NUCLEAR POWER WORLDWIDE

Hydro-Quebec has conceded that the economic case for refurbishment of Gentilly 2 is weak. The case could be further weakened by thorough assessments of the risks identified here. Those assessments could provide useful guidance, but there would be irreducible uncertainty in their findings. This climate of uncertainty is not unique to Gentilly 2, but applies to nuclear power in general. Nuclear power is now in a transitional phase. Annual, worldwide capacity additions peaked in 1985 and have been modest since 1990.[11] If construction of nuclear power plants does not resume, total capacity will decline as plants are retired. Observers view this situation in widely differing ways. Some call for a nuclear power "renaissance" in which nuclear generating capacity rises substantially. Others prefer or expect a scenario in which nuclear capacity declines, leading to eventual disappearance of the industry.

The most ambitious visions of the nuclear renaissance are exemplified by a "technology roadmap" issued under the auspices of the US Department of Energy in 2002.[12] The roadmap proposed the development and use of a range of Generation IV nuclear fission reactors that would push against engineering limits in a variety of respects. Some reactor types would produce hydrogen as well as electricity, thereby providing fuel for use in vehicles and other applications. Reactors would be deployed in such large numbers that uranium reserves would become depleted during the latter part of the 21st century. To prepare for that eventuality, large-scale reprocessing of spent nuclear fuel would begin during the next few decades, and breeder reactors would be deployed beginning in about 2030.

A less extreme but still highly ambitious vision of the nuclear renaissance is contained in a study published under the auspices of Massachusetts Institute of Technology (MIT) in 2003.[13] The authors saw no need for reprocessing or breeder reactors during at least the next 50 years. They offered an illustrative scenario for expansion of nuclear capacity using Generation III reactors whose designs would involve a comparatively small evolutionary step from the designs of present reactors. In the scenario, annual worldwide production of nuclear-generated electricity would rise by a factor of 4 to 6 between 2000 and 2050.

Many observers doubt the merits of nuclear power, and seek or expect a decline in its use.[14] Some argue that nuclear power can and should be phased out, even during an effort to dramatically reduce greenhouse gas emissions from electricity generation.[15] Others argue that scenarios for expansion of nuclear capacity are fanciful, and that the commercial nuclear industry is in terminal decline.[16]

8 CEAA, 2008, Section 2.4.
9 The term "engineered system" is used here to describe a system that is deliberately created or assembled by humans to serve specified functions.
10 Thompson, 2008a, Section 2.
11 IAEA, 2006a.
12 NERAC/GIF, 2002.
13 Ansolabehere et al, 2003.
14 Romm, 2008.
15 Makhijani, 2007; Greenpeace International, 2007.
16 Schneider and Froggatt, 2007.

STRUCTURE OF THIS REPORT

The remainder of this report has eight narrative sections and a bibliography, as shown in the table of contents. Conclusions and recommendations are provided in Section 9. All documents cited in this report are listed in the bibliography. Tables and figures, numbered according to the relevant section of the report, appear at the end of the report.

2. CANDU Nuclear Power Plants, and the Role of the CANDU 6

2.1 Scope of this Discussion

Sections 2.2 through 2.4, below, provide a brief introduction to CANDU nuclear power plants in general, and the CANDU 6 design in particular. Further details can be obtained from sources cited here.

2.2 History of the CANDU Design Concept

The term "CANDU" is a trademark of AECL that stands for CANada Deuterium Uranium. That term expresses three key features of the CANDU design concept. First, this concept is indigenous to Canada. Second, heavy water (deuterium oxide) is used as the moderator and, with exceptions discussed below, as the primary coolant. Third, the fuel is uranium. Natural (un-enriched) uranium has always been used to fuel CANDU reactors until recently. Now, the CANDU reactors at the Bruce A and B stations in Ontario are being switched to use of low-enriched uranium fuel.[17] Similar fuel would be used in the proposed ACR-1000 version of the CANDU.

RESEARCH AND DEVELOPMENT DURING WORLD WAR II

Canada acquired capabilities in nuclear science and engineering through military research and development conducted during World War II.[18] After the war, the Canadian government continued to invest in these capabilities. Work during and after the war led to the operation of three research reactors in Canada by the mid-1950s. The ZEEP reactor, with a capacity of 10 Wt, achieved criticality in 1945. The NRX reactor (42 MWt) and the NRU reactor (200 MWt) achieved criticality in 1947 and 1957, respectively. Each of these reactors was moderated by heavy water, setting a precedent that was followed by the CANDU reactors.

DEVELOPMENT OF THE CANDU

CANDU was conceived as a power reactor that could be built using Canadian technology, and whose fuel cycle would be indigenous. Canada has never had a capability to enrich uranium. Thus, natural uranium was chosen as the fuel. Heavy water was chosen as the moderator, drawing upon experience with the three research reactors mentioned above. On-line refueling was necessitated by the use of natural uranium as fuel. Those features led to a reactor configuration in which the fuel assemblies are in channels (pressure tubes) that are surrounded by a tank (the calandria) containing heavy water (the moderator). Primary coolant passes through the interior of the pressure tubes. The moderator and the primary coolant do not mix. Heavy water is used as the primary coolant, except as discussed below. With the exception of the unsuccessful Gentilly 1 plant, discussed below, the primary coolant is in the liquid phase. It is circulated through steam generators that transfer heat to the secondary coolant (light water), which is converted to steam that is fed to a turbo-generator.

A small (24 MWe) demonstration version of the CANDU – the NPD plant – entered service in 1962. The NPD plant was followed by a 220 MWe commercial prototype of the CANDU – the Douglas Point plant – that entered service in 1968. Next came the Pickering A station in Ontario, comprising four CANDU units that each had a capacity of 540 MWe. Those units entered service in the period 1971-1973. The Pickering A station employed an unusual design in which safety systems – including a containment system featuring a vacuum building – were shared among all units at the station. The same design approach was then used for another four CANDU stations that were built in Ontario. Each station has four units, with unit capacities as follows: Pickering B (540 MWe); Bruce A (900 MWe); Bruce B (915 MWe); and Darlington (935 MWe). Those units entered service during the period 1977-1993.

17 CNSC, 2008e, page 22.
18 This paragraph and the following paragraphs in Section 2.2 draw from: Brooks, 2001; and AECL, 2007.

The multi-unit stations built in Ontario represent one branch of the CANDU evolutionary tree, notable for the sharing of safety systems among units. CANDU 6 represents a related but different branch.[19] The basic reactor technology is the same as at the Pickering, Bruce and Darlington stations. However, the CANDU 6 units are primarily stand-alone plants, as are most nuclear power plants worldwide. Thus, for example, each CANDU 6 plant has its own containment, and there is no separate vacuum building. Two CANDU 6 plants have been built in Canada. The Point Lepreau plant (680 MWe) and the Gentilly 2 plant (675 MWe) both entered service in 1983. Additional CANDU 6 plants have been built in Argentina, China, Romania and South Korea, as discussed in Section 2.4, below.

THE GENTILLY 1 EXPERIENCE

Another branch of the CANDU evolutionary tree, represented by the Gentilly 1 plant, was a technical failure. In the Gentilly 1 version of the CANDU, the primary coolant was light water, which boiled inside the pressure tubes (fuel channels). The pressure tubes were vertical, whereas at all other CANDU plants the pressure tubes are horizontal. The Gentilly 1 plant was a prototype unit with a design capacity of 250 MWe. It first went critical in 1972 but never operated successfully, and was permanently shut down in 1977. The principal problems with the design related to control of the fission reaction. The reactor core had positive power and void coefficients of reactivity. As a result, boiling in a pressure tube caused an increase in power output that led to more boiling, which further increased the power output. According to a CNSC Staff presentation in 2008, "the resulting spatial power oscillations were expected to be intolerable".[20] Moreover, as discussed in Section 4.3, below, analysis performed during the construction of Gentilly 1 showed that, in the event of a loss of coolant and failure of the shutdown system, the reactor would self-destruct with sufficient violence to threaten the integrity of containment. An event of that type occurred at the Chernobyl Unit 4 plant in the USSR in 1986, although in that case the reactor was moderated by graphite rather than heavy water.[21]

RECURRING PROBLEMS WITH REACTIVITY

Curbing the potential for power excursions and oscillations has been a recurring challenge for designers and operators of Canadian reactors. That potential was vividly demonstrated at the NRX reactor in December 1952. As a result of operator errors and design deficiencies, the reactor experienced a reactivity excursion in which the power output rose briefly to 2 or 3 times the rated level. Melting of fuel assemblies occurred, accompanied by a hydrogen explosion. The core and calandria were damaged beyond repair. Those components were removed and replaced, and the reactor resumed operation about 14 months after the accident.[22]

The NRX reactor had a positive void coefficient of reactivity, as has every CANDU reactor built to date. This feature of the CANDU design, and its implications, are discussed further in Section 4.3, below. The void coefficient of reactivity is relevant to reactors cooled by water (heavy or light), as are most power reactors worldwide. A water-cooled reactor possessing a positive void coefficient has a natural propensity to undergo a power surge if voids (steam, etc.) grow in the primary coolant (water). The CANDU shares this feature with the RBMK reactor design, which was used in the USSR. Chernobyl Unit 4 was an RBMK. LWRs, which make up 80 percent of the worldwide fleet of nuclear power plants, have a negative void coefficient.

THE POSITIVE VOID COEFFICIENT AND CANDU LICENSING

The existence of a positive void coefficient has become an issue that significantly affects the licensing of CANDU reactors, in Canada and potentially elsewhere. Until recently, the Canadian nuclear industry did not attempt to address this issue by changing the basic design of the CANDU reactor. Instead, the industry relied on the use of fast-acting shutdown systems to override a CANDU reactor's propensity to undergo a power surge if coolant voids were to occur.[23] Currently, the industry has two initiatives under way that address the issue at a more basic level. One initiative is to use a new type of reactor fuel known as low void reactivity fuel (LVRF), which uses low-enriched uranium rather than natural uranium. Bruce Power intends to begin using LVRF at the Bruce stations in 2009, gradually switching all of the Bruce units to this type of fuel.[24] It appears that LVRF could be used at any existing CANDU plant. Through use of this fuel, the void coefficient of reactivity would be reduced in magnitude, but would remain positive.

19 AECL, 2005.
20 CNSC, 2008c, page 53.
21 NRC, 1987.
22 See Section 4.3.
23 See Section 4.3.
24 CNSC, 2008e, page 22.

The second initiative is AECL's effort to develop and sell a new version of the CANDU known as the ACR-1000.[25] In that version, low-enriched uranium would be used as fuel, and the primary coolant would be light water. As at other CANDUs except Gentilly 1, the primary coolant would not boil during normal operation. AECL claims that the ACR-1000 would have a negative void coefficient of reactivity. However, AECL's ability to achieve its design objectives for the ACR-1000, including a negative void coefficient, is thrown into question by the scrapping of the two MAPLE reactors at AECL's Chalk River laboratories in May 2008. These were 10 MWt, pool-type reactors employing low-enriched uranium fuel, and were intended to produce medical isotopes. They were designed and built by AECL. They first went critical in 2000 and 2003, and had been undergoing commissioning since then. AECL eventually concluded that the reactors were unfit to operate, and that their deficiencies could not be rectified within any reasonable budget and timeframe. It appears that the deficiencies included sticking of control rods and shutoff rods, and a positive power coefficient of reactivity that could produce power excursions.[26]

The CNSC has not established a clear position as to whether it would license a new nuclear power plant if the plant has a positive void coefficient of reactivity. If a positive void coefficient were unacceptable, a new CANDU 6 could not be licensed in Canada. By contrast, an ACR-1000 plant could be licensed if it achieved its design objectives. The robustness of containment is another licensing issue where an ACR-1000 might be acceptable to the CNSC while a new CANDU 6 might not be acceptable. These matters are discussed further in Section 5, below.

EXPORT OF CANDUS TO INDIA AND PAKISTAN

Canada's nuclear industry has always had a strong interest in exporting nuclear reactors, and has supplied reactors to a number of countries. That experience has clearly shown the connection between trade in nuclear technology and the proliferation of nuclear weapons. Early lessons were provided by the cases of India and Pakistan. These and other cases are discussed further in Section 4.4, below.

As part of an aid program, Canada supplied India with the CIRUS research reactor, a near copy of the NRX reactor with a capacity of 40 MWt. CIRUS entered service in 1960. Although Canada supplied CIRUS on the condition that it be used only for peaceful purposes, India produced plutonium in CIRUS for its 1974 nuclear-weapon test and for subsequent nuclear weapons.

Canada also supplied India with two CANDU nuclear power plants – RAPP 1 and RAPP 2. Their design was patterned on the Douglas Point plant, with modifications to allow supply of components by Indian manufacturers. RAPP 1 entered service in 1973. The Indian nuclear test of 1974 ended Canada's involvement with these plants. India completed RAPP 2 without Canadian help, and that plant entered service in 1983. Based on that experience, India built additional plants of indigenous manufacture, employing the CANDU design concept. At present, there are fifteen CANDU-type plants in India, including RAPP 1 and RAPP 2.[27]

Pakistan also requested CANDU technology from Canada. Their request led to construction of the KANUPP plant in Pakistan by a Canadian company, on a turnkey basis. KANUPP is a CANDU plant with a capacity of 130 MWe, which entered service in 1972 and remains operational.[28]

2.3 Design Features of the CANDU 6

A CANDU 6 plant employs the same basic technology as has been used in the Ontario CANDU plants.[29] Unlike those plants, however, the CANDU 6 does not extensively share safety systems among reactors. Notably, each CANDU 6 plant has its own containment structure, and there is no vacuum building. Each of the two CANDU 6 plants in Canada is an entirely stand-alone unit.[30]

Some design data for a typical CANDU 6 plant are shown in Table 2-1. The table also shows comparable data for an ACR-1000 plant. Notable differences between these plants include the use of natural-uranium fuel and heavy-water coolant in the CANDU 6, compared to low-enriched uranium fuel and light-water coolant in the ACR-1000. Also, in the ACR-1000, fuel would be driven to a burnup (over 20 MWt-day per kgU) substantially higher than the typical CANDU 6 fuel burnup (7.5 MWt-day per kgU).

25 AECL, 2007.
26 MacKenzie, 2008.
27 WNA, 2008, India data.
28 WNA, 2008, Pakistan data.
29 AECL, 2005.
30 Stations featuring up to four CANDU 6 plants have been built outside Canada. Those stations may employ some sharing of safety systems such as emergency diesel generators. That level of design detail was not examined during preparation of this report.

The containment of the CANDU 6 is a domed concrete cylinder with a wall thickness of about 1.1 m, and an epoxy liner. The ACR-1000 would have a similar containment, although in that case the wall thickness would be 1.8 m and there would be a steel liner. Thus, the containment of the ACR-1000 would be somewhat less vulnerable to external attack than is the CANDU 6 containment. Vulnerability of a containment to attack is discussed further in Section 3.2, below. Both containments are roughly similar, in their configuration and robustness, to the containment of a typical Generation II pressurized-water reactor (PWR). The PWR is the most common type of LWR. The Indian Point 2 plant, in New York state, is a typical Generation II PWR. Its containment is a domed concrete cylinder with a wall thickness of 1.4 m, lined by steel 1 cm thick.[31]

Table 2-2 shows the core inventories of selected radio-isotopes in the cores of the Gentilly 2 and Indian Point 2 reactors. Iodine-131 represents a category of comparatively short-lived radio-isotopes that would dominate the decay heat produced in a core during the first several days after reactor shut-down, and that would dominate the radiation dose that persons exposed offsite would receive during that period if an unplanned release to the atmosphere were to occur. Cesium-137 is a comparatively long-lived radio-isotope that would, in the event of a large unplanned release to the atmosphere, dominate the lifetime radiation dose that exposed persons would receive due to contamination of land, buildings, vegetation and water.[32] Gentilly 2 has a lower normalized core inventory of Cesium-137 than does Indian Point 2, because the Gentilly 2 fuel is driven to a lower burnup.

2.4 Existing and Potential Future CANDU 6 Plants

Table 2-3 shows the CANDU 6 nuclear power plants that are operating worldwide. AECL hopes to build new CANDU 6 plants, and is pursuing sales opportunities in Argentina, Jordan, Romania, Turkey and elsewhere.[33] AECL's corporate plan discusses the possibility of selling twenty CANDU plants around the world over the next 25 to 30 years.[34] These could be CANDU 6 or ACR-1000 plants. However, no ACR-1000 plant has been ordered to date, and completion of the ACR-1000 design has placed a heavy financial burden on AECL.[35] Thus, AECL has an incentive to sell a number of CANDU 6 plants, to improve its cash flow.

CANDU 6 AND THE POTENTIAL FOR INDIGENOUS FUEL CYCLES IN TURKEY AND JORDAN

The CANDU 6 sales opportunities in Turkey and Jordan are notable because, in each case, the host government is particularly interested in obtaining reactors that are fueled by natural uranium. The Turkish government has specifically stated that it will consider proposals for heavy-water reactors on the condition that they are fueled by natural uranium.[36] That requirement would allow a CANDU 6 plant but would exclude an ACR-1000. AECL and Jordan's Atomic Energy Commission have agreed to collaborate on studies of the feasibility of: (i) building a CANDU 6 plant in Jordan; and (ii) fabricating fuel in Jordan for that plant using locally-mined uranium.[37] A Canadian consultant has attributed Turkey's and Jordan's interest in the use of natural-uranium fuel to a desire for "fuel security", freeing these countries from dependence on outside suppliers of enrichment services.[38]

As discussed in Sections 3.3 and 4.4, below, there could be an unstated motive for the acquisition of an indigenous nuclear fuel cycle involving CANDU 6 reactors. Such a fuel cycle could establish a reserve capability to build nuclear weapons. Yet, despite Canada's experience with the CIRUS reactor in India, neither the Canadian nuclear industry nor the Canadian government has acknowledged that Turkey and Jordan are in a region with a high risk of nuclear-weapon proliferation. Three cases illustrate the risk. First, Israel has nuclear weapons. Second, Iraq sought the capacity to build such weapons until its effort was forcibly terminated by the 1991 Gulf War. Third, Iran is developing an indigenous capacity for uranium enrichment, which many observers believe is intended to provide a reserve capability to build nuclear weapons. The potential for a regional nuclear arms race is clear. Canada could inadvertently contribute to such an arms race by supplying CANDU 6 plants to the region.

31 Thompson, 2007, Section 7.5.
32 Cesium-137 accounts for most of the offsite radiation exposure of humans that is attributable to the 1986 Chernobyl reactor accident. See: DOE, 1987.
33 Stellfox, 2008.
34 AECL, 2008a.
35 Purchase, 2008.
36 TAEK, 2008.
37 AECL, 2008b.
38 Hamilton, 2008.

The Turkish government solicited bids, for submission in September 2008, for construction of the country's first group of nuclear power plants. A Russian vendor was the only bidder. AECL and other vendors declined to bid. It does not appear that concern about nuclear-weapon proliferation was a factor in AECL's response. Instead, it appears that AECL and other vendors were concerned about commercial issues, including legal constraints on the Turkish government's direct involvement with the project. The Turkish government will review its legal position and extend the bidding period into 2009.[39]

THE ENHANCED CANDU 6

AECL makes reference to a potential new plant design known as the Enhanced CANDU 6.[40] Few details are publicly available. One change from the present CANDU 6 is a relatively small increase in gross power capacity, to 740 MWe. Other design changes, also relatively small, are said by AECL to improve the safety and security of the plant. According to a January 2007 press story, a more significant design change was contemplated in the context of the possible sale of CANDU 6 plants in Ontario.[41] The change was to strengthen the containment and provide it with a steel liner, to improve resistance to aircraft impact. Apparently, that change was made at the insistence of the CNSC, creating friction between the CNSC leadership and Canada's central government. The government feared that strengthening of the containment would create delay in the possible building of CANDU 6 plants in Ontario, thus threatening AECL's market position. That confrontation illustrates the problems that arise when a government regulator (CNSC) makes decisions that affect a government-owned vendor (AECL). The issue was eventually resolved when the Ontario government announced, in March 2008, that it would accept bids for ACR-1000 plants but not Enhanced CANDU 6 plants.[42]

USE OF THE CANDU 6 IN ALTERNATIVE FUEL CYCLES

Figure 2-1 shows AECL's vision of the possible use of the CANDU plant concept in alternative nuclear fuel cycles.[43] At present, CANDUs are fueled by natural uranium. The CANDUs at the Bruce site are being switched to low-enriched uranium, and any ACR-1000 plants that are built would be fueled by low-enriched uranium.

One of the fuel cycles envisioned by AECL involves the dry processing of spent LWR fuel so that its fissile uranium and plutonium constituents can be burned in CANDUs. In this arrangement, known as the DUPIC cycle, spent LWR fuel would be crushed and heated in oxygen to remove about 40 percent of its inventory of fission products, and the residue would be formed into fuel that would be burned in CANDUs. AECL has established an agreement with the government of Ukraine to study the application of the DUPIC cycle in Ukraine. South Korea has been working since 1992 to demonstrate the DUPIC concept, and is currently cooperating with AECL in this area.[44]

A country that employs the DUPIC cycle would acquire capabilities that would assist the country to develop, if desired at a later date, a capacity for reprocessing of spent fuel. Thus, the DUPIC cycle could contribute indirectly to the proliferation of nuclear weapons.

39 Turkish Daily News, 2008; Nuclear News Flashes, 2008.
40 AECL, 2008c.
41 Hamilton, 2007.
42 Infrastructure Ontario, 2008.
43 Also see: AECL, 2005, pp 54-57.
44 WNN, 2008.

3. Risks of Operating a Nuclear Power Plant

3.1 Scope of this Discussion

This report discusses three categories of risk, defined in Section 1, above. Sections 3.2 through 3.5, below, provide a broad review of two of these categories – the risk of an unplanned release (radiological risk), and the risk of diversion of spent fuel (proliferation risk). This review applies to nuclear power plants generally. Section 4, below, continues the discussion with specific application to CANDU plants, especially the CANDU 6.

Sections 3.2 and 3.3 discuss risks that arise from typical operation of nuclear power plants. Measures that could reduce those risks are discussed in Sections 3.4 and 3.5. Each of these four sections discusses complex issues for which there is a large literature. The discussion here is at an introductory level.

3.2 Risk of an Unplanned Release of Radioactive Material

There is a large body of technical literature addressing the risk of an unplanned release of radioactive material from or within a nuclear power plant. The release would arise from accidental damage to nuclear fuel in the reactor core, the spent-fuel pool, or elsewhere in the plant. Much of the literature assesses the potential for, and offsite consequences of, an atmospheric release. Smaller, related bodies of literature consider the onsite consequences of a release, and the offsite consequences of a water-borne release.

THE ROLE OF PRA

Literature about the risk of an unplanned release typically falls under the rubric of probabilistic risk assessment (PRA). The term probabilistic safety assessment (PSA) is often used as a synonym for PRA, but can actually carry a different connotation.[45]

In the PRA field, the events that initiate an accidental release are categorized as “internal” events (human error, equipment failure, etc.) or “external” events (earthquakes, fires, strong winds, etc.). PRAs typically do not consider initiating events that involve intentional, malevolent acts, although PRA techniques can be adapted to estimate the outcomes of such acts.

PRAs for nuclear power plants are conducted at Levels 1, 2 and 3, in increasing order of completeness, as discussed below. A thorough, full-scope PRA would be conducted at Level 3, and would consider internal and external initiating events. The findings of such a PRA would be expressed in terms of the magnitudes and probabilities of a set of adverse environmental impacts, and the uncertainty and variability of those indicators. The adverse impacts would include:

(i) “early” human fatalities or morbidities (illnesses) that arise during the first several weeks after the release;

(ii) “latent” fatalities or morbidities (e.g., cancers) that arise years after the release;

(iii) short- or long-term abandonment of land, buildings, etc.;

(iv) short- or long-term interruption of agriculture, water supplies, etc.; and

(v) social and economic impacts of the above-listed consequences.

The magnitudes and probabilities of such adverse impacts would be estimated in three steps. First, a Level 1 PRA analysis would be performed. In that analysis, a set of event sequences (accident scenarios) leading to fuel damage would be identified, and the probability (frequency) of each member of the set would be estimated. The sum of those probabilities across the set would be the total estimated fuel-damage probability.[46]

45 The term PRA implies that risk will be identified, whereas the term PSA can imply that safety will be demonstrated. That difference in intent can lead to significant differences between the structure, scope and findings of PRA studies and PSA studies.

46 The term “core-damage frequency” (CDF) is often encountered. This term refers to the annual probability of severe damage to nuclear fuel in a reactor core.

Second, a Level 2 PRA analysis would be performed. In that analysis, the potential for release of radioactive material within the plant and to the environment would be examined across the set of fuel-damage sequences. A high priority of the Level 2 analysis would be to estimate the characteristics of an atmospheric release, sometimes known as the "source term". That estimate would be expressed in terms of a group of atmospheric release categories characterized by magnitude, probability, timing, isotopic composition, and other characteristics.

Third, a Level 3 PRA analysis would be performed, to yield the impact findings described above. In that analysis, the atmospheric dispersion, deposition and subsequent movement of the released radioactive material would be modeled for each of the atmospheric release groups determined by the Level 2 analysis. The dispersion modeling would account for meteorological variation over the course of a year. Then, the adverse environmental impacts of the released material would be estimated, accounting for the material's distribution in the biosphere. In a thorough analysis, onsite impacts of releases would also be assessed, as would the offsite impacts of water-borne releases if those impacts were significant.

If done thoroughly, this 3-step estimation process accounts for uncertainty and variability at each stage of the process. A thorough, full-scope, Level 3 PRA is expensive and time-consuming. It yields estimated impacts expressed as statistical distributions of magnitude and probability, not as single numbers. Even after such a thorough effort, there are substantial, irreducible uncertainties in the findings.[47]

EMPIRICAL VALIDATION OF PRA FINDINGS

Direct empirical evidence for the validity of PRA findings is limited. Worldwide operating experience of commercial nuclear power plants through 2007 is about 12,900 reactor-years (RY), and Canadian experience is about 560 RY.[48] Two events involving substantial damage to a reactor core have occurred worldwide while that experience was accruing. At Three Mile Island (TMI) Unit 2 in 1979, the reactor core was severely damaged but there was a comparatively small radioactive release to the environment. At Chernobyl Unit 4 in 1986, a substantial fraction of the core inventory of radioactive material was released to the atmosphere. This limited experience allows one to estimate the probability of a core-damage accident as 1.6 per 10,000 RY, and the probability of a large atmospheric release as 0.8 per 10,000 RY.[49]

NUREG-1150

The "high point" of PRA practice worldwide was reached in 1990 with publication by the US Nuclear Regulatory Commission (NRC) of its NUREG-1150 study, which examined five different nuclear power plants using a common methodology.[50] The study was well funded, involved many experts, was conducted in an open and transparent manner, was done at Level 3, considered internal and external initiating events, explicitly propagated uncertainty through its chain of analysis, was subjected to peer review, and left behind a large body of published documentation. Each of those features is necessary if the findings of a PRA are to be credible. There are deficiencies in the NUREG-1150 findings, which can be corrected by fresh analysis and the use of new information. The process of correction is possible because the NUREG-1150 study was conducted openly and left a documentary record.

PRA practice in the USA has degenerated since the NUREG-1150 study. Now, PRAs are conducted by the nuclear industry, and the only published documentation is a summary statement of findings. NRC formerly sponsored independent reviews of industry PRAs, but no longer does so. Thus, PRA findings have lacked credibility for at least a decade. In other countries, including Canada, PRA practice has experienced similar degeneration.[51]

47 Hirsch et al, 1989.

48 Extrapolated from Table 1 of: IAEA, 2006a. A reactor-year (RY) is equivalent to a plant-year, using this report's definition of a nuclear power plant. Both terms assume routine operation of a reactor (plant) over one calendar year.

49 2/12,900 = 1.6 per 10,000; 1/12,900 = 0.8 per 10,000.

50 NRC, 1990.

51 In Canada, PRAs are no longer available for independent review. To illustrate, Greenpeace Canada requested a copy of the PRA for the Pickering B units. CNSC has refused to order Ontario Power Generation (OPG) to provide this PRA. In so doing, CNSC has accepted OPG's argument that the PRA should be available only to OPG personnel on a "need to know" basis. See: CNSC, 2008a. This approach, although it may be well-intentioned, will inevitably create an entrenched culture of secrecy that will suppress a clear-headed understanding of risks. A more sophisticated approach could allow independent review of the PRA without disclosing information that would assist malevolent actors.

Figures 3-1 through 3-3 show some findings from the NUREG-1150 study that are relevant to this report. The findings are for a PWR plant at the Surry site, and a boiling-water reactor (BWR) plant at the Peach Bottom site. These plants typify many of the Generation II plants in the present worldwide fleet of nuclear power plants. Using the Livermore seismic estimates, the NUREG-1150 findings for these two plants are roughly comparable with the experience-derived probability estimates mentioned above – a core-damage probability of 1.6 per 10,000 RY, and a large-release probability of 0.8 per 10,000 RY.

THE POTENTIAL FOR MALEVOLENT ACTS AT NUCLEAR POWER PLANTS

No nuclear power plant now operating in the world was specifically designed to resist malevolent acts. Plants have some inherent capability to resist such acts, by virtue of their design to meet other challenges (e.g., earthquake). Over time, security measures have been introduced to provide some further protection against malevolent acts. Those measures have become more stringent since commercial aircraft were used to attack buildings in New York and Washington in September 2001. In the USA, for example, plant security measures now include fences, gates, vehicle barriers, surveillance systems, armed guards, and enhanced control of plant personnel. There has been no significant change, however, in the design features of the existing plants.

For new nuclear power plants, design options are available that could make a plant more robust against attack – from outside or inside the plant – than are the existing plants. The Canadian Nuclear Safety Commission has, in establishing criteria for the design of new nuclear power plants, included resistance to attack as a design objective.[52] To date, however, the CNSC has not specified the threats that will be considered in applying the design criteria. The Canadian Environmental Assessment Agency has established guidelines for the preparation of an environmental assessment for the construction of new nuclear power plants at the Bruce site in Ontario. Those guidelines require the consideration of environmental impacts arising from malevolent acts.[53]

A consultant to the CNSC has examined potential modes and instruments of attack on a nuclear power plant, and has recommended an approach to incorporating these threats in the design criteria for new plants.[54] Among the instruments of attack considered by the consultant were a large commercial aircraft, an explosive-laden smaller aircraft, and an explosive-laden land vehicle. Table 3-1 describes some potential modes and instruments of attack on a nuclear power plant, and also describes the defenses that are now provided at US plants. There is no defense against a range of credible attacks. Defenses at Canadian plants are no more robust than at US plants.

Among the instruments of attack mentioned in Table 3-1 is a large commercial aircraft. In September 2001, aircraft of this type caused major damage to the World Trade Center and the Pentagon. However, such an aircraft would not be optimal as an instrument of attack on a nuclear power plant. Large commercial aircraft are comparatively soft objects containing a few hard structures such as turbine shafts. They can be difficult to guide precisely at low speed and altitude. A well-informed group of attackers would probably prefer to use a smaller, general-aviation aircraft laden with explosive material, perhaps in a tandem configuration in which the first stage is a shaped charge. Table 3-2 provides some information about shaped charges and their capabilities.

There is no statistical basis for a quantitative estimate of the probability that a nuclear power plant will be attacked. However, if a given attack scenario is postulated, one can apply PRA techniques to estimate the conditional probabilities of various outcomes. The NRC took that approach in developing its vehicle-bomb rule of 1994.[55]

RADIOACTIVE RELEASES FROM STORED SPENT FUEL

At nuclear power plants in the USA and elsewhere, large amounts of spent fuel are stored under water in pools adjacent to reactors. All US pools currently employ high-density racks, to maximize the amount of spent fuel that can be stored in each pool. This practice has been adopted because it is the cheapest mode of storage of spent fuel. Unfortunately, the high-density configuration would suppress convective cooling of fuel assemblies if water were lost from a pool.

52 The design criteria are set forth in CNSC Regulatory Document RD-337. A draft version of RD-337 (CNSC, 2007a) was published in October 2007. The CNSC Staff subsequently proposed a final version of RD-337 (See: Dallaire et al, 2008) that was apparently approved by the CNSC Commissioners in June 2008 but has not been published.
53 CEAA, 2008, Section 12.
54 Asmis and Khosla, 2007.
55 NRC, 1994.

Several reputable studies have agreed that loss of water from a pool would, across a range of water-loss scenarios, lead to spontaneous ignition of the zirconium alloy cladding of the most recently discharged fuel assemblies. The resulting fire would spread to adjacent fuel assemblies and propagate across the pool. Extinguishing the fire, once it had been initiated, would be difficult or impossible. Spraying water on the fire would feed an exothermic reaction between steam and zirconium. The fire would release a large amount of radioactive material to the atmosphere, including tens of percent of the pool's inventory of Cesium-137. Large areas of land downwind of the plant would be rendered unusable for decades. Loss of water could arise in various ways as a result of an accident or an intentional, malevolent act.[56]

This author is not aware of any study on the potential for an accidental release of radioactive material from spent fuel stored at a nuclear power plant employing a CANDU reactor. Absent such a study, the potential remains unknown.

ONSITE IMPACTS OF AN UNPLANNED RELEASE

An unplanned release of radioactive material at a nuclear power plant could create adverse impacts at the plant itself, whether or not the release reaches the environment. Plant personnel could receive radiation doses that yield adverse health effects, which could be translated into monetized costs. Additional costs could arise for site cleanup, repair of damaged portions of the plant, purchase of replacement power during the period when the plant is out of service, and write-off and decommissioning of the entire plant if repair is not cost-effective.

Ontario Hydro published an analysis of the risk of onsite economic impacts from an unplanned release at one of the Darlington CANDU units. That analysis is discussed in Section 4.2, below.

3.3 Risk of Diversion of Spent Fuel and Production of Plutonium

Nuclear power plants produce large amounts of plutonium, contained within their spent fuel. As shown in Table 3-3, it has been estimated that nuclear power plants worldwide will produce about 2.1 million kg of plutonium through 2010. Plants in Canada alone will produce almost 170 thousand kg of plutonium through 2010.

PLUTONIUM AND NUCLEAR WEAPONS

For comparison with the quantities shown in Table 3-3, note that the critical mass of a bare sphere of plutonium (pure Pu-239, alpha-phase) is about 10 kg. The radius of that sphere is about 5 cm. With addition of a natural uranium reflector about 10 cm thick, the critical mass would be reduced to about 4.4 kg, comprising a sphere with a radius of about 3.6 cm, the size of an orange. The critical mass could be further reduced using implosion techniques. An implosion device built to a modern design could achieve a nuclear explosion using 2 to 3 kg of plutonium.[57]

Nuclear warheads deployed by the nuclear-weapon states each contain, on average, about 3 to 4 kg of plutonium.[58] The world's inventory of military plutonium, at the end of 1994, was about 249,000 kg, mostly held by the former USSR and the USA. About 70,000 kg of that plutonium was in operational warheads.[59]

OBTAINING WEAPON-USABLE PLUTONIUM FROM REACTORS

Plutonium in the spent fuel discharged from a reactor is not directly available for use in a nuclear warhead. Three steps would be required before the plutonium could be used in that way. First, the plutonium would be chemically separated from other constituents of the spent fuel, notably fission products, actinides, and unburned uranium. Second, the plutonium would be converted to high-purity metal components of the correct composition and shape. Third, the plutonium components would be assembled with the other components needed to make a functioning warhead.

56 Alvarez et al, 2003; National Research Council, 2006; Thompson, 2007.
57 Barnaby, 1992.
58 Albright et al, 1997, page 34.
59 Albright et al, 1997, Table 14.2.

Some of the spent fuel produced by power reactors around the world undergoes the first step. This step occurs in commercial "reprocessing" plants where plutonium is chemically separated from spent fuel for use as a reactor fuel. Currently, reprocessing does not occur in Canada, although Canada acquired experience with this technology in the 1940s. The second and third steps occur only in nuclear-weapon countries. However, all three steps are within the capabilities of many industrialized countries. The technologies are decades old, and their principles are well known.

When uranium fuel undergoes irradiation in a fission reactor, the composition of the reactor's inventory of fissile nuclei changes over time. The process is illustrated in Figure 3-4. That figure applies specifically to Magnox reactor fuel, but the same principles apply to other types of fuel including CANDU fuel. It will be noted from Figure 3-4 that higher isotopes of plutonium – including Pu-240 and Pu-241 – are increasingly formed as fuel burnup increases. Weapon designers prefer to use plutonium with a high content of Pu-239, which requires the discharge of fuel at a low burnup – typically about 0.4 MWt-day per kgU.[60] The "weapon grade" plutonium in US nuclear warheads typically contains about 93 percent Pu-239 and 6.5 percent Pu-240.[61] Nevertheless, "reactor-grade" plutonium with a Pu-239 content of 60 percent could be used to make a functioning nuclear warhead.[62]

POTENTIAL PROLIFERATION OF NUCLEAR-WEAPON CAPABILITY

There is concern that a sub-national group might obtain separated plutonium and use that plutonium in a nuclear weapon. That concern has particular relevance to current programs for separating plutonium from spent fuel, and to the management of stocks of plutonium that were separated in previous years. In this report, the focus is on the risk that spent fuel discharged from a CANDU 6 plant would be diverted for the purpose of producing separated plutonium for use in weapons. Although such diversion by a sub-national group cannot be ruled out, a more likely scenario is diversion by the government of the host country. That specific risk is addressed further in Section 4.4, below.

There is an international regime of treaties, laws, institutions and practices that seeks to limit the number of nuclear-weapon countries and to achieve nuclear disarmament. One of the regime's functions is to discourage or prevent governments from using nuclear power plants to produce plutonium for use in weapons. The central pillar of the regime is the Treaty on the Non-Proliferation of Nuclear Weapons (NPT). Another pillar is the safeguards system operated by the International Atomic Energy Agency (IAEA).[63] The regime has faced many challenges, but has succeeded in limiting the size of the nuclear-weapon club. Informed observers now fear that emerging challenges will be greater than those faced previously, potentially causing a break-up of the regime during the coming years.[64]

If the nuclear non-proliferation regime breaks up or is significantly degraded, the number of nuclear-weapon countries will increase. The scale and timing of that increase cannot be predicted. It is clear, however, that an anticipated or actual increase in the number of nuclear-weapon countries could motivate the governments of many non-nuclear-weapon countries to consider how civil nuclear capabilities could support their deployment of nuclear arsenals. A government in that position could fear being left behind in a regional arms race, leading to a decision to develop an actual or reserve capability to deploy nuclear weapons.[65] Even if a country limited its action to the development of a reserve capability, that choice could significantly influence the country's investment in the civil nuclear fuel cycle. A country developing a reserve capability to deploy nuclear weapons would typically seek to ensure that it could produce plutonium and/or highly-enriched uranium using indigenous facilities.

NUCLEAR PROLIFERATION AND THE RISK OF NUCLEAR WAR

Proliferation of nuclear arsenals would cause adverse social and economic impacts in the affected countries. It would, for example, feed authoritarian political tendencies. The dominant adverse impact would, however, be an increased risk of nuclear war. Two factors would increase the probability of such a war. First, the number of decision centers would grow, increasing the chance of inadvertent or deliberate use of nuclear weapons. Second, newly-minted nuclear arsenals would tend to have comparatively unsophisticated command systems, increasing the chance of unauthorized or accidental use. Although the war might be regional rather than global, the consequences could be severe and could extend far beyond the affected region. A group of scientists who assessed the potential consequences of a regional nuclear war, including global climate impacts from smoke generated during the war, concluded:[66]

60 Albright et al, 1997, page 21.
61 Cochran et al, 1987, page 136.
62 Barnaby, 1992.
63 Fischer and Szasz, 1985.
64 Stanley Foundation, 2006.
65 In a region of the world where countries fear their neighbors, positive-feedback effects might drive a regional arms race.
66 Toon et al, 2007, page 1225.

"The analysis summarized here shows that the world has reached a crossroads. Having survived the threat of global nuclear war between the superpowers so far, the world is increasingly threatened by the prospects of regional nuclear war. The consequences of regional-scale nuclear conflicts are unexpectedly large, with the potential to become global catastrophes."

3.4 Options for Reducing the Risk of an Unplanned Release

As discussed in Section 3.2, above, an unplanned release of radioactive material at a nuclear power plant could be caused by an accident or a malevolent act. The risk of such an event encompasses the probability and magnitude of the release. At an existing plant, limited reductions in risk might be obtained by introducing new operating practices, and by making minor changes in the plant's configuration. For example, the potential for a malevolent act might be reduced through enhanced measures of site security.

The prospects for significant risk reduction are much greater at a new plant. If risk reduction were a high priority, a new plant could be designed according to highly stringent criteria of safety and security. During the 1970s and 1980s, some plant vendors and other stakeholders sought to develop designs that could meet such criteria. One design approach was to provide a highly robust containment – which might be an underground cavity – to separate nuclear fuel from the environment. Another approach was to incorporate principles of "inherent" or "intrinsic" safety into the design. The two approaches could be complementary.

UNDERGROUND SITING

In the 1970s, there were several studies on constructing nuclear power plants underground. Those studies are exemplified by a report published in 1972 under the auspices of the California Institute of Technology (Caltech).[67] The report identified a number of advantages of underground siting. Those advantages included highly-effective confinement of radioactive material in the event of a core-damage accident, isolation from falling objects such as aircraft, and protection against malevolent acts. Based on experience with underground testing of nuclear weapons, the report concluded that an appropriately designed plant would provide essentially complete containment of the radioactive material liberated from a reactor core during a core-damage event.

The Caltech report described a preliminary design study for underground construction of an LWR power plant with a capacity of 1,000 MWe. The minimum depth of the underground cavities containing the plant components would be 150 to 200 feet. The estimated cost penalty for underground siting would be less than 10 percent of the total plant cost.

In an appendix, the Caltech report described four underground nuclear reactors that had been constructed and operated in Europe. Three of those reactors supplied steam to turbo-generators, above or below ground. The largest of those reactors and its above-ground turbo-generator made up the Chooz plant in France, which had a capacity of 270 MWe. In describing the European reactors, the report noted:[68]

"The motivation for undergrounding the plant appears to be insurance of containment of accidentally released radioactivity and also physical protection from damage due to hostile military action."

Since the 1970s, underground siting of nuclear power plants has been considered by various groups. For example, in 2002 a workshop was held under the auspices of the University of Illinois to discuss a proposed US-wide "supergrid". That grid would transmit electricity – via superconducting DC cables – and liquid hydrogen, which would provide cooling to the DC cables and be distributed as fuel. Much of the energy fed to the grid would be supplied by nuclear power plants, which could be constructed underground. Motives for placing those plants underground would include "reduced vulnerability to attack by nature, man or weather" and "real and perceived reduced public exposure to real or hypothetical accidents".[69]

67 Watson et al, 1972.
68 Watson et al, 1972, Appendix I.
69 Overbye et al, 2002.

THE PIUS REACTOR

In the 1980s the reactor vendor ASEA-Atom developed a preliminary design for an "intrinsically safe" commercial reactor known as the Process Inherent Ultimate Safety (PIUS) reactor. An ASEA-Atom official described the company's motives for developing the reactor as follows:[70]

> *"The basic designs of today's light water reactors evolved during the 1950s when there was much less emphasis on safety. Those basic designs held certain risks, and the control of those risks led to an increasing proliferation of add-on systems and equipment ending up in the present complex plant designs, the safety of which is nevertheless being questioned. Rather than to continue into this 'blind alley', it is now time to design a truly 'forgiving' light water reactor in which ultimate safety is embodied in the primary heat extraction process itself rather than achieved by add-on systems that have to be activated in emergencies. With such a design, system safety would be completely independent of operator actions and immune to malicious human intervention."*

The central goal of the PIUS design was to preserve fuel integrity "under all conceivable conditions". That goal translated to a design specification of "complete protection against core melting or overheating in case of:

- → any credible equipment failures;
- → natural events, such as earthquakes and tornadoes;
- → reasonably credible operator mistakes; and
- → combinations of the above;

and against:

- → inside sabotage by plant personnel, completely knowledgeable of reactor design (this can be considered an envelope covering all possible mistakes);
- → terrorist attacks in collaboration with insiders;
- → military attack (e.g., by aircraft with 'off-the-shelf' non-nuclear weapons); and
- → abandonment of the plant by the operating personnel".[71]

To meet those requirements, ASEA-Atom designed a light-water reactor – the PIUS reactor – with novel features. The reactor pressure vessel would contain sufficient water to cool the core for at least one week after reactor shut-down. Most of that water would contain dissolved boron, so that its entry into the core would inherently shut down the reactor. The borated water would not enter the core during normal operation, but would enter through inherent mechanisms during off-normal conditions. The reactor pressure vessel would be made of pre-stressed concrete with a thickness of 25 feet. That vessel could withstand an attack using 1,000-pound bombs. About two-thirds of the vessel would be below ground.

ASEA-Atom estimated that the construction cost of a four-unit PIUS station with a total capacity of 2,000 MWe would be about the same as the cost of a station equipped with two 1,000 MWe "conventional" light-water reactors. The PIUS station could be constructed more rapidly, which would offset its slightly lower thermal efficiency. Thus, the total generating cost would be about the same for the two stations. ASEA-Atom estimated (in 1983) that the first commercial PIUS plant could enter service in the early 1990s, if a market existed.[72] To date, no PIUS plant has been ordered.

3.5 Options for Reducing the Risk of Diversion of Spent Fuel and Production of Plutonium

In addressing the risk of diversion of spent fuel for the purpose of plutonium production, this report focuses on the potential for diversion by the host government. Such diversion would occur because the host government seeks a reserve or actual capability to deploy a nuclear arsenal. (See Section 3.3, above.) Options for reducing the risk of diversion of spent fuel must, therefore, be seen in the wider context of nuclear-weapon proliferation. Curbing such proliferation has been a major preoccupation of governments and international agencies for decades. An array of non-proliferation strategies have been employed, as succinctly summarized by William Potter:[73]

70 Hannerz, 1983, pp 1-2.
71 Hannerz, 1983, page 3.
72 Hannerz, 1983, pp 73-76.
73 Potter, 1982, page 197.

> *"Many strategies have been proposed to deal with the phenomenon of nuclear proliferation. Most can be distinguished in terms of their emphasis on affecting the* demand *for versus the* supply *of weapons. Demand-oriented approaches are intended to reduce the incentives and strengthen the disincentives of a party to acquire nuclear weapons. They include such "political-fix" strategies as security and fuel supply guarantees, conventional arms transfers, and sanctions, and arms control measures such as nuclear-free zones and a comprehensive test ban. Supply-oriented approaches, on the other hand, are designed to make it more difficult for a party seeking nuclear weapons to obtain them. Representative of this approach to nonproliferation are "technological fixes" (including export restrictions on sensitive technologies and safer fuel cycles) and international and domestic safeguards."*

The risk of diversion of spent fuel from a CANDU 6 plant should be viewed within the framework of supply-oriented approaches to curbing nuclear proliferation. This issue is discussed further in Section 4.4, below.

4. Risks of Operating CANDU Plants, Especially the CANDU 6

4.1 Scope of this Discussion

Section 3, above, provides a broad review of the risk of an unplanned release (radiological risk) and the risk of diversion of spent fuel (proliferation risk). That review applies to nuclear power plants generally. Here, the discussion is continued with specific application to CANDU plants, especially the CANDU 6. Risks and risk-reducing options are discussed.

4.2 Risk of an Unplanned Release of Radioactive Material

Section 3.2, above, discusses the role of probabilistic risk assessment in examining the potential for, and consequences of, an unplanned release of radioactive material at a nuclear power plant. PRAs, despite their limitations, are important sources of information about unplanned releases. A PRA can provide a framework for informed discussion about the risk of a release caused by an accident or a malevolent act.

Unfortunately, Canada lacks a fully developed PRA culture. PRAs performed in Canada for CANDU reactors find very low probabilities for large releases. Based on those findings, the PRAs do not estimate the radiological impacts of large releases. Yet, the low probabilities are not credible.[74] The practice of ignoring large releases deprives citizens and policy makers of needed information. For example, in a recent analysis of the radiological risk of continued operation of the Pickering B station, the largest release considered included 71 TBq of Cesium-137.[75] That is a very small fraction of the core inventory of this isotope. (See Table 2-2 for the core inventory of Cesium-137 at Gentilly 2.)

THE DARLINGTON PROBABILISTIC SAFETY EVALUATION

The high point of PRA practice in Canada was reached by Ontario Hydro in its conduct of the Darlington Probabilistic Safety Evaluation (DPSE), which was published in 1987.[76] DPSE was conducted for internal initiating events only. It was conducted to Level 3, except that the impacts of the largest releases – in Ex-Plant Release Category 0 (EPRC0) – were not evaluated. It was not subjected to an official, independent review. Thus, DPSE did not rise to the quality of NRC's NUREG-1150 study. PRA practice in Canada has degenerated since DPSE, just as PRA practice in the USA has degenerated since NUREG-1150. In both cases, PRAs are now conducted in secrecy and have lacked credibility for at least a decade. (See Section 3.2, above.)

74 Thompson, 2000; IRSS, 1992.
75 SENES, 2007, Table B.5.3-1.
76 Ontario Hydro, 1987.

A focused review of DPSE was conducted by a team led by this author.[77] Several deficiencies were revealed. For example, DPSE had failed to identify an event sequence – involving failure of service water supply – that would be familiar to analysts conducting PRAs for PWR plants. In light of that and other deficiencies in DPSE, our team concluded that a reasonable estimate of the probability of a large, accidental radioactive release to the atmosphere from the Darlington plant would be 1 per 10,000 RY. Our value is comparable to the probability derived from occurrence of the TMI and Chernobyl accidents. (As mentioned above, those events suggest a core-damage probability of 1.6 per 10,000 RY, and a large-release probability of 0.8 per 10,000 RY.) Interestingly, our value is also comparable to the 95th percentile (high-confidence) value of DPSE's estimate of the probability of release category EPRC0, adjusted to account for external initiating events. The adjusted, 95th percentile probability of EPRC0 is 1.2 per 10,000 RY.[78]

CREDIBILITY OF CONTEMPORARY PRAS IN CANADA

As an illustration of questionable claims now being made in Canada about the risk of an unplanned release, consider some findings that are said to come from the PRA conducted for the Pickering B CANDU units. The PRA itself is not available for independent review. Some findings from the PRA were published in a December 2007 document related to an environmental assessment for refurbishment and continued operation of the Pickering B units. Specifically, the document provided estimated mean frequencies (probabilities) for the occurrence of Ex-Plant Release Categories EPRC1 through EPRC9.[79] The severity of the release decreases as the number attached to the EPRC increases from 1 to 9. EPRC0 is not mentioned. The reported mean frequency of EPRC1 is 1 per 10 billion years. For EPRC2 and EPRC3, the reported mean frequency in each case is less than 1 per 100 billion years.

Those low frequencies lack any credibility. They do not account for external initiating events, such as an earthquake.[80] Nor do they account for malevolent acts. In light of human history, it is not credible to assume that a plant could operate for billions of years (millions of millennia) without undergoing significant attack from within or without. Obviously, a nuclear power plant would not operate for such a vast period, but the idea of doing so illustrates the unreality of extremely low estimates of release frequency. Over a plant's operating lifetime of perhaps 60 years, a significant attack on the plant from within or without is a real possibility.

Even if one confines the discussion solely to internal initiating events, the reported low frequencies lack a grounding in real experience with accidents in complex engineered systems. Instead, these findings reflect the examination of a theoretical (paper) plant using PRA techniques such as fault trees. Practical experience shows that design deficiencies, unanticipated common-mode failures, gross errors and other influences can be dominant determinants of accident probability.[81] An engineer who apparently has long experience in the Canadian nuclear industry, John W. Beare, has stated:[82]

> *"The issue here is whether claims of failure rates lower than 10E-3 [sic] are believable for any one system, taking into account the risk from common-mode failures and the fact that it is virtually impossible to predict every failure scenario. Design Basis Accidents and even Beyond Design Basis Accidents are simply representatives of a larger set of possible failure sequences and events. Real accidents never unfold in this manner and often involve five or more failures that would probably be considered as independent events in a Probabilistic Risk Assessment."*

The same engineer has commented on the Canadian practice of not examining the consequences of the largest potential releases, stating:[83]

> *"If the Commission [CNSC] is concerned about the cost-benefit aspects of its safety requirements it could start by completing the Severe Accident Study research project started about 1988 but never completed. The conclusion of the preliminary study is that, in the event of a catastrophic accident, a release of radioactive material proportionately as large as that from Chornobyl could not be ruled out. In the case of a water-cooled reactor like CANDU such a release could be in the form of a relatively cool aerosol and not be dispersed as much as at Chornobyl. The radiation doses close to the reactor could be higher than at Chornobyl."*

77 IRSS, 1992.
78 DPSE (Ontario Hydro, 1987) states in its Table 5-6 that the probability of EPRC0 is 4.4 per 1 million RY. Applying an uncertainty factor of 14 (see Table 5-5 of DPSE), and a multiplier of 2 to account for external initiating events, one finds a 95th percentile value for EPRC0 of 1.2 per 10,000 RY.
79 SENES, 2007, Tables 5.3-1 and 5.3-3.
80 NEA, 2007, page 324.
81 Hirsch et al, 1989.
82 Beare, 2005, paragraph 140.
83 Beare, 2005, paragraph 192.

PRA FINDINGS FOR THE POINT LEPREAU AND GENTILLY 2 PLANTS

PRAs are not publicly available for either of the Point Lepreau or Gentilly 2 CANDU 6 plants. More work has been done toward completion of a PRA in the Point Lepreau case. An AECL document published in 2002 showed some early findings of that work.[84] A study that considered only internal initiating events at the Point Lepreau plant found a core-damage frequency of 8.9 per 100,000 RY, and an external-release frequency of 8.3 per 100,000 RY. Proposed plant modifications were expected to reduce those values. A preliminary study of the Gentilly 2 plant, limited to internal initiating events, found a core-damage frequency of 5.4 to 5.9 per 100,000 RY.[85] The credibility of these studies of the Point Lepreau and Gentilly 2 plants cannot be determined until the studies are published and independently reviewed. It is interesting that these studies found core-damage and external-release frequencies roughly comparable to the values derived from worldwide experience (TMI and Chernobyl).

RISK OF ONSITE ECONOMIC IMPACTS: FINDINGS BY ONTARIO HYDRO

An unplanned release of radioactive material could cause substantial adverse impacts at the site of a nuclear power plant, whether or not there is a release to the external environment. Ontario Hydro investigated this issue in DPSE, focusing on economic impacts. The findings are set forth in Tables 4-1 and 4-2. The risk costs estimated by Ontario Hydro, expressed in cent per kWh of plant output, are substantial. Note that these findings are for a four-unit CANDU station. Risk costs at a stand-alone CANDU 6 plant could be lower.

4.3 A Positive Void Coefficient of Reactivity, and its Implications

Canada's second nuclear reactor was the NRX research reactor, which first went critical in 1947. It was moderated by heavy water, cooled by light water, and fueled by natural uranium. Its initial power capacity was 30 MWt.[86] Near copies were later built in India (as part of an aid program) and in Taiwan (as a commercial venture).[87]

THE NRX REACTIVITY EXCURSION, AND ITS LESSONS

On 12 December 1952, the NRX reactor experienced an accident involving a sharp rise in reactivity and power (a reactivity excursion). Its power reached between 60 and 90 MWt before operators dumped the moderator, thus shutting down the reactor. During the excursion a number of fuel rods melted and others ruptured. Hydrogen was generated by metal-water reactions and exploded within the calandria. The core and calandria were damaged beyond repair. They were removed and replaced, and the reactor resumed operation about 14 months after the accident. The accident was attributed to operator errors and design deficiencies. The reactor's positive void coefficient of reactivity was a major contributing factor. In a description of the accident published in 1964, MIT professor T. J. Thompson set forth six conclusions and recommendations, including:[88]

> *"The design of reactors with positive reactivity coefficients which can be rapidly brought into play during transients, or by other reasonably ordinary perturbations of the system, should be avoided or is at least open to serious question. The post-accident analysis seems to indicate that the transient would probably have been terminated with little or no damage if it had not been for the positive voiding effect. It is certainly possible to operate such reactors safely as long as their behavior is close to normal. However, if a difficulty or perturbation of normal operations should develop it is very likely to be aggravated by positive reactivity coefficients and a minor incident can easily turn into a major accident."*

84 AECL, 2002.
85 Saint-Denis et al, 2005.
86 Thompson, 1964, pp 619-622.
87 Brooks, 2001, page 7.
88 Thompson, 1964, page 622.

Leaders of Canada's effort to develop nuclear technology were aware of Thompson's recommendations, but did not accept his advice about reactivity. Instead, they sought to reduce the probability of a reactivity excursion by developing reliable shutdown systems. The Canadian approach has been described by John Beare as follows:[89]

> *"The response of advanced industrialized nations in most of the world to the NRX accident was to shun reactors with a positive void effect. The Canadian approach was to develop a defence-in-depth approach matching the characteristics of the reactor and based on tolerable risk."*

EXPERIENCE WITH THE GENTILLY 1 REACTOR

Gentilly Unit 1 was a failed version of the CANDU. It was a prototype with a design capacity of 250 MWe. It first went critical in 1972 but never operated successfully, and was permanently shut down in 1977. At Gentilly 1, the moderator was heavy water but the primary coolant was light water, which boiled inside the pressure tubes (fuel channels). The pressure tubes were vertical, whereas at other CANDU plants they are horizontal. The principal problems with the design related to control of the fission reaction. The reactor core had positive power and void coefficients of reactivity. As a result, boiling in a pressure tube caused an increase in power output that led to more boiling, which further increased the power output.[90]

The reactivity properties of Gentilly 1 created problems of two kinds. First, the reactor could not be made stable during routine operation. Feedback between power output and coolant boiling in a fuel channel could not be controlled. According to a CNSC Staff presentation in 2008, "the resulting spatial power oscillations were expected to be intolerable".[91] Second, analysis performed during the construction of Gentilly 1 showed that, in the event of a loss of coolant and failure of the shutdown system, a violent power excursion would occur. The reactor would self-destruct with sufficient energy to threaten the integrity of containment.[92]

The vulnerability of Gentilly 1 to a power excursion was not properly understood when the plant was being designed. Analysts in the Canadian nuclear industry had made what they thought was an "obviously conservative" assumption about fuel behavior during a loss-of-coolant accident.[93] They had assumed that the fuel would heat up adiabatically. During the plant's construction, however, information obtained from the UK Atomic Energy Authority showed that coolant voiding during the blow-down phase of a loss-of-coolant accident would be greater than was indicated by an adiabatic heat-up model. As a result:[94]

> *"The revised analysis of loss-of-coolant and failure of the safety shut down system showed that the reactor would self-destruct with sufficient violence to blow out the top thermal shield (the fuel channels are vertical). It was also shown that any dual failure involving failure of the safety shut down system led to a reactor runaway with similar result. The designers were not confident that the containment could be shown to be capable of withstanding such dual failures and proposed installing a second and independent safety shut-down system of diverse design (poison injection) so that the risk to the public would still be tolerable and resistant to common mode failures."*

A violent power excursion of the type described in that statement occurred at Chernobyl Unit 4 on 26 April 1986, leading to a substantial release of radioactive material to the environment. Chernobyl 4 was a graphite-moderated, water-cooled reactor of the RBMK type. Like the CANDU, that reactor had a positive void coefficient of reactivity. The Chernobyl accident scenario differed from power-excursion scenarios that might arise at a CANDU. However, the existence of a positive void coefficient was central to the Chernobyl scenario.[95]

89 Beare, 2005, paragraph 44.
90 CNSC, 2008c, page 53.
91 CNSC, 2008c, page 53.
92 Beare, 2005, paragraph 68.
93 Beare, 2005, paragraph 66.
94 Beare, 2005, paragraph 68.
95 NRC, 1987.

THE POST-GENTILLY 1 APPROACH TO THE RISK OF A REACTIVITY EXCURSION

The Gentilly 1 version of the CANDU was unique. At all other CANDUs, the pressure tubes are horizontal and the primary coolant does not boil. Nevertheless, recognition of the potential for a violent reactivity excursion at Gentilly 1 affected the development of "mainstream" CANDUs (the multi-unit CANDU stations in Ontario and the CANDU 6) in two respects. First, analyses showed that a reactivity excursion could occur at a mainstream CANDU.[96] Second, experience with Gentilly 1 spurred the development of a second emergency shutdown system, involving the injection of liquid poison into the moderator.

Those influences led the Atomic Energy Control Board (AECB) to establish, in 1977, a formal requirement that two separate emergency shutdown systems must be provided at each CANDU reactor.[97] That policy remains in effect and is now overseen by the CNSC, the AECB's successor.

The two-shutdown-system policy requires two automated, fast-acting shutdown systems that operate independently. In practice, that requirement is met by one system employing solid shutoff rods, moving vertically, and by a second system involving the injection of liquid poison into the moderator through horizontal pipes. A reactor licensee is obliged to convince the regulator (currently the CNSC) that these systems meet two major criteria. First, either system should be capable of keeping reactor power below a safe level during any design-basis accident. In this context, a "safe" level of power is one that will not cause significant damage to fuel. Second, at least one system should be available during any design-basis accident. The spectrum of design-basis accidents includes loss-of-coolant accidents and other accidents that could lead to voids in the primary coolant.

Conformance with these two criteria cannot be directly demonstrated from testing or experience. Instead, the ability of a shutdown system to keep power below a safe level during an accident is assessed analytically, using computer models and findings from small-scale experiments.[98] Also, the ability of a shutdown system to achieve a specified high level of reliability (a low rate of unavailability) is assessed by extrapolating from a limited base of experience.

The requirement to employ two emergency shutdown systems could have been supplemented by other measures to reduce the risk of a violent reactivity excursion. Notably, the regulator could have required licensees to strengthen the reactor containment so that it would withstand the loads that could arise from a large, destructive pulse of power in the reactor core.[99] No such requirement has been imposed. Thus, the containment of a CANDU is designed to accommodate the internal pressure that would arise from a loss-of-coolant accident, but not the additional loads that would arise if that accident were accompanied by failure of the shutdown systems.[100]

In the early 1970s, a consensus emerged within the Canadian nuclear industry and its regulator that the requirement of two emergency shutdown systems was a sufficient response to the risk of a reactivity excursion. That consensus fundamentally affected the risk posed by every CANDU plant that subsequently entered service, including every CANDU 6. The consensus, which was formalized by the AECB in 1977, rested on two assumptions. First, either shutdown system would be capable of preventing a destructive reactivity excursion if a loss-of-coolant accident or other relevant accident occurred. Second, the probability of both shutdown systems being unavailable when needed would be very low. The Canadian approach and its implications have been summarized by CNSC Staff analysts in 2007 as follows:[101]

> *"This essentially risk-based regulatory decision has lead [sic] to the current licensing process in which the designer and licensee are not required to demonstrate provisions for mitigation of consequences of reactivity transients with failure to shutdown. This differs significantly from the international practice where adequacy of provisions for mitigation of consequences of reactivity transients with failure to shutdown has been a design and licensing requirement."*

96 Beare, 2005, paragraph 71.
97 CNSC, 2008c, page 54.
98 A document prepared by CNSC Staff members states (Akhtar et al, 2007, page 11): "The shutdown system effectiveness in CANDU reactors is demonstrated through analyses only; furthermore, most critical parameters used in these analyses cannot be measured/confirmed under realistic power reactor conditions."
99 Relevant mechanical loads would include: (i) pressure pulses from steam generated in the core and from the explosion of hydrogen generated by metal-water reactions; and (ii) impact loads arising from the motion of dislodged objects. There would also be thermal, radioactivity and debris loads on containment systems.
100 Akhtar et al, 2007, page 5.
101 Akhtar et al, 2007, page 5.

WHAT IS THE REAL RISK OF A REACTIVITY EXCURSION?

Analyses done by the nuclear industry have found low probabilities for reactivity excursions at CANDUs. For example, AECL has estimated that the probability of this event at the Point Lepreau CANDU 6 plant is 6 per 10 billion years.[102] That estimate lacks credibility. It does not account for external initiating events such as earthquakes. Nor does it account for malevolent acts. Even within the limited sphere of scenarios initiated by internal events, that estimate does not account for gross errors or unexpected common-mode failures. In illustration of the potential for common-mode failures, the NRC has expressed concern about the sensors that would be used to trigger the emergency shutdown systems at a CANDU plant. In reviewing the design of the ACR-700, a precursor to the currently-proposed ACR-1000, the NRC noted that the same type of sensor may be used for both shutdown systems. As a result, the design might not comply with NRC requirements for system diversity.[103] That particular problem was identified, but it is imprudent to assume that every common-mode problem will be identified.

For decades, it has been clear that nuclear power plants can suffer accidents that are more severe than design-basis accidents. The art of PRA has been developed to examine the potential for such severe accidents. Yet, the Canadian nuclear industry and its regulators have never applied PRA techniques to comprehensively assess the risk of a violent reactivity excursion. The nuclear industry has repeatedly claimed that the probability of such an excursion is very low. That claim is not credible, as discussed above. Even if the claim had some credibility, the public should be provided with an estimate of the onsite and offsite consequences of an unrestrained reactivity excursion. Information about consequences would help the public and decision makers to consider the acceptability of the risk. No PRA-type study done in Canada has ever provided the needed information.

As mentioned above, confidence in the ability of the CANDU shutdown systems to curb a reactivity excursion rests upon analysis, not upon direct testing or experience. Evidence is accumulating that the confidence has been misplaced. Analysis has not dealt with the full complexity of the relevant accident situations. In part, the analytic challenge arises from the rapidity and scale of the potential pulse in reactivity. Estimates show that, during a design-basis accident, the power produced by a fuel assembly could rise, within 2 seconds, to 6 to 12 times the full power produced by the assembly during normal operation.[104] One or both shutdown systems must stop the reaction at that point. Otherwise, expanding voids in the primary coolant would cause the reactivity excursion to accelerate. The excursion would eventually be terminated by disintegration of the reactor core.

Analysis of the effectiveness of shutdown systems requires the modeling of multiple phenomena, operating interactively over a short time scale. The initial event, such as a loss-of-coolant accident, involves thermo-hydraulic behavior that is difficult to model. That modeling must be done interactively with reactivity analysis. A further complication is the potential for modification of the fuel geometry through phenomena such as cladding rupture. Overlaid on these interactive modeling tasks is the challenge of modeling the behavior of the shutdown system itself, involving rod motion or the injection of liquid poison into the moderator.

The Canadian nuclear industry and its regulators have slowly and reluctantly recognized that their understanding of these complex, interacting phenomena is limited. For example, an independent AECB-sponsored review of CANDU reactivity behavior, motivated by the 1986 Chernobyl accident but not conducted until the period 1992 to 1994, concluded that the void reactivity had an uncertainty of about 50 percent. Industry rejected that claim, but agreed to do new experiments and analysis. Eventually, this work led to a current estimate that full-core void reactivity at a typical CANDU is 14 to 18 mk, as opposed to an earlier estimate of 9 to 11 mk. Fuel temperature and power coefficients of reactivity are now thought to be slightly positive, having been previously thought to be slightly negative and zero, respectively.[105]

102 AECL, 2002, Section 4.1. That source estimated a frequency of 6.4E-10 per year for "sequences involving a rapid loss of core structural integrity due to failure of reactor to be shutdown when required."
103 Miller, 2005, page 11.
104 Schaubel, 2008, page 16.
105 CNSC, 2008d, pages 10 and 11.

Investigations in this area continue. The effectiveness of emergency shutdown systems is a significant, unresolved safety issue. Confidence in existing analysis is low, as illustrated by the following statement by CNSC Staff in July 2008:[106]

> *"The overall net effect of propagation of an unqualified void reactivity uncertainty in analysis remains indeterminate due to combined effect of other significant uncertainties, such as:*
>
> → uncertainty in initial and accident conditions,
> → predictions of voiding rate, and
> → effect of various approximations used in simulations of a very complex neutronic-thermal hydraulic transient."

4.4 Risk of Diversion of Spent Fuel and Production of Plutonium

The risk of diversion of spent fuel is discussed, from a broad perspective, in Section 3.3, above. Here, that risk is discussed with a focus on CANDU nuclear power plants. The discussion necessarily considers heavy-water reactors (HWRs) as a category. The spent-fuel diversion risk associated with CANDUs is, in many respects, also associated with other HWRs.

CANADIAN EXPERIENCE WITH INDIA AND PAKISTAN

Canada gained early experience with the risk of spent-fuel diversion through its provision of nuclear technology to India and Pakistan. (See Section 2.2, above.) The experience began when Canada supplied India with the CIRUS research reactor, a HWR with a capacity of 40 MWt. CIRUS entered service in 1960. Although Canada supplied CIRUS on the condition that it be used only for peaceful purposes, India produced plutonium in CIRUS for its 1974 nuclear-weapon test and for subsequent nuclear weapons.

After the 1974 test, Canada ceased its nuclear cooperation with India. Thereafter, India built upon the knowledge it had acquired from Canada, developing an indigenous capability to construct and operate HWRs. One of the indigenously-constructed HWRs is the Dhruva reactor, with a capacity of 100 MWt, which first entered service in 1985. Knowledgeable analysts concluded, in the late 1990s, that CIRUS and Dhruva had been the major sources of plutonium for India's nuclear weapons.[107]

Canada also supplied India with two CANDU nuclear power plants – RAPP 1 and RAPP 2. RAPP 1 entered service in 1973. After the nuclear test of 1974, India completed RAPP 2 without Canadian help, and that plant entered service in 1983. India went on to build additional plants of indigenous manufacture, employing the CANDU design concept. At present, there are fifteen CANDU-type plants in India, including RAPP 1 and RAPP 2.[108]

India could use its CANDU-type plants to produce plutonium for nuclear weapons, and has done so to some extent. However, observers conclude that India prefers, understandably, to use weapon-grade plutonium in its weapons. Production of weapon-grade plutonium requires the discharge of spent fuel at a low burnup. That mode of operation increases the cost of operation of a power reactor. It appears that, in response to this cost burden, India's CANDU-type plants have made a relatively small contribution to India's stockpile of military plutonium.[109] That finding does not eliminate the risk of diversion of spent fuel from a CANDU plant. It simply means that India has cheaper sources of weapon-grade plutonium – the CIRUS and Dhruva reactors.

Canada also supplied a CANDU plant to Pakistan. The KANUPP plant was built by a Canadian company, on a turnkey basis. KANUPP has a capacity of 130 MWe, entered service in 1972, and remains operational.[110] As shown below, knowledgeable observers suspect that spent fuel from KANUPP has been used to produce military plutonium.

106 CNSC, 2008d, page 26.
107 Albright et al, 1997, pp 266-267.
108 WNA, 2008, India data.
109 Albright et al, 1997, pp 266-267.
110 WNA, 2008, Pakistan data.

SAFEGUARDING OF CANDU PLANTS

The IAEA has agreements with many countries to safeguard nuclear facilities in those countries. At a nuclear power plant, a major purpose of the safeguards is to detect a diversion of spent fuel. Thus, the effectiveness of safeguards is one determinant of the risk of diversion of spent fuel at a CANDU plant. In a study conducted for the NRC, the following observation was made about the safeguarding of a CANDU plant:[111]

> *"The major problem associated with HWRs is the on-line refueling necessitated by the marginal reactivity of the core. IAEA openly acknowledges that it has no way of accounting for spent fuel comparable to the accounting accuracy in the LWR cycle. A HWR discharges 7 to 10 fuel bundles per day into a spent-fuel pool which is not completely visible to inspectors. The bundles may be placed in sealed containers, but verification of those containers is very difficult."*

A country seeking to divert spent fuel from a CANDU plant might exploit weaknesses in safeguards, as identified by the NRC. Alternatively, the country might accumulate a stock of spent fuel over a period of years and then break its safeguards agreement with the IAEA. To prepare for either eventuality, the country might drive some spent fuel to a very low burnup, so that it contained weapon-grade plutonium.

CANDUS, HWRS AND NUCLEAR PROLIFERATION: AN OVERVIEW

A concise summary of the contribution made by CANDUs and other HWRs to the proliferation of nuclear weapons was provided by David Fischer and Paul Szasz, both former IAEA officials, writing in 1985:[112]

> *"The proliferation record of the CANDU power reactor and of the large Canadian research reactor (also an HWR but somewhat differently designed) is much worse than that of the much more numerous and widespread LWR. An unsafeguarded Canadian research HWR reactor produced the plutonium for India's 1974 nuclear explosion; a safeguarded CANDU power reactor might be the source of spent fuel for Pakistan's pilot reprocessing plant. A safeguarded CANDU power reactor and a Canadian research HWR would probably have served the same purpose in the Republic of Korea and Taiwan respectively, had the United States not forcefully intervened and persuaded both countries to abandon their reprocessing plans. Both countries (parties to the NPT) still operate these reactors. A safeguarded CANDU or a West German-supplied natural uranium/ heavy water power reactor will probably serve as the source of spent fuel for the Argentine reprocessing plant. In the case of Israel an unsafeguarded French natural uranium-fuelled and heavy water moderated reactor also served as the source of unsafeguarded plutonium.*

In short, four of the five NNWS operating unsafeguarded 'sensitive' plants (Argentina, India, Israel and Pakistan) – including, in each case, one or more reprocessing plants – have incorporated HWRs, which can easily produce weapon-usable plutonium, in their nuclear structures. Two NPT NNWS (the Republic of Korea and Taiwan), in another region of political tension, have also done so."

111 O'Brien, 1982, page 206.
112 Fischer and Szasz, 1985, page 49.

5. Design and Siting Criteria Affecting the Risk of an Unplanned Release at a New Nuclear Power Plant

5.1 Scope of this Discussion

The risk of an unplanned release of radioactive material at a nuclear power plant is determined by various factors. The dominant factor is the physical configuration of the plant, which has two major aspects. One aspect is the nature of the site, such as the proximity of subterranean faults that could lead to seismic activity. The second aspect is the design of the plant.

Other risk-determining factors include the quality of construction, maintenance and operation of the plant. These factors could become dominant if quality were highly degraded. However, the nuclear industry is more closely regulated than most industries. Assuming a typical stringency of regulation by international standards, severe degradation in the quality of construction, maintenance and operation at a nuclear power plant is unlikely. Instead, the typical situation is one in which "normal" rates of error occur in construction, operation and maintenance. Error cannot be eliminated, due to innate qualities of the human species. That proposition holds both for routine errors and for gross errors, although the latter are less likely.

One of the sources of the risk of an unplanned release is the potential for malevolent actions. That potential is, in part, influenced by the political and social climates of the country where the plant is located, the region surrounding that country, and the world at large. Thus, social and political climates are risk-determining factors.

The planners and designers of a nuclear power plant should be aware of the normal propensity of humans to make errors. They should also be aware that malevolent acts could affect the plant. Given that the plant might operate for 60 years or more, and given the history of the human species, it would be imprudent to ignore the potential for malevolent acts. Thus, it is incumbent on planners and designers to account for human error and malevolent acts when they site and design a nuclear power plant. That is why the physical configuration of a plant – in terms of its siting and design – is the major determinant of the risk of an unplanned release at the plant.

In considering the licensing of Gentilly 2 for an extended period of operation, the CNSC will compare the risk posed by Gentilly 2 with the risk posed by a "modern" nuclear power plant. The criteria and processes to be used for that comparison are discussed in Sections 6 and 7, below. Since the risk posed by a modern plant is determined primarily by its siting and design, as explained above, the siting and design criteria applicable to new plants are highly relevant to the licensing of Gentilly 2 for extended operation. Also, those criteria would be central to the licensing of a new CANDU 6 plant.

Section 5.2, below, briefly reviews the criteria established by the IAEA and the CNSC for siting and design of new nuclear power plants.[113] Then, Section 5.3 outlines alternative criteria that could greatly reduce the risk of an unplanned release.

5.2 International and CNSC Criteria for Siting and Design of New Plants

IAEA CRITERIA

In 2000, the IAEA published its document NS-R-1, titled *Safety of Nuclear Power Plants: Design*, which set forth safety-related criteria for the design of a nuclear power plant.[114] Criteria for plant siting were subsequently set forth by the IAEA in its document NS-R-3, published in 2003.[115]

NS-R-1 reflected the consensus of IAEA Member States at the time. It addressed events that are "very unlikely", such as severe accidents that result in large releases of radioactive material, but it did not address "extremely unlikely" events such as the impact of a meteorite.[116] NS-R-1 did not discuss intentional, malevolent acts. That omission presumably reflects the consensus of Member States in 2000. The IAEA has considered malevolent acts in documents published more recently, as discussed below.

113 For a more detailed review by this author, see: Thompson, 2008a.
114 IAEA, 2000.
115 IAEA, 2003.
116 IAEA, 2000, pp 1-2.

"DESIGN-BASIS" AND "BEYOND-DESIGN-BASIS" ACCIDENTS

NS-R-1 articulated a set of safety objectives that are summarized in Table 5-1. A general objective was supported by specific objectives relating to radiation protection and technical safety. The technical safety objectives embraced a concept that is currently employed in the reactor-safety field worldwide. The concept is that certain potential accidents are taken into account in designing a nuclear power plant, while others are not. Accidents in the first category are known as "design-basis" accidents, and would not involve core damage if the plant functioned as designed. Accidents in the second category are known as "severe" accidents or "beyond-design-basis" accidents. Those terms are used interchangeably in NS-R-1. Accidents in the second category would involve core damage.

The practice of dividing potential reactor accidents into two categories has been so widely adopted that many persons now working in the nuclear industry and its regulators may be unaware of the practice's origins. Those origins date from the first two decades of the commercial nuclear power industry (roughly, 1953-1975), when the foundations of the industry were laid. The basic designs of the present fleet of nuclear power plants were established at that time.

Until 1975, the nuclear industry and its regulators, with some limited exceptions, equated design-basis accidents with credible accidents. It was assumed that accidents of greater severity, involving significant damage to a reactor core, were non-credible.[117] That assumption became untenable when the Reactor Safety Study was published in 1975.[118] The TMI accident of 1979 and the Chernobyl accident of 1986 demonstrated empirically that core-damage accidents are indeed credible. At that point, the industry could have gone back to the drawing board, and developed new, safer types of reactor. Indeed, ASEA-Atom took that step, developing and attempting to market the PIUS design in the early 1980s. The nuclear industry as a whole took a different path, and regulators participated in that decision. The formerly "non-credible" accidents became "beyond-design-basis" accidents. PRAs were performed to estimate the "risk" of a beyond-design-basis accident, and that risk was deemed "acceptable" if its estimated value was below some threshold. NS-R-1 reflected that paradigm.

PRAs have yielded useful, practical knowledge. They have identified deficiencies in the design, operation and maintenance of nuclear power plants. Some of those deficiencies have been corrected, thereby reducing the probability of a radioactive release. PRA findings have guided the development of capabilities for offsite emergency response. Nevertheless, it should not be forgotten that the need for PRAs derives from fundamental weaknesses in design. The present fleet of commercial reactors, and the proposed Generation III reactors, are unable to ride out a variety of credible events outside their design basis. If subjected to such an event, one of these reactors would experience core damage and, potentially, a release of radioactive material to the environment.

NS-R-1 did not specify any quantitative target for the risk of a beyond-design-basis accident. Instead, it specified qualitative targets. For example, as shown in Table 5-1, NS-R-1 called for a plant to be designed such that "the likelihood of accidents with serious radiological consequences is extremely low". NS-R-1 did not provide further guidance about implementing that objective.

IAEA RECOMMENDATIONS AND REQUIREMENTS REGARDING DESIGN FEATURES

NS-R-1 set forth general recommendations and specific requirements regarding the design features of nuclear power plants, from a safety perspective. The general recommendations are exemplified by Table 5-2, which shows a recommended hierarchy of preference in selecting a plant design feature. There was merit in the hierarchy. It called for choosing an inherently safe design as the first preference, or a passively safe design as the second preference. That recommendation is significant for the CANDU 6 design, as discussed below. Nevertheless, the hierarchy was deficient in important respects. It ranked continuously-operating, active safety systems on the same level as passive safety features, which is a serious deficiency. It stated that a preference should be exercised if "that can reasonably be achieved", but provided no criterion for determining what is reasonable.

Two examples illustrate the specific design requirements that were set forth in NS-R-1. First, NS-R-1 stated that "Structures, systems and components important to safety shall generally not be shared between two or more reactors in nuclear power plants".[119] That requirement appears to rule out the plant designs used for the present CANDU stations in Ontario. At those stations, up to eight reactors share safety systems, including containment and core-cooling systems. The second example is the statement in NS-R-1 that "The means for shutting down the reactor shall consist of at least two different systems to provide diversity".[120]

117 Okrent, 1981.
118 NRC, 1975.
119 IAEA, 2000, page 24.
120 IAEA, 2000, page 30.

The overall pattern of design recommendations and requirements in NS-R-1 was to set forth vague, elastic recommendations about plant performance, but precise, rigid requirements regarding particular aspects of plant design, such as the number of reactor shut-down systems. That approach is exactly opposite to the approach that would be taken if a regulator were seeking to maximize creativity in plant design. To maximize creativity and safety, a regulator would set forth precise, highly-demanding performance requirements, but would say comparatively little about design details.

IAEA CONSIDERATION OF MALEVOLENT ACTS

NS-R-1, which was published in 2000, did not discuss malevolent acts. IAEA documents published more recently have discussed such acts. For example, in 2006, the IAEA published a study on advanced nuclear power plant design options to cope with external events.[121] The study involved the participation of plant designers from a number of Member States. Various design options were considered, including options that would improve a plant's ability to resist malevolent acts. The study did not yield specific design requirements for new nuclear power plants.

IAEA PROMOTION OF HIGHER STANDARDS OF SAFETY AND SECURITY

One of the functions of the IAEA is to facilitate nuclear power production worldwide. As part of its pursuit of that function, the IAEA is promoting safety and security standards that are more stringent than previous standards, and that are more uniform across national regulators. This promotional effort is intended to reduce the likelihood that an accident or an attack at a nuclear power plant in one country will feed public opposition to nuclear power in other countries. The effort is illustrated by an October 2008 speech by the IAEA Director General on the occasion of the 50th anniversary of the OECD Nuclear Energy Agency.[122]

One thrust of the Director General's speech was to call upon the vendors of nuclear-power technology to employ high standards. That call has implications for Canada's sale of the CANDU 6 design. The Director General stated:

> *"Naturally, we stress that the primary responsibility to ensure safety and security lies with the countries concerned. However, we also make the companies – and countries – which supply the equipment and expertise aware of their responsibility. In most cases, that means OECD companies and countries. This is because failures of either safety or security can have consequences stretching well beyond national borders, as the Chernobyl accident demonstrated. Suppliers of nuclear technology owe a duty of care to the recipients and to the world at large. Overall, safety is much better than it was 10 years ago, but we still have vulnerabilities in safety, as well as in security."*

The Director General also called for more stringent regulation at the national level, stating:

> *"In some countries, we see a troubling combination of old reactors and weak regulators. This could be a ticking bomb. It is in all our interests to ensure that the highest safety standards are upheld everywhere."*

CNSC GUIDANCE DOCUMENTS RD-337 AND RD-346

In October 2007, the CNSC published draft versions of its guidance documents RD-337 and RD-346, setting out design and siting criteria for new plants. RD-337 was titled *Design of New Nuclear Power Plants.*[123] RD-346 was titled *Site Evaluation for New Nuclear Power Plants.*[124] In each case, the stated purpose of the document was to "set out the expectations" of the CNSC regarding design and siting.

In May 2008, the CNSC Staff published a document containing revised versions of RD-337 and RD-346, which the Staff submitted to the CNSC Commissioners for approval at the Commission meeting of 10 June 2008.[125] It appears that the Commissioners approved the revised versions. However, final versions of RD-337 and RD-346 have not been published by the CNSC.

121 IAEA, 2006b.
122 ElBaradei, 2008.
123 CNSC, 2007a.
124 CNSC, 2007b.
125 Dallaire et al, 2008.

COMPARISON OF RD-337 AND NS-R-1

One could reasonably expect RD-337 to be generally consistent with NS-R-1 and with design standards in leading industrialized countries. The CNSC has encouraged that expectation by stating that its regulatory framework is aligned with "international standards and best practices", and that new nuclear power plants built in Canada "will meet the highest standards".[126]

Standards set forth in IAEA documents have been described by a Canadian author as "lowest common denominator" standards.[127] Thus, in comparing NS-R-1 and RD-337, one would expect the latter to have additional requirements and be generally more demanding. That is true in some respects, but not in all, as discussed below.

RD-337 shares with NS-R-1 the same basic paradigm, in which potential accidents are in two categories – those within, and those beyond, the design basis. Accidents in the former category are addressed deterministically, while those in the latter category are addressed probabilistically. As discussed above, this two-tier accident paradigm reflects fundamental deficiencies in the design of nuclear power plants, including the proposed Generation III plants.

CONSIDERATION OF MALEVOLENT ACTS IN RD-337

NS-R-1 did not consider malevolent acts. By contrast, RD-337 does consider such acts, stating:[128]

> *"The design shall include provisions that promote security and robustness in response to malevolent acts, in accordance with applicable regulations and modern standards and codes."*

Having articulated that goal, RD-337 proceeds to introduce the concept of design-basis threats (DBTs) and beyond-design-basis threats (BDBTs), which are analogous to the two categories of accident mentioned above. DBTs are described as "credible malevolent acts", while BDBTs are described as "severe" DBTs. That terminology is reminiscent of pre-1975 practice regarding accidents, when design-basis accidents were equated with credible accidents.

RD-337, in its draft and apparent final versions, does not characterize either category of threat. A consultant to the CNSC has examined the issue of designing nuclear power plants to resist malevolent acts, and has offered recommendations intended to "facilitate the development of regulatory requirements".[129] The consultant postulated one type of DBT that is similar to aspects of the DBT employed in the USA by the NRC, and a second type of DBT that could include "a common large commercial aircraft at speeds which can reasonably be achieved at low altitudes or an executive jet or a personal aircraft with a load of explosives taking off from an unregulated airfield". The consultant also provided examples of potential BDBTs, including "a large malevolent, explosive laden, vehicle or a LPG tanker gaining access past the physical protection barriers by stealth, deceit or force".[130]

The consultant recommended that the CNSC be the "prime developer" of the DBTs and BDBTs. Implementation of that recommendation would exclude citizens from participating in the determination of DBTs and BDBTs. Such exclusion would be antithetical to the principles of sustainability, and would be unnecessary. Citizens could be engaged in dialogue on this issue without broad dissemination of detailed technical information (e.g., computer models describing the response of a structure to blast) that might assist persons with malevolent intent.

QUANTITATIVE SAFETY GOALS IN RD-337

NS-R-1 did not set forth quantitative safety goals. By contrast, RD-337 sets forth the quantitative safety goals shown in Table 5-3. The table itself shows the quantitative goals set forth in the draft version of RD-337, and the notes show how those goals were weakened in the apparent final version. The goals were significantly weakened in two respects. First, the quantitative goals in the "should be less than" category were abandoned. Second, the quantitative goals in the "shall not exceed" category were retained, but with different language. The final version of RD-337 states that the sum of frequencies of all event sequences that can lead to a specified outcome "is less than" a numerical value.

126 CNSC, 2006, page 3.
127 Harvie, 2004, page 3.
128 CNSC, 2007a, page 36.
129 Asmis and Khosla, 2007.
130 Asmis and Khosla, 2007, pp 66-67.

The only logical explanation for the weakening of the safety goals is a determination by the CNSC that plant designs being proposed for construction in Canada could not meet the goals set forth in the draft version of RD-337. The present version of the goals allows broad latitude in interpretation. For example, the final version of RD-337 does not say whether the estimated sum of event-sequence frequencies is a mean value, a median, the 95th percentile value, or some other value. Also, it does not say if the estimate includes external initiating events or accidents when the reactor is in an operating mode other than full power.

Core damage or a large release would be instances of a beyond-design-basis accident. RD-337 does not explain how the probability of a beyond-design-basis threat would relate to the quantitative safety goals. There is, at present, no statistical basis for a quantitative estimate of the probability of a postulated malevolent act. Nevertheless, in a policy or planning context, judgment could be used to assign minimum probabilities to postulated acts.

Consideration of the potential for BDBTs could prevent the CNSC from determining compliance with the quantitative safety goals in RD-337. If that potential were arbitrarily set aside, determining compliance could still be difficult or impossible. The determination would rely on PRAs. Yet, as explained in Section 4.2, above, the findings of Canadian PRAs are not credible.

RECOMMENDATIONS AND REQUIREMENTS IN RD-337 REGARDING DESIGN FEATURES

Like NS-R-1, RD-337 sets forth general recommendations and specific requirements regarding the design features of nuclear power plants, from a safety perspective. NS-R-1, as shown in Table 5-2, set forth a recommended hierarchy of preference in selecting plant design features. By contrast, RD-337 describes a similar set of design features, but does not place them in a hierarchy of preference. Instead, RD-337 states that features should be adopted where "that can be reasonably achieved".[131] That approach is a significant retreat from the safety standard established by NS-R-1.

As shown in Table 5-2, NS-R-1 called for "inherent safety" as the first priority in plant design. Interpreted strictly, that recommendation would preclude the licensing of a plant possessing a positive void coefficient of reactivity. Thus, a CANDU 6 plant could not be licensed. By contrast, RD-337 does not preclude or discourage the licensing of a plant with a positive void coefficient of reactivity. Such a provision did appear in a CNSC draft document that preceded the draft RD-337 of October 2007. The preceding document was issued in April 2005. Reportedly, its Clause 6.1 stated that the reactor design must be such that:[132]

> *"…The fission chain reaction shall be controlled, and, when necessary, terminated under all credible circumstances, and* Priority *shall be given to [sic] nuclear reactor's inherent negative feedbacks that shall mitigate any rapid increase in reactivity and reactor power."*

A consultant to the CNSC has commented on the implications of that provision, as follows:[133]

> *"*Priority *per se is not a standard. However, AECL believes that one interpretation of this clause would drive the design towards a negative coolant and void reactivity; AECL's opinion is that this is unnecessary. They believe this criterion may reflect on the current CANDU 6 design with a negative impact on the marketing prospects of the CANDU 6. In addition, clause 6.1 will have broad implications on existing power reactor licences, refurbishment projects, and on the LVRF project."*

AECL's position prevailed. RD-337 does not call for inherent negative feedback on reactivity as a priority in reactor design. Clearly, the CNSC was persuaded that the marketing prospects of the CANDU 6 and the continued operation of CANDU plants are more important considerations than establishing stringent standards of reactor safety.

This case is a classic illustration of two chronic and interrelated problems with nuclear power – political influence on regulation, and the locking-in of obsolete technology. The Canadian government is both the overseer of the CNSC and the owner of AECL, and thus has conflicting motives. After the NRX reactor accident of 1952, the Canadian nuclear industry had an opportunity to change course, by switching to a reactor design with negative void and power coefficients of reactivity. That opportunity was not taken. Now, more than half a century later, AECL has inherited a business whose continuation depends on CNSC's continuing acceptance of positive void and power coefficients, at a time when trends in reactor design are moving toward inherent safety. The CNSC, no doubt influenced by its governmental masters, has worded RD-337 to accommodate AECL's needs.

131 CNSC, 2007a, page 13.
132 Jarman, 2007, page 8.
133 Jarman, 2007, page 8.

5.3 Alternative Criteria that Could Reduce the Risk of Unplanned Release

Section 3.4, above, provides a brief review of design options for reducing the risk of an unplanned release at a nuclear power plant. The PIUS version of the LWR, developed by ASEA-Atom, is a notable example.

New, stringent criteria for design and siting could drive the nuclear industry to build plants with a substantially lower risk of unplanned release. Criteria of that type are outlined here, as an alternative to the IAEA and CNSC criteria discussed above.

Table 5-4 describes the proposed criteria. They are purely deterministic. All the events that they are intended to accommodate are within the plant's design basis. Thus, they offer a clear alternative to the paradigm employed by the IAEA and CNSC, in which design-basis accidents are addressed deterministically and beyond-design-basis accidents are addressed probabilistically. The proposed criteria reject that two-tier paradigm, and also reject the nuclear industry's traditional concept of risk and its acceptability.

The criteria set forth in Table 5-4 are not definitive. At various points, they state that a parameter would be "specified", but they leave that specification open or suggest a tentative value for consideration. The intention is that the final parameters would be determined by public processes, as discussed below.

Table 5-4 provides design-basis criteria for a plant's safety performance under two conditions – reactor operation, and reactor refueling. The criteria for reactor operation are similar to those articulated by ASEA-Atom for the PIUS reactor. The criteria for reactor refueling reflect an expectation that the plant's containment would be somewhat compromised during refueling. The maximum release specified under the refueling criteria could be linked to the frequency of refueling for a particular plant design.

Both of these sets of criteria are performance-based. They would encourage creativity in plant design, providing an opportunity to move beyond the present designs, whose basic features were established in the 1950s and 1960s. Compliance with the criteria for reactor operation could be demonstrated, to a substantial extent, by testing of the actual plant. For example, the plant's ability to ride out a loss of power and normal heat sinks, and abandonment by operators, could be tested directly. Other aspects of compliance would be established through conservative modeling and analysis.

Table 5-4 provides deterministic siting criteria, expressed in terms of maximum radiological impacts from design-basis events. Those criteria would translate into permissible distributions of population and land use in regions surrounding a plant. Compliance would be established through conservative modeling and analysis.

PUBLIC PROCESSES FOR DECIDING ON FINAL CRITERIA

The criteria set forth in Table 5-4 provide a point of departure for public processes that could yield final criteria for the design and siting of new nuclear power plants. That transition could occur in two steps. First, the general structure of the criteria would be debated, and modified as appropriate. Second, final specifications would be established for the various parameters that appear in Table 5-4, or the analogous parameters that would appear in a modified structure.

Suitable public processes would engage local and provincial governments, and a broad range of other groups of stakeholders, in dialogue about citizens' preferences regarding the safety and sustainability of nuclear power. That dialogue should be informed by technical analyses that respond to stakeholder questions. All aspects of the dialogue should occur openly, even when the dialogue addresses the potential for malevolent acts. An essential feature of any sustainable energy system is that it should be robust against malevolent acts by virtue of its inherent properties, and should not require protection through secrecy. Indeed, secrecy and related measures, such as surveillance of the population, are antithetical to sustainability.

6. CNSC Risk Objectives for Life Extension of Gentilly 2

In February 2008, the CNSC published a guidance document that describes requirements and procedures related to the licensing of an existing Canadian nuclear power plant for an extended period of operation. The document is designated RD-360, and is titled *Life Extension of Nuclear Power Plants.*[134]

RD-360 does not set forth any clear objective for the risk of an unplanned release, or for the risk of diversion of spent fuel. The closest that RD-360 comes to providing a risk objective is in discussing licensee responsibilities for preparation of an Integrated Safety Review (ISR). RD-360 characterizes the ISR as follows:[135]

> *"Performed by the licensee, the ISR involves an assessment of the current state of the plant and plant performance to determine the extent to which the plant conforms to modern standards and practices, and to identify any factors that would limit safe long-term operation. Operating experience in Canada and around the world, new knowledge from research and development activities, and advances in technology, are taken into account. This enables determination of reasonable and practical modifications that should be made to systems, structures, and components, and to management arrangements, to enhance the safety of the facility to a level approaching that of modern nuclear power plants, and to allow for long term operation."*

That statement allows great latitude in interpretation. The nearest the statement comes to specificity is in stating that the safety of the plant being considered should "approach" the level of safety achieved by a "modern" nuclear power plant. Apparently, the modern plant in this comparison would not necessarily comply with CNSC criteria for a new plant, which are discussed in Section 5.2, above.

AECL, apparently acting as agent for New Brunswick (NB) Power, has adopted targets for the risk of an unplanned release at the refurbished Point Lepreau CANDU 6 plant. Refurbishment is under way at that plant. The target for the frequency of core damage is 1 per 10,000 RY, and the target for a large release is 1 per 100,000 RY. A "large release" is not defined.[136] Hydro-Quebec has adopted the same risk targets for the refurbished Gentilly 2, except that a large (important) release is defined in that case. Such a release would involve the escape to the environment of more than 1 percent of the core inventory of Cesium-137.[137]

Neither AECL nor Hydro-Quebec says whether the estimated frequency of core damage or release is a mean value, a median, the 95th percentile value, or some other value. Also, neither says if the estimate includes external initiating events or accidents when the reactor is in an operating mode other than full power. Finally, neither AECL nor Hydro-Quebec discusses malevolent acts and their relationship to the risk targets.

134 CNSC, 2008b.
135 CNSC, 2008b, Section 6.2.
136 AECL, 2004, Section 2.2.2.
137 Pageau, 2004, pages 10 and 11 of attached document.

7. CNSC Process for Considering Risks Associated with Life Extension of Gentilly 2

As discussed in Section 6, above, RD-360 does not set forth any specific objective for the risk of an unplanned release during the extended life of a CANDU plant. Nevertheless, the CNSC does have a process for considering that risk. The process is not explained clearly in RD-360 or other CNSC documents, and its actual functioning must be inferred from fragmentary information. In the remainder of this section, the process is reviewed with particular attention to the regulatory risk associated with life extension of the Gentilly 2 plant.

The CNSC has no evident process or criterion for assessing the risk of diversion of spent fuel from a CANDU plant, whether in Canada or elsewhere.

LICENSEE EFFORTS TO REDUCE REGULATORY RISK

Refurbishment of a CANDU plant typically requires several years of preparation and perhaps two years of execution. The licensee must estimate the costs that will be incurred over the period of preparation and execution, and the economic benefits of continued operation. Comparison of the estimated costs and benefits will establish the economic case for refurbishment.

Experience has shown that substantial cost overruns can occur during CANDU refurbishment. For example, the refurbishment of Pickering Unit 1 cost more than double the original estimate. That experience led the Ontario government to permanently close Pickering Units 2 and 3, instead of refurbishing them. Refurbishment of Bruce A Units 1 and 2 was expected to cost Can$2.75 billion. As of April 2008, while that refurbishment was under way, the estimated cost of its completion had risen to Can$3.1 to 3.4 billion.[138]

The licensees of the two CANDU 6 plants in Canada are undoubtedly aware that cost overruns could occur during these plants' refurbishment. In both instances, the potential for cost overruns is especially significant because the economic case for refurbishment is weak. In September 2002, the Public Utilities Board of New Brunswick concluded that refurbishment of the Point Lepreau plant would not be in the public interest, because refurbishment did not provide a significant economic advantage over alternatives. Despite that finding, NB Power and the New Brunswick government eventually went ahead with the refurbishment, which is now under way. Their decision to refurbish took account of "non-economic considerations".[139] In January 2004, in a communication to the CNSC, Hydro-Quebec conceded that the economic case for refurbishment of Gentilly 2 is weak.[140]

One cause of cost overruns for a refurbishment project could be the introduction of new CNSC requirements and objectives during the preparation or execution phase of the project. The potential for cost overruns of that type is a form of "regulatory risk". A prudent licensee will pay careful attention to the regulatory risk associated with refurbishment, especially if the economic case for refurbishment is weak.

In 2000, NB Power sought advice from the CNSC regarding regulatory risk for the Point Lepreau refurbishment project. The CNSC Staff responded in October 2000 as follows:[141]

> *"CNSC has no regulatory requirements or policies [sic] in place which specifically covers plant refurbishment for the purpose of life extension. The lack of such specific requirements reflects a regulatory risk to your project. Following are two ways in which this risk might be minimized:*
>
> 1. CNSC prepares policies and requirements that clearly state in a prescriptive manner what should be done in such situations. Experience has shown, however, that introducing new requirements of this type with the attendant need for extensive consultation with both the industry and the public is not an activity which can be completed in the short term. It seems unlikely that the time required would be consistent with your proposed schedule for decision making.
> 2. New Brunswick Power anticipates CNSC future requirements and voluntarily account [sic] for changes which have occurred since original licensing as well as changes which we can see are likely to occur in future. We suggest that if you decide to follow this approach, one of the key activities would be the early completion of a Periodic Safety Review similar to those carried out in a number of Western Countries and as described in IAEA 50-SG-012, *Periodic Safety Review of Operational Nuclear Power Plants*. If this were done, we believe that the prospects for regulatory certainty for the whole period of the life extension would be much improved."

138 WNA, 2008, Canada data.
139 NB Power, 2006, page 8.
140 Pageau, 2004, pages 4 and 5 of attached document.
141 Harvie, 2000.

That response from the CNSC should be viewed in the context of CNSC document RD-360, which was published in February 2008. As shown in Section 6, above, RD-360 does not set forth any specific risk objective regarding life extension. Thus, it is clear that the CNSC has eschewed the first regulatory approach outlined in its October 2000 response to NB Power. That approach, involving the establishment of firm requirements through stakeholder involvement, was eschewed because it could inconvenience licensees. Instead, the CNSC has employed a regulatory approach involving separate, ad hoc negotiations with each licensee. Those negotiations are conducted almost entirely in secret. An interested citizen cannot obtain specific information about risk-related actions that the CNSC requires from a licensee, and the rationale for those actions.

Hydro-Quebec, like NB Power, has sought to limit its regulatory risk by conducting its own, ad hoc negotiations with the CNSC regarding life extension for Gentilly 2. In a January 2004 submission to the CNSC, Hydro-Quebec conceded that the economic case for refurbishment and continued operation of Gentilly 2 is weak, and dependent upon assumptions about the regulatory requirements imposed by the CNSC. As a result, said Hydro-Quebec, an agreement with the CNSC about the scope of regulatory requirements would be essential if the refurbishment were to proceed.[142] Hydro-Quebec proposed a regulatory regime that would provide a basis for that agreement.[143]

In August 2008, Hydro-Quebec announced its plan to proceed with refurbishment of Gentilly 2. Thus, it can be presumed that Hydro-Quebec's negotiations with the CNSC led Hydro-Quebec to conclude that the regulatory risk for this project is known and limited. It is also likely that Hydro-Quebec was encouraged by the example of the Point Lepreau plant. Refurbishment of that plant, which is similar to Gentilly 2, is proceeding. However, Hydro-Quebec's judgment about regulatory risk could be faulty, as discussed below.

A COMPARISON BETWEEN THE LIFE EXTENSION PROCESSES FOR THE PICKERING B UNITS AND GENTILLY 2

Ontario Power Generation (OPG) is examining the merits of life extension for the four CANDU units at the Pickering B station. As part of that examination, OPG is engaged in negotiations with the CNSC regarding the risk of an unplanned release, and the actions needed to address that risk. The OPG-CNSC negotiations provide experience that is relevant to Gentilly 2.

OPG's negotiations with the CNSC are ad hoc and largely conducted in secret, just as for the Point Lepreau and Gentilly 2 cases. Nevertheless, enough information is publicly available to show that significant safety issues remain unresolved. The CNSC is requiring lengthy and expensive analyses of those issues by OPG. Findings from the new analyses could compel the CNSC to require the implementation of plant modifications as a condition of life extension for the Pickering B units.[144] The direct costs of the plant modifications, and the costs arising from associated delays in the planning and execution of refurbishment, could be substantial, thus weakening the economic case for life extension.

Section 4.3, above, shows that the ability of a CANDU reactor's shutdown systems to curb a reactivity excursion during a design-basis accident is now in question. The CNSC Staff raised this issue in an April 2008 review of a portion of OPG's Integrated Safety Review for the Pickering B units.[145] The Staff directed OPG to address the issue through lengthy, complex studies that include "a comprehensive revisit of the safety case supporting the shutdown systems [sic] effectiveness for AOOs and DBAs", and "a condition assessment and reconstitution of the reactor core nuclear design".[146] Those are demanding assignments. Other tasks required from OPG include substantial additional work on the PRA for Pickering B, to account for external initiating events, uncertainty, and other factors.[147]

Remarkably, the CNSC Staff has demanded that OPG assess the conformance of the Pickering B units with modern standards in the following respect:[148] "The reactor core incorporates inherent and/or passive safety features having a significant bearing on the probability or consequences of accidental release of radioactive materials."

142 Pageau, 2004, pages 4 and 5 of attached document.
143 Pageau, 2004.
144 Relevant plant modifications might include the use of low-enriched uranium (LVRF) fuel.
145 Schaubel, 2008.
146 Schaubel, 2008, page 14 and Appendix 1, page 1.
147 Schaubel, 2008, Sections 5.2.9 and 5.2.10.
148 Schaubel, 2008, Appendix 1, page 1.

That standard, if interpreted strictly, would rule out the continued operation of the Pickering B units or any other existing CANDU plant. Each of those plants has the inherent property that it will experience a violent reactivity excursion if shutdown systems fail during certain design-basis accidents.[149] Also, the shutdown systems are not fully passive. Thus, in responding to the Staff's demand, OPG must concede that the Pickering B units cannot conform to the specified standard. At that point, the CNSC must either retreat from the principles of inherent and passive safety, or refuse license extensions for the Pickering B units. The latter position would logically apply to all existing CANDU plants.

In theory, an environmental assessment (EA) is performed in connection with any proposed license extension. In practice, consistent with the CNSC's ad hoc approach to the approval of license extensions, licensees perform EAs to widely differing extents. For example, OPG has published risk-related documents as part of its EA process for Pickering B life extension.[150] Those documents provide a partial picture of OPG's work to assess the risk of an unplanned release. No comparable document is available for Gentilly 2. Indeed, no PRA has been completed for Gentilly 2.[151] Yet, it appears that the EA process for the life extension of Gentilly 2 was concluded in November 2006, when the CNSC accepted an EA screening report.[152] Apparently, the EA process will be much more thorough and informative in the Pickering B case than in the Gentilly 2 case.

The preceding paragraphs show that the CNSC is requiring OPG to perform lengthy studies of safety issues related to Pickering B life extension. There is a substantial likelihood that these studies will lead to the imposition of costly measures of risk mitigation, thereby weakening the economic case for life extension. OPG's approach to decision making suggests that OPG is aware of the potential for cost escalation. OPG has deferred a decision about refurbishing the Pickering B units until relevant studies are completed. By contrast, Hydro-Quebec reached a decision to refurbish Gentilly 2 without waiting for similar studies to be conducted.

Evidently, Hydro-Quebec believes that the studies being conducted by OPG, or similar studies that the CNSC might require in the context of Gentilly 2, have no significant bearing on the regulatory risk associated with refurbishment of Gentilly 2. What is Hydro-Quebec's basis for that belief? Three explanations seem plausible. First, Hydro-Quebec has conducted its own particular set of ad hoc negotiations with the CNSC, and may believe that these negotiations will shield Gentilly 2 from the imposition of new regulatory requirements. Second, Hydro-Quebec may believe that the CNSC will treat Gentilly 2 in the same manner as it has treated the Point Lepreau plant, even if new studies reveal significant safety problems. Third, Hydro-Quebec may be confident that new studies will not reveal any safety problem that motivates the CNSC to impose costly regulatory requirements.

These three explanations imply, in differing ways, that Hydro-Quebec does not regard the CNSC as a fully professional regulator. Such a regulator would relentlessly pursue unresolved safety issues such as the effectiveness of emergency shutdown systems. If significant problems were found, the regulator would not hesitate to require appropriate plant modifications and other risk-mitigating measures, regardless of any previous discussion with a licensee.

Hydro-Quebec may be correct in believing that there are no significant, unresolved safety problems at CANDU 6 plants. Also, Hydro-Quebec may be correct in believing that the CNSC is not a fully professional regulator. Neither belief represents a prudent approach to assessing regulatory risk.

PROFESSIONALISM OF THE CNSC

Observation of the CNSC reveals a current tension within the organization between its traditional regulatory approach, which has ad hoc and incestuous qualities, and a more modern and professional approach. If the latter approach prevails, the economic case for continued operation of Gentilly 2 is likely to become weaker.

One piece of evidence that illustrates the deficiencies of the traditional CNSC approach is the progressive weakening of the document RD-337 as it moved from pre-draft to draft to final versions. (See Section 5.2, above.) Similar pieces of evidence include the vagueness of the life extension requirements in RD-360 and the ad hoc, secretive manner in which life extensions are approved. Contrasting evidence, indicating the existence of professionalism within the CNSC, includes the CNSC Staff's insistence that OPG conducts detailed studies of safety issues related to life extension of the Pickering B units.[153]

149 Available information suggests that this outcome would still occur if LVRF fuel were used.
150 SENES, 2007.
151 NEA, 2007, page 324.
152 Keen, 2006.
153 Schaubel, 2008.

Tension between the CNSC's traditional regulatory approach and a more professional approach is illustrated by events in late 2007 and early 2008 related to AECL's operation of the NRU reactor at Chalk River.[154] That reactor produces a substantial fraction of the radioisotopes used for medical tests and procedures worldwide. Its continued operation is particularly important in light of AECL's failure to make the MAPLE reactors operational. In November 2007, the CNSC ordered AECL to cease operation of the NRU reactor, pending the upgrading of safety systems. CNSC had been dissatisfied with AECL's progress in making the upgrade. In December 2007, the Canadian parliament voted to override the CNSC's order, and the NRU reactor was re-started. Continuing conflict between the CNSC President and the Canadian government led to the President's dismissal in January 2008. An independent evaluation of these events led to findings including the following:[155]

> *"Based on a review of these events, and related internal and external communications of both organizations, a fundamental observation of the Talisman Team is that the CNSC regulatory program and the AECL regulatory compliance program are 'expert based' and not 'process based'. The regulatory effectiveness of both organizations can be significantly improved by developing and implementing formal processes, to be used for establishing and complying with regulatory requirements."*

8. Adapting the CNSC Process to Allow a Fuller Consideration of Risks

The current CNSC process for considering license extensions for nuclear power plants does not provide adequate assessments of the risk of an unplanned release, or of the options for reducing that risk. Also, the process provides no assessment of the risk that spent fuel will be diverted and used to produce plutonium. Finally, the process does not provide an adequate assessment of regulatory risk.

The licensing process could be adapted to provide thorough assessments of the risks in all three of these categories. Actions would be required from the CNSC, the Canadian government, and Hydro-Quebec. Each of those entities would engage closely with a full range of stakeholders while conducting an assessment. The assessments would be published, with limited exceptions for information that is sensitive from a security perspective.

The CNSC would require licensees to perform full-scope PRAs that examine unrestrained reactivity excursions and other fuel-damage scenarios. Complementary studies would be performed to assess the risks of unplanned releases caused by malevolent acts. Those PRAs and studies would be available for independent review. Licensees would be required to identify and characterize a range of risk-reducing options, including the use of low-enriched uranium fuel. Descriptions of the options and their effects on risk would be published. Much of the analysis and stakeholder engagement could be done within an EA framework, replacing the superficial and uninformative EAs that have been done to date.

The government of Canada would direct its relevant agencies, including the CNSC, to assess the risk that international marketing of the CANDU 6 will contribute to the risks of nuclear-weapon proliferation and nuclear war. The assessment would be published, with limited exceptions for sensitive information.

Hydro-Quebec would support the assessments being done by the CNSC and the Canadian government, by providing relevant information and analyses. Also, Hydro-Quebec would independently assess the regulatory risk associated with refurbishment of Gentilly 2, and the risk of onsite economic impacts from fuel-damage events. Those assessments would inform a review of the costs and benefits of refurbishing Gentilly 2. The risk assessments and the cost-benefit review would be published.

Legislators in Quebec and across Canada could promote and facilitate the performance of the assessments described above, through appropriate initiatives. If the CNSC, the Canadian government and Hydro-Quebec do not perform adequate assessments, legislators could sponsor alternative actions such as the conduct of independent hearings and studies.

Citizens could demand that governments and legislators at provincial and central levels perform, require and support the needed assessments. Citizen groups could conduct independent hearings and studies to examine risks associated with CANDU 6 plants.

154 CBC News, 2008.
155 Talisman, 2008, page i.

9. Conclusions and Recommendations

MAJOR CONCLUSIONS OF THIS REPORT ARE AS FOLLOWS:

C1. Operation of any nuclear power plant creates risks. Plants of the CANDU 6 design pose additional risks that arise from basic features of the design, especially the use of natural uranium as fuel and heavy water as moderator. Those features create additional risks in two respects. First, a CANDU 6 reactor could experience a violent power excursion, potentially leading to containment failure and a release of radioactive material to the environment. Second, spent fuel discharged from a CANDU 6 plant could be diverted and used to produce plutonium for nuclear weapons.

C2. An unplanned release of radioactive material at a CANDU 6 plant could be caused by an accident or a malevolent act. The release could cause adverse impacts within the plant and at offsite locations. The release scenario could involve a violent power excursion or other event sequences that damage nuclear fuel. The risk of an accidental release can be examined using techniques of probabilistic risk assessment, although the findings will contain irreducible uncertainty. PRA techniques can be adapted to examine the risk of a release due to malevolent action.

C3. Canada's nuclear industry has conducted probabilistic studies, but the country lacks a fully developed PRA culture. Notably, releases are not estimated for the most severe fuel-damage scenarios at CANDU plants, including a violent power excursion. Industry studies find very low probabilities for the releases that are considered. Those findings do not account for external events (e.g., earthquake), malevolent acts and other factors, and are not credible. The industry studies are not available for independent review, and thus do not satisfy a fundamental requirement of scientific discourse.

C4. The Canadian nuclear industry and the CNSC have assumed for three decades that the probability of a violent power excursion at a CANDU plant is very low, due to the use of two fast-acting shutdown systems. Recent investigations have shown that the analysis used to assess the performance of the shutdown systems is flawed. That analysis is crucial because the systems cannot be tested under representative conditions. Thus, the systems' effectiveness is questionable. The CNSC is now requiring licensees to perform new analyses and implement mitigation measures. That effort will be costly. One mitigation measure, scheduled for implementation at the Bruce station, is to use low-enriched uranium fuel.

C5. For three decades, Canada's nuclear safety regulator has regarded the risk of a violent power excursion as acceptable, because the risk is controlled by active systems. That position increasingly diverges from regulatory approaches in other countries. There is an emerging trend among nuclear safety regulators to prefer the use of passive and inherently safe systems. If that trend continues, a new CANDU 6 plant may no longer be licensable in many countries, especially if it uses natural uranium fuel. AECL may try to sell the ACR-1000 version of the CANDU in such countries, if the ACR-1000 functions as envisioned. The ACR-1000 is meant to have a negative void coefficient of reactivity, reducing its propensity to suffer a power excursion.

C6. Another emerging trend among nuclear regulators is to require that new nuclear power plants have some capability to ride out malevolent actions. The CNSC, for example, says that it will impose requirements of this type. One manifestation of the requirements will be a containment structure that is more robust against attack. The CANDU 6 design does not have specific provisions to resist attack. Thus, if the trend continues, a new CANDU 6 may no longer be licensable in many countries.

C7. The CNSC has established design criteria for licensing new nuclear power plants in Canada. Those criteria have a high degree of flexibility, and do not assign preference to passive and inherent safety. The CNSC criteria for approving license extensions for existing Canadian plants are vague. It is difficult to determine the stringency with which the CNSC applies those criteria to license extensions, and the extent to which all plants seeking license extensions will be treated equally. That uncertainty reflects a current tension within the CNSC between its traditional regulatory approach, which has ad hoc and incestuous qualities, and a more modern and professional approach. If the professional approach prevails, then any Canadian licensee seeking a CANDU license extension will be obliged to conduct lengthy and expensive studies on shutdown-system effectiveness and other matters. The licensee could be required to implement risk-mitigating measures, which could be costly.

C8. In view of the considerations set forth in Conclusion C7, Hydro-Quebec faces significant regulatory uncertainty regarding the extension of the Gentilly 2 operating license. No PRA is currently available for Gentilly 2. Also, the CNSC is requiring lengthy and expensive safety analyses in the context of the Pickering B license extension, and it appears that no comparable analyses have been done for Gentilly 2. Thus, if the CNSC takes a uniform, professional approach to all license extensions, Hydro-Quebec will be obliged to conduct lengthy safety studies and may be required to implement costly modifications of the Gentilly 2 plant. Also, delays pursuant to CNSC requirements could arise during the refurbishment of Gentilly 2. Hydro-Quebec has already stated that the economic case for refurbishment and life extension of Gentilly 2 is weak. Accounting for regulatory uncertainty could further weaken that case. Additional weakening could come from consideration of the risk of onsite economic impacts from fuel-damage events.

C9. A national government seeking an actual or reserve capability to deploy nuclear weapons would develop indigenous sources of plutonium and/or highly-enriched uranium. A proven option in this regard is to develop a nuclear fuel cycle in which natural uranium is used to fuel reactors that employ on-line refueling. CANDU 6 reactors would meet this requirement. A country possessing CANDU 6 reactors would always have the option of diverting spent fuel from these reactors to produce plutonium for nuclear weapons. That option would be much less readily available to a country possessing light-water reactors.

C10. Canada has the capability to deploy nuclear weapons, but is unlikely to do so. Thus, continued operation of Gentilly 2 would not directly contribute to the proliferation of nuclear weapons. However, AECL would undoubtedly use the refurbishment and continued operation of Gentilly 2 as an asset in its worldwide marketing of the CANDU 6. Success in that endeavor could contribute to an increase in the number of countries possessing nuclear weapons. That trend would, in turn, increase the probability of nuclear war. Accordingly, when citizens of Quebec and other Canadian provinces weigh the costs and benefits of refurbishing Gentilly 2, they should consider the contribution of the refurbishment to the risk of nuclear war. At present, there is no public process to consider that risk.

BASED ON THE PRECEDING CONCLUSIONS AND THE BODY OF THIS REPORT, RECOMMENDATIONS FOR ACTION ARE OFFERED HERE AS FOLLOWS:

R1 (to the CNSC). In considering the approval of license extensions for nuclear power plants, the CNSC should apply stringent criteria of safety and security, and should do so uniformly for all licensees. Also, the CNSC should require licensees to perform full-scope PRAs that examine unrestrained reactivity excursions and other fuel-damage scenarios. Complementary studies should be performed to assess the risks of unplanned releases caused by malevolent acts. Those PRAs and studies should be available for independent review. Licensees should be required to identify and characterize a range of risk-reducing options, including the use of low-enriched uranium fuel. Descriptions of the options and their effects on risk should be published.

R2 (to the Canadian government). The government of Canada should direct its relevant agencies, including the CNSC, to assess the risk that international marketing of the CANDU 6 will contribute to the risks of nuclear-weapon proliferation and nuclear war. That assessment should be published, with limited exceptions for sensitive information.

R3 (to Hydro-Quebec). Hydro-Quebec should support recommendations R1 and R2 by taking appropriate actions. Also, Hydro-Quebec should independently assess the regulatory risk associated with refurbishment of Gentilly 2, and the risk of onsite economic impacts from fuel-damage events. Those assessments should inform a review of the costs and benefits of refurbishing Gentilly 2. The risk assessments and the cost-benefit review should be published.

R4 (to legislators). Legislators in Quebec and across Canada should support recommendations R1 through R3 through appropriate initiatives. If the CNSC, the Canadian government and Hydro-Quebec do not act on those recommendations, legislators should consider sponsoring alternative actions such as the conduct of independent hearings and studies.

R5 (to citizens). Citizens should demand that governments and legislators at provincial and central levels support recommendations R1 through R4. Citizen groups should consider conducting independent hearings and studies to examine risks associated with CANDU 6 plants.

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Table 2-1

Selected Data on Design of CANDU 6 and ACR-1000 Nuclear Power Plants

DESIGN FEATURE	CANDU 6	ACR-1000
Thermal output (MWt)	2,064	3,187
Electrical Output (MWe)	728 (gross), 666 (net)	1,165 (gross), 1,085 (net)
Fuel type	Natural UO2	Low-enriched UO2
Fuel burnup (MWt-day per kgU)	7.5	>20
Heavy water inventory, moderator (Mg D2O)	265	250
Heavy water inventory, coolant (Mg D2O)	192	0
Coolant pressure at reactor outlet (MPa)	9.9	11.1
Coolant temperature at reactor outlet (deg. C)	310	319
Containment type	Pre-stressed concrete cylinder/ dome, epoxy liner	Pre-stressed concrete cylinder/ dome, steel liner
Containment wall thickness (m)	1.07	1.8
Containment inside diameter (m)	41.4	56.5
Containment height, base slab to top of dome (m)	51.2	74.0

SOURCE: Petrunik, 2007.

Table 2-2

Estimated Inventory of Selected Radioactive Isotopes in the Core of the Gentilly 2 or Indian Point 2 Reactor at Full Power, Steady State

RADIOACTIVE ISOTOPE	Core Inventory (thousand TBq)		Normalized Core Inventory (TBq per MWt)	
	Gentilly 2	Indian Point 2	Gentilly 2	Indian Point 2
Iodine-131 (Half-life = 8 days)	2,400	3,200	1,163	995
Cesium-137 (Half-life = 30 years)	51	420	24.8	129
Plutonium-239 (Half-life = 24,000 years)	?	1.2	?	0.38

NOTES:

(a) Gentilly 2 data are from: ISR, 2003, Annex B, Table 35. That source does not provide a core inventory of Plutonium-239.

(b) Indian Point data are from: Entergy, 2007, Appendix E, Table E.1-13.

(c) Here, it is assumed that the Gentilly 2 reactor has a power capacity of 2,064 MWt, and the Indian Point 2 reactor has a capacity of 3,216 MWt.

(d) The higher normalized inventory of Iodine-131 for Gentilly 2, compared with Indian Point 2, presumably reflects a combination of differences in the neutron spectrum, the fissile composition of the core, and the estimation model.

(e) Data provided by Entergy suggest that Indian Point 2 fuel is typically driven to a burnup of about 59 MWt-day per kgU.

Table 2-3

CANDU 6 Nuclear Power Plants in Operation Worldwide

NAME OF UNIT	Location	In-Service Date	Gross Output (MWe)
Point Lepreau	Canada	February 1983	680
Wolsong 1	Korea	April 1983	679
Gentilly 2	Canada	October 1983	675
Embalse	Argentina	January 1984	648
Cernavoda 1	Romania	December 1996	706
Wolsong 2	Korea	July 1997	715
Wolsong 3	Korea	July 1998	715
Wolsong 4	Korea	October 1999	715
Qinshan 1	China	December 2002	728
Qinshan 2	China	July 2003	728
Cernavoda 2	Romania	October 2007	706

SOURCE: AECL website, http://www.aecl.ca/Reactors/CANDU6/CANDU6-Units.htm, accessed on 16 July 2008.

Table 3-1

Some Potential Modes and Instruments of Attack on a Nuclear Power Plant

ATTACK MODE/INSTRUMENT	Characteristics	Present Defenses at US Plants
Commando-style attack	→ Could involve heavy weapons and sophisticated tactics → Successful attack would require substantial planning and resources	Alarms, fences and lightly-armed guards, with offsite backup
Land-vehicle bomb	→ Readily obtainable → Highly destructive if detonated at target	Vehicle barriers at entry points to Protected Area
Small guided missile (anti-tank, etc.)	→ Readily obtainable → Highly destructive at point of impact	None if missile launched from offsite
Commercial aircraft	→ More difficult to obtain than pre-9/11 → Can destroy larger, softer targets	None
Explosive-laden smaller aircraft	→ Readily obtainable → Can destroy smaller, harder targets	None
10-kilotonne nuclear weapon	→ Difficult to obtain → Assured destruction if detonated at target	None

NOTES:

(a) This table is adapted from: Thompson, 2007, Table 7-4. Further citations are provided in that table and its supporting narrative.

(b) Defenses at Canadian plants are no more robust than at US plants. See: Frappier, 2007.

Table 3-2

The Shaped Charge as a Potential Instrument of Attack

CATEGORY OF INFORMATION	Selected Information in Category
General information	→ Shaped charges have many civilian and military applications, and have been used for decades → Applications include human-carried demolition charges or warheads for anti-tank missiles → Construction and use does not require assistance from a government or access to classified information
Use in World War II	→ The German MISTEL, designed to be carried in the nose of an un-manned bomber aircraft, is the largest known shaped charge → Japan used a smaller version of this device, the SAKURA bomb, for kamikaze attacks against US warships
A large, contemporary device	→ Developed by a US government laboratory for mounting in the nose of a cruise missile → Described in detail in an unclassified, published report (citation is voluntarily withheld here) → Purpose is to penetrate large thicknesses of rock or concrete as the first stage of a "tandem" warhead → Configuration is a cylinder with a diameter of 71 cm and a length of 72 cm → When tested in November 2002, created a hole of 25 cm diameter in tuff rock to a depth of 5.9 m → Device has a mass of 410 kg; would be within the payload capacity of many general-aviation aircraft
A potential delivery vehicle	→ A Beechcraft King Air 90 general-aviation aircraft will carry a payload of up to 990 kg at a speed of up to 460 km/hr → A used King Air 90 can be purchased in the US for $0.4-1.0 million

SOURCE: Thompson, 2007, Table 7-6. Further citations are provided in that table and its supporting narrative.

Table 3-3

Estimated Discharge of Plutonium from Nuclear Power Reactors, 1961-2010: Selected Countries and World Total

COUNTRY	Cumulative Discharge of Plutonium (kg)		
	1961-1993	1994-2010	1961-2010
ARGENTINA	5,970	12,200	18,170
BRAZIL	520	4,400	4,920
CANADA	67,230	99,270	166,500
INDIA	4,500	21,120	25,620
KOREA (SOUTH)	14,670	49,870	64,540
PAKISTAN	410	780	1,190
SOUTH AFRICA	2,340	5,600	7,940
WORLD TOTAL	846,200	1,278,760	2,124,960

SOURCE: Albright et al, 1997, Tables 5.3 and 5.4.

Table 4-1

Ontario Hydro Estimate of the Risk of Onsite Economic Impacts from Fuel-Damage Events at the Darlington Nuclear Power Plants (Existing CANDU Plants)

FUEL DAMAGE CATEGORY	Est. Mean Probability (Uncertainty Factor)	Est. Onsite Economic Impacts (million 2008 Can $)	Risk of Onsite Economic Impacts (million 2008 Can $ per RY)	
			Using Mean Estimate of FDC Probability	Using 95th Percentile Estimate of FDC Probability
FDC0	3.8E-06 per RY (UF = 6)	?	?	?
FDC1	2.0E-06 per RY (UF = 6)	6,400 to 11,500	0.013 to 0.023	0.077 to 0.14
FDC2	8.0E-05 per RY (UF = 6)	5,800 to 10,200	0.46 to 0.82	2.80 to 4.90
FDC3	4.7E-04 per RY (UF = 4)	3,400 to 5,900	1.60 to 2.80	6.40 to 11.10
FDC4	3.0E-05 per RY (UF = 10)	3,400 to 6,200	0.10 to 0.19	1.02 to 1.90
FDC5	1.0E-04 per RY (UF = 10)	2,700 to 5,200	0.27 to 0.52	2.70 to 5.20
FDC6	2.0E-03 per RY (UF = 10)	1,900 to 3,700	3.80 to 7.40	38.0 to 74.0
FDC7	3.0E-03 per RY (UF = 5)	790 to 2,500	2.40 to 7.50	11.90 to 37.5
FDC8	2.0E-03 per RY (UF = 10)	120 to 600	0.24 to 1.20	2.40 to 12.0
FDC9	2.3E-02 per RY (UF = 3)	390 to 700	8.97 to 16.10	26.9 to 48.3
TOTAL RISK			**17.9 to 36.6**	**92.2 to 195.0**

NOTES:

(a) Estimates are from the Darlington Probabilistic Safety Evaluation (DPSE). See: Ontario Hydro, 1987, Tables 5-2, 5-8 and 5-9. For additional data from the full version of DPSE, see: IRSS, 1992, Volume 2, Annex IV.

(b) DPSE provided cost estimates in 1985 Can $. These are adjusted here to 1991 Can $ by a multiplier of 1.25 (see: IRSS, 1992, Volume 2, Annex IV), and from 1991 Can $ to 2008 Can $ by a multiplier of 1.36 (CPI inflator from Bank of Canada). The combined multiplier is 1.70.

(c) DPSE did not estimate the risk of onsite economic impacts for FDC0.

(d) These estimates are limited to fuel damage in a reactor core or a fueling machine, caused by accidents initiated by internal events.

(e) Replacement power is the dominant component of the estimated onsite economic impacts. The other component considered by DPSE is the cost of decontamination and repair.

(f) The range of estimated onsite economic impacts is from a “best estimate” (lower bound) to a “probable maximum” (upper bound).

(g) The Darlington station has four CANDU units (plants) that share many safety and support systems (e.g., fueling duct and vacuum building), which means that a fuel-damage event at one unit could readily lead to adverse impacts on the other units. DPSE determined that accidents in categories FDC1 through FDC9 would lead to forced outage of all four units. For example, given the occurrence of an FDC1 accident, the estimated duration of the forced outage would be 45-72 months for all four units, and an additional 65-126 months for the unit that suffered fuel damage.

(h) The uncertainty factor (UF) in the second column is DPSE’s estimate of the ratio of the 95th percentile value to the mean value.

Table 4-2

Risk Costs of Onsite Impacts of Fuel-Damage Events at Existing CANDU Plants in Ontario, Using an Ontario Hydro Estimate of the Risk of Economic Impacts at the Darlington Plants

INDICATOR	Value of Indicator	
	Using Mean Estimate of Probabilities of Fuel Damage Categories	Using 95th Percentile Estimate of Probabilities of Fuel Damage Categories
Risk of onsite economic impacts	17.9 to 36.6 (million 2008 Can $ per RY)	92.2 to 195.0 (million 2008 Can $ per RY)
Risk costs of onsite economic impacts (OH estimate for internal initiating events only)	0.26 to 0.53 (2008 Can cent per kWh)	1.33 to 2.81 (2008 Can cent per kWh)
Risk costs of onsite economic impacts (internal initiating events + external events + malevolent acts)	0.5 to 1.1 (2008 Can cent per kWh)	2.7 to 5.6 (2008 Can cent per kWh)

NOTES:

(a) Ontario Hydro considered the occurrence of accidents involving Fuel Damage Categories FDC1 through FDC9, but not the most severe Category (FDC0).

(b) Ontario Hydro considered fuel damage in a reactor core or a fueling machine, caused by accidents initiated by internal events.

(c) Values in the first row are from Table 4-1. Values in the second row are calculated from the first row.

(d) Values in the third row are adjusted upward from values in the second row by a factor of 2, to account for accidents initiated by external events, and for malevolent acts.

(e) Each Darlington plant has a capacity of 0.88 GWe. A capacity factor of 0.9 is assumed here.

(f) This table also appears in: Thompson, 2008b.

Table 5-1

Safety Objectives for New Nuclear Power Plants, as Specified in IAEA Safety Standards Series Document NS-R-1

OBJECTIVE	Characteristics of Objective
General Nuclear Safety Objective	→ Protect individuals, society and the environment from harm by establishing and maintaining effective defenses against radiological hazards
Radiation Protection Objective	→ Ensure that, in all operational states, radiation exposure within the plant, or due to any planned release of radioactive material from the plant, is kept below prescribed limits and as low as reasonably achievable → Ensure mitigation of the radiological consequences of any accident
Technical Safety Objective	→ Take all reasonably practicable measures to prevent accidents and to mitigate their consequences should they occur → Ensure, with a high level of confidence, that the radiological consequences of any accident taken into account in designing the plant would be minor and below prescribed limits → Ensure that the likelihood of accidents with serious radiological consequences is extremely low

SOURCE: International Atomic Energy Agency, *Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1*.
See: IAEA, 2000, pages 3 and 4.

Table 5-2

Hierarchy of Nuclear Power Plant Design Characteristics Relevant to Safety, as Specified in IAEA Safety Standards Series Document NS-R-1

PREFERENCE IN SELECTING A PLANT DESIGN FEATURE	Design Characteristics Relevant to Safety The expected plant response to any postulated initiating eventshall be those of the following that can reasonably be achieved ……
First Preference	No significant safety-related effect, or a change toward a safe condition by virtue of inherent characteristics of the plant
Second Preference	The plant is rendered safe by passive safety features, or by the action of safety systems that are continuously operating
Third Preference	The plant is rendered safe by the action of safety systems that are brought into service in response to the initiating event
Fourth Preference	The plant is rendered safe by specified procedural actions

SOURCE: International Atomic Energy Agency, *Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. NS-R-1*.
See: IAEA, 2000, page 11.

Table 5-3

Safety Goals for a New Nuclear Power Plant, as Specified in CNSC Draft Regulatory Document RD-337

TYPE OF OUTCOME	Safety Goals	
	Sum of frequencies of all event sequences that can lead to this outcome ……..	
	Should be less than	Shall not exceed
Small Release to the Environment (more than 1,000 TBq of Iodine-131)	1 per 1 million plant-years	1 per 100,000 plant-years
Large Release to the Environment (more than 100 TBq of Cesium-137)	1 per 10 million plant-years	1 per 1 million plant-years
Core Damage (significant core degradation)	1 per 1 million plant-years	1 per 100,000 plant-years

NOTES:

(a) The table as shown describes the safety goals set forth by the CNSC in October 2007 in the document: *Design of New Nuclear Power Plants, RD-337, Draft*. See: CNSC, 2007a, page 5.

(b) On 27 May 2008, the CNSC Staff published a document (Dallaire et al, 2008) containing a revised version of RD-337, which the Staff submitted to the CNSC Commissioners for approval at the Commission meeting of 10 June 2008. At page 5 of the revised RD-337, revised safety goals are set forth, exhibiting the following changes from the table above. First, the numerical goals in the "should be less than" category are abandoned. Second, the numerical goals in the "shall not exceed" category are retained, but with different language. The revised RD-337 states that the sum of frequencies of all event sequences that can lead to a specified outcome "is less than" a numerical value. Each of these changes represents a significant weakening of the safety goals.

Table 5-4

Proposed Safety Criteria for Design and Siting of a New Nuclear Power Plant

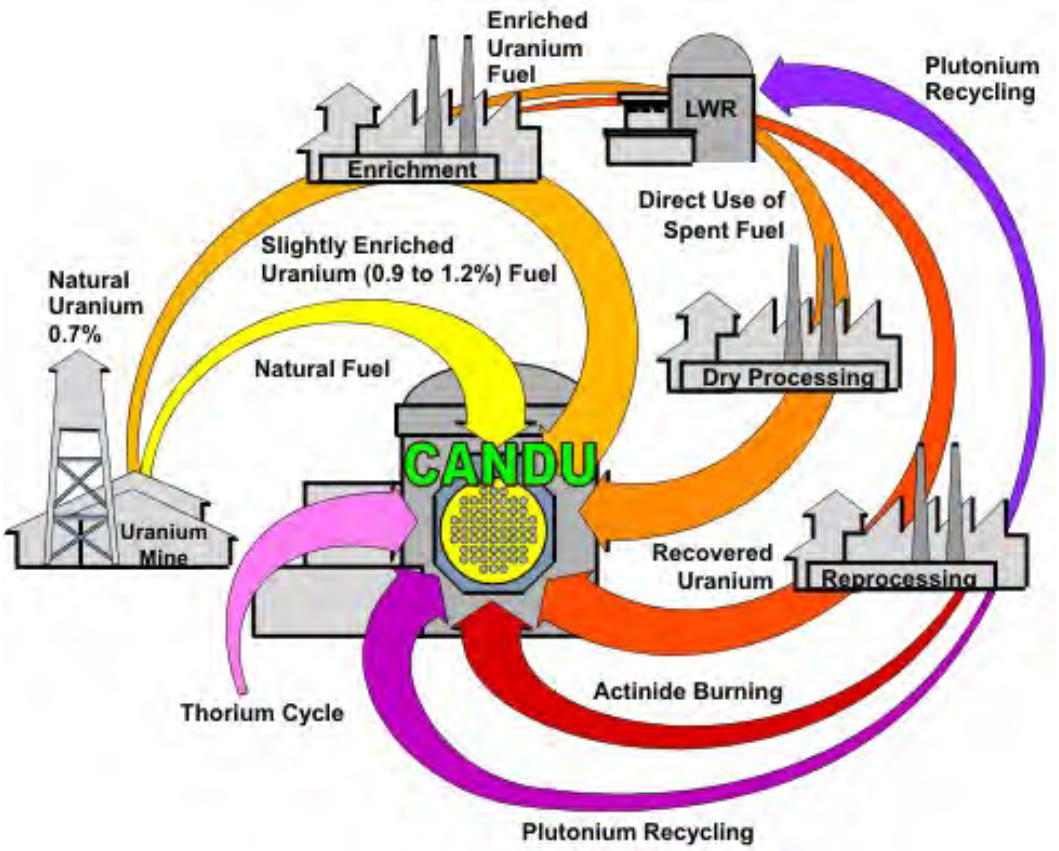
APPLICATION OF CRITERIA	Criteria
Safety performance of the plant during reactor operation (design-basis criteria)	No significant damage of the reactor core or adjacent stored spent fuel in the event of: → Loss of all electrical power (AC & DC), compressed air, other power sources, and normal heat sinks for an extended period (e.g., 1 week); → Abandonment of the plant by operating personnel for an extended period (e.g., 1 week); → Takeover of the plant by hostile, knowledgeable persons who are equipped with specified explosive devices, for a specified period (e.g., 8 hours); → Military attack by specified means (e.g., 1,000-pound air-dropped bombs); → An extreme, specified earthquake; → Conceivable erroneous operator actions that could be accomplished in a specified period (e.g., 8 hours); or → Any combination of the above.
Safety performance of the plant during reactor refueling (design-basis criteria)	A specified maximum release of radioactive material to the accessible environment in the event of: → Loss of reactor coolant at a specified time after reactor shut-down, with replacement of the coolant by fluid (e.g., air, steam, or un-borated water) creating the chemical and nuclear reactivity that would maximize the release of radioactive material, at a time when the plant's containment is most compromised; and → Any combination of the events specified above, in the context of reactor operation.
Site specification (radiological-impact criteria)	In the event of the maximum release of radioactive material specified above, in the context of reactor refueling, radiological impacts would not exceed specified values regarding: → Individual dose; → Population dose; and → Land areas in various usage categories that would be contaminated above specified levels.

NOTE:

The criteria in the first two rows of this table would apply to spent fuel stored adjacent to the reactor core. Separate criteria would apply to an independent facility for storing spent fuel, whether onsite or offsite.

Figure 2-1

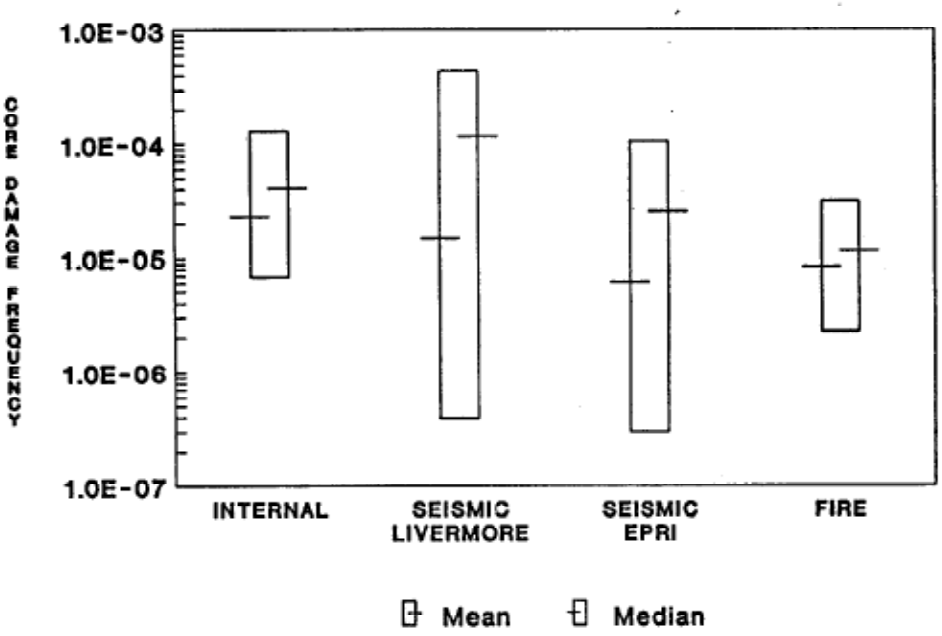
Application of the CANDU Reactor to Various Fuel Cycles, as Envisioned by AECL



SOURCE: AECL.

Figure 3-1

Core Damage Frequency for Accidents at a Surry PWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



NOTES:

(a) This figure is adapted from Figure 8.7 of: NRC, 1990.

(b) The bars range from the 5th percentile (lower bound) to the 95th percentile (upper bound) of the estimated core damage frequency (CDF). CDF values shown are per reactor-year (RY).

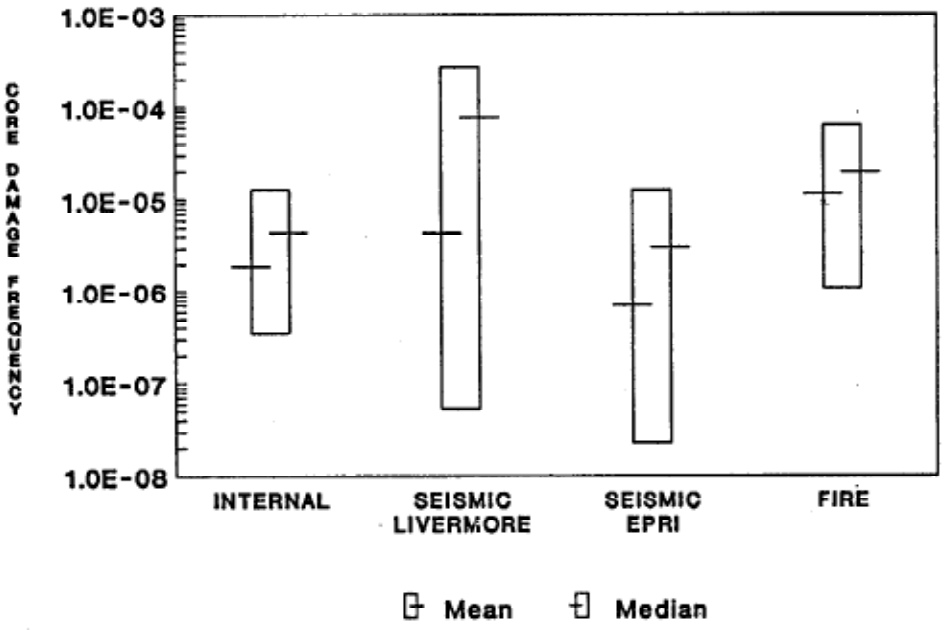
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One is from Lawrence Livermore National Laboratory (Livermore), the other is from the Electric Power Research Institute (EPRI).

(d) CDFs are not estimated for external initiating events other than earthquakes and fires.

(e) Malevolent acts are not considered.

Figure 3-2

Core Damage Frequency for Accidents at a Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



NOTES:

(a) This figure is adapted from Figure 8.8 of: NRC, 1990.

(b) The bars range from the 5th percentile (lower bound) to the 95th percentile (upper bound) of the estimated core damage frequency (CDF). CDF values shown are per reactor-year (RY).

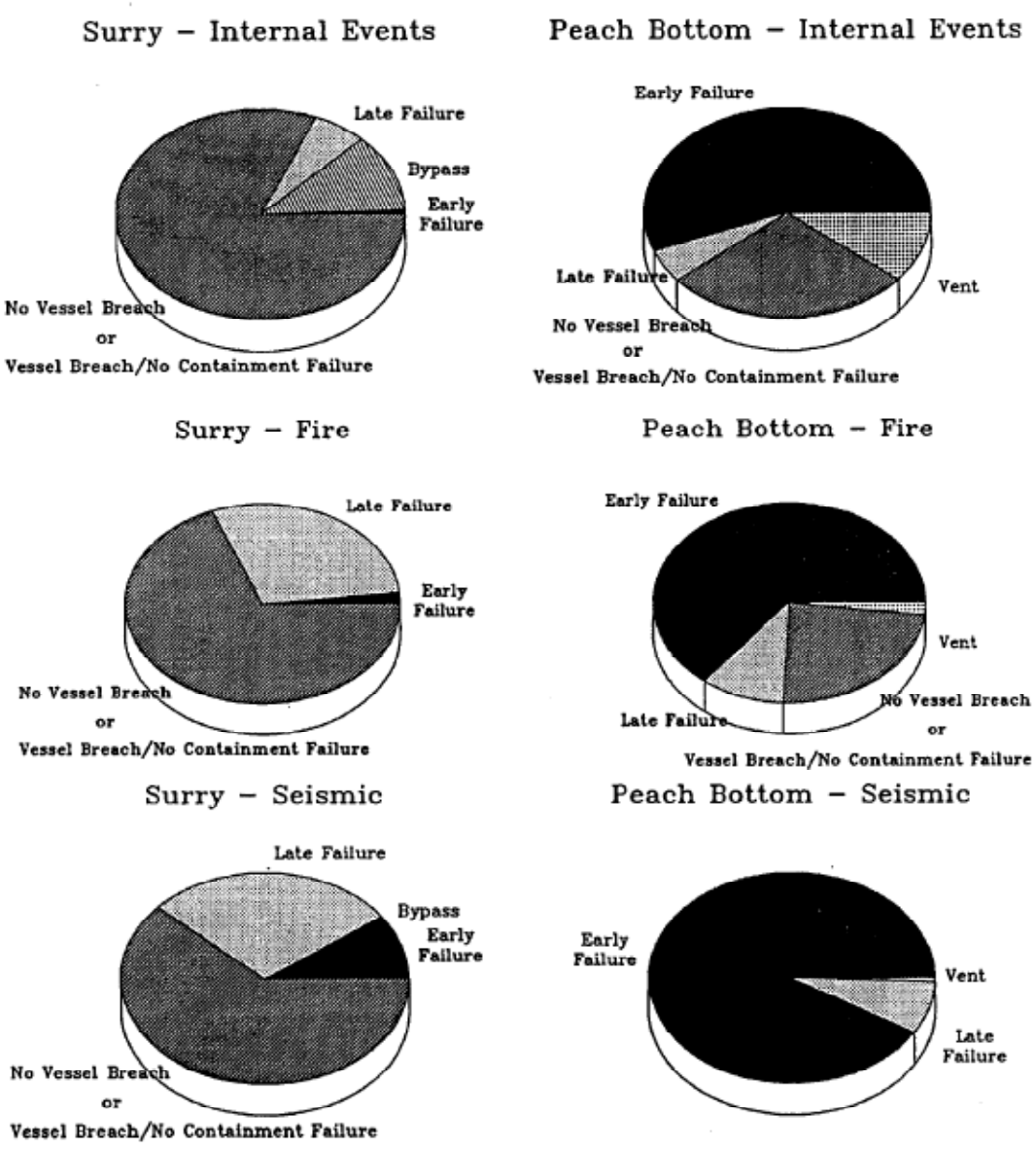
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One is from Lawrence Livermore National Laboratory (Livermore), the other is from the Electric Power Research Institute (EPRI).

(d) CDFs are not estimated for external initiating events other than earthquakes and fires.

(e) Malevolent acts are not considered.

Figure 3-3

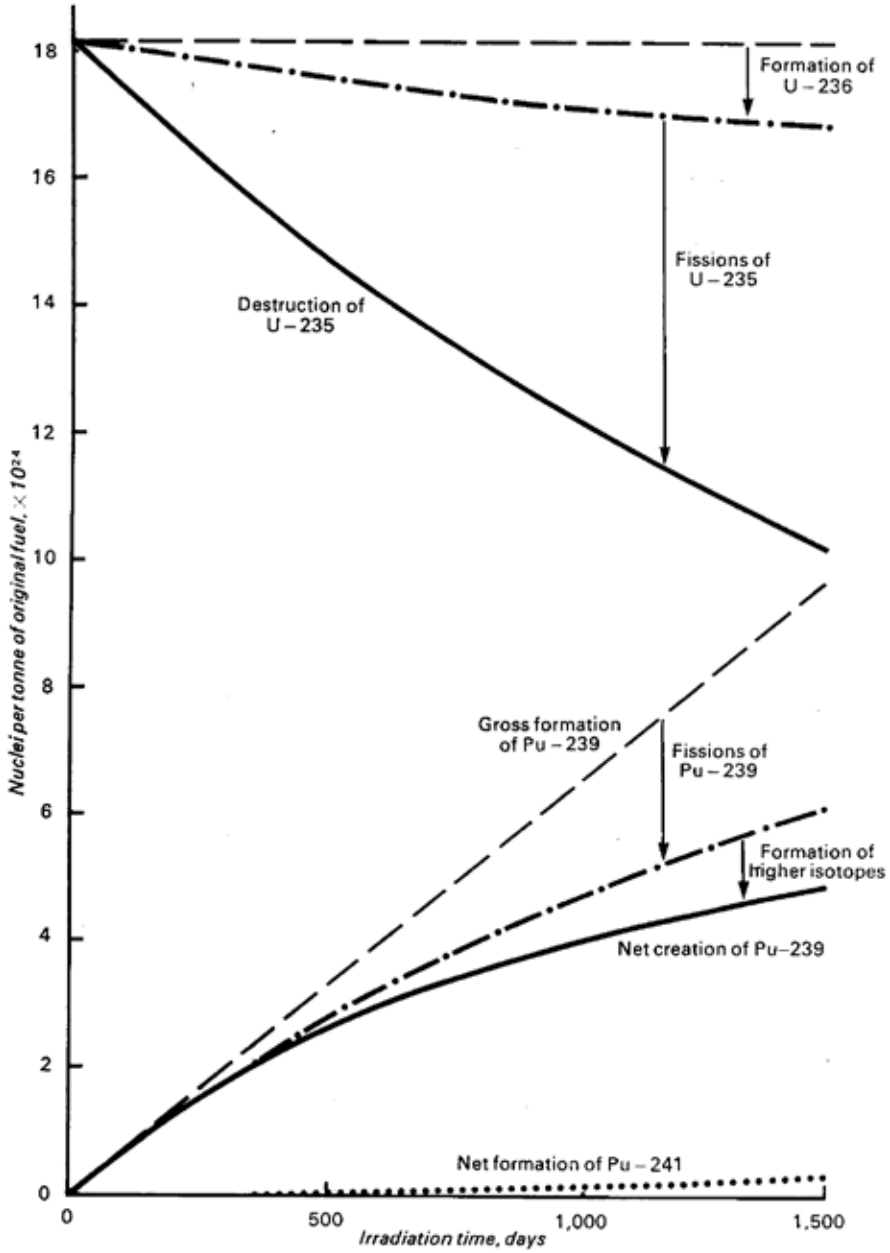
Conditional Probability of Containment Failure Following a Core-Damage Accident at a Surry PWR or Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



NOTE: This figure is adapted from Figure 9.5 of: NRC, 1990.

Figure 3-4

Trends in Numbers of Fissile Nuclei During Irradiation of Magnox Fuel



The changes in numbers of fissile nuclei in Magnox fuel during irradiation, showing the contribution made by plutonium- 239

SOURCE: Flowers et al, 1976, Figure 5.

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Backgrounder on Probabilistic Risk Assessment

Printable Version

On this page:

- Background
- Risk
- Risk Assessment Methods
- Types of Risk Assessments
- Risk Assessment Results
- NRC Uses of PRA

Background

The Nuclear Regulatory Commission's responsibilities include ensuring U.S. nuclear power plants and other licensed facilities operate with minimal risk to public health and safety. The NRC uses the probabilistic risk assessment – often

called PRA – techniques to examine a complex system's potential risk and identify what problems could have the most impact on public health and safety.

The NRC lists technical and regulatory requirements on plant design and operations in Title 10 of the Code of Federal Regulations (10 CFR). The requirements are often written in terms of engineering practices that add "safety margins" and "defense-in-depth" during plant design, construction, and operation. PRAs can evaluate the effectiveness of safety margins and defense-in-depth, and in some cases improve them.

Risk

Risk comes from three factors:

- What can go wrong?
- How often can something go wrong (how likely is it?)?
- What happens if it goes wrong (consequence)?

We reduce risk by making a bad event less likely or by reducing its consequences. The NRC and the nuclear industry use PRA as one way to evaluate and reduce overall risk.

TOP

Risk Assessment Methods

Performing a PRA requires several steps:

Specify the **hazard** – the outcome(s) to be prevented or reduced. For nuclear power plants, the focus is reducing the chance of damaging the fuel in the reactor core.

Identify **initiating events (i.e, what can go wrong)?** – things that could possibly cause the hazard (e.g., breaking a pipe carrying water to cool the core).

Estimate the **frequency (i.e., how likely is it)?** of each initiating event by answering questions such as, "How often do we expect a pipe of this size to break?"

What happens if the hazard happens (i.e., what are the consequences)?

For each initiating event, risk analysts identify combination of failures (e.g., pump failure and valve failure) that lead to a specific outcome. Analysts then calculate the likelihood of each combination of events and add up the probabilities of all the

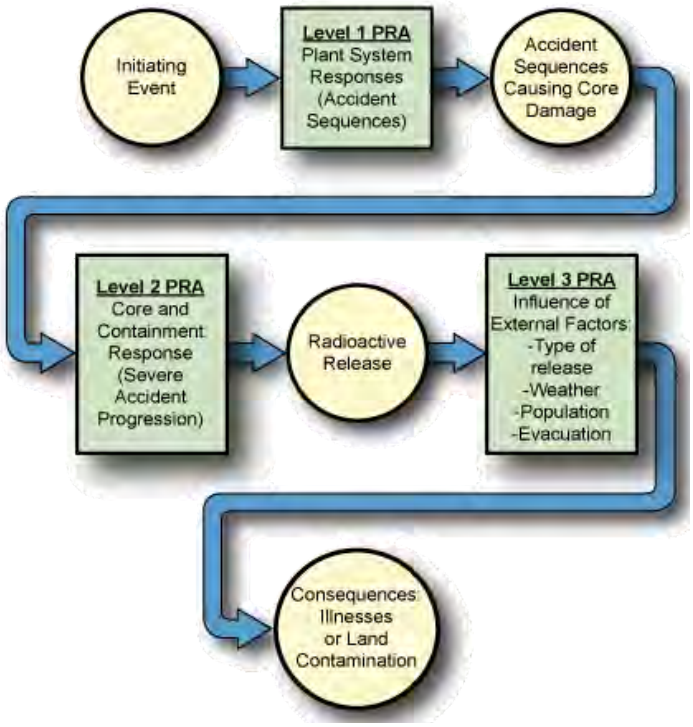
sequences that lead to the same outcome. The likelihood of the outcome equals the sequence probabilities multiplied by the frequency of the initiating event(s).

PRA uses several specific techniques to accomplish this analysis:

- *Event trees* model the plant response to each initiating event.
- *Fault trees* model plant systems in detail so that analysts can identify the combinations of failures that disable an overall system. The fault tree logic also calculates the overall failure probability. Analysts pay particular attention to problems that can fail more than one component at the same time.
- *Human reliability analysis* evaluates human errors that are important to the outcome of an event. Analysts assess the probability of human mistakes based on factors such as training, procedures, and expected conditions during an event.
- *Monte Carlo* methods are a variety of computer modeling that are used, if necessary, to help account for uncertain parts of the analysis. These methods consider variations in each factor of the analysis, imperfect knowledge and the many possible ways the factors can interact.

TOP

Types of Risk Assessments



Developing even the most basic PRA model requires a lot of effort. Modern computers have enough speed and power, however, so that analysts use, re-use, and refine PRA models to address many questions.

Nuclear power plant PRAs deal with “internal events” – those that start inside the power plant or the electric system it serves – and “external events” such as earthquakes and hurricanes. PRAs can also address unique situations such as spent nuclear fuel storage cask design or the geology of a potential site for permanently storing high-level radioactive waste.

Nuclear power industry PRAs fall into three levels:

- A Level 1 PRA estimates how often a reactor core could be damaged. It starts with well-understood conditions, usually a reactor operating at full power. A Level 1 PRA also models a reactor’s protective systems. These systems are so well understood that a Level 1 PRA has relatively small uncertainties.
- A Level 2 PRA estimates how much radioactive material reaches the environment, and how quickly the material could be released. Level 2 PRAs are less precise, due to more uncertainties such as how much water

or steam escapes the reactor systems (and how violently), as well as variations in how the reactor's containment structure responds.
- A Level 3 PRA estimates the health effects and economic losses that might result if radioactive material reaches the environment. Level 3 PRAs have the largest uncertainties, since they include highly variable factors such as wind speed and direction.

TOP

Risk Assessment Results

PRA results are complex and can't be reduced to a single number. Instead, PRAs provide a spectrum of possible outcomes. The frequency for each outcome is provided as a *distribution* of values. PRAs help understand how much larger or smaller an outcome's actual risks might be.

PRA results are uncertain because reality is more complex than any computer model, because analysts have imperfect information, and partly because of chance. Analysts and regulators maintain confidence that PRAs help achieve adequate safety in one of two ways:

1. Imposing strict enough safety margin and defense-in-depth to account for the uncertainties in estimated risks; or
2. Increasing the PRA's certainty so that a smaller margin can provide the same (or better) confidence of safety.

Analysts reduce uncertainty by a) enhancing their models to more accurately reflect the real world; b) incorporating research results to improve the physical processes they model; or c) collecting additional data to improve model precision. The NRC does all of these.

TOP

NRC Uses of PRA

The NRC first developed nuclear power plant PRAs in the 1970s. Since then the agency has refined its methods and developed new risk insights. The NRC combines these insights with traditional engineering methods to make regulatory decisions about power plants, medical uses of nuclear materials, and the handling of nuclear waste. This "risk-informed" approach to regulation includes:

- Plants use PRA for integrated plant evaluations that discover and correct subtle vulnerabilities, resulting in significant improvements to reactor safety.

- Inspections use PRA insights to focus on plant systems, operations and human performance that are most important to safety.
- The Reactor Oversight Program's significance determination process uses plant-specific PRA models to assess the safety impact when a piece of equipment fails or is taken out of service for maintenance. The NRC increases its inspection and oversight as nuclear plant problems increase in risk importance.
- The NRC often uses PRA to confirm that new or revised rules are rigorous enough to cover uncertainties – and to justify new requirements.
- The NRC has used special PRAs to assess issues such as spent fuel storage cask safety.

The nuclear industry uses PRA to:

- Enhance existing plant designs by reducing vulnerabilities.
- Reduce risk when multiple systems are being maintained.
- Focus resources on the most safety-significant systems and components, through risk-informed technical specifications, risk-informed in-service inspection programs, and risk-informed categorization of structures, systems, and components.
- Analyze and enhance new reactor designs before asking the NRC to certify them.

The NRC expects wider PRA use should result in more predictable and timely agency actions.

The NRC's website has a more detailed discussion on PRA.

January 2024

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Page Last Reviewed/Updated Tuesday, August 11, 2026

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Perspective

An Overview of Probabilistic Safety Assessment for Nuclear Safety: What Has Been Done, and Where Do We Go from Here?

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Abstract: The paper provides an introduction to the concept of Probabilistic Safety Assessment, an evaluation of its recent developments, and perspectives on the future research directions in this area. To do so, a conceptual understanding to safety assessment is first provided, followed by an introduction to what Probabilistic Safety Assessment is about. From this, the historical background and development of Probabilistic Safety Assessment in the context of nuclear safety are discussed, including a brief description and evaluation of some methods implemented to perform such analysis. After this, the paper reviews some of the recent research developments in Probabilistic Safety Assessment in the aspects of multi-unit safety assessment, dynamic Probabilistic Safety Assessment, reliability analysis, cyber-security, and policy-making. Each aspect is elaborated in detail, with perspectives provided on its potential limitations. Finally, the paper discusses research topics in six areas and challenges within the Probabilistic Safety Assessment discipline, for which further investigation might be conducted in the future. Hence, the objectives of the review paper are (1) to serve as a tutorial for readers who are new to the concept of Probabilistic Safety Assessment; (2) to provide a historical perspective on the development of the Probabilistic Safety Assessment field over the past seven decades; (3) to review the state-of-the-art developments in the use of Probabilistic Safety Assessment in the context of nuclear safety; (4) to provide an evaluative perspective on the methods implemented for Probabilistic Safety Assessment within the current literature; and (5) to provide perspectives on the future research directions that can potentially be explored, thereby also targeting the wider research community within the nuclear safety discipline towards pushing the frontiers of Probabilistic Safety Assessment research.

Keywords: nuclear safety; probabilistic safety assessment; risk assessment; core damage; large release; multi-unit PSA; dynamic PSA; reliability analysis; cyber-security; policy-making

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Citation: Lye, A.; Chang, J.; Xiao, S.; Chung, K.Y. An Overview of Probabilistic Safety Assessment for Nuclear Safety: What Has Been Done, and Where Do We Go from Here? *J. Nucl. Eng.* **2024**, *5*, 456–485. https://doi.org/10.3390/jne5040029

Academic Editor: Dan Gabriel Cacuci

Received: 4 July 2024
Revised: 6 September 2024
Accepted: 11 October 2024
Published: 16 October 2024

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1. Introduction

1.1. Background

In recent years, the option of nuclear energy is increasingly being considered or implemented to provide an alternative energy source to meet the increasing global energy demand. Coupled with the pressing issue of global warming and climate change, the use of nuclear energy has become an increasingly promising option, given that it is carbon-free and has the potential to allow countries to reduce emissions while meeting their domestic energy demand. In fact, as of April 2024, there have been 417 operational nuclear reactors across the world with another 58 of such reactors under construction. Of the operational nuclear reactors, 27.10% are situated within Northern America, 30.46% are situated within Asia, and 40.29% are situated within Europe [1].

However, this option comes with risks, as nuclear reactors require the highest reliability, down to their smallest components and in the human/organisational factors that

manage them. History has shown the potential impacts of nuclear accidents as seen from the Windscale fire accident in 1957 [2,3], Three Mile Island accident in 1979 [4], Chernobyl accident in 1986 [5], and the recent Fukushima-Daiichi accident in 2011 [6]. Details on each accident are found in the respective references. Hence, this brings forth the need for nuclear safety, security, and safeguards (i.e., the nuclear 3S concept) which is promoted by the International Atomic Energy Agency (IAEA) and its statute, while the responsibility of managing the nuclear 3S falls on its member states [7–9]. The focus of the paper is on the nuclear safety aspect, which is of critical importance within the nuclear industry [10]. As the saying goes, "Safety is freedom, freedom from unaffordable harm, and, thus, a human right" (Zio, 2018) [11].

A key approach towards evaluating the safety level of the nuclear reactor technology is Probabilistic Safety Assessment, also referred to as Quantitative Risk Analysis [12–14]. Such analysis seeks to address three key questions [15]:

1. What could possibly happen? This involves identifying the accident scenario(s) leading to core damage and source term release (e.g., loss of coolant, steam-generator pipe rupture, and station black-out).
2. How likely is the accident event? This involves quantifying the associated probability of the accident scenario(s).
3. What is/are the consequence(s) of the accident? This involves identifying the outcome(s) of the accident scenario for the environment and life forms.

In doing so, it facilitates the risk-informed decision-making process in the reactor design, as well as in the human factors and emergency mitigation so as to minimise the associated risks of a severe accident. This makes Probabilistic Safety Assessment an informative and useful technique for risk analysis [16,17].

It is to be highlighted that there are key differences in the safety approaches between the nuclear fission and the nuclear fusion technology installations, details of which are found in [18–20]. In the interest of the work presented, the context of the paper is on the nuclear fission reactor technologies.

1.2. Objectives and Scope

In the interest of the paper, the technique of Probabilistic Safety Assessment is studied and reviewed with the following objectives: (1) to serve as a tutorial for readers who are new to the concept of Probabilistic Safety Assessment; (2) to provide a historical perspective on the development of the Probabilistic Safety Assessment field over the past seven decades; (3) to provide an evaluative perspective on the methods implemented for Probabilistic Safety Assessment in the literature; (4) to review the state-of-the-art developments in the field of Probabilistic Safety Assessment for nuclear safety; and (5) to provide perspectives over the future research directions that can be potentially explored, thereby serving as a reference for the wider nuclear safety research community towards developing the state-of-the-art in the field of Probabilistic Safety Assessment.

To achieve the objectives, the scope of the paper is as follows: (1) the objectives of safety assessment; (2) the need for Probabilistic Safety Assessment in nuclear safety and a brief history of its developments; (3) the tools for Probabilistic Safety Assessment and an evaluation of the methods; (4) the recent developments in Probabilistic Safety Assessment and perspectives; and (5) the research gaps that present an opportunity for future research works. As such, the paper is structured as follows: Section 2 outlines the concept of safety assessment and its objectives, followed by an overview of the concept of Probabilistic Safety Assessment. After, Section 3 provides a historical perspective of Probabilistic Safety Assessment in the nuclear industry, and its rise to prominence in the field of nuclear safety, as well as evaluating the tools implemented to perform such assessment. From there, Section 4 proceeds to discuss and review the state-of-the-art developments in the following aspects of Probabilistic Safety Assessment in nuclear safety: (1) multi-unit Probabilistic Safety Assessment; (2) dynamic Probabilistic Safety Assessment; (3) reliability analysis; (4) cyber-security; and (5) policy-making. Penultimately, Section 5 presents a series of

research gaps identified which can serve as open research problems for future research works. Finally, Section 6 summarises the contents presented in the paper before drawing the paper to a close.

To illustrate the organisational structure of the paper and the categorisation of the sections based on their respective coverage, a flow-chart is presented in Figure 1.

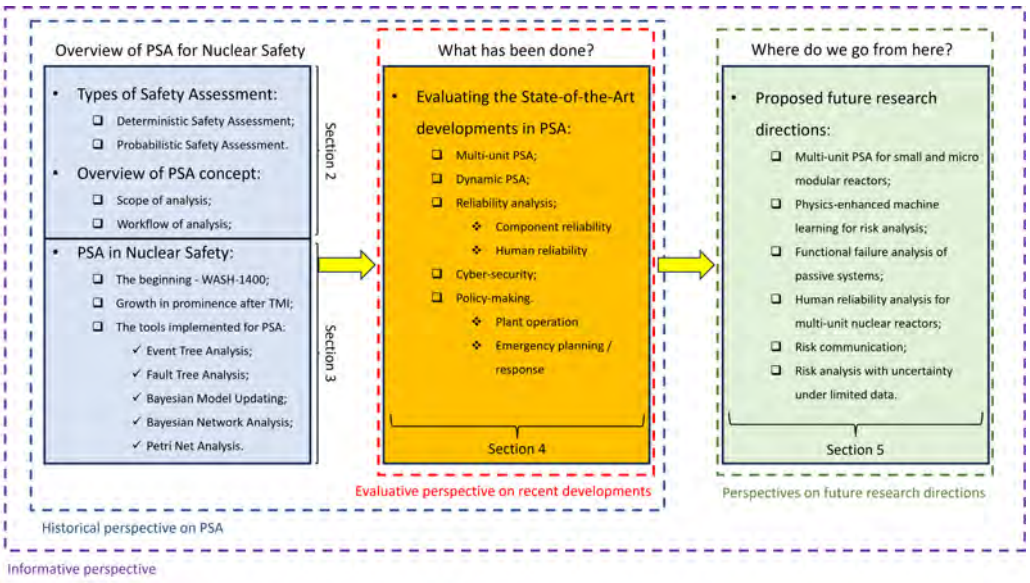

Figure 1. The flow-chart summarising the scope, organisational structure, and the categorisation of the respective sections.

2. What Is Safety Assessment

2.1. Concept and Philosophy

The general concept of safety assessment can be understood as a procedure to assess and evaluate the potential hazards associated with the design, operation, decommissioning, or maintenance of a facility [21]. In practice, such a procedure is mostly carried out before and during the operational lifetime of complex industrial facilities such as the nuclear reactors, where the relevance of safety assessment is seen in the evaluation of the facility design, construction and/or operational licensing, and life extension [22–24]. The safety assessment procedure usually involves modelling the physical phenomena (e.g., the loss of coolant) and analysing the response by the safety systems in place. This is to ensure that the safety requirements are met by the plant facility in dealing with the postulated accident scenarios. As such, the objective of performing such assessments is to verify whether the risks—defined as the probability that an initiating event (IE) eventually leads to an undesirable outcome (e.g., the atmospheric release of the source terms)—are of acceptable levels [22]. This makes such an assessment equivalent to a form of risk assessment [25].

There are multiple considerations through which the safety assessments can be performed, including the following:

- **As Low As Reasonably Achievable:** The understanding that risks are not completely reducible, as there are either no viable options available (i.e., impossible to reduce), or they are not cost-effective to reduce (i.e., the cost to reduce risk outweighs any potential benefits of reducing, either because cost is too high, or the risk is already negligible) [26];
- **Defence-in-Depth:** A safety philosophy achieved by multiple layers of protection (i.e., lines of defence) rather than relying solely on one layer for the plant to operate safely. Those layers may be provided by either redundant means or diverse means, thereby reducing the failure probability [27,28];

- **Design Basis Accidents:** A set of postulated accidents which the design of the facility must account for such that it is resilient towards these postulated accidents without any loss to the Structures, Systems, or Components (SSC) [29];
- **Hazards:** Anything that has the potential to cause an undesired event (e.g., such as a Loss of Coolant Accident (LOCA)) or condition that leads to equipment damage [30];
- **Single Failure:** A sole failure occurrence which leads to the loss in the capability of the nuclear reactor to operate safely [21]. This can be prevented through the introduction of multiple barriers (i.e., Defence-in-Depth) [22].

Based on the safety standards set by the IAEA, the scope of the safety assessment should encompass the following [31]: (1) the consideration of all sources of radioactive material, which includes the reactor core, irradiated fuel in transit, irradiated fuel in storage, and stored radioactive waste (e.g., spent fuel); (2) the assessment and evaluation of the plant performance under different plant states (i.e., operational and accidental conditions) against the deterministic and probabilistic criteria for safety and the risk of source term release; (3) the assessment of the design and safety features of the plant in fulfilling the required safety functions in response to the identified hazards; (4) the demonstration of the plant meeting the safety requirements and the hazard risks associated with the plant being acceptably low; and (5) the evaluation of the proposed plant design improvements by identifying potential design weakness and then highlighting the plant conditions along with the IEs which were not considered adequately a priori in the proposed design. In fact, these five points listed above are especially essential when assessing the safety of the Generation II to IV reactor designs. Many countries seek to improve existing safety system designs through the adoption of passive safety systems so as to improve the safety performance of such reactors and to mitigate a potential severe nuclear accident (e.g., NuScale's reactor design). As such, the existence of the nuclear industry is heavily dependent on nuclear safety standards—nuclear safety is the lifeline of the nuclear industry.

Broadly speaking, the safety assessment can be performed through two approaches: (1) Deterministic Safety Assessment (DSA) and (2) Probabilistic Safety Assessment (PSA). In line with the scope of the paper, an overview and explanation of the concept of PSA is provided in Section 2.2. Readers may refer to references [32–34] for details on DSA.

2.2. Overview of Probabilistic Safety Assessment

The PSA approach is a form of data-centric stochastic risk modelling and, hence, is also commonly referred to as a probabilistic risk assessment [35]. Its main purposes are to provide an understanding of the undesired consequences (e.g., core damage or the atmospheric release of source terms) by modelling such events with their associated probabilities and consequences using available information, usually based on the operating history of reactors of interest. In doing so, the analysis serves to provide a systematic and sophisticated analysis of the different risk management approaches, which are supported by the computed probabilities [31,36]. Hence, the objectives of PSA are [15] (1) identifying and delineating the different scenarios that could possibly result in a severe accident; (2) assessing the expected probability of occurrence for each of the corresponding combination of events; and (3) evaluating the consequences associated with each combination of events.

To achieve the aims and objectives, the scope of the analysis by the PSA approach is usually categorised into three distinct levels described as follows [37,38]:

- **Level 1 PSA:** The scope of the assessment is on the nuclear reactor design and its operating states, with the main focus being the accident sequences from a given IE that could result in core damage. This level of analysis serves to evaluate the strengths and weaknesses in the plant design, thereby facilitating the development of the necessary modifications in the safety systems and/or human factors towards the prevention of core damage and subsequent large release of source terms [36].
- **Level 2 PSA:** The scope of the assessment includes that of the Level 1 PSA and the phenomenon of the core damage accident, with the main focus being the response of the containment structure(s) to the expected load and the eventual release of the

radioactive materials into the environment. This level of analysis serves to reflect the information pertaining to the associated probabilities of the source term releases, thereby highlighting the relative importance of the events pertaining to the primary safety concerns due to the potential atmospheric releases, and allowing for the identification of actions towards mitigating the consequences of such accidents [39].

- **Level 3 PSA:** The scope of the assessment includes that of the Level 2 PSA and the eventual atmospheric release of the source terms. A full Level 3 PSA serves to investigate the dispersion of the radioactive nuclides into the surrounding environment and analyse the potential environmental and health consequences of such a release [40].

To illustrate the relationship between the three levels of PSA, a flow-chart is provided in Figure 2.

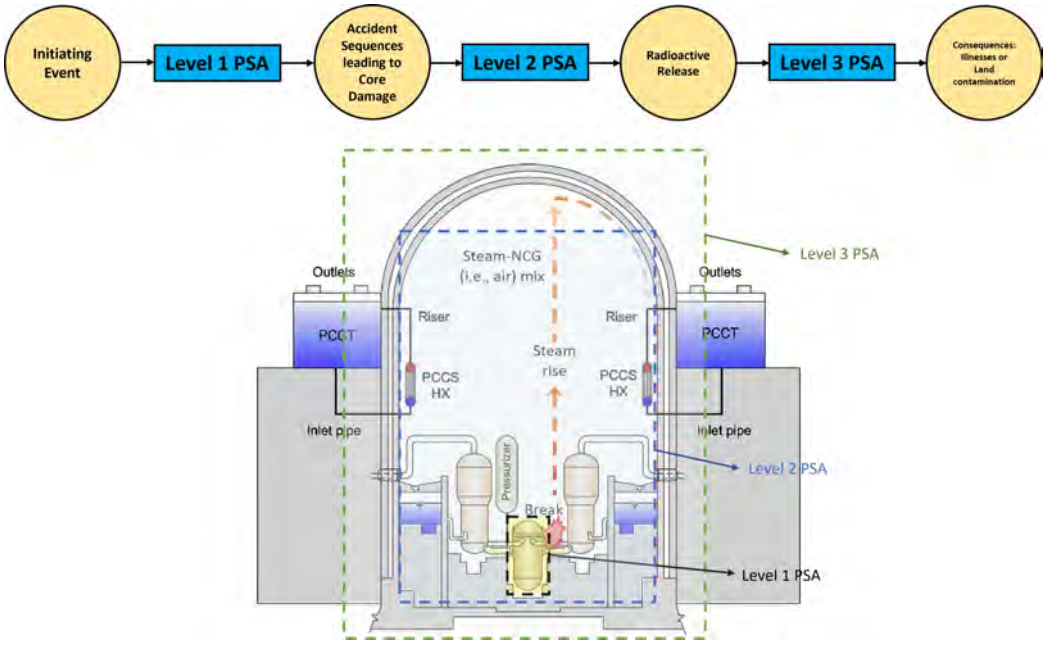

Figure 2. Flow-chart and schematic diagram illustrating the relationship between the different levels of PSA. Reprinted from Bhowmik et al. [41,42] with permission from Elsevier.

Broadly speaking, there are two types of analyses that can be achieved through PSA [43]. The first type of analysis is called the a posteriori analysis. Such an analysis is performed on the existing nuclear reactors with operating histories based on the plant-specific data (e.g., the component failure data) obtained from operating experience. Should such data be lacking, generic non-plant-specific data can also be used in their place. Based on information from past operating experience, key accident events are identified and analysed to determine their associated conditional probabilities of progression towards the identified accident of interest (i.e., the Top event). From the derived conditional probabilities, an estimate of the Core Damage Frequency (CDF) can be obtained. Hence, the a posteriori analyses is performed on the basis of the total operating experience of the plant of interest to identify the aspects of operational vulnerability and propose improvements from there [43]. The second type of analysis is called the a priori analysis. Such an analysis is performed on a nuclear reactor with no operating history (i.e., one that is in the stage of conceptual design or under regulatory review) from which a prediction is made on the probability of a severe accident such as a core damage or the atmospheric release of source terms. Due to the absence of any plant-specific data, generic non-plant-specific data are used for the study [43].

A simplified and general procedure towards performing PSA is as follows [44]: The analysis is initiated by identifying the IE of interest, the hazard(s), and defining the scope of analysis (i.e., Level 1, Level 2, or Level 3 PSA). This serves as the basis for the safety

assessment. Next, the IE and the hazard(s) are analysed, for which the subsequent possible sequences of events leading up to the consequence(s) (e.g., core damage or large atmospheric release of source terms) are identified. This is the scenario modelling stage, from which the plant response is analysed by evaluating the design reliability of the safety system and/or the human reliability in response to each event within the accident sequence. Such an analysis is performed using the information provided by the existing operational practices and the availability of plant-specific data in the case of the a posteriori analysis, or the availability of generic non-plant-specific data in the case of the a priori analysis. Based on such analysis, the information pertaining to the event sequence frequency and its consequence are obtained, from which an evaluation on the resulting risk of the severe accident scenario is performed against a defined safety goal. Should this safety goal not be met, the analyst may propose modifications to the equipment design and/or improvements for the operating procedures or technical competencies of the plant operators. Given the proposed modifications and the same initial conditions (i.e., the IE and scope of the analysis), the plant response is then re-evaluated against the same safety goal. Otherwise, the PSA procedure terminates, and the current/updated design and operational requirement of the plant are defined to be sufficiently robust in complying with the safety standards given the IE and the scope of the analysis. As a summary, a flow-chart outlining the work-flow for PSA is illustrated in Figure 3.

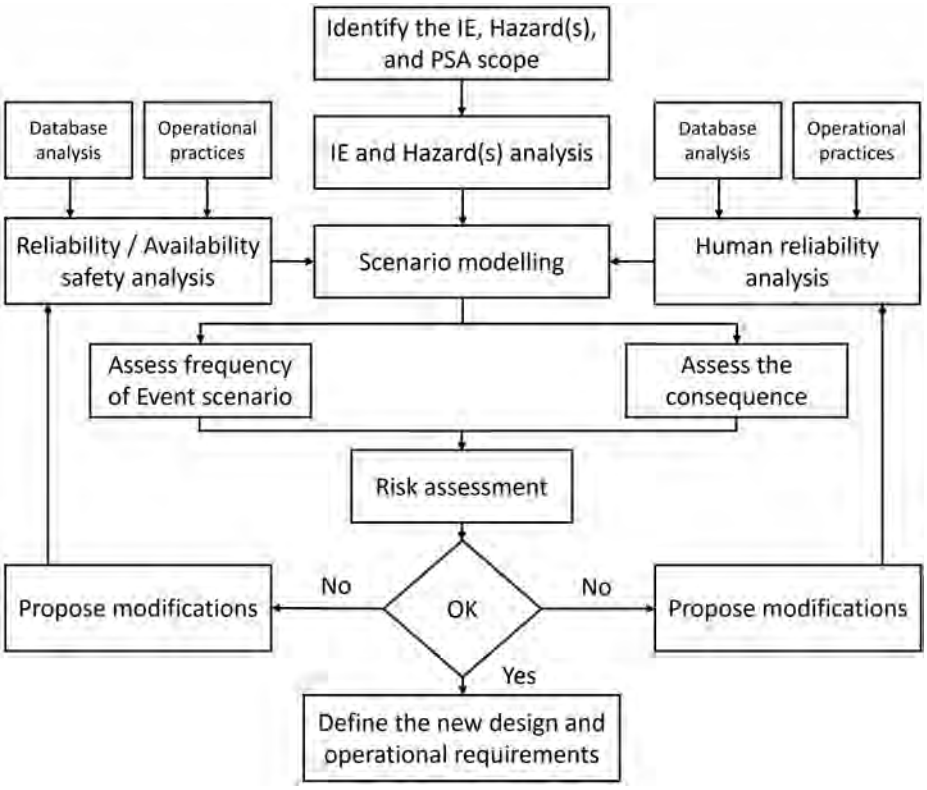

Figure 3. Flow-chart summary of the PSA work-flow. Adapted from Vasconcelos et al. [22] with permission from Elsevier.

There are three notable strengths of the PSA approach [12]: (1) It can quantify the severity and the frequency of the severe accident, thereby providing a more comprehensive and systematic safety analysis. (2) It accounts for the uncertainties associated with the data, the approach used in modelling the event scenarios, and the completeness of the analysis. In doing so, it proves to be robust towards uncertainty propagation. (3) Finally, it considers thousands of different accident scenarios, including those not considered by DSA, involving multiple failures, which facilitates the understanding of the different system failure modes. Owing to such strengths, the PSA method thus serves as a cost-effective tool

for (1) risk-based decision-making towards plant design improvements and operational support (e.g., risk-based maintenance and repair strategies; this motivates the NRC to encourage its implementation in complementing the DSA approach in the safety analysis requirements in 1995 [45]); (2) facilitating emergency planning and preparedness; and (3) serving as a tool to aid risk communication and outreach to various stakeholders (e.g., government, regulatory board, and the general public).

However, there remain two significant drawbacks to the PSA approach: First, its robustness and capabilities are limited by the availability and quality of the necessary data required for such an analysis to be possible [46]. This drawback becomes especially pronounced when performing an a priori analysis with available generic non-plant-specific data that may not be applicable to the plant design of interest. And second, it remains relatively challenging to model human errors and organisational factors (e.g., safety culture) and to account for them in the overall analysis, although there is significant ongoing research being conducted to try to overcome this challenge [47,48].

3. Historical Perspective of Probabilistic Safety Assessment in the Nuclear Industry

The concept of reactor safety gained significant attention and importance between the late 1960s and early 1970s when an increasing number of nuclear reactors were either completed or under construction [49]. This attention was accompanied by increased public and political opposition towards nuclear reactors, due to the perception that the US Atomic Energy Commission (AEC) safety criteria for the licensing of such plants were insufficiently robust relative to the apparent safety significance of the various SSCs within the plants [50]. Furthermore, there was a debate through the years as to whether the emergency core cooling system would work, as well as an increased demand for the risk estimate in the failure of the safety systems. These factors motivated the AEC to initiate a much more comprehensive study on the safety evaluation of the light water reactors designs and hence the Reactor Safety Study (RSS) [51].

The analysis considered the following two key areas: (1) the failure in the major systems such as the engineered safety systems; and (2) the resilience of the plant containment system towards preventing the atmospheric release of the source terms in the event of an accident. To perform the analysis, the RSS looked at six specific LOCA events as the IE [49]:

- Small pipe breaks (i.e., less than 2 inches in diameter);
- Intermediate pipe breaks (i.e., between 2 and 6 inches in diameter);
- Large pipe breaks (i.e., larger than 6 inches in diameter);
- Large disruptive reactor vessel ruptures;
- Gross steam generator ruptures;
- Ruptures in systems that interface with the reactor coolant system.

In addition to this, the RSS also considered the different possible reactor transients and IEs. In the context of the study, a transient refers to the situation whereby the key reactor operating parameter shows a significant deviation from the normal operating value. This includes all non-LOCA events, such as equipment failure or human error, which can lead to a reactor shut-down to minimise any potential damage to the fuel. The three main areas for transients considered in the RSS are instances where there is an increase in the reactor power, a decrease in the flow of the coolant, or an increase in the pressure of the coolant. Broadly speaking, the transients can be categorised into two distinct categories: (1) anticipated transients (e.g., loss of off-site power and loss of feedwater); and (2) unanticipated transients (e.g., vessel rupture, turbine missiles, and sabotage). However, through some preliminary analysis of the frequency and the consequences, it was found that the unanticipated transients' contribution towards the overall risk was small compared to that of the anticipated transients, which produced the same consequences. This led to the removal of the unanticipated transients and some of the relatively low-frequency anticipated transients from consideration, allowing for a significant reduction in the number of possible IEs, leading to a core damage and radioactive release to be considered.

To perform the PSA, the industrial data were utilised on the component failure rates along with expert elicitation on aspects for which data were unavailable. In order to account for the epistemic uncertainties (i.e., uncertainty due to a lack of knowledge associated with the availability of limited data) [52,53], the probabilities of the key events in a given the accident sequence were modelled to follow a Log-normal distribution instead of point estimates. Based on the available techniques and information provided, the study was able to quantify that the CDF for a Generation II PWR is of the order 10^{-5} per reactor year [49]. This result was significantly higher than that obtained from previous studies of such reactors, which yielded failure probabilities of the order 10^{-6} per reactor year. The reason for this is that the RSS managed to determine that small-break LOCAs had the highest contribution towards the overall failure probabilities through the modelling techniques implemented, while previous analyses would usually neglect the contribution of small-break LOCA towards the eventual core damage [49]. This allowed for the RSS to produce relatively accurate and realistic results compared to prior studies, effectively making the RSS the first in implementing the PSA framework within the nuclear industry.

Despite such robust results presented by the RSS, the initial series of hearings held in June 1976 by the Committee on Interior and Insular Affairs of the US House of Representatives found the study to be misleading in its conclusions. Subsequently, a series of meetings held between August 1977 and September 1978 was conducted by the Risk Assessment Review Group (i.e., the Lewis Committee) aimed at performing a detailed review over the RSS [54]. While the committee found the RSS to be generally commendable in the completeness and depth of its analysis, the review raised numerous concerns, including [54] (1) the lack of “scrutability" in the calculation procedure; (2) the lack of accurate data to perform the risk analysis; (3) the lack of consideration of some external events (e.g., earthquakes, fires, and human accident initiation) towards the overall risk quantification; (4) the scepticism over the peer review process of the RSS by the US NRC; (5) the difficulty in finding the content discussing the health impact of radiation release; and (6) the poor communication of information within the Executive summary. Following the criticism by the Lewis Committee, the NRC withdrew its support towards the RSS and focused on the continued implementation of the DSA approach towards performing safety assessments of nuclear reactors, effectively casting the PSA approach and the RSS aside.

In March 1979, there was revived interest towards the RSS and the PSA approach by the NRC in the wake of the Three Mile Island accident in Unit 2 [49]. This was due to the fact that the RSS had considered an event sequence similar to that of the Three Mile Island accident in the PSA analysis for another another plant, while the accident also validated the conclusion of the RSS that a small-break LOCA is more risk significant than a large-break one. Subsequently, the NRC began investing heavily towards expanding the PSA implementation within the nuclear industry [55]. This resulted in two follow-up PSA studies which were conducted between 1979 and 1982 known as (1) the Reactor Safety Study Methodology Application Program, which aimed at applying the methodology presented in the RSS to other plant designs [56,57], and (2) the Interim Reliability Evaluation Program which aimed at developing and standardising the reliability methods implemented towards reliability and safety assessments [58–60]. Since then, significant efforts have been invested by the NRC and the nuclear industry into improving and developing the state-of-the-art PSA approaches and implementing extensive PSA methods for the licensing of nuclear reactors [49]. A result of such efforts was the definition of the Safety Goals by the NRC in 1990 pertaining to the CDF and the Large Release Frequency (LRF), where the following values were set as the “benchmark”: 10^{-4} for the CDF and 10^{-5} for the LRF.

To date, numerous risk modelling approaches have been implemented to perform PSA on the nuclear reactor designs, for which some examples, along with their corresponding references to their implementation, are presented in Table 1. An overview of each of the PSA approaches listed in the table are presented in Sections 3.1–3.3.

Table 1. Examples of some of the approaches implemented for PSA.

Approach	Acronym
Fault Tree Analysis	FTA
Event Tree Analysis	ETA
Bayesian Model Updating	BMU
Bayesian Network Analysis	BNA
Petri Net Analysis	PNA

3.1. Fault Tree Analysis

The FTA is a commonly implemented approach to perform PSA and was first utilised for such a purpose in 1975. It provides a systematic analysis of the cause of an identified Top event through the use of a graphical and logical model (i.e., the Fault Tree) to express the parallel and serial combinations of faults leading to the eventual Top event. As such, the Fault Tree itself illustrates the logical inter-relationships (e.g., the "AND" or "OR" logic) of the basic/root events that result in the Top event. Based on the constructed Fault Tree, the minimal cut sets are identified, which highlight the smallest combination of basic event and represent the most important event sets which will lead to the Top event [61,62]. From this, a quantitative analysis on the Fault Tree is performed by computing the probability of occurrence of the Top event from the probability of occurrence of the basic events. The computation process, along with the subsequent uncertainty, sensitivity, and importance analyses, can be supported using computer code [63–65]. This is especially so when performing the analysis on complex Fault Trees, which may be the case when analysing the risk of CDF or LRF for a nuclear reactor. Hence, the FTA is a useful technique in carrying out the reliability and the availability analysis on a nuclear reactor design [66].

The main advantages of the FTA approach are threefold:

- It can model many hazards from different event combinations;
- It can be used to identify common cause failures;
- It provides a clear and logical presentation of the cause of a Top event.

However, its main disadvantages are twofold:

- Such an analysis can only be performed for one Top event at a time;
- The construction of the Fault Tree and the subsequent analysis can be time-consuming and complicated for complex/complicated sub-systems of the nuclear reactor.

3.2. Event Tree Analysis

The ETA is another commonly implemented approach to perform PSA, and the first recorded implementation of such a methodology was in an IAEA study on the containment and siting of nuclear reactors by Farmer in 1967 [67]. The approach was based on the decision analysis field and its subsequent implementation in the RSS stemmed from the realisation that the FTA approach for the entire nuclear reactor was deemed too complex at the time of the study [49]. Such a decision proved to be worthwhile, as the ETA approach was able to address the time and resource constraints associated with the FTA approach [51].

Like the FTA approach, the ETA approach provides an alternative way to perform a systematic analysis of the sequence of events starting from a given IE all the way to the eventual Top event. However, the difference between the FTA and the ETA is that the latter is performed through the use of a branched graph model which illustrates all possible sequence of the plant states, operating performance, emergency response, and their corresponding probability of occurrence. The individual branch denotes a particular Defence-in-Depth level, and its associated failure probability is obtained from either the historical data or previous studies such as reliability analysis. Hence, an Event Tree allows for a logical presentation of the different end states of the given nuclear reactor that is studied along with the associated probability of occurrence for each end state. This makes the approach useful towards estimating the risk and consequence of each accident scenario given an identified IE [49].

The main advantages of the ETA approach are threefold:

- It simplifies the accident sequence through clear and logical presentation;
- It is applicable towards a wide range of hazards in qualitative risk analysis;
- It can diagnose both equipment-related events and those related to human reliability.

However, its main disadvantages are fourfold:

- It is inefficient in modelling accident sequences where many events have to occur in combination, as it yields many redundant branches;
- The independence assumptions between distinct events can lead to missing systematic and common-mode failures;
- The analysis is limited to only one initiating event at a time;
- The binary logic (i.e., Yes or No) becomes inapplicable when involving elements of uncertainty such as human error or adverse weather conditions.

3.3. Bayesian Model Updating

The BMU approach provides a quantitative method to update one's subjective knowledge over the failure parameter(s) (e.g., component failure probability) as new information/data become available and has seen numerous applications in the field of engineering [68], especially nuclear [17,69–71]. The earliest recorded implementation of such an approach for PSA in NRC was in the NUREG/CR-2300 technical report published in 1983 [72] and is based on Bayes' theorem, which is mathematically defined as [73,74]:

$$P(\boldsymbol{\theta}|\boldsymbol{D}, M) = \frac{P(\boldsymbol{\theta}|M) \cdot P(\boldsymbol{D}|\boldsymbol{\theta}, M)}{P(\boldsymbol{D}|M)} \tag{1}$$

where $\boldsymbol{\theta}$ is the vector of failure parameter(s) to be estimated, $\boldsymbol{D}$ is the vector of data used to infer the failure rates, and M is the model relating $\boldsymbol{\theta}$ to $\boldsymbol{D}$. The terms in Equation (1) are as follows [75,76]:

- $P(\boldsymbol{\theta}|M)$ is the prior distribution denoting the a priori knowledge of $\boldsymbol{\theta}$ before observing $\boldsymbol{D}$ through the given model M;
- $P(\boldsymbol{D}|\boldsymbol{\theta}, M)$ is the likelihood function quantifying how likely it is that $\boldsymbol{\theta}$ represents the observed $\boldsymbol{D}$ through the given model M;
- $P(\boldsymbol{D}|M) = \int_{-\infty}^{\infty} P(\boldsymbol{\theta}|M) \cdot P(\boldsymbol{D}|\boldsymbol{\theta}, M) \cdot d\boldsymbol{\theta}$ is the numerical normalisation constant;
- $P(\boldsymbol{\theta}|\boldsymbol{D}, M)$ is the posterior distribution denoting the a posteriori knowledge on $\boldsymbol{\theta}$ after observing $\boldsymbol{D}$.

Generally speaking, the integral of $P(\boldsymbol{\theta}|\boldsymbol{D}, M)$, and therefore the mean and variance estimates on $\boldsymbol{\theta}$ may not be obtained analytically. As such, the estimates on $\boldsymbol{\theta}$ are performed numerically by sampling from $P(\boldsymbol{\theta}|\boldsymbol{D}, M)$ using sampling tools such as the Markov Chain Monte Carlo or the Sequential Monte Carlo samplers [77–79]. However, there exists a specific case where the posterior distribution is of the same distribution class as the prior distribution given a likelihood function. In this case, the posterior and prior distributions are conjugate distributions, and a closed-form solution of the posterior distribution can be obtained analytically.

The main advantages of the BMU approach are threefold:

- It yields a distribution estimate of the inferred parameter(s) characterising its uncertainty;
- It is applicable and efficient in performing estimates on the inferred parameter(s) when dealing with a limited or small dataset;
- It can also perform online learning on the inferred parameter(s) when the data are obtained sequentially.

However, its main disadvantages are twofold:

- High computational cost is incurred in cases where the model M is computationally expensive;

- The choice of the prior is subjective to the analyst, which affects the probabilistic estimates of the inferred parameter(s), especially when the data are scarce, which makes the estimates on the inferred parameter(s) highly dependent on such a choice.

3.4. Bayesian Network Analysis

The BNA methodology was developed by Pearl in 1986 and was first implemented in the domain of artificial intelligence before subsequently being implemented in the areas of information technology, industrial, reliability, and safety fields [80]. The approach was first implemented for PSA in 2010 [81] and involves the construction of a graphical model consisting of nodes and edges to illustrate the polymorphism and the failure dependencies between the root, intermediate and the Top events along with their associated uncertainties. There are two features to a given Bayesian network with N nodes [81]. The first feature is the mode structure. Such a structure is illustrated a N-node directed acrylic graph with a given nodes set $V = \{V_1, \ldots, V_N\}$, which represents the variables such as the equipment state and personnel operations. The causal relationship between the variables is represented by the directed edges between the node pairs. For a given pair of nodes $\{V_i, V_j\}$ (for $i \neq j$) related by a directed edge from V_i to V_j, node V_j is known as the parent node of V_i, and conversely, node V_i is known as the child node of V_j. A node without an associated parent node is a root node, while a node without an associated child node is the leaf node. A parent node set of V_i is denoted as $\mathrm{Pa}(V_i)$. The second feature is the associated probability of the root node and the conditional probabilities between the nodes. Under the conditional independence assumption, the conditional probability distribution which quantifies the association between the node and its parent nodes is denoted as $P(V_i|\mathrm{Pa}(V_i))$. By the same assumption, the joint distribution of the nodes $P(V_1, \ldots, V_N)$ is defined as

$$P(V_1, \ldots, V_N) = \prod_{i=1}^{N} P(V_i|\mathrm{Pa}(V_i)) \tag{2}$$

The main advantages of the BNA approach are threefold:

- It is able to update the joint and conditional probabilities as more data and observations are made;
- It can illustrate graphically the causal relationships between the root and intermediate events leading to the Top event, making such approach useful in risk communication;
- It can quantify and propagate the uncertainties in the conditional probability estimates though the network.

While its main disadvantages are twofold:

- Complex Bayesian networks with large number of nodes can incur high computational costs;
- It is unable to handle continuous data well, implying the need for the analyst to discretise the initial continuous data before the analysis.

3.5. Petri Net Analysis

The PNA approach involves the use of a graph-based structure referred to as a Petri Net which was proposed by Petri during his PhD [82] and was first implemented for PSA in 1986 [83]. Like the Bayesian Network, the structure itself is made up of nodes and edges, which are referred to in this context as places and transitions. As such, the Petri Net is able to simulate the dynamic behaviour of a given nuclear reactor system through the continuous "firing" of enabled transitions, which allow for tokens to be removed or inserted into the various places. In doing so, it is able to capture the information pertaining to the impact of the response, maintenance, and recovery processes for the different modes of failure associated with the plant. In fact, the approach is applicable to systems which are either concurrent, asynchronous, distributed, or parallel, which provides flexibility to the PNA framework. This allows the PNA to serve two purposes in PSA: (1) to provide a mathematical representation of the system via state equations, algebraic equations, and

numerous mathematical models describing the dynamics of the system being studied; and (2) serving as a tool to aid in the visual communication when communicating the risk of a severe accident to the various target audiences, such as the general public, policy-makers, and the government personnel.

The main advantages of the PNA approach are twofold:

- It is useful in modelling the transition dynamics between the different operating states of the nuclear reactor being studied;
- It illustrates graphically the causal relationships between the root and intermediate events leading to the Top event, making such an approach useful in risk communication.

However, its main disadvantage is as follows:

- The configuration of the Petri-net can be inherently complex, especially when studying the relatively complex sub-systems of a nuclear reactor.

4. Review of the State-of-the-Art Developments

In recent years, the implementation of the PSA techniques described in Sections 3.1–3.5 has increased significantly owing to the development of robust computational techniques which facilitate the study of complex nuclear reactor designs and compute the small failure probabilities (i.e., the order of 10^{-6} and below). To provide an understanding and an evaluation of such recent developments, the relevant works in the literature published between 2022 and 2024 are reviewed and are categorised according to the aspect of PSA being studied. The scope of the review covers the following five main aspects:

1. Multi-unit PSA;
2. Dynamic PSA;
3. Reliability analysis, which can be further sub-classified into component, and human reliability;
4. Cyber-security;
5. Policy-making, which can be further sub-classified into plant inspection/maintenance policy, human/organisational factors, and emergency planning/response.

A chart is presented in Figure 4 to illustrate the structuring of the main aspects and the sub-aspects (if any), while Table 2 presents the categorisation of the literature in their respective aspects and sub-aspects along with the list of the implemented approach(es) and the scope of analysis (i.e., PSA level).

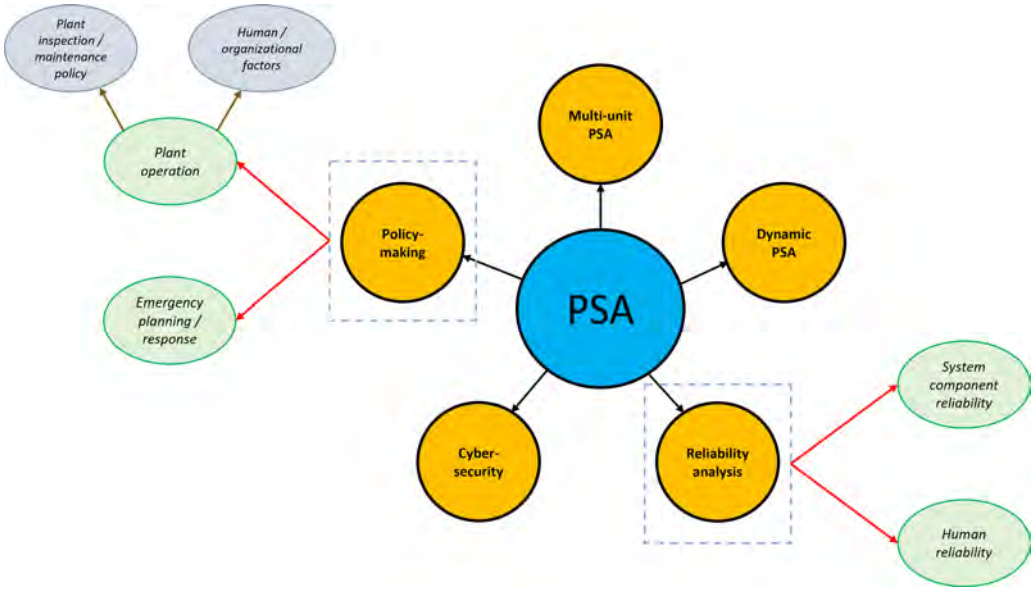

Figure 4. The five different aspects of PSA to be discussed in the section.

Table 2. Examples of recent works published between 2022 and 2024 categorised according to the respective aspects of the PSA reviewed, approaches implemented, and scope of the analysis.

Aspect	Sub-Aspect	Approach	PSA Level	Reference
Multi-unit PSA	-	FTA	3	[84]
		FTA/ETA/BNA	1	[85]
		ETA	2	[86]
		ETA	3	[87]
		ETA	1	[88]
Dynamic PSA	-	FTA/BNA	1	[89]
		ETA	1	[90,91]
		PNA	1	[92,93]
Reliability analysis	Component reliability	BMU	1	[94–96]
		BMU/BNA	1	[97]
		BNA	1	[98]
	Human reliability	BMU	1	[99,100]
		BNA	1	[101]
		ETA	1	[102]
		ETA/BNA	2	[103]
Cyber-security	-	PNA	1	[104,105]
		ETA	1	[106,107]
		BNA	1	[108]
Policy-making	Plant inspection/maintenance policy	BMU	1	[109–111]
		ETA/FTA	1	[112]
		PNA	1	[113]
	Human/Organisational factors	BNA	1	[114]
		BMU/BNA	1	[115]
		ETA	1	[116]
		FTA	1	[117]
		ETA/FTA	1	[118]
	Emergency planning/response	FTA/ETA	2	[119]
		BNA	2, 3	[120]
		BMU	3	[121,122]
		FTA/ETA	3	[123]

4.1. Multi-Unit Probabilistic Safety Assessment

Traditionally, PSA studies have been restricted to single reactor units and are referred to as single-unit PSAs [124]. Such studies consider only the accident scenarios which are exclusive to one specific reactor unit and this is performed under the assumption that the accident impact by the other reactor units is less critical. For this reason, the single-unit PSAs account only for the dependencies between the SSCs within the single reactor unit of interest whose failure(s) may lead to a station black-out or a common cause failure, for example [125]. This is known as intra-unit dependency.

Multi-unit PSAs, on the other hand, account for the dependencies across the multiple different reactor units co-located at the same site, and such dependencies are referred to as inter-unit dependencies. Examples of such inter-unit dependencies may include [126] (1) IEs which occur across multiple reactor units simultaneously; (2) the occurrence of a transient event in one reactor unit which also affects some or all of the other reactor units of the multi-unit nuclear power plant; (3) the proximity of the individual reactor units to one another; and (4) the shared SSCs such as batteries, diesel generators, and common operation practices. Hence, this brings to light the importance of characterising such dependencies properly, as well as the site-level dependencies that are essential towards obtaining an accurate and robust risk profile of the nuclear power plant site.

The history of multi-unit PSA dates back to the early 1980s when the first of such studies was conducted on the Indian Point Station which investigated the dual-unit atmospheric source term releases as a result of seismic and high wind hazards (i.e., Level 3 PSA) [127]. Despite the risk, the question of nuclear safety among multi-unit plants was initially deemed to be relatively insignificant by the nuclear community and was not raised again until the Fukushima-Daiichi accident, which demonstrated the following [128]:

- There were no accident management plans for multi-unit accidents then;
- The recovery of Unit 2 was delayed by the hydrogen explosion occurring at the adjacent Unit 1;
- The hydrogen gas production in Unit 3 led to a hydrogen explosion of the unit and the subsequent delayed recovery of Unit 4.

This made the PSA community re-evaluate the current safety regulations to consider inter-unit dependencies, which are critical in severe accident risks, such as multiple core damages, spent fuel pool damages, and damages to the other radioactive waste storage facilities.

Since then, the relevance of multi-unit PSA has continued to gain significant importance in light of the increasing number of multi-unit nuclear power plants on a global scale [129]. In fact, the IAEA reported in 2020 that 73% of the world's existing nuclear power plants are multi-unit plants [130] and that 61.40% of these existing multi-unit plants are based within the United States [131]. This motivated the investment of research efforts towards developing methods for multi-unit PSA as seen in the recent works: (1) the implementation of FTA to perform a Level 3 PSA study on the adverse effects of inter-unit radioactive release on operator actions at a multi-unit site [84]; (2) the implementation of ETA, FTA, and BNA to perform a Level 1 PSA study on the effects of the varying correlation in spatial ground motion on the extent of core damage within a hypothetical dual-unit reactor site [85]; (3) the implementation of ETA to perform a Level 2 PSA study on a high-temperature gas-cooled reactor via a multi-unit event sequence modelling method accounting for the dependencies between events of single-unit event trees, aiming at computing the LRF of each release category [86]; (4) the implementation of ETA to perform a Level 3 PSA study via a plume segmentation optimisation method to analyse the subsequent off-site consequences of a multi-unit accident [87]; and (5) the implementation of ETA to perform a Level 1 PSA study on a dual-unit reactor under the loss of offsite power initiating event using multi-unit event trees derived via a combinatorial computation [88].

Based on the review of the above literature, the following insights were obtained: First, ETA is the most implemented approach towards performing multi-unit PSA, owing to its simplicity in modelling accident sequences, diagnosing both equipment-related events and those related to human reliability, and the dependencies between individual reactor units through numerous approaches, such as linking [84] or combinatorial [88] approaches. However, the ETA method mostly assumes independence between the individual events of the accident sequence, which can result in an underestimation of the accident probability. Second, while the multi-unit PSA analyses have generally managed to characterise and model the inter-unit dependencies of the reactor facilities, they fail to accurately model and account for the physical site configurations, such as the physical distance/layout between the individual reactor and the wind directions when performing a Level 3 PSA. Such shortcomings may compromise the interpretation of the PSA results, as it is not representative of the physical site conditions. Finally, the analysis is performed mostly in a deterministic manner, in that point values are used in the ETA computation without accounting for the variability due to uncertainties or characterising the uncertainty in the resulting risk estimates. This presents a limitation in such analysis towards a risk-based and risk-informed decision-making pertaining to emergency planning or response.

4.2. Dynamic Probabilistic Safety Assessment

The dynamic nature of the operating behaviour/state is an important characteristic of many engineering systems, especially that of a nuclear reactor. This emphasises the need to also model the time-dependent interactions between the system components and the reactor units, and subsequently quantifying the associated risk. This gives importance to the aspect of dynamic multi-unit PSA [132–134].

Unlike the traditional approach towards PSA, the dynamic PSA approach couples the implementation of the safety analysis codes such as the ASTEC and MELCOR along with the stochastic simulation methods such as the Risk Analysis and Virtual ENvironment (RAVEN) [135] or Monte Carlo methods to complement the logic structures such as the Event

Trees and the Fault Trees (i.e., see Section 2.2) [136,137]. In doing so, it allows for an increased robustness and realism in the safety assessment by accounting for the time-varying effects on the plant system and the dynamics of the physics-based models, thereby providing a real-time update on the analysis [138]. Hence, the dynamic PSA approach becomes favourable in modelling the numerous component behaviour of the reactor which include [139] (1) the thermal–hydraulic behaviour and (2) the operators' responses towards the accident. From these, the probabilistic aspect of the analysis is implemented by defining a set of stochastic parameters characterising the time-dependent accident progression [133,134,140,141].

Recent developments within the aspect of dynamic PSA include (1) the implementation of FTA and BNA to perform a Level 1 PSA study in the form of a dynamical reliability analysis of the emergency diesel generator within a boiling water reactor [89]; (2) the implementation of ETA to perform a Level 1 PSA study in the form of a risk analysis of a LOCA for hypothetical three-loop pressurised water reactor via a proposed dynamic integrated consequence evaluation approach [90]; (3) the implementation of ETA to perform a Level 1 PSA study in the form of a risk analysis of a small-break LOCA by incorporating the operator action timing within the human reliability evaluation method and considering such effects on the conditional core damage probability [91]; (4) the implementation of PNA to perform a Level 1 PSA study involving the dynamical reliability analysis of the reactor's liquid poison injection system [92]; and (5) the implementation of PNA to perform a Level 1 PSA study involving the dynamical resilience analysis of nuclear reactors under the threat of natural hazards [93].

Based on the review of the above literature, the following insights are obtained: First, the analysis is mostly focused on Level 1 PSA involving the reliability computation of SSCs and the human factors under dynamic scenarios. Such an analysis can be extended towards Level 3 PSA, where the dynamical feature, in the form of the radioactive nuclide dispersion evolution, becomes more pronounced. Second, the PNA method is shown in [93] to be capable in not only computing the CDF under different accident scenarios (e.g., LOCA or station black-outs) but also predicting how soon the reactor system recovers from the different scenarios considered, an aspect which should also be of interest in PSA. Such predictive capability is yet to be explored with the ETA, FTA, and the BNA approaches. Finally, the BNA method is gaining interest among researchers and experts, as it serves as an effective tool to capture causal relationships between the events, although such a method still falls short when defining the acyclic relationship between the events.

4.3. Reliability Analysis

Reliability analysis is a fundamental aspect of PSA and the objective is to assess the likelihood of the key components of a nuclear reactor, or the reactor as a whole, functioning as intended (or otherwise) during the operation time. Such an analysis is crucial for the following reasons: First, it allows for the reliability quantification of the critical equipment and systems, from which the operators and the key decision-makers (e.g., reactor engineers and policy-makers) can estimate the probability of accidents or failures occurring. Second, it identifies the most critical SSCs within the reactor via a sensitivity analysis, allowing for its prioritisation for maintenance, upgrades, or replacement. Third, such an analysis evaluates the effectiveness of safety systems and barriers within the reactor facility to mitigate the consequences of accidents or failures. And fourth, such an analysis is often required by regulators as part of the overall reactor safety assessments to instil confidence and trust to the public, and the relevant stake-holders that the reactor complies with the safety standards.

4.3.1. System Component Reliability

Recent developments within the aspect of the reactor system component reliability include (1) the implementation of BMU to update the Thermo-Hydro-Mechanical and Leakage model based on the strains, global leakage, and local leak measurements toward inferring the model parameters associated with the representative structural volume of

a nuclear containment building and assessing the structural component's reliability [94]; (2) the implementation of BMU to infer the uncertain parameters of the Thermo-Hydro-Mechanical model, based on the uncertain strain predictions, towards providing a reliable prognosis of the nuclear containment building structural mechanical state [95]; (3) the implementation of BMU to calibrate the finite element pre-stressed concrete containment vessel, based on the displacement measurements, towards assessing the structural reliability of the vessel under different internal pressure conditions [96]; (4) the implementation of BNA and BMU to estimate and update the reliability parameters for a helium circulator of a high-temperature gas-cooled reactor [97]; and (5) the implementation of BNA to develop a reactor reliability model and a reliable life analysis method for the Stirling integrated space reactor which is robust to manipulation and can obtain system reliability indicators with small confidence intervals for engineering decisions [98].

Based on the review of the above literature pertaining on the use of PSA to perform a reactor system component reliability, the following insights were obtained: First, most of the reliability analysis involves the use of a physics-guided machine learning approach, where a physics-based model is used to perform a model updating and reliability analysis on the system component. This provides the element of interpretability and explainability for the reliability results. Second, the reliability analysis thus far via the BMU approach is performed in an offline manner, where the updating of the inferred parameters is performed with all the available data at once. Such an analysis can be extended towards cases where the data are obtained across different time-steps, thereby paving way for an online BMU approach to be implemented towards a real-time dynamical reliability analysis on the system components. Finally, the Bayesian approaches (i.e., BMU and BNA methods) fall short under extreme data scarcity with large epistemic uncertainty, although this can be addressed via Monte Carlo simulation or expert judgement.

4.3.2. Human Reliability

Besides ensuring the reliability of the reactor's system mechanical components, the human reliability analysis is also a critical component of PSA. It focuses on evaluating and quantifying the likelihood and consequences of human errors in nuclear facilities, recognising that human factors play a significant role in overall safety performance.

Recent developments within the aspect of human reliability include (1) the implementation of BMU to quantitatively compare human reliability analysis methods of interest using both prior knowledge and human performance data [99]; (2) the implementation of BMU to update prior human reliability estimates with the simulator evidence to obtain the posterior human error probability to study the risk of postulated accidents on an advanced reactor via a plant simulator [100]; (3) the implementation of BNA to assess the team situational awareness aspect of the plant operators during a steam generator tube rupture accident scenario within a Chinese digital magnetically controlled reactor [101]; (4) the implementation of ETA to facilitate the process of removing operator actions that do not contribute to the risk and identify all key operator actions that are critical to the safety of the Xe-100 high-temperature gas-cooled pebble-bed reactor design [102]; and (5) the implementation of ETA and BNA to facilitate the dependency analysis of human failure events and accurately evaluating the probability of human failure events leading to a large release as part of a Level 2 PSA study on a reactor [103].

Based on the review of the above literature pertaining on the use of PSA to perform a human reliability analysis, the following insights were obtained: First, the analysis is mostly performed with quantitative data such as the human error probability data provided by the experts. A way to improve the level of information on the human reliability analysis would be to incorporate the qualitative data (e.g., good operating procedural practices) into the analysis. Second, like in the case of the system component reliability analysis, the human reliability analysis is limited by the scarce availability of relevant data, on which the Bayesian approach can be used to update the prior information. Although there are databases available such as the Scenario Authoring, Characterisation and Debriefing

Application (SACADA) [142] and the Human Reliability data Extraction (HuREX) [143], the human error data may not necessarily be sufficiently informative or relevant across all types of reactors. Finally, the human reliability analysis is thus far performed before (i.e., early design stages of advanced reactors) and during the reactor operation. An area of potential interest would also be the human reliability analysis for the Level 3 PSA during the decommissioning phase of the reactor.

4.4. Cyber-Security

With the increasing demand and importance of the engineering system performance of the nuclear power plants, it brings the need to optimise the development life cycle of nuclear reactors if it is to remain cost competitive in the energy market [144]. An approach towards enhancing such competitiveness is to develop advanced and robust digital technologies and integrate such technologies with the plant's physical system such as the reactor protection systems, the engineering safety features actuation systems, the safety instrumentation systems, the safety monitoring systems, and the information processing and monitoring systems [144,145]. Such integration allows for improved performance by the numerous critical infrastructures such as the smart grids and the plant itself, as well as improving the efficiency, reliability, cost competitiveness, and the capability of remote supervision on the physical systems of the plant [144]. This leads to an increasing popularity towards the digitalisation of the plant operating systems and the implementation of digital instrumentation and control systems especially in the era of Industry 4.0 [144].

However, the increased need and reliance on such digital technologies comes with the increased risk of the critical infrastructures and the plant system being subjected to the numerous cyber-threats or attacks that would severely compromise the safety in the operation of the nuclear power plant. Generally speaking, such attacks can be classified into four categories [145]:

- **Type I:** Direct attack;
- **Type II:** Indirect attack;
- **Type III:** Operator failure; and
- **Type IV:** Initiating event

A Type I cyber-attack involves the hacking of the digital systems such as the reactor protection systems and the engineering safety features actuation systems. This results in the unavailability or the abnormal operations of such systems, leading to the failure of the system to respond to an accident [146]. An example to such would be the Stuxnet worm attack on June 2010, which occurred at the Iranian nuclear facility at Natanz [147]. A Type II cyber-attack involves the hacking of the digital controllers which control the analog components such as the pumps and valves, which may lead to the operational failure or the physical damage of such components [148]. Examples include the Slammer worm attack on January 2003 which occurred at the Davis-Besse nuclear power plant in Ohio [149] and the cyber-attack on August 2006 leading to the subsequent shut-down of Unit 3 of Browns Ferry nuclear power plant in Alabama [150]. A Type III cyber-attack is one that compromises the human–machine interface, which in turn leads to operator failures due to poor or incorrect information (i.e., error of commission). An illustrative example would be to consider a Type II cyber-attack on the digital control system, which results in the operators receiving incorrect information on the operating state of a component, and subsequently taking the incorrect response actions, leading to an accident such as a LOCA. As a result, the safety injection system is switched off when in fact it should remain switched on [145]. A Type IV cyber-attack is one that causes initiating events such as LOCA or a station black-out. This can also occur as a result of Type I and II cyber-attacks. An illustrative example would be to consider a Type II cyber-attack on the letdown valves or the safety depressurisation valves such that they remain open, which could eventually lead to a LOCA [145].

The significance of the consequence(s) of such cyber-attacks on the operating state and safety levels of the nuclear power plant motivates the need to develop PSA approaches

towards assessing and improving the resilience, reliability, and the response measures against such attacks (i.e., cyber-security). Recent developments within the aspect of cyber-security include (1) the implementation of PNA to perform a resilience assessment of the digital feed water control system of a pressurised water reactor and mitigate the risk of Type I, II, and III cyber-attacks [104]; (2) the implementation of PNA to analyse the effect of integrating the preventive and reactive measures on a digital feedwater control system against Type I and II cyber-attacks, and study the subsequent impact such attacks on the sub-system's control node disturbance and manual recovery probability [105]; (3) the implementation of ETA to develop a risk analysis framework to evaluate the vulnerabilities of the instrumentation and control systems within the TRIGA research reactor and improve its resilience against a Type IV cyber-attack [106]; (4) the implementation of ETA to improve the reliability of a fission battery system in transmitting operational data securely to a monitoring facility and minimising the risk of a Type III cyber-attack [107]; and (5) the implementation of BNA in developing a flow intrusion model used to evaluate the resilience of the reactor's security system against a Type I cyber-attack [108].

Based on the review of the above literature, the following insights were obtained: First, the analysis thus far is limited to Level 1 PSA involving the reliability and resilience analysis of SSCs against cyber-attacks. Such an analysis can be extended towards Level 2 and 3 PSA to further understand the impact(s) of such attacks on the environment due to the risk of source term release. Second, the dynamic modelling approaches such as PNA and BNA are favoured for such an analysis, owing to their capability in modelling the time evolution of the accident sequence initiated by a cyber-attack, and facilitating the subsequent analysis of the recovery time of the SSCs. Finally, the modelling approaches implemented assume independence between the different components. Such an assumption may not always hold true in general and such an analysis could incorporate the information on the varying dependencies between the components.

4.5. Policy-Making

A key aspect of PSA is its capability to quantify the risks and consequences associated with the identified severe accident scenario as highlighted in Section 2.2. By doing so, it provides a tool and basis for risk-informed decision-making (i.e., risk management), an important step beyond the risk assessment phase towards managing risk. In fact, the NRC supports such a stance, and this is reflected in its policy statement that the use of PSA technology in its regulatory activities should be increased and supported by the state-of-the-art PSA methods and data in a manner that complements the NRC deterministic safety analysis [151]. This stance is further reinforced with its endorsement of the updated technical requirements published by the Nuclear energy Institute in 2019, which looks at the risk-informed performance-based technology extending towards non-light water reactors [152]. Parallel efforts were invested by the American Society of Mechanical Engineers which published its updated standard on the PSA methods on non-light water reactors in 2013 [153].

Risk-informed decision-making involves the process of evaluating and accounting for the risk associated with the identified accident scenario, based on its probability of occurrence, and using such information to make informed choices and come up with policies towards accident response, mitigation, and prevention [154]. In the context of nuclear safety, the implementation of such policies towards risk management can be broadly targeted towards numerous area which include the following:

- Plant inspection/maintenance policy;
- Human/organisational factors;
- Emergency planning/response.

4.5.1. Plant Inspection/Maintenance Policy

Recent work pertaining to the implementation of PSA to devise plant inspection/maintenance policy include (1) the implementation of BMU to develop an improved

Bayesian filtering framework, predict the remaining useful life, and devise a maintenance schedule of the reactor's electric gate valves [155]; (2) the implementation of BMU to develop a reliability analysis framework to monitor the deterioration state of the reactor's pipe, and devise the maintenance planning of the component [110]; (3) the implementation of BMU to develop an adaptive maintenance policy optimisation for a reactor pump component under imperfect knowledge of the degradation model and partial observation of system states [111]; (4) the implementation of ETA and FTA to develop a safety inspection methodology on an APR1400 based on the Defence-in-Depth principle [112]; and (5) the implementation of PNA to develop a dynamic risk-aware maintenance framework, combining both condition-based and risk-based maintenance principle, for the pump and valve components of the reactor high-pressure injection system [113].

Based on the review of the above literature pertaining on the use of PSA to devise plant inspection/maintenance policy, the following insights were obtained. First, the online Bayesian model updating framework is mostly implemented owing to its capability in allowing the analyst to sequentially update the degradation model of SSCs with data coming in across different time-steps. This allows for a real-time update of the degradation model as well as the operating state of the SSCs which, in turn, allows for a data-driven update on the predictive maintenance policy and improving the reliability of such SSCs. Second, the works of [110,155] present a physics-guided Bayesian updating framework while that of [111] presents a more data-driven Bayesian updating framework towards updating the degradation model. The latter approach is limited to the quality and availability of the data and hence, the extrapolation of the degradation model needs to be treated with due caution due to the lack of information on the physics of the system (e.g., the material's physical property). Third, the implementation of the ETA along with the FTA approach in [112] provides an added advantage in this context in that it allows for the analyst to better visualise the accident sequence and its root causes, as well as identifying the vulnerable components and the areas where redundancy is required. This facilitates a risk-informed maintenance policy planning, although it needs to be noted that such an approach still assumes independence between the accident events when computing the Top event probability. Finally, the PNA approach, while useful in performing a dynamic risk modelling of the operating state as part of the risk-aware maintenance framework, is only as good as the knowledge of the SSCs' degradation profile.

4.5.2. Human/Organisational Factors

Recent works pertaining to the implementation of PSA to evaluate the human/organisational factors include (1) the implementation of BNA to assess technical capabilities of the operator during the start-up and shut-down of the reactor, and identify the tasks needing attention and to be addressed via training or improved operating procedures [114]; (2) the implementation of BMU and BNA to perform a technical assessment with scarce data and expert judgement towards quantifying the level of decision-related "Errors of Commission", and thus, the procedural expertise of the operators [115]. (3) the implementation of ETA to perform a dynamical core damage risk assessment, and evaluate the technical capabilities of the operator in responding to a steam line break accident [116]; (4) the implementation of FTA towards quantifying the effectiveness of the risk mitigation measures by decision-makers and plant operators on the safety level of the VVER-1000 systems [117]; and (5) the implementation of ETA and FTA to identify human failure events and crew failure modes and evaluate the preparation and response level towards external flooding events in a reactor [118].

Based on the review of the above literature pertaining on the use of PSA to evaluate the human/organisational factors, the following insights were obtained: First, the BNA approach implemented in [114] evaluates the human factors on individual operator actions during the start-up and shut-down of reactors but is yet to account for the case where multiple operators are involved in such procedures (i.e., the case in most reactors) which provides a more realistic assessment of the human factors. Second, the ETA approach imple-

mented in [116] identified a significant correlation between the human error probability and the reactor's available time during a main steam line break accident for a pressurised water reactor. However, such a conclusion is yet to be verified against other types of accident such as a pressuriser failure. Finally, the analyses in the work presented thus far were performed with the available dataset. None of them has yet considered an online approach towards evaluating the human factors in real-time while data collection is ongoing.

4.5.3. Emergency Planning/Response

Recent works pertaining to the implementation of PSA to facilitate emergency planning/ response include: (1) the implementation of ETA and FTA to estimate the containment failure frequency of a reactor and develop a framework to model severe accident management guidelines based on a Level 2 PSA [119]; (2) the implementation of BNA to develop an integrated assessment method for the real-time source term release for a high temperature gas-cooled reactor and facilitate emergency planning [120]; (3) the implementation of BMU to update the turbulent dispersion model and subsequently infer/track the radioactive leakage location to facilitate an early emergency response [121]; (4) the implementation of BMU to infer the height of the Iodine-131 release source and the wind direction via an Artificial Neural Network surrogate model to facilitate the emergency response after a nuclear accident [122]; and (5) the implementation of ETA and FTA to develop a risk-informed comprehensive path-planning method for the road transportation of radioactive materials based on the accident radiological consequence assessment [123].

Based on the review of the above literature pertaining on the use of PSA to facilitate emergency planning and response, the following insights were obtained: First, while the work in [119] has provided a detailed Level 2 PSA analysis and severe accident management guidelines, the analysis can be extended towards a Level 3 PSA to provide a more comprehensive approach for accident mitigation and emergency planning. Second, the BMU-based method in [121] provides a data-driven deep-learning framework towards inferring radioactive leakage location. Such an approach, however, is subjected to the availability and the quality of the data available. As an extension to such work, a physics-informed neural network can be implemented to improve the inference and prediction performance by the prediction model. Finally, the analyses in the works presented thus far have only been performed for a single-unit reactor. Such studies can be extended towards the case of a multi-unit reactor under different physical configurations and uncertainties in the dependencies between the individual reactor unit.

5. Future Research Directions

Following from the literature review presented in Section 4, the subsequent subsections identify and provide perspectives on the latest research challenges in six areas within the PSA discipline on which further investigations can be embarked.

5.1. Multi-Unit PSA of Small and Micro Modular Reactors

Since the Fukushima-Daiichi accident, there have been more stringent requirements in place for the improved passive safety systems for licensing, which has increased the operational costs of commercial reactors [156]. This has motivated the nuclear industry to look into small modular reactors which refer to advanced reactors with a power capacity of up to 300 MW per unit [157,158]. Some key traits of a small modular reactor include [156] (1) factory built; (2) modular in design; (3) can be easily transported to the plant site; and (4) the capability to provide additional on-demand capacity additions [157]. Examples, along with their design status and corresponding references, are presented in Table 3.

As most of the recent research works have focused on performing a single-unit PSA, there have been relatively few works that have looked into the multi-unit PSA on the small modular reactors under different physical configurations or site settings. Such work would be of relevance given that a plant site would typically implement multiple such reactors in

numerous configurations (e.g., 12-module configuration in the case of NuScale's reactor design, and the 6-module configuration in the case of HTR-PM600 design [159]).

Table 3. Examples of some of the small modular reactors and their operational statuses (at the time of writing). Details on the respective reactor are found in [157].

Full Name	Acronym	Status
Boiling Water Reactor X-300	BWRX-300	Conceptual Design
High Temperature Gas-cooled Reactor - Pebble-bed Module	HTR-PM	In Operation
NUWARD	NUWARD	Conceptual Design
Advanced Lead Fast Reactor European Demonstrator	ALFRED	Under Design
KLT-40S	KLT-40S	Under Construction
NuScale SMR	NuScale	Under Regulatory Review
Xe-100	Xe-100	Conceptual Design
SMR-300	SMR-300	Seeking UK Licensing
ACP-100 Linglong One	ACP-100	Under Construction
CANada Deuterium Uranium SMR	CANDU SMR	Conceptual Design
System-integrated Modular Advanced ReacTor	SMART	Conceptual Design
Kairos Power Fluoride Salt-Cooled High-Temperature Reactor	KP-FHR	Seeking US Licensing

In addition, the PSA procedure can be made more robust by deviating from the independence assumption and accounting for the inter-modular dependencies or characterising the uncertainty in the dependencies between the individual modules while computing the overall probability of a core damage or source term release. The robustness of such an analysis can be further reinforced by accounting for the updated PSA standards set by the Nuclear Energy Institute and the American Society of Mechanical Engineers (i.e., see Section 4.5), given that some small modular reactors are non-light water reactors by design such as the HTR-PM and the Xe-100. This remains an open research topic.

Finally, the multi-unit PSA on small modular reactors can be extended to incorporate the time-varying aspect of the risk analysis as the accident progresses, thereby providing an opportunity to develop real-time dynamic PSA approaches as information pertaining to the accident is provided to the analyst sequentially across different time sequences.

It is to be highlighted that the proposed investigation can be extended towards micro modular reactors (e.g., the Ultra Safe Nuclear Corporation micro modular reactor), which are a sub-category of small modular reactors with a power capacity of up to 20 MW per unit [160].

5.2. Functional Failure Analysis of Passive Systems

The functional failure analysis of passive safety systems is an ongoing research topic within the area of reliability analysis for PSA being heavily studied [161–164] and reviewed [165–167]. While not a new research area, its importance is becoming of greater significance with the increased interest and safety analysis of small modular reactor designs in recent years. Through the literature review process, however, it has been identified that most such analyses were performed without considering the dynamic nature of the reliability of the passive system (i.e., accounting for the system degradation across the operational lifetime). As such, future research efforts can be dedicated towards treating such a system as a multi-state system and performing a dynamic PSA via a time-varying reliability analysis.

In addition, the literature survey presented in Table 2 found that most component reliability analyses are performed through time-independent Bayesian model updating. Such an analysis can be extended to time-varying reliability analysis through the implementation of sequential Bayesian filtering technique involving Sequential Monte Carlo

methods to infer the time-varying physical parameters of the passive safety system such as the water pressure and temperature parameters. This generalises the reliability analysis.

Penultimately, a recent work by [161] highlighted the logical "AND" relationship between the functional failure and the system fault/configuration when calculating the failure probability of the passive safety system via FTA. However, such an analysis was performed under the independence assumption between these two events. In reality, such an analysis may not hold true, and future research efforts may be invested in considering the uncertain dependencies between these two events in the eventual imprecise failure probability computation of the passive safety system.

Finally, further investigations can be conducted to study the dependencies between the reliability of the safety-related passive safety systems and that of the active safety systems, especially during nuclear transients (e.g., LOCA and station black-out). This would improve the robustness of the functional failure analysis of the passive safety systems, making the result of the analysis more realistic due to the relaxation of the independence assumption between the passive and active safety systems. The independence assumption can lead to an underestimation of the failure probabilities, which is undesired.

5.3. Physics-Enhanced Machine Learning for Risk Analysis

Based on the literature survey, a data-driven approach is implemented in most of the research to perform the necessary PSA, thereby being a data-driven risk analysis. However, such an approach is only as effective as the availability and the quality of the data (e.g., the component reliability data) and may fail to fully characterise the physics-based performance of the system. The latter is noteworthy given that the physical phenomena such as the heat flux and the fluid dynamics within the system can affect the system performance/functional reliability. This becomes of greater importance when studying the passive safety systems of small modular reactors or other advanced reactors which rely heavily on convection current and gravity to ensure the circulation of the coolant without the need of mechanical pumps.

To address this, a physics-enhanced machine learning approach can be implemented to perform a physics-informed risk analysis on the passive safety systems or the reactor steam generator sub-system. This can be achieved through the use of physics-informed neural networks developed by Raissi (2019) [168], which have recently been implemented to develop a reduced-order model for the digital-twinning of a reactor [169] as well as to perform a physics-enhanced risk assessment on a Westinghouse three-loop 900 MW pressurised water reactor. Such works can be extended towards small modular reactors and other advanced reactor designs to perform a real-time health monitoring of the reactor during its operational lifetime—i.e., an online Level 1 PSA of the reactor.

In addition, such an approach can also be applied towards performing a Level 3 PSA in predicting the time-evolving dispersion profile of the radioactive nuclide release. This provides an improved model interpretation and prediction explanation, which facilitates the risk-based emergency planning and response to such accident scenarios. The physics-enhanced machine learning approach for this purpose is more favourable should the analyst deviate from the use of the simplified Gaussian plume model and instead opt for a more complex physics-based differential equation, which can only be solved numerically, to describe the fluid dynamics of the radioactive nuclide release.

5.4. Human Reliability Analysis for Multi-Unit Nuclear Reactors

In operating a nuclear reactor facility, the psychological confidence level of the plant operators is of importance. A high psychological confidence level leads to less technical error or decision error associated with the operator which leads to an increased human reliability and the overall reactor reliability. However, such a confidence level would differ between operating a single-unit plant and a multi-unit plant, given the significant difference in the operating conditions and the operational configuration of the plant. This is especially so given that the configuration of a multi-unit plant is relatively more complex and its operating states require greater attention by the plant operators on site.

In this aspect, further works can be invested towards performing a dynamic PSA to assess the psychological confidence levels on the operators of the multi-unit plant. In doing so, it not only serves to identify the factors which contribute to the psychological confidence of the operators but also assists decision-makers in devising the necessary training or operating procedures aimed towards improving the technical expertise on a multi-unit plant as well as the mental well-being of the operators who may experience a higher degree of stress associated with having to operate a more complex system. The analysis can also be extended towards improving the control room design with user-friendly digital instrumentation and control configuration.

5.5. Risk Communication

An essential and challenging aspect of nuclear-related communication would be the risk communication to the general public. Such an aspect is of great importance if it is targeted towards an audience seeking to understand the impact of radiation on their daily lives and to broaden their perspectives on nuclear energy and its risks [170].

To the best of the authors' knowledge based on the literature survey, there not been any investigation on employing the PSA methods studied here (e.g., FTA, BNA, and PNA) as educational tools towards such outreach efforts. As such, future studies can delve deeper into studying the effectiveness of such PSA methods as visual aid for nuclear risk communication. In essence, the interest is to investigate how effective such tools are in educating the layman on the benefits and risk of nuclear energy, and what impact such an approach has on the eventual public perception of the nuclear technology.

5.6. Risk Analysis with Uncertainty under Limited Data

At the conclusion of Section 2.2, it is identified that one key challenge in performing PSA is the lack of data. The issue can arise in two ways: (1) when the data are incomplete (i.e., filled with missing data and gaps); or (2) when the volume of data is limited due to the limited experiment campaigns conducted. Such a problem still persists at present, and remains an active research challenge [171].

To address such a research challenge, uncertainty quantification approaches can be implemented, such as interval arithmetic and approximate Bayesian computation [79]. This allows for one to quantify the epistemic uncertainties associated with the estimate on the probability of the Top event or the small failure probability of the nuclear reactor system (i.e., best estimate plus uncertainty). This presents opportunities to further develop numerical/computational techniques, and allows for the propagation of intervals and Bayesian updating procedures on the reliability models of the nuclear reactor system. Such research would be beneficial especially when there is a high number of correlated uncertain model parameters to infer from the limited data and the need for the computational costs to be minimised.

6. Conclusions

The review paper has attempted to address the following three questions pertaining to research efforts in Probabilistic Safety Assessment for nuclear safety:

1. What is Probabilistic Safety Assessment in the context of nuclear safety and how did it come to be?
2. What has been done thus far in this area?
3. Where do we go from here in terms of future research efforts?

To address the first question, a conceptual introduction on Probabilistic Safety Assessment, its historical development in nuclear safety, and a description and evaluation on the standard tools implemented for such an analysis are provided in Sections 2 and 3. This aims to provide the readers with an overview and a historical perspective of Probabilistic Safety Assessment within the nuclear industry.

To address the second question, a literature survey of the recent research developments between 2022 and 2024 is provided in Section 4 across the five different aspects of

Probabilistic Safety Assessment: (1) multi-unit Probabilistic Safety Assessment; (2) dynamic Probabilistic Safety Assessment; (3) reliability analysis; (4) cyber-security; and (5) policy-making. For each aspect, a review of the recent literature is provided, highlighting what has been performed and the limitations/gaps still yet to be addressed in the research. This provides an evaluative perspective on the recent developments.

To address the third question, the paper proceeds to provide suggestions on future research developments in Section 5 based on the research gaps identified from the literature survey. Such future research works are proposed for the following six areas: (1) multi-unit Probabilistic Safety Assessment of small and micro modular reactors; (2) functional failure analysis of passive systems; (3) physics-enhanced machine learning for risk analysis; (4) human reliability analysis for multi-unit nuclear reactors; (5) risk communication; and (6) risk analysis with uncertainty under limited data. For each area, the perspective(s) on the proposed research direction is provided for potential consideration to address the current research limitations/gaps.

Hence, the review paper has sought to provide historical and evaluative perspectives, as well as a prospective on the area on Probabilistic Safety Assessment. Through such coverage, the paper targets readers who are new to the topic and would require a comprehensive introduction; those who are at the literature review phase and striving to understand the current state-of-the-art developments; and the wider research community within the nuclear safety discipline towards pushing the frontiers of Probabilistic Safety Assessment research.

To conclude the paper, a timeline is illustrated in Figure 5, summarising the historical development of Probabilistic Safety Assessment in nuclear safety and the key milestones over the past seven decades.

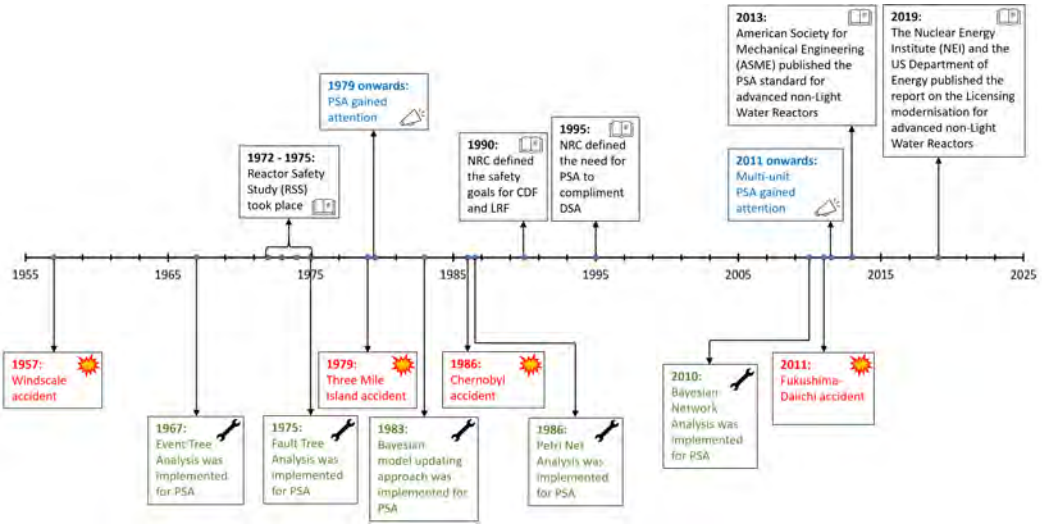

Figure 5. The timeline outlining the key milestones and developments in PSA.

Author Contributions: Conceptualisation, A.L. and S.X.; Methodology, A.L. and S.X.; Formal analysis, A.L., J.C. and S.X.; Investigation, A.L., J.C. and S.X.; Resources, A.L. and J.C.; Writing—original draft preparation, A.L.; Writing—review and editing, S.X. and J.C.; Supervision, S.X.; Project administration, A.L. and S.X.; Funding acquisition, S.X. and K.Y.C. All authors have read and agreed to the published version of the manuscript.

Funding: This research was funded by the National Research Foundation Singapore (Grant number: A-0001360-06-00).

Acknowledgments: The authors would like to acknowledge the support of the Singapore Nuclear Research and Safety Institute (SNRSI) in providing the necessary resources towards realising the work.

Conflicts of Interest: The authors declare no conflicts of interest.

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Regulatory Role of PSA in Canada and lessons learned from its use in advanced reactor facilities

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Abstract: In Canada, Probabilistic Safety Assessment (PSA) is formally integrated in the licensing and regulatory oversight of nuclear power plants throughout the whole life cycle. The Canadian Nuclear Safety Commission (CNSC) uses the information provided by the PSA to support the licensing, regulatory oversight program, to reexamine the efficacy of regulatory systems, to enhance emergency preparedness and, to develop a risk-informed compliance verification process.

The CNSC regulatory document REGDOC-2.4.2, "Probabilistic Safety assessment (PSA) for Reactor Facilities" [1] sets high level requirements which call for the development and application of Level 1 and Level 2 PSA.

The CNSC's current licensing strategy for the new builds is established based on the practice and experiences from the licensing of CANDU Nuclear Power Plants (NPPs). As part of an application for a licence to construct, CNSC-REGDOC 1.1.2 "Licence Application Guide: Licence to Construct a Reactor Facility" [2] requires that the application shall include a PSA conducted in accordance with REGDOC-2.4.2. It is expected that the PSA scope and level of detail, at this stage, should be sufficient to provide preliminary risk results and insights, and to demonstrate that the PSA objectives established in REGDOC 2.4.2 and REGDOC 2.5.2 "Physical Design – Design of Reactor Facilities" [3] are met.
The CNSC REGDOC 2.5.2 requires that the PSA shall be developed to demonstrate that the safety objectives including safety goals, as well as other safety system performance indicators have been achieved. It also requires that the application describe how the results of the PSA have been used to identify plant vulnerabilities to severe accident, to support the Structures, systems and components (SSCs) classification, and how the results of the PSA are used to provide insights into the accident management program.

This paper aims to provide an overview of current status, application and the regulatory aspects of PSA in Canada. In addition, it will provide a brief background on the recent regulatory use of PSA for advanced reactor facilities and discuss its technical challenges and lessons learned.

Keywords: PRA, PSA, SMR, advanced reactor

1. INTRODUCTION

The probabilistic safety assessment (PSA) is a comprehensive, structured and logical analysis aimed at identifying, assessing and quantifying the risks in complex technological or natural systems. It is recognized as one of the key approaches to assess the safety critical aspects for the existing plants and for the new designs. PSA provides insights into the strengths and weaknesses of the design and operation of a nuclear power plant as it assists plant management and the regulator to target resources where the largest benefit for plant safety can be obtained.

In Canada, PSA is formally integrated in the regulatory process of nuclear power plants throughout the whole life cycle of a nuclear power plant starting from the early design phase and continuing through the construction and operation phases up to the decommissioning phase. High level regulatory requirements for

the development and application of PSA are established in the Canadian Nuclear Safety Commission (CNSC) regulatory document REGDOC-2.4.2 "Probabilistic Safety assessment (PSA) for Reactor Facilities" [2]. The current CNSC regulatory requirements on PSA are adequate for new reactor designs including SMRs, however more guidance is needed on some topics such as the scope and level of detail of PSA at each licensing stage, as well as the applicability of the numerical safety goals. Guidance is also needed on some technical topics such as human reliability analysis, and the modelling of passive safety features SSCs and safety functions.

2. BRIEF HISTORY ON PSA USES IN CANADA

The use of PSA in, or as an adjunct to, plant licensing has had a long history in Canada, going back to the first commercial Canada Deuterium Uranium (CANDU) prototype (Douglas Point) ([4], [5]).

The early designs of the CANDU reactors were based solely on deterministic rules and criteria such as, single/dual failure[1], and reliable design features such as redundancy, and diversity. The continuous effort for safety improvements, specifically by regularly trying to answer the "what if?" question, has led to concerns regarding the independence of the mitigating systems from the process systems based on knowing that the failure of the safety support systems could result in common cause (cross-link) failures of process and special safety systems. To address these concerns, Atomic Energy of Canada Limited (AECL) developed, in the late 1970's, an analysis approach called the Safety Design Matrix methodology. This is considered as the predecessor to PSA.

The first comprehensive PSA for CANDU reactors was completed by Ontario Hydro for the Darlington Station in 1987. As a pre-operational study, the Darlington Probabilistic Safety Evaluation (DPSE) played a significant role in optimizing the design of the station. A number of vulnerabilities were identified and corrected, for example.

Following the completion of DPSE, Ontario Hydro began a program of Probabilistic risk assessments (PRAs) for Pickering A and Pickering B nuclear power plants (NPPs). The Pickering A station PRA, completed in 1995, identified a number of single failures and changes were made to the moderator system during the Pickering A prior to Return to Service.

The Bruce B station PRA, completed in 1999, identified the need for improving the powerhouse venting system. AECL used its PSA methodology during the design process of its next product line, the CANDU 9 design, which incorporated a number of improved design features.

Currently, all licensees have completed the Level 1 and Level 2 PSAs for internal and external events during at-power and shutdown states. Internal hazards include hazards originating from the sources located on the site of the reactor facility that proceeds from a human error or from a failure of a SSC. External hazards are those hazards of natural or human-induced origin that originate outside the site (e.g., earthquakes, lightning, tornadoes, fires and floods, aircraft crash).

3. PSA IN REGULATORY FRAMEWORK

PSA is referred to in licensees' Nuclear Power Reactor Operating Licence (PROL) as a licence condition under the Safety and Control Area (SCA) "Safety Analysis". The objectives of the PSA are listed in section 3 of CNSC REDOC-2.4.2. PSA is also referred to in a number of CNSC regulatory documents as an approach that can be used to support other SCAs and regulatory programs, such as: Physical Design; Deterministic Safety Analysis (DSA); Reliability Program; Minimum Shift Complement; Maintenance Program; Aging Management; Accident Management, and Emergency Management.

[1] A "single failure" is a failure of any one process system. Dual failures, a much less likely event, are defined as a single failure coupled with the unavailability of one special safety system (one shutdown system, the emergency core cooling system, or impairments in containment).

3.1. PSA regulatory requirements

The CNSC issued the regulatory document REGDOC-2.4.2 [1] in April 2014 as an amendment of the previous CNSC standard S-294 [7] in response to CNSC Fukushima Task Force recommendations [8]. This regulatory document sets high level requirements which call for the development of Level 1 and Level 2 PSA by applying a formal quality assurance process for conducting the PSA. The scope of the PSA shall include internal and external events, both at-power and shutdown operational states, multi-unit impacts, as well as the inclusion of sensitivity analysis, uncertainty analysis, and importance measures. It also requires the licensees to seek CNSC acceptance of the methodology and computer codes to be used for the PSA, which means that the methodology and the computer codes have to be formally accepted by the CNSC prior to the submission of the PSA reports. As a basis for methodology acceptance, REGDOC-2.4.2 refers to the IAEA specific safety guides SSG-3 [9] and SSG-4 [10].

REGDOC 2.4.2, version 2 [1] was issued in May 2022 to take into account the recent developments of new nuclear and small modular reactor technologies. This document retained the PSA requirements and guidance included in the previous version with the clarification that these requirements are set out for a licence to construct or operate a reactor facility. It also clearly specifies that the PSA requirements are applicable to reactor facilities, including nuclear power plants, small reactor facilities, or non-power reactors (research and test reactors), using a graded approach.

The Canadian Standards Association (CSA) Group standard on PSA, CSA-290.17 [11] provides guidance as to how a licensee can achieve requirements currently found in REGDOC-2.4.2. This standard includes clear requirements for the Licensees to develop and implement a systematic process to update a PSA.

3.2. Safety Goals, Criteria and Risk Metrics

At the CNSC, the ultimate output of safety is the prevention of unreasonable risk as endorsed in the Nuclear Safety and Control Act [12]. Safety Goals, as introduced in REGDOC 2.5.2, are defined in a more comprehensive way by combining both qualitative goals and quantitative goals.

Two qualitative safety goals are established based on the principle that nuclear risks posed by NPP operation should not be a significant addition to other societal risks. The qualitative safety goals are:

- Individual members of the public shall be provided a level of protection from the consequences of reactor facility operation, such that there is no significant additional risk to the life and health of individuals.
- Societal risks to life and health from reactor facility operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks.

The quantitative safety goals are formulated in addition to the dose acceptance criteria and to deterministic design requirements to ensure that risk to the public originating from accidents outside the design basis is considered. Quantitative safety goals established in REGDOC 2.5.2 include:

- Core damage frequency (CDF): The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 10^{-5} per reactor year.
- Small release frequency (SRF): The sum of frequencies of all event sequences that can lead to any release to the environment that requires temporary evacuation of the local population or a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year.
- Large release frequency (LRF): The sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the population or a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year.

The Safety Goals for existing NPPs are adopted by the licensees in conformance with International best practice, specifically the IAEA Safety Series No 75-INSAG-3 (1988) [13] which was updated in 1999 as INSAG-12 [14]. Canadian licensees use different terms to characterize these safety goals: Safety Goal;

Safety Goal target; Administrative Safety Goal. However, all these definitions come down to the same general principle of establishing safety goal not to be exceeded, and a Target "administrative safety goal" to strive for.

Safety goals adopted by Canadian licensees are shown in Table 1 below:

Table 1. Canadian Licensees definition of Safety Goals

	Safety Goal (per reactor.year)	Administrative Safety Goal "Target" (per reactor.year)
Severe Core Damage Frequency	10^{-4}	10^{-5}
Large Release Frequency	10^{-5}	10^{-6}

The industry application of Safety Goals to implement design upgrades is defined as follows:

- If assessed risk metrics are above safety goal: modifications are necessary.
- If assessed risk metrics are between safety goal and administrative safety goal: modifications are necessary if practicable.
- If assessed risk metrics are below the administrative safety goal: modifications are expected if practicable.

3.3. PSA regulatory role

In 2018, CNSC issued a regulatory fundamentals document REGDOC-3.5.3 "Regulatory Fundamentals" [15] which describes risk informed approach to licensing and compliance activities where the focus is put on issues of higher risk for effectiveness and efficiency, but it does not refer to the use of PSA risk insights (qualitative and/or quantitative results), nor to the safety goals. This REGDOC has been updated in 2023 to expand information on the CNSC's risk-informed and graded approaches to nuclear regulating, particularly in regulation of new technologies such as small modular reactors (SMRs).

In addition, all REGDOCs prefaces evoke the use of the "Risk informed approach". In this approach, the PSA results can be used to complement the deterministic approach, and Risk Informed Decision Making (RIDM) key principles (PSA limitations and uncertainty should be considered). Figure 1. below illustrates the CNSC RIDM process.

The RIDM process:

- helps in prioritizing goals and activities,
- ensures that all aspects of a given issue are identified and considered when making decisions,
- is easily understandable, well documented and structured,
- is adaptable to various decision-making situations,
- enables decision-makers to explain decisions,
- respects stakeholder consultation, and
- is founded on open communication

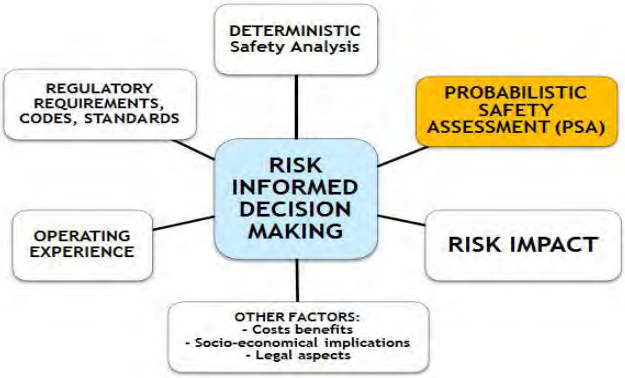

Figure 1. CNSC RIDM Process

As a comparison of PSA uses in other countries, US as an example, US NRC issued different Regulatory Guides and other RIDM processes which promote the use of the PSA results and provide guidance on the thresholds for change in core damage frequency (ΔCDF) and Large Early Release Frequency (ΔLERF) for characterizing the risk significance of issues. As an example,

- RG 1.174 "*An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*" [20],
- NRR Office Instruction LIC-504 "*Integrated Risk-Informed Decision-making Process for Emergent Issues"* [21].

In both these documents, acceptance guidelines regarding incremental risk calculated by the PSA (ΔCDF and ΔLRF) are provided. As an example, $\Delta CDF < 1x10^{-6}$ per reactor year and $\Delta LERF < 1x10^{-7}$ per reactor year serve as thresholds to show that issues are of very low risk significance.

The regulatory role of the PSA, using the RIDM approach, can be grouped in the following broad regulatory activities:

3.3.1 Licensing: comparison of PSA results against safety goals

For the CNSC, Safety Goals are quantitative indicators of the overall safety of the plant. The safety Goals form part of the licensing basis to identify design improvements to enhance safety and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

The main tool for demonstrating that the NPP design meets Safety Goals is PSA; Level 1 for the assessment of plant failures and responses of reactor systems (core damage frequency), and Level 2 for the assessment of containment response (large release frequency). The consequences to the public (doses) and to the environment are assessed using the Level 3 PSA. Consistently with the international practice, this level of assessment is not required by the CNSC.

The PSA results are compared against the established safety goals. However, it is internationally accepted that the probabilistic safety goals should not be used as the sole basis for regulatory decisions. It is also recognized that a firm regulatory limit cannot be stated on a risk metric that is assessed using a purely probability-based approach.

3.3.2 Regulatory oversight

CNSC staff use the information provided by the PSA to support the regulatory oversight program. Indeed, as part of the ongoing risk-informed, performance-based approach in support of the regulatory compliance verification program, including site inspection activities, CNSC staff completed an internal project, called the Risk Handbook, to summarize the licensees' PSA results, important risk insights and the performance of important safety systems for CNSC inspectors. The project has two main objectives: to provide introductory

PSA training to CNSC site inspectors, and to develop a user-friendly, web-based tool summarizing licensees' PSA results and insights.

The handbook can be used by CNSC inspectors to:

1. Optimize inspection planning and improve efficiency:
    - Focus on risk significant SSCs, Initiating Events, and Human Actions, Specific hazard information
2. Provide risk insights to define the scope of existing regulatory inspection procedures:
    - Electrical/Mechanical Systems Inspections
    - Human performance
    - environmental qualification (EQ), and seismic qualification inspections
    - Fire and flood hazards inspections
3. Evaluate inspection results, and provide an understanding of the risk impacts in situations where a SSC is taken out of service, or a system experiences an impairment level
4. Help in determining the safety significance of operational events

3.3.3 Operational event evaluation and abnormal plant configurations

CNSC staff use the PSA in combination with the DSA to evaluate the potential consequences of an operational event. PSA-based event analysis provides a quantitative assessment of the risk significance of the event by calculating the risk increase induced by the event. Operational event review is used to provide an input into the consideration of what changes could be made to reduce the likelihood of recurrence of such operating events.

PSA is also used as an integrated model to evaluate the risk increase due to an abnormal plant configuration where multiple SSCs unavailability has occurred. This unavailability could be due to equipment random failures or to situations where SSCs are taken out of service for maintenance activities.

3.3.4 Changes to licensing basis

CNSC staff use the PSA results and insights in combination to the DSA to assess the impacts of licensees' proposed changes to the licensing basis. These changes may include:

- Changes in test/maintenance interval
- Changes in Operating Policies and Procedures (OP&Ps)
- Request for extending allowed outage time of an SSC.

3.3.5 PSA use in Severe Accident Management and emergency preparedness.

Since the severe core damage accidents at Three Mile Island, Chernobyl, and Fukushima, the identification and mitigation of severe accidents has become a major part of nuclear safety. PSA provides most of the input to this process: it systematically identifies severe accidents, provides a frequency so that mitigation measures can be applied in a risk-effective manner, and predicts the consequences (through PSA support analyses) so there is a sound basis for on-site accident management and off-site emergency planning.

In Canada, the CNSC is promoting the use of PSA insights in defining the strategies to cope with the consequences of severe accidents. Indeed, the Level 2 PSA has been used to identify accident management measures that could be carried out to mitigate the effects of a severe accident. This has led to the implementations of generic or plant specific Severe Accident Management Guidelines (SAMG) to guide operators in the event of a severe accident.

In addition, the source terms and frequencies produced by the Level 2 PSA have been used as the basis for Emergency Preparedness Drills and Exercises.

4. CNSC REGULATORY USE OF PSA FOR ADVANCED REACTOR FACILITIES

4.1. CNSC Licensing process and PSA Regulatory Requirements for advanced reactor facilities

The Canadian nuclear industry has shown strong interest in the development and deployment of advanced and small modular reactors (SMRs) within Canada[2].

The CNSC has been preparing for the regulation and licensing of SMRs and advanced reactors for over a decade and is continuing to invest efforts towards readiness activities. The CNSC is also currently reviewing license applications for advanced reactors/SMRs.

The CNSC licensing process for all reactor facilities include the following:

- Licence to prepare a site: This includes the preparation of an environmental assessment.
- Licence to construct: a preliminary safety report (PSAR) as well as a PSA shall be prepared as part this application.
- Licence to operate: a final safety report (FSAR), a PSA reflecting the as-built as-operated Conditions, and a reliability program for systems important to safety, should be included as part of this application.
- Licence to Decommission.
- Licence to Abandon.

CNSC also provides a pre-licensing service known as Vendor Design Reviews (VDR) [16] to provide early feedback on how the vendor may comply with CNSC regulatory requirements. Several SMR vendors have engaged the CNSC through the VDR Program.

As per the CNSC Regulatory document REGDOC 2.5.2, a safety analysis of the plant design shall include hazard analysis (HA), DSA, and PSA. The safety analysis shall demonstrate achievement of all levels of defence in depth and confirm that the design can meet the applicable expectations, dose acceptance criteria and safety goals. The PSA shall be conducted in accordance with the requirements specified in REGDOC-2.4.2 [1].

As part of an application for a licence to construct, REGDOC 1.1.2 requires that the application shall include a PSA conducted in accordance with REGDOC-2.4.2 . It also requires that the application should describe how the results of the PSA have been used to identify any vulnerabilities, and how the results of the PSA are used to provide insights into the severe accident management program, and how these results meet the safety goals.

For a licence to operate application, REGDOC 2.4.2 [1] requires that the PSA shall reflect the as-built as-operated conditions. A reliability program, in accordance with REGDOC 2.6.1 [17], is also required for the purpose of identifying systems important to safety (SIS), establishing reliability targets, and reporting annually to the CNSC on the performance of these SISs.

4.2. Regulatory Review of PSA Technical Elements and challenges for SMRs

SMRs incorporate novel aspects and features that represent a departure from traditional CANDU technology. The design of these reactors encompasses a wide array of technologies, most of which have limited operational experience.

PSA for SMRs is recognized as an important approach to achieve improved safety for the future nuclear facilities. However, the application of PSA to SMRs, which have limited operational experience, encounters concurrent challenges which can include a lack of design detail, a lack of empirical data and the possibility of failure scenarios that differ in character from those treated in CANDU PSAs. These challenges may lead to PSA results that do not reflect the future as-built, as-operated plant.

While the current CNSC regulatory requirements on PSA are adequate for SMRs, additional guidance is needed on some PSA elements. The following sections include a discussion about the PSA technical

[2] The term SMR used throughout the paper includes advanced nuclear reactors which may not be small in size.

elements and the challenges associated with their applicability to SMRs, and the CNSC staff guidance and expectations regarding each PSA technical element.

4.2.1 PSA scope and level of detail

For a licence to construct application, the applicant is required to seek CNSC acceptance of the methodology and computer codes used for the PSA. The scope and level of detail should be consistent with the risk impact, using a graded approach.
The challenge encountered is whether a full scope and detailed PSAs should be required for all reactor types and sizes at the different licensing stages, and the extent to which a graded approach can be used.
As an example, for some designs, the relatively low consequences may lead to the use of simple PSA models, to provide an indicative risk estimate. The scope may also be limited to at-power state only, and some internal and External Hazards.
The expectation from a regulatory perspective is that the applicant should document and justify the scope reduction of the PSA and to seek CNSC staff acceptance.

4.2.2 PSA Quantitative Safety Goals

The challenge about the application of the current quantitative safety goals, established in REGDOC 2.5.2, is that for designs using pebble-bed fuel (TRISO), or liquid fuel, CDF, SRF and LRF may not be considered to be applicable. The regulatory expectation is that whenever the quantitative safety goals are shown to be not applicable, the applicant shall demonstrate that the qualitative safety goals are met, including the surrogates for SRF and LRF quantitative safety goals, i.e., the temporary evacuation and the long-term relocation of the local population, respectively. CNSC is also considering the development of a Quantitative Health Objective (QHO) based on cancer incidence, as well as establishing a dose acceptance criteria for the Design Extension Conditions (DEC) which would be applicable to all nuclear power plants and SMR designs.

4.2.3 Initiating Event Analysis

The applicant is required to apply a systematic approach to provide a reasonably complete list of postulated initiating events (PIEs). For SMRs and unique designs, the challenge is the lack of operating experience (OPEX) and the absence of similar reactors to compare the list of PIEs to ensure completeness of the list and its use as a reference for deriving PIEs frequencies. The challenge may also include the lack of understanding of certain phenomena that could result in omitting some initiating events.
As indicated in IAEA Safety Report Series 25 [18], it is recognized that it is not possible to demonstrate completeness for first of a kind projects. However, by using a combination of inductive and deductive analysis methods, it should be possible to gain confidence that the contribution to the risk from initiating events which have not been identified would be small.
The regulatory guidance and expectations suggest the use of both a top-down approach, such as the Master Logic Diagram (MLD) and the bottom-up approach such as the Failure Mode and Effects Analysis (FMEA), and to provide the rationale for screening out any initiating event. It may also be necessary to use of Phenomena Identification and Ranking Technique (PIRT) to rank the importance of some phenomena.

4.2.4 Accident Sequence Analysis and System Analysis

The objective of the accident sequence analysis is to adequately model the systems responses and operator actions that affect the key safety functions for each initiating event.
For SMRs and unique designs, the lack of experience and reference studies may result in an incomplete understanding of the mitigation systems' responses and success criteria, Operator actions, as well as certain phenomena in the event sequences. There could be a challenge in modelling the plant specific dependencies, and the passive safety features, including:

- Identification of failure modes from natural phenomena.
- Modelling a suitable range of environmental and operational conditions due to less driving forces.
- Phenomena-based parameters (e.g., heat sink temperature, non-condensable gas fraction, convective heat transfer, etc.), including interdependencies.

It is, therefore, expected that the safety functions required to prevent core damage for the individual initiating events are identified, along with specific plant systems required to perform those safety functions, the systems success criteria, and related operating procedures. It is also recommended to develop event-sequence

diagrams (ESDs) for each group of initiating events to facilitate the collection and display of information required for developing system event trees, especially if the large event tree/small fault tree approach is used. Furthermore, the qualified safety analysis tools may not be yet available to support Level 1 and Level 2 PSA, specifically for the source term quantification.
The challenge also includes the PSA end states definition and whether there is a need for a separate Level 1 and Level 2 PSA.
For system analysis, the challenges include the fault tree development for new systems and system level failure criteria, modelling of passive systems, Digital I&C, mission times, time windows for operator actions, and the modelling of intersystem and intra-system dependencies.
It is expected that the applicant will identify and fully document the boundaries of each system, and to include direct functional dependencies that are incorporated directly into the fault or event trees, and the dependencies with no direct causal relationships. This task can be facilitated by the development of dependency matrices.

4.2.5 Human Reliability Analysis (HRA)

SMRs present new concepts of operation enabling a smaller operating crew for operating a single reactor or, multiple reactor units in parallel. As noted in Ref [19], the unique aspects of the modern human-system interface (HSI) in the SMR cannot be readily accounted for in the Technique for Human Error Rate Prediction (THERP), and newer HRA methods may more readily generalize to the types of technologies and concepts of operation found in SMRs.
The general challenges in HRA for SMRs include the lack of HRA data, and the lack of understanding of human actions' impact on system unavailability and the dependency between human actions, specifically at the design stage or during an application for a licence to construct.
The applicant is expected to document the HRA methodology used for the identification of the human failure events (HFEs) as well as the quantification of the human error probabilities (HEPs) taking into consideration the specific design features of SMRs such as the modern HIS, digital controls and displays, computerized procedures, and automated actions. The methodology should also describe instances where the operator may need to take recovery actions when passive safety system function is impaired.

4.2.6 Data analysis

Data analysis includes the statistical analysis of raw information, the use of generic and specific data, and/or the use of expert judgment data. The necessary data include the frequencies of initiating events, component-failure rates, common-cause probabilities, and operational data such as repair times, test frequencies and test downtimes.
The challenges related to data analysis for SMRs and unique designs include the lack of operational data to derive initiating events' frequencies and component or system unavailabilities specifically for first of a kind (FOAK) components.
The applicant is expected to document the data analysis methods. The data may be generic industry data or plant-specific data, or a combination of both, and engineering judgment supplemented with an uncertainty characterisation.

5. CONCLUSION

The Canadian Nuclear Safety Commission (CNSC) uses risk-informed approaches to support its regulatory activities. CNSC staff use the licensees' PSA results and insights to support regulatory activities, such as licensing, regulatory oversight, review and evaluation of operational events and abnormal plant configurations.

PSA for SMRs has been used from the very early stage of SMR design phase as an important approach to achieve improved safety for the future reactor facilities.

The current CNSC regulatory requirements on PSA are adequate for SMRs, however more guidance is needed on some PSA topics such as the scope and level of detail of PSA at each licensing stage, as well as the applicability of the numerical safety goals. Guidance is also needed on some technical topics such as human reliability analysis and the modelling of passive safety features (SSCs and safety functions). This new guidance will be forthcoming once more experience is gained. Indeed, the CNSC applies a risk-informed

approach to regulation focusing on the safety risks for each licensing phase. The CNSC has the regulatory tools and legislative power to make decisions as new guidance is being developed in an evolving environment.

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Emergency Mitigating Equipments – Post Fukushima Actions at Canadian Nuclear Power Plants – Portable AC Power Sources

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Abstract

In response to the Fukushima Daiichi NPP accident in 2011, the Canadian Nuclear Safety Commission set up a Task Force to evaluate operational, technical and regulatory implications on Canadian NPPs. While accepting that the risk from beyond-design-basis accidents (BDBA) at Canadian NPPs is very low, the Task Force identified a number of areas where additional improvements or confirmatory assessments would further enhance safety.

As a result, a set of 36 Fukushima Action Items (FAIs) were assigned to the licensees.

This paper focuses on the FAI related to electrical power system enhancements to address a total loss of all AC Power leading to a possibility of loss of heat sinks (i.e. Station Blackout). This required the licensees to implement the following:

- Additional back up power supplied by portable diesel generator(s) to allow key instrumentation and control equipment and key electrical loads to remain operable;
- Provisions for a storage and timely transportation and connection of the portable generator(s) to the applicable units ;
- Provisions for testing of the portable generator;
- Provisions for fuelling of portable generators;
- Provisions such as panels, receptacles, and connectors to quickly deploy the portable generators to plant system, and separate feeder cables route to avoid a common mode failure;
- Load shedding strategy to extend the existing station's battery life to ensure that the connection of portable generators can be completed before the batteries are depleted;
- Provisions to supply water to steam generators and Irradiated Fuel Bay using portable pumps;

The paper will also provide a brief description of Electrical power systems of the Canadian NPPs designed to satisfy the high safety and reliability requirements for nuclear systems, which are based on the following:

- 2 group design philosophy (Group 1 and Group 2 Electrical Power Systems)
- 2 separate groups of onsite emergency generators (Class III Standby generators and Emergency Power generators)
- 4-7 days' supply of fuel for standby and emergency generators,
- The capability to transfer power between units in a multi-unit plant

1. Introduction

Canada has five Nuclear Power Plants (NPPs) in three provinces that houses 22 CANDU reactors; 20 located in Ontario at three multi-unit stations ; Pickering, Darlington and Bruce. There are also single-

unit CANDU stations in Gentilly, Québec and Point Lepreau, New Brunswick. All currently operating reactors are CANDU (Canadian Deuterium –uranium) reactors.

Following the Fukushima Daiichi nuclear accident, the Canadian Nuclear Safety Commission (CNSC) requested all the Canadian Nuclear Power Plants (NPPs) to conduct a review of the initial lessons learned from Fukushima, under subsection 12(2) of the General Nuclear Safety and Control Regulations. In April 2011, the CNSC convened a task force to review the licensees' responses to the 12(2) request and evaluate the operational, technical and regulatory implications of the Fukushima Daiichi nuclear accident for the Canadian NPPs. While acknowledging that the probability of a beyond-design-basis accident (BDBA) at Canadian NPPs is very low, the Task Force identified a number of areas where additional improvements would further enhance safety. To address the Task Force recommendations, the CNSC developed a draft CNSC Action Plan. The document established a four-year plan, for both licensees and CNSC staff, to strengthen reactor defence in depth, enhance emergency response, improve regulatory oversight and crisis communication capabilities, and enhance international collaboration.

In parallel to the task force the President of the CNSC established an External Advisory Committee (EAC) to review the CNSC's process in responding to the Fukushima crisis and in developing proposed changes to its processes and regulatory framework. The EAC concluded that the process followed by the CNSC in response to the accident was appropriate, and identified a number of complementary areas for further enhancements.

CNSC Staff Action Plan was developed taking into consideration the comments received from the public and stakeholders on the CNSC Fukushima Task Force (FTF) Report. The CNSC Staff Action Plan identified a number of actions to address the Task Force Report recommendations. The actions are grouped in the following three categories:

– Strengthening reactor defence in depth
– Enhancing emergency response
– Improving regulatory framework and processes

This paper discusses the findings, and recommendations of the FTF with respect to the strengthening of defence in depth of CANDU Electrical Power Systems (EPS).

2. CANDU Electrical Power Systems Design

CANDU EPS is designed to satisfy the high safety and reliability requirements of the nuclear power plants. Notably, this is achieved through grouping and separation.

2.1 Group Separation

The concept of grouping and separation of safety related systems has been integral to CANDU plants. This concept provides physical and functional separation of safety related systems to ensure that common cause events do not impair the capability to perform essential safety functions. In this concept, the plant can be shut down, decay heat removed, and the plant conditions monitored independently from systems and components of either one of two groups, known as Group 1 and Group 2. The EPS is separated into Group 1 and Group 2 in accordance with the two groups' separation philosophy. Group 2 power is qualified to operate following the design basis common-mode events.

2.2 Division Separation

The design provides two electrically independent and physically separated power distribution divisions. One is designated odd, the other even. The two divisions provide power to redundant station loads and prevent failures or damage of one divisions cascading to the other.

2.3 Classes of Power

There are four distinct classes of power, namely Class IV, III, II and I, as well as Emergency Power Supply. Class IV, III, II and I are Group 1 supplies while Emergency Power Supply is Group 2 power.

The Class IV power system is an AC power that supplies the unit auxiliary loads during normal plant operation. The Class IV power can be supplied from the turbine generator or from the off-site power and may be subject to long term interruptions. Under certain circumstances the loads may be transferred from one source to others. The class IV distribution arrangement is a radial dual bus scheme (i.e. Class IV is split into two divisions; even and odd. CANDU turbine generators units are designed to survive 100 % load rejection from the external grid and continue to supply their own house load. Most of the CANDU reactors are capable of quickly reducing reactor power to 60 % of full power, holding at reduced power, and then returning more slowly to full power using their adjuster rods. The reactors have provisions to dump the steam to condenser and can remain at 60 % full power indefinitely ready for reconnection to the grid or for an orderly shut down. The unit electrical output would be held to around 10-15 % full power to supply the house loads.

Class III power is an AC system that is normally supplied from the Class IV system. Upon loss of Class IV power it is automatically supplied from on-site stand-by generators (SGs) under the control of the emergency transfer scheme (ETS). Each SG is capable of black starting and supplying the Class III requirements. The SGs are provided with an on-site fuel oil supply. Uninterruptible control power required for switching and monitoring following loss of Class IV and until Class III restoration is provided from Class I and II.

Class II is an AC system which supplies uninterruptible AC power to the essential loads, which cannot tolerate power interruptions that may occur in the Class III system. The Class II power is normally supplied from Class I batteries through inverters with alternate supply from the Class III system.

The Class I is a DC system which supplies uninterruptible DC power to the essential auxiliary services. Class I power is normally supplied from Class III power through rectifiers and is backed up by batteries to ensure uninterruptible supply to Class I and Class II loads. The overall system requirement is that if AC supplies fail than the battery will supply the emergency loads without interruption for the required time. Batteries are capable of providing uninterruptible power to the full Class II and Class I loads for a minimum period of 40 minutes following a total loss of Class III and IV power. After this time, all class I and II loads will be lost if class III power cannot be restored. Table – 1 summarizes the attributes of classes of power.

Table 1 – Classes of power in a CANDU Nuclear Power Plant

RELIABILITY ↓		
	Class IV	• Normal Alternating Current (AC) – Supplied from turbine generator and/or grid
	Class III	• AC system – Normally supplied from Class IV – Upon loss of Class IV, from Standby generator
	Class II	• AC System – Normally supplied form Class I DC system – Can be supplied from Class III
	Class I	• DC system – Normally supplied from Class III via rectifiers – Backup battery supply

2.4 The Emergency Power Supply

Emergency Power Supply is a separate power system consisting of its own on-site power generation and AC and DC power distribution, including batteries. Emergency Power Supply is the Group 2 power. The purpose of Emergency Power Supply is to provide power to selected nuclear safety-related loads following a common-mode incident such as earthquake, should all the Group 1 supplies fail. The whole Emergency Power Supply is seismically qualified to withstand a design basis earthquake. Two Emergency Power Generators (EPGs) are provided, each capable of black starting and supplying the minimum power requirements for safe plant shutdown. The EPGs are required to be manually started within 30 minutes of the occurrence of a common-mode event. Figure 1 Block diagram illustrating typical separation of Class I, II, III and IV electrical systems into independent divisions with a separate emergency power supply.

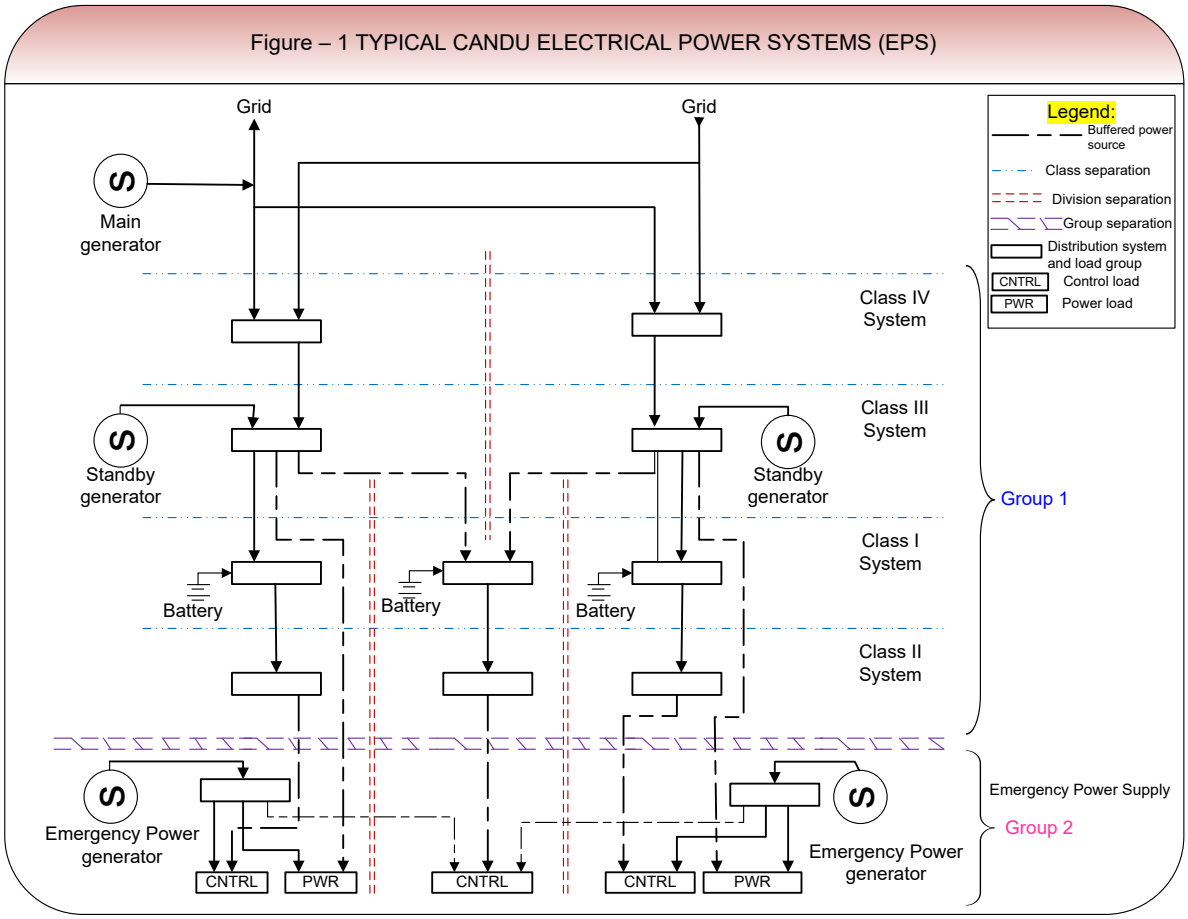


Figure – 1 TYPICAL CANDU ELECTRICAL POWER SYSTEMS (EPS)

3. Lessons learned from Fukushima

The lessons learnt from Fukushima required the licensees to analyze station capability to respond to the Station Blackout (SBO).The CNSC also reviewed the level of preparedness of Canadian NPPs to respond to a sustained SBO event. Immediately after the Fukushima event, the CNSC staff performed walkdowns at Canadian NPPs to verify the licensees' emergency preparedness for external hazards and severe accidents so that the CNSC could reassure the Commission and Canadian public that certain aspects that had contributed to the events in Japan had been specifically verified. One of the aspects verified was availability of backup power.

3.1 Fukushima Task Force Findings

A major class of beyond-design-basis accidents are those involving a loss of all heat sinks. The most challenging of these are caused by loss of electrical power. The Fukushima Daiichi accident was of this type and was the focus of the CNSC Fukushima Task Force (FTF).

One of the focus areas of FTF review was strengthening reactor defence in depth. In the event of a loss of all normal, backup and emergency AC power the batteries are the only electrical stored energy source available. However, they will be depleted in approximately 40 minutes. After that time, most control and instrumentation functions such as reactor start up monitoring, containment parameter monitoring, lighting, motorized valve operation etc, will be lost unless an alternative power source is established well before the batteries are depleted.

3.2 Fukushima Action Items

The CNSC Task Force found that licensees should explore options to extend the duration of power supplies to instrumentation and control equipment that may be needed to mitigate beyond-design-basis accidents, including severe accidents. Consequently, all licensees have performed an evaluation to determine if the 40 minute mission time for batteries can be extended to at least 8 hours. This time is required between total loss of AC power and connection of additional power sources i.e. the portable generators.

Fukushima Action Item (FAI) 1.10 required all licensees to investigate means of extending the availability of power for key instrumentation and control (I&C) needed in accident management actions following a loss of all AC power.

FAI 1.11 required all licensees to procure, as quickly as possible, emergency mitigating equipment (EME) and other resources that could be stored offsite and brought onsite to mitigate a severe accident. EME consists of portable generators, portable UPS, cables, receptacles, panels, portable diesel driven pumps and fire hoses.

3.3 Implementation of Action Items

Deliverables of FAI 1.10 implementation required the licensees to evaluate the requirements and capabilities of electrical power for key instrumentation and control (I&C). Licensees were asked to identify practicable upgrades that would extend the availability of key I&C, if needed. Licensees were asked to present a plan and schedule for deployment of identified upgrades. A target of 8 hours without the need for offsite support was required.

Deliverables of FAI 1.11 implementation required the licensees to provide a plan and schedule for procurement of emergency equipment and other resources that could be stored offsite and brought onsite to mitigate a severe accident.

To comply with the CNSC requirements the following actions were performed by the licensees:

- Operational guides were developed for shedding of non-priority Class 1 and 2 loads to extend the battery discharge time.

- Station specific list of priority loads was developed which should powered by portable generators[6] / portable uninterruptable power supplies (UPS)

- Calculated the extended life of the batteries after load shedding to ensure that it exceeds the time required to deploy and connect the portable generators.

- Procured and deployed portable generators / portable UPS. The size and number of portable generators/UPS is site specific. In addition to the above, the panels, receptacles, and cable/connectors to quickly deploy the portable generators to unit systems are also site specific.

[6] Some stations have provisions to rapidly deploy Portable UPS for key I&C followed by the AC portable generators.

- Confirmed through tests/drills that there is sufficient time to invoke the load shedding strategy and extend the battery life prior to deploying, connecting and operating the portable generators / portable UPS.

- Developed a number operational and maintenance guidelines for deployment of EME equipment.

- Developed and provided training for the Emergency Response Team personnel.

- Built EME storage facility near the station to house the EME equipment.

- The lessons learned during the tests/drills were used to improve the emergency response time during a total loss of AC power event

4. Conclusion

Fukushima Task Force review acknowledged that the probability of a BDBA at a Canadian NPPs is very low. However CNSC staff identified a number of areas where additional improvements would further enhance safety. Implementation of the action items has strengthened the defence in depth. Among other improvements this has resulted in an enhancement of the already robust CANDU EPS. It has been confirmed that the essential safety functions such as control, cool, contain and monitor will be maintained during a SBO event. Fukushima events led the CNSC to consider enhancing the regulatory requirements of EPS which includes requirement for Alternate AC power. This requirement has been included in the draft REGDOC 2.5.2 which will replace the current regulatory document RD-337.

References:

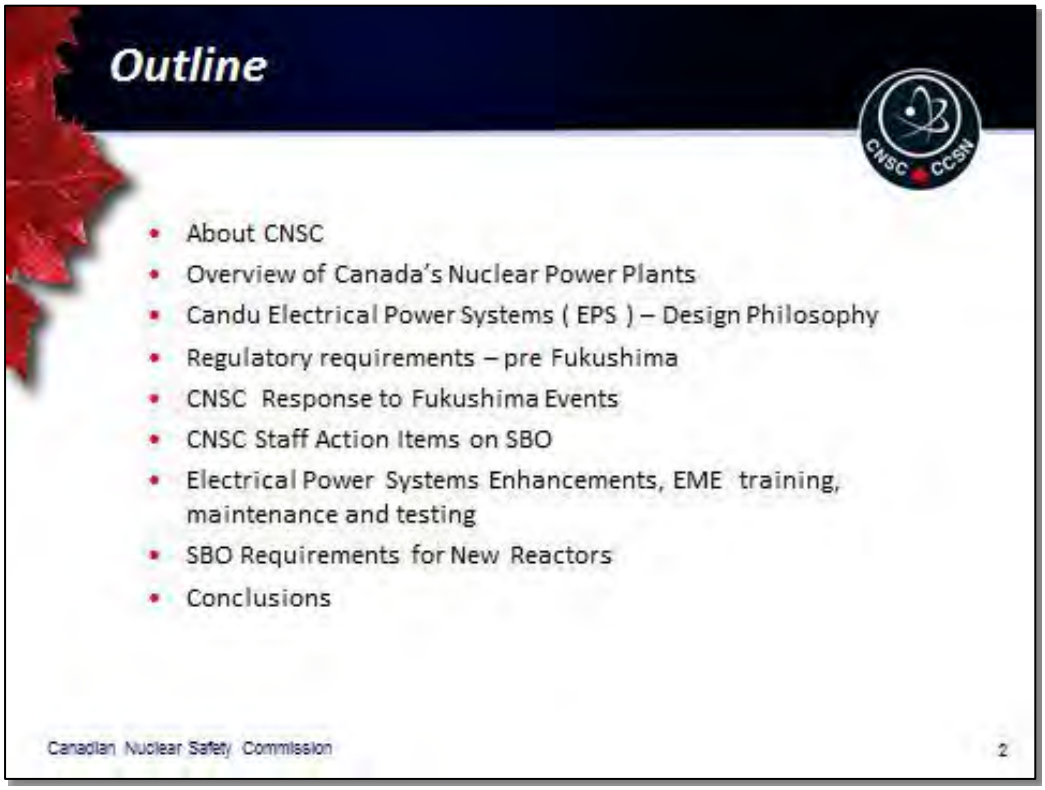
1. CNSC Fukushima Task Force Report, CNSC INFO-0824
http://nuclearsafety.gc.ca/pubs_catalogue/uploads/October-2011-CNSC-Fukushima-Task-Force-Report_e.pdf

2. CNSC Management Response to the CNSC Fukushima Task Force Recommendations, INFO-0825,
http://nuclearsafety.gc.ca/pubs_catalogue/uploads/October-2011-Management-Response-to-Fukushima-Task-Force-Report_e.pdf

3. CNSC Staff Action Plan on the CNSC Fukushima Task Force Recommendations INFO-0828
http://www.nuclearsafety.gc.ca/pubs_catalogue/uploads_unapproved/REVISED-INFO-0828-Draft-CNSC-Staff-Action-Plan-on-Fukushima-Dec-2011_e.PDF

4. CNSC REGDOC-2.5.2 Physical design: design of reactor Facilities: Nuclear Power plants (Draft) [Former RD-337]
http://nuclearsafety.gc.ca/pubs_catalogue/uploads/RD-337_e.pdf

Canadian Nuclear Safety Commission
Commission canadienne de sûreté nucléaire
CNSC CCSN
Emergency Mitigating Equipments
Post Fukushima actions at Canadian NPPs
Portable AC power sources
Presented by Jasmina Vucetic
nuclearsafety.gc.ca
Presented at the CSNI International Workshop on Robustness of Electrical Systems of NPPs in Light of the Fukushima Dai-Ichi Accident
Paris, April 1-3, 2014
Canada


Outline
CNSC CCSN
About CNSC
Overview of Canada's Nuclear Power Plants
Candu Electrical Power Systems (EPS) – Design Philosophy
Regulatory requirements – pre Fukushima
CNSC Response to Fukushima Events
CNSC Staff Action Items on SBO
Electrical Power Systems Enhancements, EME training, maintenance and testing
SBO Requirements for New Reactors
Conclusions
Canadian Nuclear Safety Commission
2

Canadian Nuclear Safety Commission

- Established May 2000, under *Nuclear Safety and Control Act (NSCA)*
- Replaced Atomic Energy Control Board, established in 1946 under *Atomic Energy Control Act*

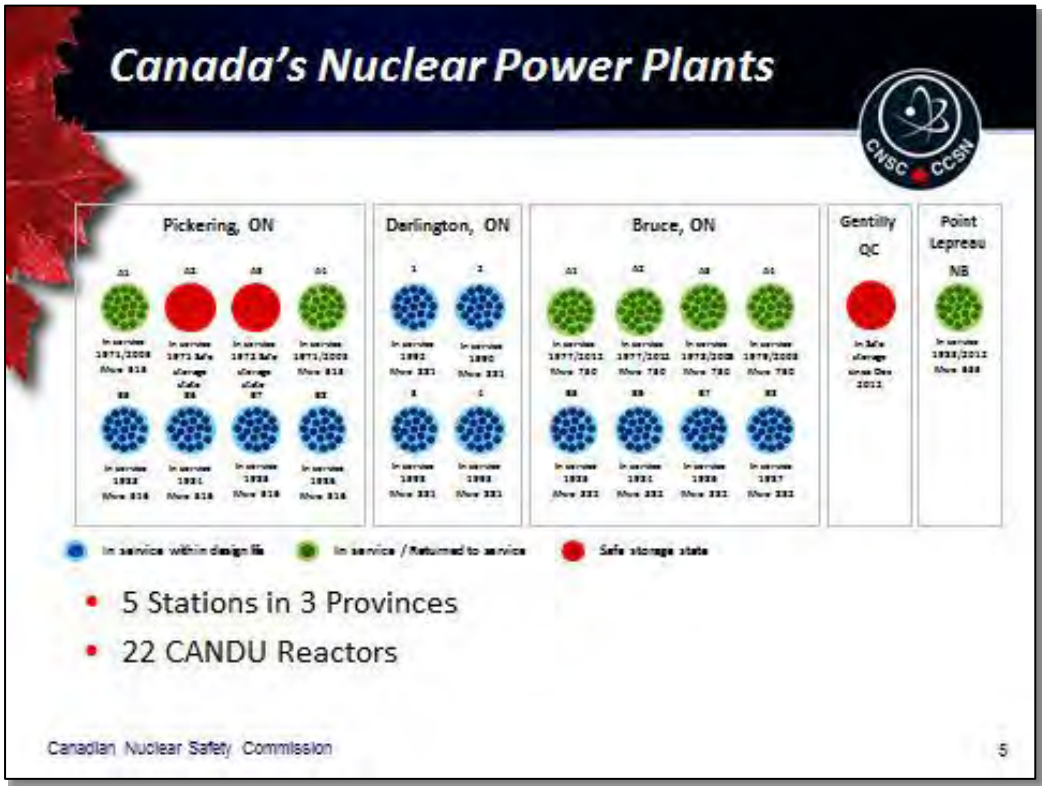
Canada's independent nuclear regulator
Over 67 years of experience

Canadian Nuclear Safety Commission 3

CNSC Mission

- To regulate the use of nuclear energy and materials so that the health, safety and security of Canadians and the environment are protected, and to implement Canada's international commitments on the peaceful use of nuclear energy.
- Canada's nuclear watchdog

Canadian Nuclear Safety Commission 4


Canada's Nuclear Power Plants
Pickering, ON
Darlington, ON
Bruce, ON
Gentilly QC
Point Lepreau NB
In service within design life
In service / Returned to service
Safe storage state
5 Stations in 3 Provinces
22 CANDU Reactors
Canadian Nuclear Safety Commission
5

Pickering Nuclear Power Plant
Multi-unit station (8 units)
Six units in operation
Units #2 and #3 under safe storage
Canadian Nuclear Safety Commission
6

Bruce Nuclear Power Plant

- Multi-unit station
- 2X4 Units (Bruce – A and Bruce – B)
- All units in operation

Canadian Nuclear Safety Commission 7

Darlington Nuclear Power Plant

- Multi-unit station (4 units)
- All units in operation

Canadian Nuclear Safety Commission 8

Pt. Lepreau Nuclear Power Plant

- Single unit station
- Unit in operation following refurbishment

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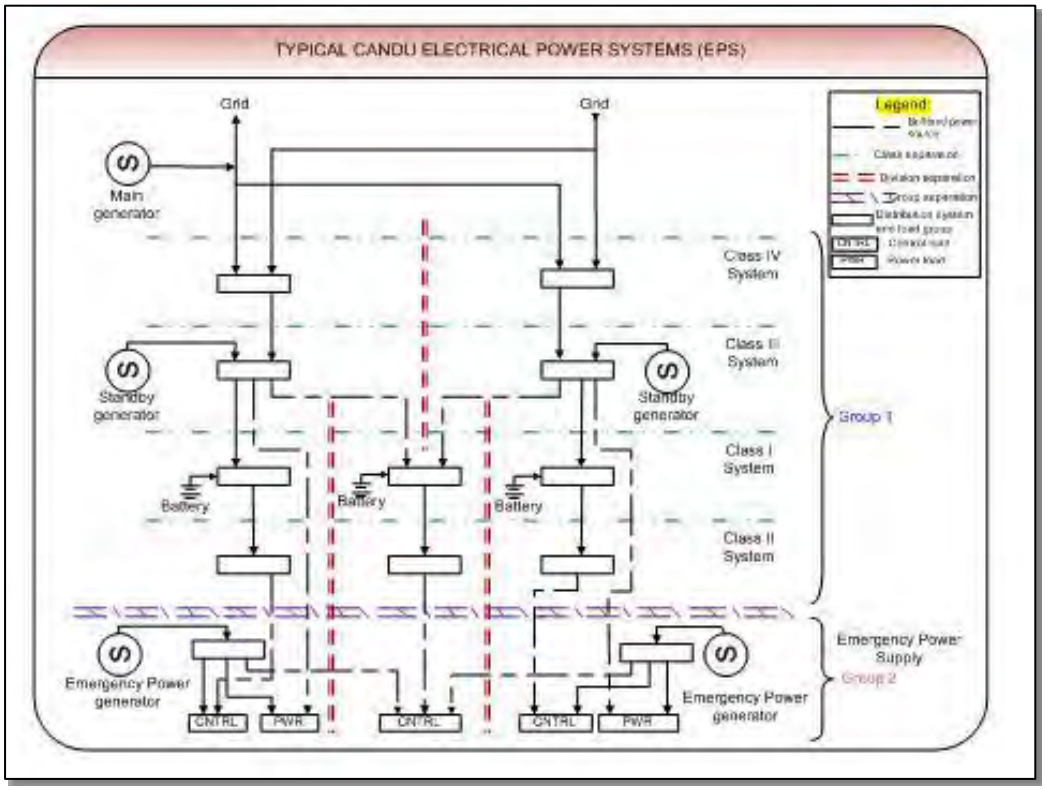

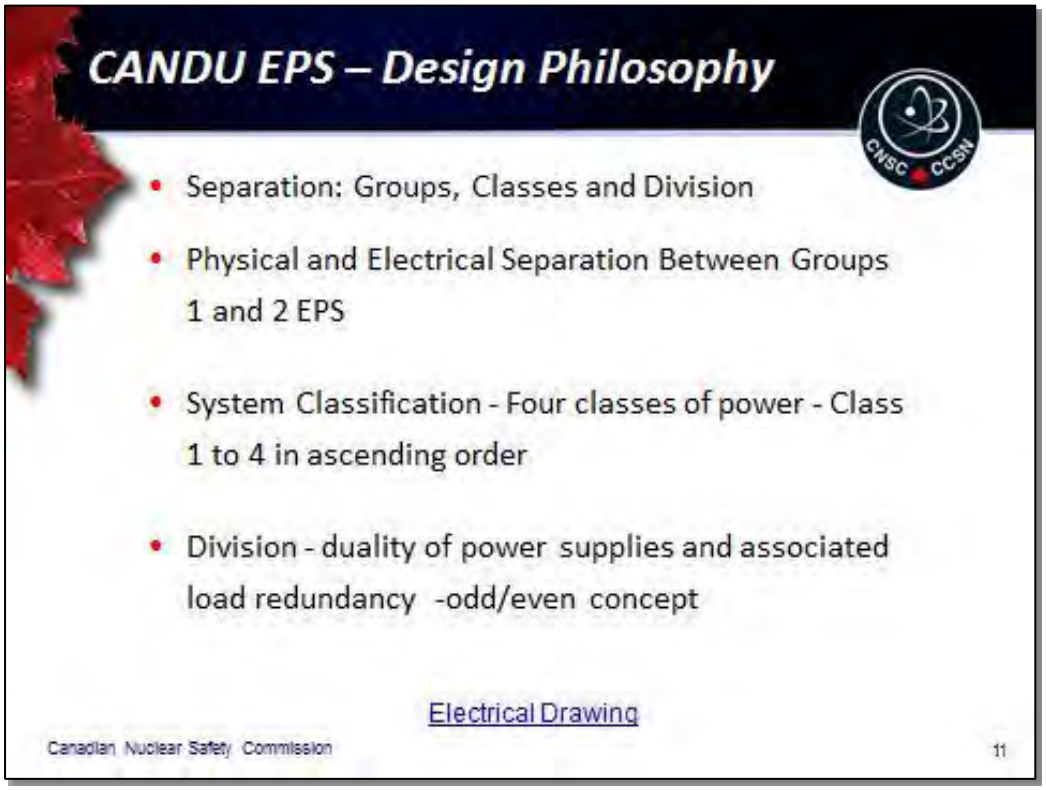
CANDU EPS – Design Philosophy
CNSC CCSN
• Separation: Groups, Classes and Division
• Physical and Electrical Separation Between Groups 1 and 2 EPS
• System Classification - Four classes of power - Class 1 to 4 in ascending order
• Division - duality of power supplies and associated load redundancy -odd/even concept
Electrical Drawing
Canadian Nuclear Safety Commission
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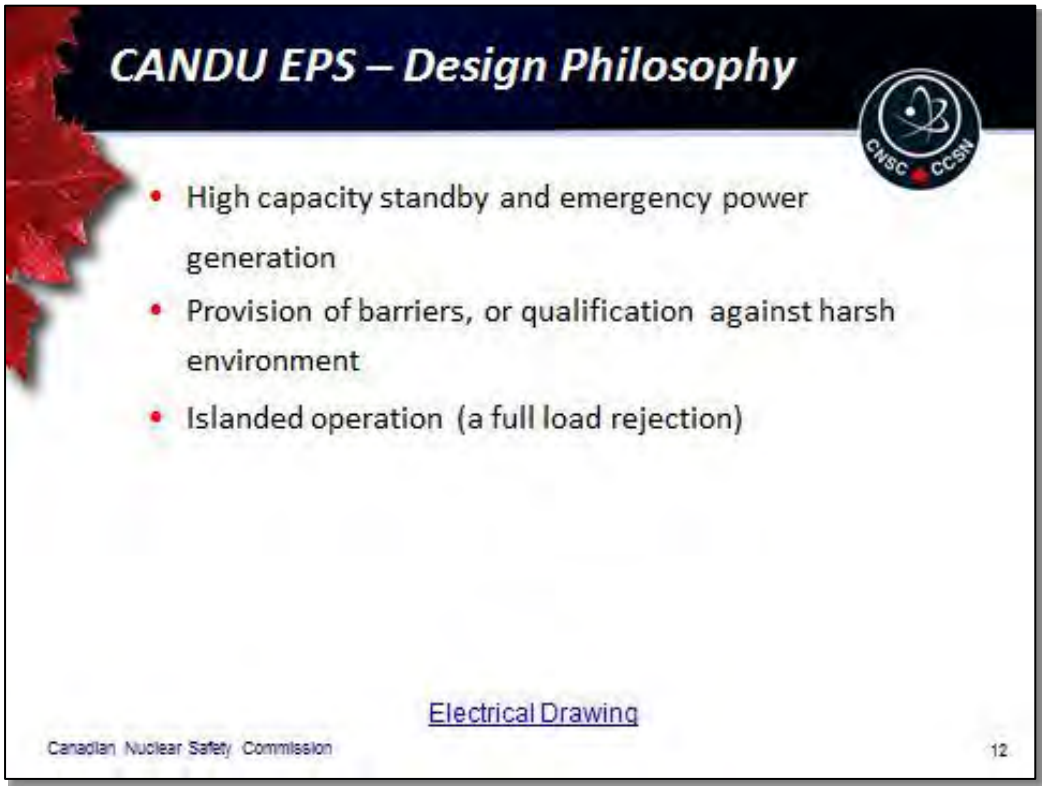
CANDU EPS – Design Philosophy
CNSC CCSN
• High capacity standby and emergency power generation
• Provision of barriers, or qualification against harsh environment
• Islanded operation (a full load rejection)
Electrical Drawing
Canadian Nuclear Safety Commission
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CANDU EPS – Classes of power

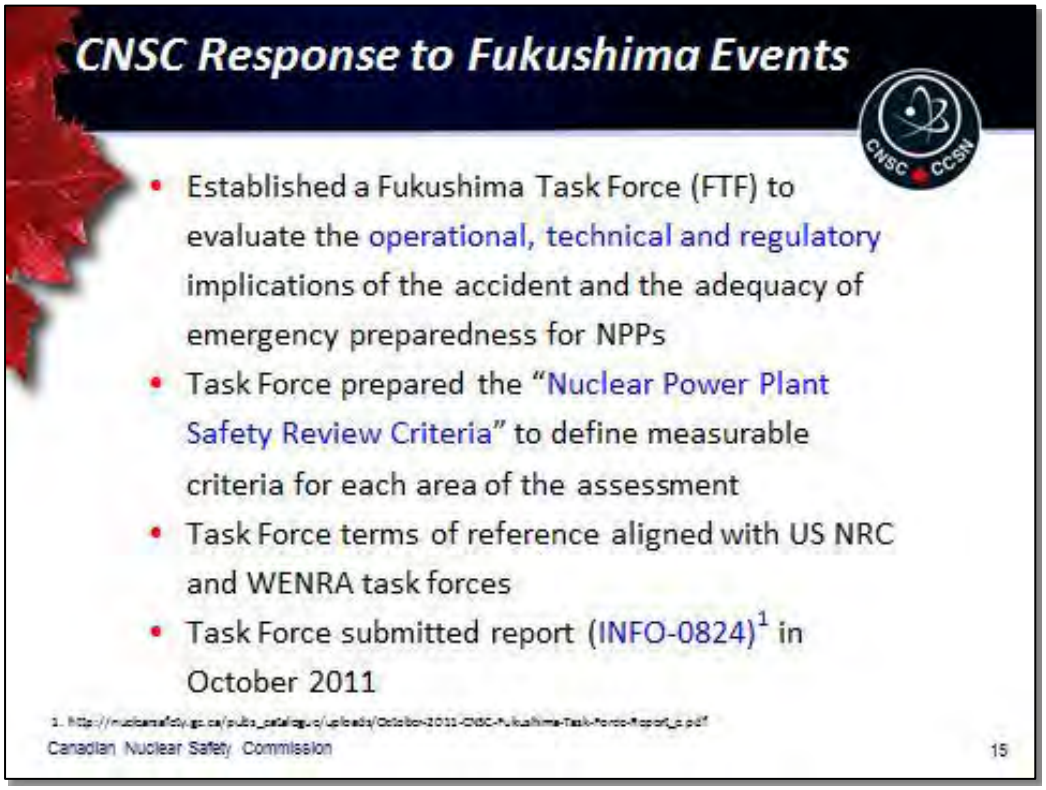
RELIABILITY

Class IV	•Normal Alternating Current (AC) –Supplied from turbine generator and/or grid
Class III	•AC system –Normally supplied from Class IV –Upon loss of Class IV, from Standby generator
Class II	•AC System –Normally supplied form Class I DC system –Can be supplied from Class III
Class I	•DC system –Normally supplied from Class III via rectifiers –Backup battery supply

Electrical Drawing

Regulatory requirements - Pre Fukushima

- Loss of Class IV - analyzed in Safety Reports as part of Design Basis Accidents (DBA)
- Station Blackout (SBO) was considered a Beyond Design accident (BDBA) with an extremely low probability of occurrence
- No explicit requirements for SBO response
- Some stations analyzed loss of Class IV combined with loss of Class III power. No failure predicted
- The lessons learnt from Fukushima required licensees to analyze station capability to respond to SBO.


CNSC Response to Fukushima Events
CNSC CCSN
• Established a Fukushima Task Force (FTF) to evaluate the operational, technical and regulatory implications of the accident and the adequacy of emergency preparedness for NPPs
• Task Force prepared the "Nuclear Power Plant Safety Review Criteria" to define measurable criteria for each area of the assessment
• Task Force terms of reference aligned with US NRC and WENRA task forces
• Task Force submitted report (INFO-0824)[1] in October 2011
1. http://nuclearsafety.gc.ca/pubs_catalog/uploads/October-2011-CNSC-Fukushima-Task-Force-Report_e.pdf
Canadian Nuclear Safety Commission
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CNSC Response to Fukushima Events
CNSC CCSN
• CNSC management committed to implement FTF recommendations in a timely and transparent manner (INFO-0825)[1]
• CNSC established an External Advisory Committee (EAC)[2], to provide an independent assessment of response to the Fukushima Daiichi incident
• The CNSC Integrated Action Plan[3] was based on the findings and recommendations of the FTF Report and EAC as well as comments from public and stakeholders
• Specific CNSC Staff Action Plan developed (INFO-0828)[4]
1. CNSC Management Response to Fukushima Task Force Report
2. External Advisory Committee Final Report
3. Fukushima Task Force CNSC Integrated Action Plan
4. CNSC Staff Action Plan on Fukushima
Canadian Nuclear Safety Commission
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CNSC Regulatory Documents

- INFO-0824, INFO-0828 & REGDOC-2.5.2

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Fukushima Action Items on SBO

- FAI – 1.10

  Licensees should investigate means of extending the availability of power for key instrumentation and control (I&C) needed in accident management actions following a loss of all AC power

  – Deliverables:

  1. An evaluation of the requirements and capabilities for electrical power for key I&C. The evaluation should identify practicable upgrades that would extend the availability of key I&C, if needed.
  2. A plan and schedule for deployment of identified upgrades. A target of eight hours without the need for offsite support should be used.

- Current status: closed for all NPPs in 2012

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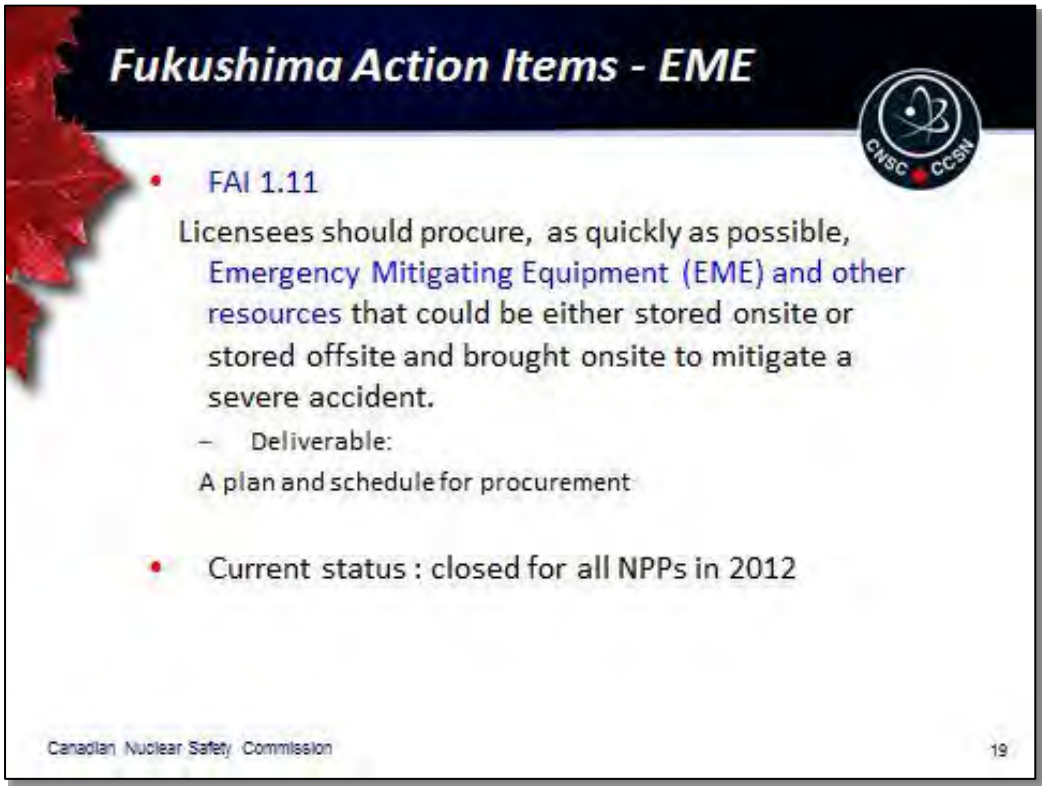
Fukushima Action Items - EME
FAI 1.11
Licensees should procure, as quickly as possible, Emergency Mitigating Equipment (EME) and other resources that could be either stored onsite or stored offsite and brought onsite to mitigate a severe accident.
– Deliverable:
A plan and schedule for procurement
Current status : closed for all NPPs in 2012
Canadian Nuclear Safety Commission
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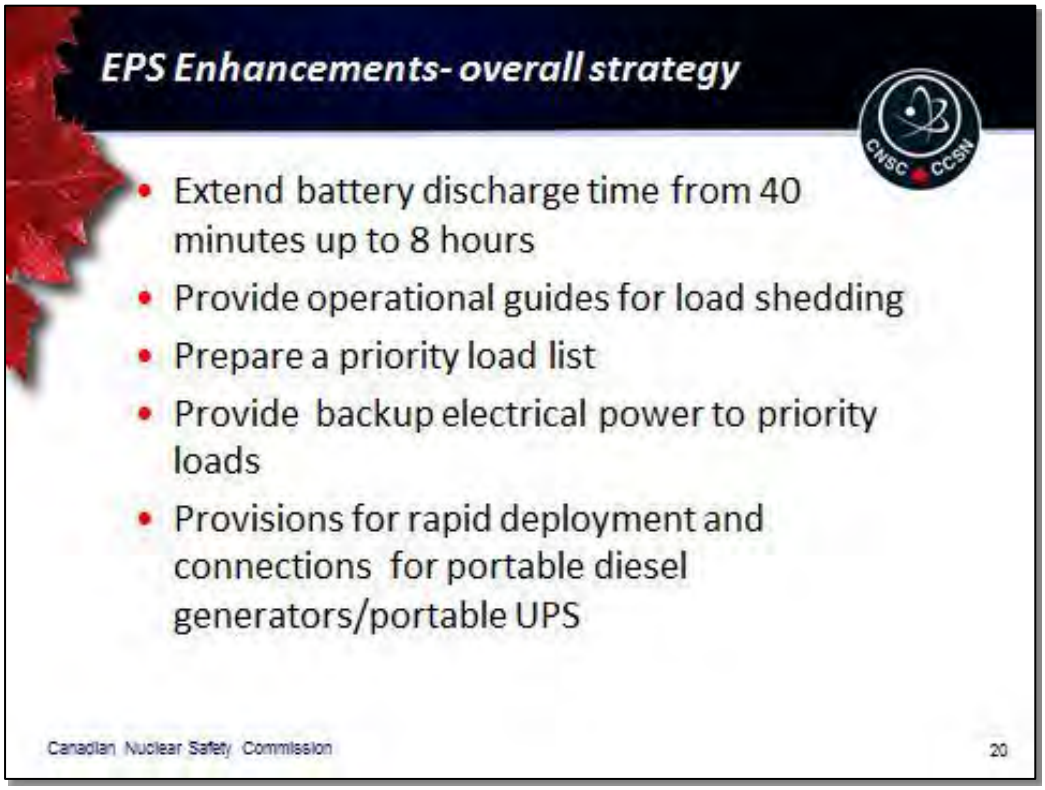
EPS Enhancements- overall strategy
Extend battery discharge time from 40 minutes up to 8 hours
Provide operational guides for load shedding
Prepare a priority load list
Provide backup electrical power to priority loads
Provisions for rapid deployment and connections for portable diesel generators/portable UPS
Canadian Nuclear Safety Commission
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EME Training

- A training needs analysis conducted
- Station-specific Emergency Mitigating Equipment Guides (EMEGs) produced
- The EME training delivered to all ERT personnel who will be involved in emergency response
- Drills conducted to test the response effectiveness
- Focus on improving the capability to respond in time during the SBO event

New regulatory requirements

- REGDOC 2.5.2 Design of Reactor Facilities: Nuclear Power Plants supersede RD-337-2008
- Electrical Power System requirements has been expanded in REGDOC 2.5.2
- Provision for mitigating the complete loss of on-site and off-site AC power are required
  - onsite portable, transportable or fixed power sources, or/ and
  - offsite portable or transportable power sources

Emergency Mitigating Equipment

100 kW Generators

Canadian Nuclear Safety Commission 23

Emergency Mitigating Equipment

Mobile Operations Centre

Canadian Nuclear Safety Commission 24

Emergency Mitigating Equipment

Panel for 400kW AC generators

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Emergency Mitigating Equipment

400 kW AC Generator

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Conclusions

- Defence in Depth has been strengthened for the already robust CANDU EPS
- Controlling, cooling, containing and monitoring functions will be maintained during a SBO scenario
- For future NPPs to be built in Canada, Alternate AC power to mitigate SBO will be a regulatory requirement. (REGDOC 2.5.2)

CNSC Video presentation on SBO

For those interested, this CNSC video shows the progression of an accident scenario involving a total station blackout at a Canadian nuclear power plant.

http://www.youtube.com/watch?v=vggzl9OngaM&feature=player_detailpage

Questions ???
CNSC CCSN
THANK YOU
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No. GSR Part 7

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PREPAREDNESS AND RESPONSE FOR A NUCLEAR OR RADIOLOGICAL EMERGENCY

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each in Arabic, Chinese, English, French, Russian and Spanish versions.
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INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2015

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Printed by the IAEA in Austria
November 2015
STI/PUB/1708
https://doi.org/10.61092/iaea.3dbe-055p

IAEA Library Cataloguing in Publication Data

Preparedness and response for a nuclear or radiological emergency : general safety requirements. — Vienna : International Atomic Energy Agency, 2015.
p. ; 24 cm. — (IAEA safety standards series, ISSN 1020–525X ; no. GSR part 7)
STI/PUB/1708
ISBN 978–92–0–105715–0
Includes bibliographical references.

1. Nuclear accidents — Management. 2. Ionizing radiation — Accidents. 3. Emergency management. I. International Atomic Energy Agency. II. Series.

IAEAL 15–00987

FOREWORD

by Yukiya Amano
Director General

The IAEA's Statute authorizes the Agency to "establish or adopt… standards of safety for protection of health and minimization of danger to life and property" — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

PREFACE

BACKGROUND

Organizations responsible for the management of emergencies (including conventional emergencies) recognize that good preparedness in advance of any emergency can substantially improve the emergency response. One of the most important elements of emergency preparedness is the coordination of arrangements among the different bodies involved to ensure clear lines of responsibility and authority.

The Convention on Early Notification of a Nuclear Accident ('Early Notification Convention') and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency ('Assistance Convention'), both adopted in 1986[1], place specific obligations in relation to a nuclear or radiological emergency on the States Parties to the Conventions and on the IAEA.

The practical implementation of the various articles of these Conventions, as well as the fulfilment of certain obligations under Article 16 of the Convention on Nuclear Safety[2] and Article 25 of the Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management[3], warrants the establishment of appropriate arrangements for emergency preparedness and response.

The Inter-Agency Committee on Radiological and Nuclear Emergencies (IACRNE) has been established as an inter-agency coordination mechanism to ensure that arrangements for emergency preparedness and response at the international level are consistent. IACRNE, which comprises relevant international intergovernmental organizations (hereafter, international organizations), maintains the Joint Radiation Emergency Management Plan of the International Organizations. IACRNE also contributes to the development of consistent international standards on emergency preparedness and response and to their practical application.

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Convention on Early Notification of a Nuclear Accident and Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, Legal Series No. 14, IAEA, Vienna (1987).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Convention on Nuclear Safety, Legal Series No. 16, IAEA, Vienna (1994).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Joint Convention on the Safety of Spent Fuel Management and the Safety of Radioactive Waste Management, IAEA International Law Series No. 1, IAEA, Vienna (2006).

In March 2002, the IAEA's Board of Governors established safety requirements for preparedness and response for a nuclear or radiological emergency as an IAEA safety standard. The Safety Requirements publication, Preparedness and Response for a Nuclear or Radiological Emergency (IAEA Safety Standards Series No. GS-R-2), was issued in November 2002 with joint sponsorship by seven international organizations: the Food and Agriculture Organization of the United Nations (FAO), the IAEA, the International Labour Organization (ILO), the OECD Nuclear Energy Agency (OECD/NEA), the Pan American Health Organization (PAHO), the United Nations Office for the Coordination of Humanitarian Affairs (OCHA) and the World Health Organization (WHO).

Since its publication in 2002, States have been using the IAEA Safety Requirements publication No. GS-R-2 in establishing and enhancing their arrangements for emergency preparedness and response. The 55th General Conference of the IAEA in 2011, in resolution GC(55)/RES/9, emphasized "the importance for all Member States to implement emergency preparedness and response mechanisms and develop mitigation measures at a national level, consistent with the Agency's safety standards, for improving emergency preparedness and response, facilitating communication in an emergency and contributing to harmonization of national criteria for protective [actions] and other actions".

The 56th General Conference of the IAEA in 2012, in resolution GC(56)/RES/9, requested the IAEA Secretariat, Member States and relevant international organizations "to address compatibility issues in the development of national and international emergency response mechanisms and procedures consistent with the Agency's safety standards".

The 59th General Conference of the IAEA in 2015, in resolution GC(59)/RES/9, emphasized "the importance of the establishment, implementation, regular exercise and continuous improvement of national emergency preparedness and response measures, taking into account the IAEA safety standards", and encouraged Member States "to strengthen their national, bilateral, regional and international emergency preparedness and response mechanisms, as appropriate, to facilitate timely information exchange during a nuclear emergency, and [to] improve bilateral, regional and international cooperation to that effect".

REVISION PROCESS FOR THE SAFETY REQUIREMENTS

In 2011, the IAEA Secretariat, relevant international organizations and Member States began the review of IAEA Safety Requirements publication No. GS-R-2 on the basis of lessons identified in exercises and from the response

to emergencies since its publication in 2002 (including the response to the accident at the Fukushima Daiichi nuclear power plant in Japan in March 2011), and in due consideration of recommendations of the International Commission on Radiological Protection (ICRP).

The revision of IAEA Safety Requirements publication No. GS-R-2 commenced with a series of drafting meetings organized on the basis of thematic areas, as well as a series of review meetings of IACRNE. Representatives of IAEA Member States and of relevant international organizations, including representatives of the Sponsoring Organizations, then considered the draft text at a technical meeting held in November 2012.

On the basis of recommendations from these meetings, a revised draft text was prepared and was submitted for a first review by the IAEA Safety Standards Committees[4] and Nuclear Security Guidance Committee in the first half of 2013. In July 2013, the draft text was submitted for comment to IAEA Member States and relevant international organizations. On the basis of comments received, a revised draft text was prepared and submitted for a second review by the Safety Standards Committees and the Nuclear Security Guidance Committee in the first half of 2014. The revised draft text was approved by the Safety Standards Committees and the Nuclear Security Guidance Committee in July 2014 and was endorsed by the Commission on Safety Standards in November 2014.

At its meeting on 3 March 2015, the Board of Governors of the IAEA established the draft Safety Requirements (in English) "as an Agency safety standard — in accordance with Article III.A.6 of the Statute[5]" and authorized the Director General "to promulgate these Safety Requirements and to issue them as a Safety Requirements publication in the IAEA Safety Standards Series".

The international organizations that had expressed an interest in becoming sponsoring organizations of the revision of IAEA Safety Requirements publication No. GS-R-2 and that had actively participated in the revision process were invited to sponsor the new Safety Requirements publication. By July 2015, 13 international organizations had responded positively to the invitation, having followed their respective processes for approval or confirmation.

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[4] Nuclear Safety Standards Committee, Radiation Safety Standards Committee, Transport Safety Standards Committee and Waste Safety Standards Committee.

[5] Statute of the International Atomic Energy Agency, IAEA, Vienna (1990).

The revision of IAEA Safety Requirements publication No. GS-R-2 is hereby issued in the IAEA Safety Standards Series as General Safety Requirements Part 7 under joint sponsorship by 13 international organizations: the FAO, the IAEA, the International Civil Aviation Organization (ICAO), the ILO, the International Maritime Organization (IMO), INTERPOL, the OECD/NEA, PAHO, the Preparatory Commission for the Comprehensive Nuclear-Test-Ban Treaty Organization (CTBTO), the United Nations Environment Programme (UNEP), OCHA, the WHO and the World Meteorological Organization (WMO).

The IAEA, on behalf of all the Sponsoring Organizations, wishes to express its great appreciation to all those who assisted in the preparation of this Safety Requirements publication and in the process of reaching a consensus.

APPLICATION OF THE SAFETY REQUIREMENTS

These safety requirements are binding on the IAEA Secretariat in relation to its own operations and on Member States in relation to operations assisted by the IAEA. They are recommended for use by Member States and by national authorities in relation to their own activities.

The Sponsoring Organizations are committed to continuously improving emergency preparedness and to coordinating their response to a nuclear or radiological emergency in line with these safety requirements, their respective mandates and, as appropriate, the Joint Radiation Emergency Management Plan of the International Organizations. These safety requirements are to be applied by the Sponsoring Organizations to their own operations in line with their respective mandates. States that are member states of the Sponsoring Organizations other than the IAEA may adopt these safety requirements, at their own discretion, or in accordance with their membership obligations, for application to their own activities.

All international organizations, irrespective of whether or not they are members of IACRNE, are encouraged to consider these safety requirements in establishing or enhancing their own emergency arrangements.

THE SPONSORING ORGANIZATIONS

The Food and Agriculture Organization of the United Nations (FAO)

In 1943, 44 governments committed to founding a permanent organization for food and agriculture at a meeting in Hot Springs, Virginia, USA. The formal foundation of the FAO of the United Nations took place at the first session of the

FAO Conference at Chateau Frontenac in Quebec, Canada, in 1945. The FAO works in partnership with the IAEA and other international organizations through the Joint FAO/IAEA Division of Nuclear Techniques in Food and Agriculture in preparing for and responding to nuclear or radiological emergencies in line with the Joint Radiation Emergency Management Plan of the International Organizations.

The FAO can provide assistance according to its Constitution[6] and as a Party to the Early Notification and Assistance Conventions and in line with its strategic objective to increase the resilience of livelihoods to threats and crises. It collects, analyses, interprets and disseminates information relating to nutrition, food and agriculture (including fisheries, marine products, and forestry and primary forestry products). The FAO also promotes and, where appropriate, recommends national and international action with regard to the improvement of the processing, marketing and distribution of food and agricultural products and the adoption of international policies with regard to arrangements concerning agricultural commodities.

The International Atomic Energy Agency (IAEA)

The IAEA was established in 1957. Its statutory objective is to seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world. One of the IAEA's statutory functions is "to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety… (including such standards for labour conditions), and to provide for the application of these standards".

Deriving from this function, in the area of preparedness and response for a nuclear or radiological emergency, the IAEA develops safety standards and technical tools, supports its Member States in strengthening their emergency arrangements, provides for capacity building in its Member States, and performs, at the request of Member States, peer reviews on established emergency arrangements (such as Emergency Preparedness Review missions).

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[6] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, Basic Texts of the Food and Agriculture Organization of the United Nations, Volumes I and II, 2013 Edition, FAO, Rome (2013).

The role of the IAEA in the response to a nuclear or radiological emergency derives, primarily, from the Early Notification Convention and the Assistance Convention. It includes notification and the exchange of official information, assessment of the potential consequences of an emergency and prognosis of its possible progression, the provision of assistance to Member States on request, and the provision of information to the public. The IAEA maintains its own emergency arrangements to fulfil its role in emergency response.

The IAEA also provides the secretariat of IACRNE, coordinates the inter-agency response in a nuclear or radiological emergency, and is the main coordinating body for the development and maintenance of the Joint Radiation Emergency Management Plan of the International Organizations.

The International Civil Aviation Organization (ICAO)

ICAO is an organization based on the Convention on International Civil Aviation[7], signed in 1944. It became a specialized agency of the United Nations in 1947, with a mission to serve as the global forum of States for international civil aviation. ICAO is a cosponsoring organization of the Joint Radiation Emergency Management Plan of the International Organizations. The activities of ICAO[8] include, for example, the reception and dissemination to aircraft in flight of information concerning releases of radioactive material into the atmosphere and provision of the regulatory framework for the safe transport of radioactive material as cargo on aircraft. ICAO also works with international partners to facilitate coordinated and consistent preparedness and response in the event of a nuclear or radiological emergency that may have an impact on international civil aviation.

The International Labour Organization (ILO)

The ILO was established in 1919 by the Treaty of Versailles to bring governments, employers and trades unions together for united action in the cause of social justice and better living conditions everywhere. It is a tripartite organization, with representatives of workers and employers taking part in its

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[7] Convention on International Civil Aviation, Chicago (1944).

[8] Convention on International Civil Aviation, Chicago (1944), Annex 3 — Meteorological Service for International Air Navigation, Annex 11 — Air Traffic Services, and Annex 15 — Aeronautical Information Services. Additionally, provisions related to the transport of radioactive and nuclear material are contained in Annex 18 — The Safe Transport of Dangerous Goods by Air and the associated Technical Instructions for the Safe Transport of Dangerous Goods by Air.

work on equal status with representatives of governments. The ILO became the first specialized agency of the United Nations in 1946. One of the main features of the ILO, in addition to its tripartite structure, is its standard-setting activity. Some 60 international conventions and recommendations concern the protection of workers against occupational hazards.

Radiation protection is part of the ILO's action on the protection of workers against sickness, disease and injury arising out of employment, as mandated by the Constitution of the International Labour Organization[9]. In 1949, the ILO published a set of practical international standards on radiation protection which were revised and considerably extended in 1957 and were incorporated into the ILO Manual of Industrial Radiation Protection. In June 1960, the International Labour Conference adopted the Radiation Protection Convention, 1960 (No. 115), and its accompanying Recommendation (No. 114).

The Convention applies to all activities involving the exposure of workers to ionizing radiations in the course of their work and provides that each Member of the ILO which ratifies it shall give effect to its provisions by means of laws or regulations, codes of practice or other appropriate means. In 1986, the ILO Governing Body approved the publication of a Code of Practice for the radiation protection of workers (ionizing radiations) which gives practical guidance on the implementation of a radiation protection programme at the enterprise level.

The International Maritime Organization (IMO)

The IMO, established in 1948, works in partnership with the IAEA and other international organizations through its Marine Environment and Maritime Safety Divisions in preparing for, and responding to, nuclear or radiological emergencies in line with the Joint Radiation Emergency Management Plan of the International Organizations. The IMO has general responsibilities in relation to emergency preparedness, response and cooperation in accordance with the OPRC Convention[10] and the OPRC-HNS Protocol[11].

The OPRC-HNS Protocol, in particular, by its implicit definition of hazardous and noxious substances, would also normally extend to incidents of marine pollution involving nuclear or radioactive material occurring at sea or in

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[9] INTERNATIONAL LABOUR OFFICE, Constitution of the International Labour Organization, Geneva (1919), as amended up to 1974.

[10] International Convention on Oil Pollution Preparedness, Response and Co-operation (1990), as amended.

[11] Protocol on Preparedness, Response and Co-operation to Pollution Incidents by Hazardous and Noxious Substances (2000).

port. In addition, the IMO has developed safety codes, standards and guidelines for the transport of radioactive material as cargo and for nuclear powered ships for the prevention of such incidents, which, in certain cases, also cover elements of preparedness (see chapters VII and VIII of the SOLAS Convention)[12].

INTERPOL

INTERPOL, founded in 1914, is the world's largest international police organization. It facilitates cross-border police cooperation, and supports and assists all organizations, authorities and services whose mission it is to prevent or combat international crime. INTERPOL provides law enforcement officials in the field with support in an emergency and in operational activities, especially in its priority crime areas of fugitives, public safety and terrorism, drugs and organized crime, trafficking in human beings and financial and high technology crime.

When necessary, INTERPOL can deploy an Incident Response Team to support a country or countries in tasks as requested. The Command and Coordination Centre operates 24 hours a day and works in INTERPOL's four official languages, namely English, French, Spanish and Arabic, and serves as the first point of contact for any member country faced with a crisis. The Command and Coordination Centre can also assume a coordination role if an attack or disaster involves several member countries or if a member country's own ability to assume such a role has been compromised.

The OECD Nuclear Energy Agency (OECD/NEA)

The OECD/NEA was established in 1958. Its mission is to assist its member countries in maintaining and further developing, through international cooperation, the scientific, technological and legal bases required for a safe, environmentally benign and economical use of nuclear energy for peaceful purposes. Emergency preparedness and emergency management have long been main areas of focus for the OECD/NEA's Committee on Radiation Protection and Public Health. Following the Three Mile Island accident in 1979, and in particular following the Chernobyl accident in 1986, the OECD/NEA has worked to share national experience in emergency preparedness and emergency management, and to identify and address emerging issues. The OECD/NEA's International Nuclear Emergency Exercise series, INEX, has been an important mechanism for this work.

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[12] International Convention for the Safety of Life at Sea (1974), as amended.

While the OECD/NEA member countries have assigned no statutory functions to the OECD/NEA for preparedness and response to a nuclear or radiological emergency, its work in this area has provided numerous useful resources to its members. The OECD/NEA is a founding and active member of IACRNE, and a cosponsoring organization of the Joint Radiation Emergency Management Plan of the International Organizations.

The Pan American Health Organization (PAHO)

PAHO, founded in 1902, is the specialized health agency of the Inter-American System, serves as the Regional Office for the Americas of the World Health Organization (WHO), and is a member of the United Nations system. According to the PAHO constitution[13], its governing bodies set the organization's mandates. PAHO has technical expertise at its headquarters and at a number of country offices and scientific centres.

Among other functions, PAHO provides technical cooperation support in alerts and response to epidemics, in disaster preparedness, in health systems and services, in environmental health, in health legislation, in access to medicines and technologies, and in regulatory capacity. Several resolutions have been approved in relation to emergencies, including the formulation of regional plans of action for various types of disaster, and in relation to radiation safety standards that address nuclear and radiological emergencies. PAHO is a Party to the International Health Regulations[14], the legally binding instrument that establishes the global health security framework, covering biological, chemical and radiation related hazards.

The Preparatory Commission for the Comprehensive Nuclear-Test-Ban Treaty Organization (CTBTO)

The Comprehensive Nuclear-Test-Ban Treaty[15] (CTBT) bans nuclear explosions by all States Parties to the Treaty at any place under their jurisdiction or control, in the atmosphere, underwater and underground. In preparation for the entry into force of the Treaty, the CTBTO was tasked in 1996 with establishing the International Monitoring System (IMS) for monitoring for signs of a nuclear

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[13] PAN AMERICAN HEALTH ORGANIZATION, Constitution of the Pan American Health Organization, PAHO, Buenos Aires (1947) as amended up to 1999.

[14] WORLD HEALTH ORGANIZATION, International Health Regulations, WHO, Geneva (2005).

[15] Comprehensive Nuclear-Test-Ban Treaty, United Nations, New York (1996).

weapon test explosion or any other nuclear explosion. The IMS includes a number of globally distributed radionuclide stations that continuously, and in near real time, report spectral measurements with high sensitivity for a large set of fission and activation products.

The response task for the CTBTO in a nuclear or radiological emergency is to provide monitoring data on radionuclides and noble gases in real time, including confirmation of non-detection. Advice on predictions in relation to atmospheric transport and dispersion may also be provided, as appropriate. Following an emergency, the CTBTO provides all relevant results on air concentrations of radionuclides from the global monitoring network and contributes relevant expertise.

The United Nations Environment Programme (UNEP)

UNEP[16] was established in 1972 with headquarters in Nairobi, Kenya, and with six regional offices and various country offices. UNEP, as the leading global environmental authority, sets the global environmental agenda, promotes the coherent implementation of the environmental dimension of sustainable development within the United Nations system and serves as an authoritative advocate for the global environment. UNEP's disasters and conflicts subprogramme seeks to minimize environmental threats to human well-being from the environmental causes and consequences of conflicts and disasters. In addition, the UNEP/OCHA Joint Environment Unit based in Geneva, Switzerland, mobilizes and coordinates the international emergency response to acute environmental risks caused by conflicts, natural disasters and industrial accidents.

Further, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), which was established by the General Assembly[17] in response to widespread concerns about the effects of radiation on human health and the environment, evaluates radiation doses, effects and risks worldwide. The UNSCEAR secretariat is based in Vienna and operates under the auspices of UNEP.

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[16] Institutional and Financial Arrangements for International Environmental Cooperation, United Nations General Assembly Resolution A/RES/27/2997, UN, New York (1972).

[17] United Nations General Assembly Resolution 913 (X), UN, New York (1955).

The United Nations Office for the Coordination of Humanitarian Affairs (OCHA)

OCHA, established in 1991 as an entity within the United Nations Secretariat, is responsible for bringing together humanitarian actors to ensure a coherent response to emergencies. The mandate of OCHA encompasses the coordination of humanitarian response, policy development and humanitarian advocacy. OCHA's activities include, for example, support in the planning of inter-agency response, mobilization and monitoring of humanitarian funding, reporting, analysis and information sharing, and deployment of rapid response teams for emergency relief.

OCHA also promotes preparedness to lessen the impact of disasters on vulnerable communities, especially in disaster prone countries. OCHA works with national governments, regional bodies and other agencies to develop, to test and to implement measures that help save lives in an emergency. OCHA also provides tools for contingency planning, hazard mapping and early warning reports, for example. With its international partners, OCHA works to ensure that the international community is ready to respond to humanitarian emergencies without delay and with the right assistance.

The World Health Organization (WHO)

The WHO, established in 1948, is the United Nations' specialized agency for health, and is the directing and coordinating authority on international health in the United Nations system. With 194 Member States, the WHO carries out its mandates through its headquarters in Geneva, six world regional offices and 150 country offices — and through close collaboration with national and international partners worldwide. Under its Constitution[18], the resolutions of its governing body, and as a Party to both the Early Notification Convention and the Assistance Convention, the WHO provides technical support to its member states in relation to public health and medical response to nuclear and radiological emergencies.

Under the International Health Regulations, which are legally binding upon States, the WHO also conducts global public health surveillance, receives and assesses notifications and reports from States as required under the International Health Regulations, supports member states, and coordinates the public health response to a broad range of health events and risks, including biological,

[18] Constitution of the World Health Organization adopted by the International Health Conference, New York (1946), as amended up to 2005.

chemical and radiation related hazards. It also supports countries in developing the necessary core public health capacities for these hazards.

The World Meteorological Organization (WMO)

The WMO is the authoritative organization in the United Nations system on the state and behaviour of the Earth's atmosphere, its interaction with the oceans, the climate it produces and the resulting distribution of water resources. The International Meteorological Organization, founded in 1873, was the predecessor of the WMO, which was formally created when the WMO Convention[19] came into force in 1950. The WMO became a specialized agency of the United Nations in 1951 with a mandate covering the areas of meteorology (weather and climate), operational hydrology and related geophysical sciences.

The WMO has an operational infrastructure comprising a number of World Meteorological Centres, Global Processing Centres for Long Range Forecasts, Regional Climate Centres and Regional Specialized Meteorological Centres. Some of these Regional Specialized Meteorological Centres support the IAEA through the provision of atmospheric transport modelling outputs.

The WMO promotes cooperation among its members in the creation and maintenance of meteorological, climatological, hydrological and geophysical observation networks, the exchange of data, and the processing and standardization of data, and in order to provide assistance, where necessary, with technology transfer, training and research. It also fosters collaboration between the national meteorological and hydrological services of its members.

The WMO fosters the application of meteorology in public weather services in order to better serve weather sensitive sectors — for example, agriculture, aviation, shipping, the environment and water resource management — and to mitigate the impacts of natural disasters. In addition, the WMO facilitates the free and unrestricted exchange of weather and climate related data and information products and services in real time or near real time on matters relating to public safety and security, economic welfare and environmental protection.

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[19] Convention of the World Meteorological Organization adopted by the Washington Conference, Washington, DC (1947), as amended up to 2007.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered 'overarching' requirements, are expressed as 'shall' statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

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[1] See also publications issued in the IAEA Nuclear Security Series.

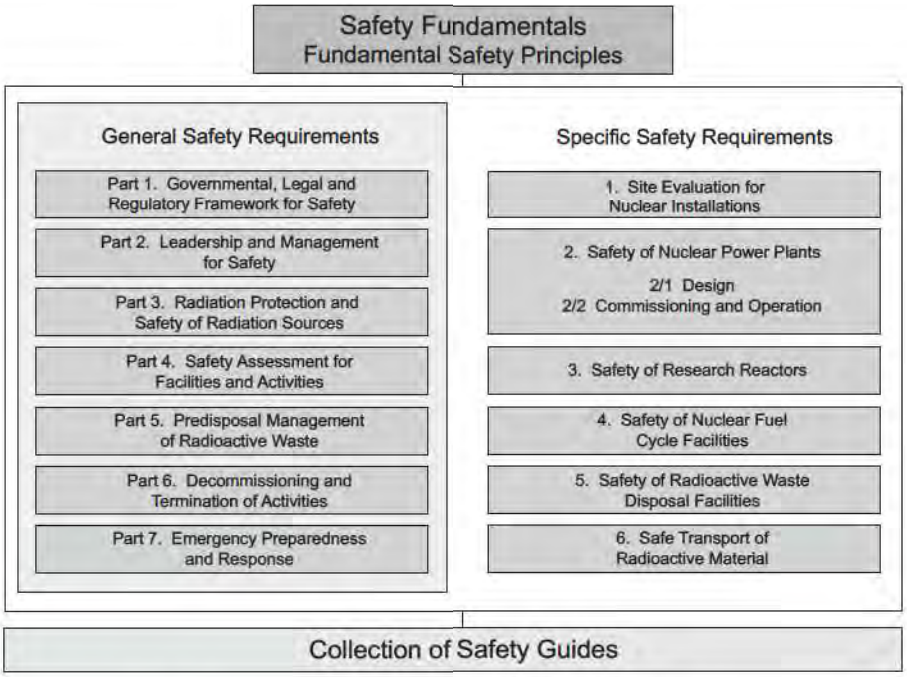

FIG. 1. The long term structure of the IAEA Safety Standards Series.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as 'should' statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five safety standards committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of

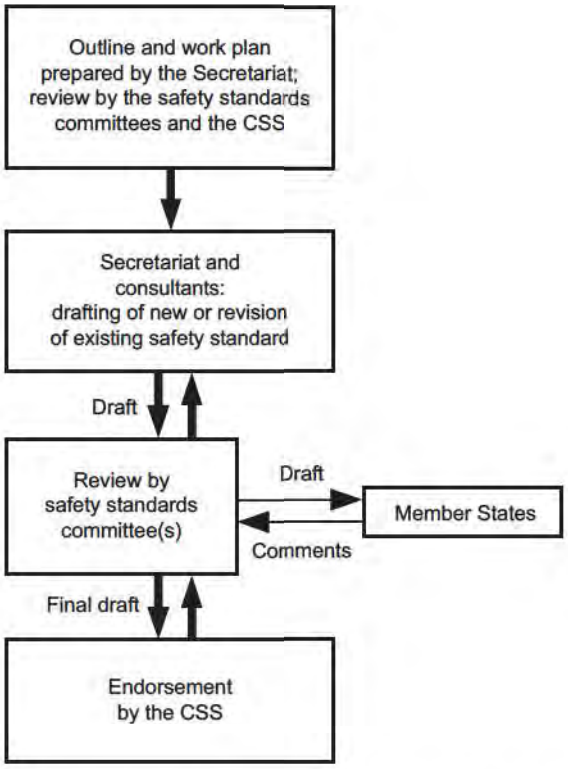

FIG. 2. The process for developing a new safety standard or revising an existing standard.

the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection

(ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see http://www-ns.iaea.org/standards/safety-glossary.htm). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

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1. INTRODUCTION

BACKGROUND

1.1. This IAEA Safety Requirements publication is governed by the fundamental safety objective and the fundamental safety principles established in the IAEA Safety Standards publication Fundamental Safety Principles (SF-1) [1]. In particular, this publication addresses Principle 9, which is concerned with the arrangements that must be made for preparedness and response for a nuclear or radiological emergency [1].

1.2. This publication also allows for consistency with Essential Element No. 11 of the IAEA Nuclear Security Fundamentals [2], which is concerned with the planning for, preparedness for and response to a nuclear security event. It therefore addresses the emergency arrangements that must be in place irrespective of the initiator of the emergency, which could be a natural event, a human error, a mechanical or other failure, or a nuclear security event.

1.3. In 2002, the IAEA published the Safety Requirements publication, Preparedness and Response for a Nuclear or Radiological Emergency (GS-R-2)[1], jointly sponsored by seven international organizations (the Food and Agriculture Organization of the United Nations (FAO), the IAEA, the International Labour Organization (ILO), the OECD Nuclear Energy Agency (OECD/NEA), the Pan American Health Organization (PAHO), the United Nations Office for the Coordination of Humanitarian Affairs (OCHA) and the World Health Organization (WHO)). This Safety Requirements publication is a revised edition of IAEA Safety Standards Series No. GS-R-2, updated to take into account developments and experience gained since 2002. In the revision process, due consideration has been given to — but was not limited to — experience gained from the response to the accident at the Fukushima Daiichi nuclear power plant and to recommendations of the International Commission on Radiological Protection (ICRP) [3]. The IAEA Safety Guides Criteria for Use in Preparedness

[1] FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS OFFICE FOR THE COORDINATION OF HUMANITARIAN AFFAIRS, WORLD HEALTH ORGANIZATION, Preparedness and Response for a Nuclear or Radiological Emergency, IAEA Safety Standards Series No. GS-R-2, IAEA, Vienna (2002).

and Response for a Nuclear or Radiological Emergency (GSG-2) [4] and Arrangements for Preparedness for a Nuclear or Radiological Emergency (GS-G-2.1) [5] elaborate on the requirements established in GS-R-2 and provide recommendations and guidance on their implementation. In addition, Planning and Preparing for Emergency Response to Transport Accidents Involving Radioactive Material (TS-G-1.2 (ST-3)) [6] provides guidance on planning and preparing for emergency response to transport accidents involving radioactive material.

1.4. This Safety Requirements publication addresses the requirements for preparedness and response for a nuclear or radiological emergency (including requirements for the transition to an existing exposure situation). Other Safety Requirements publications refer to and are consistent with these requirements in relation to emergency preparedness and response.

1.5. The response to a nuclear or radiological emergency may involve many national organizations (e.g. the operating organization and response organizations at the local, regional and national levels) as well as international organizations. The functions of many of these organizations may be the same for the response to a nuclear or radiological emergency as for the response to a conventional emergency. However, the response to a nuclear or radiological emergency might also involve specialized agencies and technical experts. Therefore, in order to be effective, the response to a nuclear or radiological emergency has to be well coordinated, and emergency arrangements have to be appropriately integrated with arrangements for the response to a conventional emergency and with the response measures for a nuclear security event.

1.6. Safety measures and security measures have in common the aim of protecting human life and health and protecting the environment. Paragraph 1.10 of Ref. [1] states that "Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security." This emphasizes the importance of effective coordination between safety measures and security measures in relation to the response to a nuclear or radiological emergency.

1.7. This publication also provides guidance for (1) preparedness and response for a nuclear or radiological emergency by the relevant international organizations and (2) the inter-agency coordination performed through the Inter-Agency Committee on Radiological and Nuclear Emergencies (IACRNE).

1.8. It is assumed that States applying these requirements have in place an infrastructure for the purpose of regulating the safety of facilities and activities that could pose radiation risks. This includes laws and regulations governing the safe operation of facilities and the safe conduct of activities, and an independent regulatory body with responsibilities for establishing and enforcing rules for safe operation and safe conduct. In this context, the IAEA has issued General Safety Requirements publications on the Governmental, Legal and Regulatory Framework for Safety (GSR Part 1) [7] and on Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards (GSR Part 3) [8].

1.9. In addition, it is assumed that States applying these requirements have in place an infrastructure for the purpose of regulating the nuclear security of nuclear material and other radioactive material, associated facilities and associated activities, as well as nuclear security measures for nuclear material and other radioactive material out of regulatory control. This also includes an independent regulatory body as well as other competent authorities with responsibilities for regulating nuclear security. In this context, IAEA Nuclear Security Series publications [9–11] provide recommendations.

OBJECTIVE

1.10. The present publication establishes the requirements for an adequate level of preparedness and response for a nuclear or radiological emergency. The application of these requirements is also intended to mitigate the consequences of a nuclear or radiological emergency if such an emergency arises despite all efforts made to prevent it.

1.11. The fulfilment of these requirements will contribute to the harmonization worldwide of arrangements for preparedness and response for a nuclear or radiological emergency.

1.12. These requirements are intended to be applied by the government at the national level by means of adopting legislation and establishing regulations, and by making other arrangements, including assigning responsibilities (e.g. to the operating organization or the operating personnel of a facility or an activity, local or national officials, response organizations or the regulatory body) and verifying their effective fulfilment.

1.13. The requirements are also intended for use by response organizations, operating organizations and the regulatory body in respect of preparedness and response for a nuclear or radiological emergency, as well as by authorities with responsibilities for emergency preparedness and response at the local and regional level and, as appropriate, by relevant international organizations at the international level.

SCOPE

1.14. The requirements apply for preparedness and response for a nuclear or radiological emergency in relation to all those facilities and activities, as well as sources, with the potential for causing radiation exposure, environmental contamination or concern on the part of the public warranting protective actions and other response actions.

1.15. The requirements also apply to preparedness and response for a nuclear or radiological emergency in relation to off-site jurisdictions that may need to take protective actions and other response actions.

1.16. The requirements apply for preparedness and response for a nuclear or radiological emergency irrespective of the initiator of the emergency, whether the emergency follows a natural event, a human error, a mechanical or other failure, or a nuclear security event[2]. The requirements do not cover preparedness for, or response measures that are specific to, nuclear security events, for which recommendations are provided in Refs [9–11]. Such response measures include activities for the identification, collection, packaging and transport of evidence contaminated with radionuclides, nuclear forensics and related actions in the context of investigation into the circumstances surrounding a nuclear security event. The requirements established here do provide for a coordinated and integrated approach to preparedness and response for a nuclear or radiological emergency arising from a nuclear security event that necessitates protective actions and other response actions to be taken for protection of members of the public, workers and emergency workers, helpers in an emergency and patients.

[2] A 'nuclear security event' is an event that has potential or actual implications for nuclear security that must be addressed. Such events include criminal or intentional unauthorized acts involving or directed at nuclear material, other radioactive material, associated facilities or associated activities. A nuclear security event, for example, sabotage of a nuclear facility or detonation of a radiological dispersal device, may give rise to a nuclear or radiological emergency.

STRUCTURE

1.17. This publication comprises six sections. Section 2 provides for the interpretation and entry into force of the requirements. Section 3 establishes the goals of emergency preparedness and response. Section 4 establishes the general requirements that are to be met before effective emergency arrangements can be made, defines by using a graded approach the emergency preparedness categories for which the requirements have been established and elaborates on the development of a protection strategy on the basis of the hazards assessed. Section 5 establishes the requirements to be met for performing the functions critical for an effective emergency response. Section 6 establishes requirements for the infrastructure necessary to develop and maintain adequate arrangements for preparedness. Guidance values for restricting exposure of emergency workers in a nuclear or radiological emergency are provided in Appendix I. Generic criteria for use in emergency preparedness and response are provided in Appendix II. Annex I presents the applicability of paragraphs in the text for each emergency preparedness category.

2. INTERPRETATION, RESOLUTION OF CONFLICTS AND ENTRY INTO FORCE

DEFINITIONS

2.1. Terms used in this publication have the meanings given under 'Definitions' on page 79. If not otherwise defined under Definitions, terms are used as defined in the IAEA Safety Glossary, 2007 Edition [12].

INTERPRETATION

2.2. Except as specifically authorized by the statutory governing body of a Sponsoring Organization, no interpretation of this standard by any officer or employee of the Sponsoring Organization other than a written interpretation by the Director General of the Sponsoring Organization shall be binding on the Sponsoring Organization.

RESOLUTION OF CONFLICTS

2.3. The requirements of this standard are established in addition to and not in place of other applicable requirements, such as those of relevant binding conventions and national laws and regulations.

2.4. In cases of conflict between the requirements of this standard and other applicable requirements, the government or the regulatory body, as appropriate, shall determine which requirements are to be enforced.

2.5. Nothing in this standard shall be construed as restricting any actions that may otherwise be necessary for protection and safety or as relieving the parties referred to in this standard from complying with applicable laws and regulations.

ENTRY INTO FORCE

2.6. The Secretariat envisages that, for the IAEA's own operations and for those operations assisted by the IAEA, arrangements will be made to meet these requirements within a period of no more than one year from the date of publication of this standard.

2.7. This standard shall come into force within a period of no more than one year from the date of publication of this standard for all the Sponsoring Organizations in accordance with their respective mandates.

2.8. If a State decides to adopt this standard, this standard shall come into force at the time indicated in the formal adoption by that State, and preferably within a period of no more than one year from the date of its publication.

3. GOALS OF EMERGENCY PREPAREDNESS AND RESPONSE

GOAL OF EMERGENCY PREPAREDNESS

3.1. The goal of emergency preparedness is to ensure that an adequate capability is in place within the operating organization and at local, regional and national levels and, where appropriate, at the international level, for an effective

response in a nuclear or radiological emergency. This capability relates to an integrated set of infrastructural elements that include, but are not limited to: authority and responsibilities; organization and staffing; coordination; plans and procedures; tools, equipment and facilities; training, drills and exercises; and a management system.

GOALS OF EMERGENCY RESPONSE

3.2. In a nuclear or radiological emergency, the goals of emergency response are:

(a) To regain control of the situation and to mitigate consequences;
(b) To save lives;
(c) To avoid or to minimize severe deterministic effects;
(d) To render first aid, to provide critical medical treatment and to manage the treatment of radiation injuries;
(e) To reduce the risk of stochastic effects;
(f) To keep the public informed and to maintain public trust;
(g) To mitigate, to the extent practicable, non-radiological consequences;
(h) To protect, to the extent practicable, property and the environment;
(i) To prepare, to the extent practicable, for the resumption of normal social and economic activity.

4. GENERAL REQUIREMENTS

Requirement 1: The emergency management system

The government shall ensure that an integrated and coordinated emergency management system for preparedness and response for a nuclear or radiological emergency is established and maintained.

4.1. The government shall ensure that an emergency management system is established and maintained on the territories of and within the jurisdiction of the State for the purposes of emergency response to protect human life, health, property and the environment in the event of a nuclear or radiological emergency.

4.2. The emergency management system shall be designed to be commensurate with the results of the hazard assessment (see paras 4.18–4.26) and shall enable an effective emergency response to reasonably foreseeable events (including very low probability events).

4.3. The emergency management system shall be integrated, to the extent practicable, into an all-hazards emergency management system (see paras 5.6 and 5.7).

4.4. The government shall ensure the coordination of and consistency of national emergency arrangements with the relevant international emergency arrangements[3].

Requirement 2: Roles and responsibilities in emergency preparedness and response

The government shall make provisions to ensure that roles and responsibilities for preparedness and response for a nuclear or radiological emergency are clearly specified and clearly assigned.

General

4.5. The government shall make adequate preparations to anticipate, prepare for, respond to and recover from a nuclear or radiological emergency at the operating organization, local, regional and national levels, and also, as appropriate, at the international level. These preparations shall include adopting legislation and establishing regulations for effectively governing the preparedness and response for a nuclear or radiological emergency at all levels (see para. 1.12).

4.6. The government shall ensure that arrangements are in place for effectively governing the provision of prompt and adequate compensation of victims for damage due to a nuclear or radiological emergency.

[3] Arrangements set under the Assistance Convention and under the Early Notification Convention [13] are examples of international emergency arrangements that are relevant for States Parties to these Conventions.

4.7. The government shall ensure that all roles and responsibilities for preparedness and response for a nuclear or radiological emergency are clearly allocated in advance among operating organizations, the regulatory body and response organizations[4].

4.8. The government shall ensure that response organizations, operating organizations and the regulatory body have the necessary human, financial and other resources, in view of their expected roles and responsibilities and the assessed hazards, to prepare for and to deal with both radiological and non-radiological consequences of a nuclear or radiological emergency, whether the emergency occurs within or beyond national borders.

4.9. The government shall ensure that operating organizations, response organizations and the regulatory body establish, maintain and demonstrate leadership in relation to preparedness and response for a nuclear or radiological emergency [14].

Coordinating mechanism

4.10. The government shall establish a national coordinating mechanism[5] to be functional at the preparedness stage, consistent with its emergency management system, with the following functions:

(a) To ensure that roles and responsibilities are clearly specified and are understood by operating organizations, response organizations and the regulatory body (see para. 4.7);
(b) To coordinate the hazard assessment within the State (see paras 4.18–4.26) and periodic reviews of the assessed hazards (see para. 4.25);
(c) To coordinate and ensure consistency between the emergency arrangements of the various response organizations, operating organizations and the regulatory body at local, regional and national levels under the all-hazards approach, including those arrangements for response to relevant nuclear security events, and, as appropriate, those arrangements of other States and of international organizations;

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[4] This also includes the allocation of roles and responsibilities, as appropriate, among members of the government.

[5] The mechanism for ensuring coordination may differ for different tasks. It may involve an existing body or a newly established body (e.g. a committee consisting of representatives from different organizations and bodies) that has been given the authority to ensure the necessary coordination.

(d) To ensure consistency among requirements for emergency arrangements, contingency plans and security plans of operating organizations specified by the regulatory body and by other competent authorities with responsibilities for regulating nuclear security, as relevant, and to ensure that these arrangements and plans are integrated (see para. 4.14(b));
(e) To ensure that appropriate emergency arrangements are in place, both on the site and off the site, as appropriate, in relation to facilities and activities under regulatory control, both within the State and, as relevant, beyond its borders, and also for sources that are not under regulatory control[6];
(f) To coordinate arrangements made for enforcing compliance with the national requirements for emergency preparedness and response as established by legislation and regulations (see paras 1.12, 4.5 and 4.12);
(g) To coordinate a subsequent analysis of an emergency, including analysis of the emergency response (see Requirement 19);
(h) To ensure that appropriate and coordinated programmes of training and exercises are in place and implemented, and that training and exercises are systematically evaluated;
(i) To coordinate effective communication with the public in preparedness for a nuclear or radiological emergency.

Regulatory body

4.11. The government shall ensure that arrangements for preparedness and response to a nuclear or radiological emergency for facilities and activities under the responsibility of the operating organization are dealt with through the regulatory process.

4.12. The regulatory body is required to establish or adopt regulations and guides to specify the principles, requirements and associated criteria for safety upon which its regulatory judgements, decisions and actions are based [7]. These regulations and guides shall include principles, requirements and associated criteria for emergency preparedness and response for the operating organization (see also paras 1.12 and 4.5).

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[6] Examples of sources not under regulatory control are sources that have been abandoned, lost or stolen and sources under governmental control but not under regulatory control. Examples also include radioactive material that is out of regulatory control as discussed in Ref. [11].

4.13. The regulatory body shall require that arrangements for preparedness and response for a nuclear or radiological emergency be in place for the on-site area for any regulated facility or activity that could necessitate emergency response actions. Appropriate emergency arrangements shall be established by the time the source is brought to the site, and complete emergency arrangements shall be in place before the commencement of operation of the facility or commencement of the activity. The regulatory body shall verify compliance with the requirements for such arrangements.

4.14. Before commencement of operation of the facility or commencement of the activity, the regulatory body shall ensure, for all facilities and activities under regulatory control that could necessitate emergency response actions, that the on-site emergency arrangements:

(a) Are integrated with those of other response organizations, as appropriate;
(b) Are integrated with contingency plans in the context of Ref. [9] and with security plans in the context of Ref. [10];
(c) Provide, to the extent practicable, assurance of an effective response to a nuclear or radiological emergency.

4.15. The regulatory body shall ensure that the operating organization is given sufficient authority to promptly take necessary protective actions on the site in response to a nuclear or radiological emergency that could result in off-site consequences.

Operating organization

4.16. The operating organization shall establish and maintain arrangements for on-site preparedness and response for a nuclear or radiological emergency for facilities or activities under its responsibility, in accordance with the applicable requirements (see paras 1.12, 4.5 and 4.12).

4.17. The operating organization shall demonstrate that, and shall provide the regulatory body with an assurance that, emergency arrangements are in place for an effective response on the site to a nuclear or radiological emergency in relation to a facility or an activity under its responsibility.

Requirement 3: Responsibilities of international organizations in emergency preparedness and response

Relevant international organizations shall coordinate their arrangements in preparedness for a nuclear or radiological emergency and their emergency response actions.[7]

Requirement 4: Hazard assessment

The government shall ensure that a hazard assessment is performed to provide a basis for a graded approach in preparedness and response for a nuclear or radiological emergency.

4.18. Hazards shall be identified and potential consequences of an emergency shall be assessed to provide a basis for establishing arrangements for preparedness and response for a nuclear or radiological emergency. These arrangements shall be commensurate with the hazards identified and the potential consequences of an emergency.

4.19. For the purposes of these safety requirements, assessed hazards are grouped in accordance with the emergency preparedness categories shown in Table 1. The five emergency preparedness categories (hereinafter referred to as 'categories') in Table 1 establish the basis for a graded approach to the application of these requirements and for developing generically justified and optimized arrangements for preparedness and response for a nuclear or radiological emergency.

[7] The Inter-Agency Committee on Radiological and Nuclear Emergencies and its Joint Radiation Emergency Management Plan of the International Organizations are examples of such coordination.

TABLE 1. EMERGENCY PREPAREDNESS CATEGORIES

Category	Description
I	Facilities, such as nuclear power plants, for which on-site events[a, b] (including those not considered in the design[c]) are postulated that could give rise to severe deterministic effects[d] off the site that would warrant precautionary urgent protective actions, urgent protective actions or early protective actions, and other response actions to achieve the goals of emergency response in accordance with international standards[e], or for which such events have occurred in similar facilities.
II	Facilities, such as some types of research reactor and nuclear reactors used to provide power for the propulsion of vessels (e.g. ships and submarines), for which on-site events[a, b] are postulated that could give rise to doses to people off the site that would warrant urgent protective actions or early protective actions and other response actions to achieve the goals of emergency response in accordance with international standards[e], or for which such events have occurred in similar facilities. Category II (as opposed to category I) does not include facilities for which on-site events (including those not considered in the design) are postulated that could give rise to severe deterministic effects off the site, or for which such events have occurred in similar facilities.
III	Facilities, such as industrial irradiation facilities or some hospitals, for which on-site events[b] are postulated that could warrant protective actions and other response actions on the site to achieve the goals of emergency response in accordance with international standards[e], or for which such events have occurred in similar facilities. Category III (as opposed to category II) does not include facilities for which events are postulated that could warrant urgent protective actions or early protective actions off the site, or for which such events have occurred in similar facilities.
IV	Activities and acts that could give rise to a nuclear or radiological emergency that could warrant protective actions and other response actions to achieve the goals of emergency response in accordance with international standards[e] in an unforeseen location. These activities and acts include: (a) transport of nuclear or radioactive material and other authorized activities involving mobile dangerous sources such as industrial radiography sources, nuclear powered satellites or radioisotope thermoelectric generators; and (b) theft of a dangerous source and use of a radiological dispersal device or radiological exposure device[f]. This category also includes: (i) detection of elevated radiation levels of unknown origin or of commodities with contamination; (ii) identification of clinical symptoms due to exposure to radiation; and (iii) a transnational emergency that is not in category V arising from a nuclear or radiological emergency in another State. Category IV represents a level of hazard that applies for all States and jurisdictions.

Please see table notes on following page

TABLE 1. EMERGENCY PREPAREDNESS CATEGORIES (cont.)

Category	Description
V	Areas within emergency planning zones and emergency planning distances[g] in a State for a facility in category I or II located in another State.

[a] That is, on-site events involving an atmospheric or aquatic release of radioactive material, or external exposure (due, for example, to a loss of shielding or a criticality event), that originates from a location on the site.

[b] Such events include nuclear security events.

[c] This includes events that are beyond the design basis accidents and, as appropriate, conditions that are beyond design extension conditions.

[d] See 'deterministic effect' under Definitions.

[e] See the goals of emergency response in para. 3.2 and the generic criteria in Appendix II.

[f] A radiological dispersal device is a device to spread radioactive material using conventional explosives or other means. A radiation exposure device is a device with radioactive material designed to intentionally expose members of the public to radiation. They could be fabricated, modified or improvised devices.

[g] See para. 5.38.

4.20. The government shall ensure that for facilities and activities, a hazard assessment on the basis of a graded approach is performed. The hazard assessment shall include consideration of:

(a) Events that could affect the facility or activity, including events of very low probability and events not considered in the design;
(b) Events involving a combination of a nuclear or radiological emergency with a conventional emergency such as an emergency following an earthquake, a volcanic eruption, a tropical cyclone, severe weather, a tsunami, an aircraft crash or civil disturbances that could affect wide areas and/or could impair capabilities to provide support in the emergency response;
(c) Events that could affect several facilities and activities concurrently, as well as consideration of the interactions between the facilities and activities affected;
(d) Events at facilities in other States or events involving activities in other States.

4.21. The government shall ensure that the hazard assessment identifies those facilities and locations at which there is a significant likelihood of encountering a dangerous source that is not under control.[8]

4.22. The government shall ensure that the hazard assessment includes consideration of the results of threat assessments made for nuclear security purposes [9–11].[9]

4.23. In the hazard assessment, facilities and activities, on-site areas, off-site areas and locations shall be identified for which a nuclear or radiological emergency could — with account taken of the uncertainties in and limitations of the information available — warrant any of the following:

(a) Precautionary urgent protective actions to avoid or to minimize severe deterministic effects by keeping doses below levels approaching the generic criteria at which urgent protective actions and other response actions are required to be undertaken under any circumstances, with account taken of Appendix II;
(b) Urgent protective actions and other response actions to avoid or to minimize severe deterministic effects and to reduce the risk of stochastic effects, with account taken of Appendix II;
(c) Early protective actions and other response actions, with account taken of Appendix II;
(d) Other emergency response actions such as longer term medical actions, with account taken of Appendix II, and emergency response actions aimed at enabling the termination of the emergency (see Requirement 18); or
(e) Protection of emergency workers in accordance with Requirement 11 and with account taken of Appendix I.

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[8] Examples of such facilities and locations are: scrap metal processing facilities, border crossing points, seaports, airports and abandoned military facilities or other facilities where dangerous sources might have been used in the past.

[9] This includes consideration of 'strategic locations', i.e. locations of high security interest in the State which are potential targets for attacks using nuclear and other radioactive material and locations for detection of nuclear and other radioactive material that is out of regulatory control, in line with Ref. [11].

4.24. The government shall ensure that the hazard assessment also identifies non-radiation-related hazards[10] to people on the site and off the site that are associated with the facility or activity and that may impair the effectiveness of the response actions to be taken.

4.25. The government shall ensure that a review of the hazard assessment is performed periodically with the aims of: (a) ensuring that all facilities and activities, on-site areas, off-site areas and locations where events could occur that would necessitate protective actions and other response actions are identified, and (b) taking into account any changes in the hazards within the State and beyond its borders, any changes in assessments of threats for nuclear security purposes, the experience and lessons from research, operation and emergency exercises, and technological developments (see paras 6.30, 6.36 and 6.38). The results of this review shall be used to revise the emergency arrangements as necessary.

4.26. The government through the regulatory body shall ensure that operating organizations review appropriately and, as necessary, revise the emergency arrangements (a) prior to any changes in the facility or activity that affect the existing hazard assessment and (b) when new information becomes available that provides insights into the adequacy of the existing arrangements.[11]

Requirement 5: Protection strategy for a nuclear or radiological emergency

The government shall ensure that protection strategies are developed, justified and optimized at the preparedness stage for taking protective actions and other response actions effectively in a nuclear or radiological emergency.

4.27. The government shall ensure that, on the basis of the hazards identified and the potential consequences of a nuclear or radiological emergency, protection strategies are developed, justified and optimized at the preparedness stage for taking protective actions and other response actions effectively in a nuclear or radiological emergency to achieve the goals of emergency response.

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[10] Examples of non-radiation-related hazards are the release of toxic chemicals, e.g. uranium hexafluoride (UF_6), fires, explosions and floods.

[11] Examples of such changes and available information include the movement of irradiated nuclear fuel to a new location, projected flooding, and information on storms or other meteorological hazards.

4.28. Development of a protection strategy shall include, but shall not be limited to, the following:

(1) Consideration shall be given to actions to be taken to avoid or to minimize severe deterministic effects and to reduce the risk of stochastic effects. Deterministic effects shall be evaluated on the basis of relative biological effectiveness (RBE) weighted absorbed dose to a tissue or organ. Stochastic effects in a tissue or organ shall be evaluated on the basis of equivalent dose to the tissue or organ. The detriment associated with the occurrence of stochastic effects in individuals in an exposed population shall be evaluated on the basis of the effective dose.

(2) A reference level expressed in terms of residual dose shall be set, typically as an effective dose in the range 20–100 mSv, acute or annual, that includes dose contributions via all exposure pathways. This reference level shall be used in conjunction with the goals of emergency response (see para. 3.2) and the specific time frame in which particular goals are to be achieved.[12]

(3) On the basis of the outcome of the justification and the optimization of the protection strategy, national generic criteria for taking protective actions and other response actions, expressed in terms of projected dose or of dose that has been received, shall be developed with account taken of the generic criteria in Appendix II. If the national generic criteria for projected dose or received dose are exceeded, protective actions and other response actions, either individually or in combination, shall be implemented.

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[12] The application solely of the reference level for effective dose would not be sufficient to develop the protection strategy. Consideration needs to be given to the particular goal to be met in the response, the time to allow for actions to be taken effectively, and the appropriate dose quantity to be used to ensure that organ doses will be kept below those at which protective actions and other response actions are justified (see para. 4.28 (1)). For example, actions to avoid or to minimize severe deterministic effects are to be taken urgently when projected doses expected to be received within a short period of time exceed those given in Table II.1 of Appendix II for the RBE weighted absorbed dose to a tissue or organ. In this case, if such doses are received, then prompt and appropriate medical actions are necessary. Moreover, selection of a particular value (to be used for optimization purposes and for retrospective assessment of the effectiveness of actions and strategy taken) within the proposed range of 20–100 mSv acute or annual effective dose would depend on the phase of the emergency, the practicality of reducing or preventing exposures, and other factors. In the urgent phase of an emergency, an effective dose of 100 mSv, acute or annual, might be justified as one of the dosimetric bases for implementing and optimizing a protection strategy. In the later phases, such as during the transition, an effective dose of 20 mSv per year may be justified as one of the dosimetric bases for implementing and optimizing a protection strategy to enable the transition to an existing exposure situation to be made.

(4) Once the protection strategy has been justified and optimized and a set of national generic criteria has been developed, pre-established operational criteria (conditions on the site, emergency action levels (EALs) and operational intervention levels (OILs)) for initiating the different parts of an emergency plan and for taking protective actions and other response actions shall be derived from the generic criteria[13]. Arrangements shall be established in advance to revise these operational criteria, as appropriate, in the course of a nuclear or radiological emergency, with account taken of the prevailing conditions as they evolve.

4.29. Each protective action, in the context of the protection strategy, and the protection strategy itself shall be demonstrated to be justified (i.e. to do more good than harm), with account taken not only of those detriments that are associated with radiation exposure but also of those detriments associated with impacts of the actions taken on public health[14], the economy, society and the environment.

4.30. The government shall ensure that interested parties are involved and are consulted, as appropriate, in the development of the protection strategy.

4.31. The government shall ensure that the protection strategy is implemented safely and effectively in an emergency response through the implementation of emergency arrangements, including but not limited to:

(a) Promptly taking urgent protective actions and other response actions with account taken of Appendix II to avoid or to minimize severe deterministic effects, if possible, on the basis of observed conditions and before any exposure occurs;
(b) Taking early protective actions and other response actions to reduce the risk of stochastic effects with account taken of Appendix II;
(c) Providing for registration, health screening and longer term medical follow-up, as appropriate, with account taken of Appendix II;
(d) Taking actions to protect emergency workers, with account taken of guidance values provided in Appendix I;

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[13] The operational criteria (i.e. operational intervention levels) need to be derived for a representative person with account taken of those members of the public that are most vulnerable to radiation exposure (i.e. pregnant women and children).

[14] Examples of such impacts include possible deaths among patients evacuated without the necessary medical care and possible reduced life expectancy due to resettlement.

(e) Taking actions to mitigate non-radiological consequences, with account taken of Appendix II;
(f) Assessing the effectiveness of the actions taken and adjusting them as appropriate on the basis of prevailing conditions and available information as well as the reference level expressed in terms of residual dose;
(g) Revising the protection strategy as necessary and its further implementation;
(h) Discontinuing protective actions and other response actions when they are no longer justified.

5. FUNCTIONAL REQUIREMENTS

GENERAL

5.1. The requirements established in this section address the functions that are essential for the emergency response in a nuclear or radiological emergency to be effective and for achieving the goals of emergency response (see para. 3.2).

Requirement 6: Managing operations in an emergency response

The government shall ensure that arrangements are in place for operations in response to a nuclear or radiological emergency to be appropriately managed.

5.2. For facilities in categories I, II and III, arrangements shall be made for the on-site emergency response to be promptly executed and managed without impairing the performance of the continuing operational safety and security functions both at the facility and at any other facilities on the same site. The transition from normal operations to operations under emergency conditions on the site shall be clearly specified and shall be effectively made. The responsibilities of all personnel who would be on the site in an emergency shall be designated as part of the arrangements for this transition. It shall be ensured that the transition to the emergency response and the performance of initial response actions do not impair the ability of operating personnel (such as operating personnel in the control room) to ensure safe and secure operation while taking mitigatory actions.

5.3. For facilities in categories I, II and III, and, where appropriate, for activities in category IV, arrangements shall be made for an off-site emergency response to be promptly executed, effectively managed and coordinated with an on-site emergency response.

5.4. For a site where several facilities in categories I and II are collocated, adequate arrangements shall be made to manage the emergency response at all the facilities if each of them is under emergency conditions simultaneously. This shall include arrangements to manage the deployment of and the protection of personnel responding on and off the site (see Requirement 11).

5.5. For facilities and activities in categories I, II, III and IV, arrangements have to be made, as far as practicable, so that the facility or activity has a nuclear security system or systems [9–11] that would be functional in a nuclear or radiological emergency.

5.6. Arrangements for response to a nuclear or radiological emergency shall be coordinated and integrated with arrangements at the local, regional and national levels for response to a conventional emergency and to a nuclear security event.[15] These arrangements shall take into consideration the fact that the initiator of the nuclear or radiological emergency may not be known early in the response.

5.7. Arrangements shall be made for the establishment and use of a clearly specified and unified command and control system for emergency response under the all-hazards approach as part of the emergency management system (see paras 4.1–4.3). The command and control system shall provide sufficient assurance for effective coordination of the on-site and off-site response. The authority and responsibility for directing the emergency response and for making decisions on emergency response actions to be taken shall be clearly assigned. The responsibility for directing the emergency response and for decision making on emergency response actions to be taken shall be promptly discharged following a notification of an emergency.

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[15] The coordination and integration of arrangements for response to a nuclear or radiological emergency with arrangements for response to a nuclear security event includes coordination with and integration of arrangements for response measures such as identification, collection, packaging and transport of evidence contaminated with radionuclides, nuclear forensics and related activities in the context of an investigation into the circumstances surrounding a nuclear security event.

5.8. Arrangements shall be made for obtaining and assessing the information necessary for making decisions on the allocation of resources for all response organizations throughout a nuclear or radiological emergency.

5.9. For facilities in category I or II and areas in category V, arrangements shall be made for coordinating the emergency response between response organizations (including those of other States) within the emergency planning zones and emergency planning distances (see para. 5.38) and for providing mutual support.

5.10. Arrangements shall be made with other States, as appropriate, for coordinated response to a radiological emergency.

Requirement 7: Identifying and notifying a nuclear or radiological emergency and activating an emergency response

The government shall ensure that arrangements are in place for the prompt identification and notification of a nuclear or radiological emergency and for the activation of an emergency response.

5.11. An off-site notification point[16], or more than one, shall be established to receive notification of an actual or potential nuclear or radiological emergency. The notification point(s) shall be maintained in a state of continuous availability to receive any notification or request for support and to respond promptly, or to initiate a preplanned and coordinated off-site emergency response appropriate to the emergency class or the level of emergency response. The notification point(s) shall be able to initiate immediate communication by suitable, reliable and diverse means with the response organizations that are providing support.

5.12. For facilities in categories I and II and for areas in category V, the notification point shall be able to initiate immediate communication with the authority that has been assigned the responsibility to decide on and to initiate precautionary urgent protective actions and urgent protective actions off the site (see also para. 5.7).

5.13. For facilities and locations at which there is a significant likelihood of encountering a dangerous source that is not under control (see para. 4.21), arrangements shall be made to ensure that the on-site managers of operations and

[16] This may be the notification point used to receive notification of and to initiate an off-site emergency response to an emergency of any type (conventional, or nuclear or radiological).

other personnel are aware of the indicators of a potential radiological emergency, the appropriate notification, and protective actions and other response actions that are immediately warranted in an emergency. For facilities and locations for which there is a significant likelihood of encountering a dangerous source that is not under control and for an emergency at an unforeseen location, arrangements shall be made to ensure that the local officials responsible for the response and first responders are aware of the indicators of a potential radiological emergency, the appropriate notification, and protective actions and other response actions that are warranted to be taken immediately in an emergency.

5.14. The operating organization of a facility or activity in category I, II, III or IV shall make arrangements for promptly classifying, on the basis of the hazard assessment, a nuclear or radiological emergency warranting protective actions and other response actions to protect workers, emergency workers, members of the public and, as relevant, patients and helpers in an emergency, in accordance with the protection strategy (see Requirement 5). This shall include a system for classifying all types of nuclear or radiological emergency[17] as follows:

(a) *General emergency* at facilities in category I or II for an emergency that warrants taking precautionary urgent protective actions, urgent protective actions, and early protective actions and other response actions on the site and off the site. Upon declaration of this emergency class, appropriate actions shall promptly be taken, on the basis of the available information relating to the emergency, to mitigate the consequences of the emergency on the site and to protect people on the site and off the site.
(b) *Site area emergency* at facilities in category I or II for an emergency that warrants taking protective actions and other response actions on the site and in the vicinity of the site. Upon declaration of this emergency class, actions shall promptly be taken: (i) to mitigate the consequences of the emergency on the site and to protect people on the site; (ii) to increase the readiness to take protective actions and other response actions off the site if this becomes necessary on the basis of observable conditions, reliable assessments and/or results of monitoring; and (iii) to conduct off-site monitoring, sampling and analysis.
(c) *Facility emergency* at facilities in category I, II or III for an emergency that warrants taking protective actions and other response actions at the facility and on the site but does not warrant taking protective actions off the site.

[17] The emergency classes may differ from those specified in (a)–(e) provided that emergencies of all these types are included.

Upon declaration of this emergency class, actions shall promptly be taken to mitigate the consequences of the emergency and to protect people at the facility and on the site. Emergencies in this class do not present an off-site hazard.

(d) *Alert* at facilities in category I, II or III for an event that warrants taking actions to assess and to mitigate the potential consequences at the facility. Upon declaration of this emergency class, actions shall promptly be taken to assess and to mitigate the potential consequences of the event and to increase the readiness of the on-site response organizations.

(e) *Other nuclear or radiological emergency*[18] for an emergency in category IV that warrants taking protective actions and other response actions at any location. Upon declaration of this emergency class and the level of emergency response, actions shall promptly be taken to mitigate the consequences of the emergency on the site, to protect those in the vicinity (e.g. workers and emergency workers and the public) and to determine where and for whom protective actions and other response actions are warranted.

5.15. For facilities in category I, II or III and for category IV, arrangements shall be made to review the declared emergency class in the light of any new information and, as appropriate, to revise it.

5.16. The emergency classification system for facilities and activities in categories I, II, III and IV shall take into account all postulated emergencies, including those arising from events of very low probability. The operational criteria for classification shall include emergency action levels and other observable conditions (i.e. 'observables') and indicators of the conditions at the facility and/or on the site or off the site. The emergency classification system shall be established with the aim of allowing for the prompt initiation of an effective response in recognition of the uncertainty of the available information. It shall be ensured that any process for rating an event on the International Nuclear and Radiological Event Scale (INES) [15] does not delay the emergency classification or emergency response actions.[19]

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[18] This class covers broad types of emergency (see Table 1 and paras 4.21 and 4.22). A graded approach may need to be taken when postulating emergencies and expected consequences within this class in order to determine the level of emergency response warranted.

[19] The emergency classification system is not to be confused with the INES. The INES is a scale developed for use by States solely for the purpose of communicating with the public on the safety significance of events associated with sources of radiation. The INES is not to be used as a basis for emergency response actions.

5.17. For facilities and activities in categories I, II and III, and for category IV, arrangements shall be made: (1) to promptly recognize and classify a nuclear or radiological emergency; (2) upon classification, to promptly declare the emergency class and to initiate a coordinated and preplanned on-site response; (3) to notify the appropriate notification point (see para. 5.11) and to provide sufficient information for an effective off-site response; and (4) upon notification, to initiate a coordinated and preplanned off-site response, as appropriate, in accordance with the protection strategy. These arrangements shall include suitable, reliable and diverse means of warning persons on the site, of notifying the notification point (see paras 5.41–5.43, 6.22 and 6.34) and of communication between response organizations.

5.18. In the event of a transnational emergency, the notifying State shall promptly notify[20,21] the IAEA of the emergency and, either directly or through the IAEA, those States that could be affected by it. The notifying State shall provide information on the nature of the emergency and on its potential transnational consequences, and shall respond to requests from other States and from the IAEA for information for the purposes of mitigating any consequences.

5.19. The State shall make known to the IAEA and to other States, directly or through the IAEA, its single warning point responsible for receiving emergency notifications and information from other States and information from the IAEA. This warning point shall be maintained in a state of continuous availability to receive any notification, request for assistance or request for verification and to promptly initiate a response or verification. The State shall promptly inform the IAEA and shall inform other States, directly or through the IAEA, of any changes that occur in respect of the warning point. The State shall make arrangements for promptly notifying and for providing relevant information, directly or through the IAEA, to those States that could be affected by a transnational emergency.

5.20. The notifying State shall have arrangements in place for promptly responding to requests from other States or from the IAEA for information in respect of a transnational emergency, in particular with regard to minimizing

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[20] Such a notification is in accordance with the State's obligations under the general principles and rules of international law and, for the case of a transboundary release that could be of radiological safety significance for another State, it is in accordance with the Early Notification Convention [13].

[21] A transnational emergency that is considered to represent a public health emergency of international concern may also be expected to be notified in accordance with the International Health Regulations [16].

any consequences. These arrangements shall include making known to the IAEA and to other States, directly or through the IAEA, the notifying State's designated organization(s) for so doing.

5.21. Arrangements shall be made for promptly and directly notifying any State within the emergency planning zones and emergency planning distances (see para. 5.38) within which urgent protective actions and early protective actions and other response actions could be required to be taken.

5.22. Appropriate emergency response actions shall be initiated in a timely manner upon the receipt of a notification from another State or of information from the IAEA on a notification relating to an actual or potential transnational emergency that could have impacts on the State or its nationals.

Requirement 8: Taking mitigatory actions

The government shall ensure that arrangements are in place for taking mitigatory actions in a nuclear or radiological emergency.

5.23. The operating organization of a facility or activity in category I, II, III or IV shall promptly decide on and take actions[22] on the site that are necessary to mitigate the consequences of a nuclear or radiological emergency involving a facility or an activity under its responsibility.

5.24. Off-site emergency services shall be made available for the purpose of, and shall be capable of, supporting the on-site emergency response at facilities and activities in category I, II, III or IV.[23]

5.25. For facilities in category I, II or III, arrangements shall be made for mitigatory actions to be taken by the operating personnel, in particular:

(a) To prevent escalation of an emergency;
(b) To return the facility to a safe and stable state;

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[22] Such actions may include actions with off-site consequences such as discharge of radioactive material to the environment, provided that the appropriate off-site organizations are notified in advance.

[23] This is not to be understood as diminishing the responsibility of the operating organization to have adequate capabilities to respond to an emergency arising in the facility or activity under its responsibility.

(c) To reduce the potential for, and to mitigate the consequences of, radioactive releases or exposures.

These arrangements shall take into account the full range of possible conditions affecting the emergency response, including those resulting from conditions in the facility and those resulting from impacts of postulated natural, human induced or other events and affecting regional infrastructure or affecting several facilities simultaneously. Arrangements shall include emergency operating procedures and guidance for operating personnel on mitigatory actions for severe conditions (for a nuclear power plant, as part of the accident management programme [17]) and for the full range of postulated emergencies, including accidents that are not considered in the design and associated conditions. As far as practicable, the continued functionality of nuclear security system(s) (see Refs [9–11]) needs to be considered in these arrangements.

5.26. The operating organization of a facility or activity in category I, II, III or IV shall assess and determine, at the preparedness stage, when and under what conditions assistance from off-site emergency services may need to be provided on the site, consistent with the hazard assessment and the protection strategy.[23]

5.27. For facilities in category I, II or III, arrangements shall be made, in particular by the operating organization, to provide technical assistance to the operating personnel. On-site teams for mitigating the consequences of an emergency (e.g. damage control, firefighting) shall be available and shall be prepared to perform actions at the facility. Paragraph 5.15 of Safety of Nuclear Power Plants: Design (SSR-2/1) [18] states that:

> "Any equipment that is necessary for actions to be taken in manual response and recovery processes shall be placed at the most suitable location to ensure its availability at the time of need and to allow safe access to it under the environmental conditions anticipated."

The operating personnel directing mitigatory actions shall be provided with information and technical assistance to allow them to take actions effectively to mitigate the consequences of the emergency. Arrangements shall be made to obtain support promptly from the emergency services (e.g. law enforcement agencies, medical services and firefighting services) off the site. Off-site emergency services shall be afforded prompt access to the facility, and shall be informed of on-site conditions and provided with instructions and with means for protecting themselves as emergency workers.

5.28. Arrangements shall be made for the operating organization of an activity in category IV, first responders in an emergency at an unforeseen location, and those personnel at locations where there is a significant likelihood of encountering a dangerous source that is not under control (see para. 4.21) to take promptly all practicable and appropriate actions to mitigate the consequences of a nuclear or radiological emergency. These arrangements shall include providing basic instructions and training in the means of mitigating the potential consequences of a nuclear or radiological emergency (see para. 5.44).

5.29. Arrangements shall be made to provide expertise and services in radiation protection promptly to local officials, first responders in an emergency at an unforeseen location and specialized services (e.g. law enforcement agencies) responding to emergencies involving activities and acts in category IV, and to those personnel at locations where there is a significant likelihood of encountering a dangerous source that is not under control (see para. 4.21). This shall include arrangements for on-call advice or other appropriate mechanisms and arrangements to dispatch to the site an emergency team capable of assessing radiation hazards, mitigating radiological consequences and managing the exposure of emergency workers. In addition, arrangements shall be made to determine whether and when additional assistance is necessary and to determine how to obtain such assistance (see paras 5.24 and 5.94).

5.30. Arrangements shall be made to initiate a prompt search in the event that a dangerous source could possibly be in the public domain as a result of its loss or unauthorized removal (see para. 5.47).

Requirement 9: Taking urgent protective actions and other response actions

The government shall ensure that arrangements are in place to assess emergency conditions and to take urgent protective actions and other response actions effectively in a nuclear or radiological emergency.

5.31. Arrangements shall be made so that the magnitudes of hazards and the possible development of hazardous conditions are assessed initially and throughout a nuclear or radiological emergency in order to promptly identify, characterize or anticipate, as appropriate, new hazards or the extent of hazards and to revise the protection strategy.

5.32. The operating organization of a facility in category I, II or III shall make arrangements to promptly assess and anticipate:

(a) Abnormal conditions at the facility;
(b) Exposures and radioactive releases and releases of other hazardous material;
(c) Radiological conditions on the site and, as appropriate, off the site;
(d) Any exposures or potential exposures of workers and emergency workers, the public and, as relevant, patients and helpers in an emergency.

5.33. These assessments as stated in para. 5.32 shall be used:

(a) For deciding on mitigatory actions to be taken by the operating personnel;
(b) As a basis for emergency classification (see para. 5.14);
(c) For deciding on protective actions and other response actions to be taken on the site, including those for the protection of workers and emergency workers;
(d) For deciding on protective actions and other response actions to be taken off the site;
(e) Where appropriate, to identify those individuals who could potentially have been exposed on the site at levels requiring appropriate medical attention in accordance with Appendix II.

5.34. These arrangements as stated in para. 5.32 shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)) and provision for access to instruments displaying or measuring those parameters that can readily be measured or observed in a nuclear or radiological emergency. In these arrangements, the expected response of instrumentation and of structures, systems and components at the facility under emergency conditions shall be taken into account.

5.35. The operating organization for activities in category IV shall make arrangements to assess promptly the extent and/or the significance of any abnormal conditions on the site, any exposures or any contamination. These assessments shall be used:

(a) For initiating the mitigatory actions;
(b) As a basis for protective actions and other response actions to be taken on the site;
(c) For determining the level for emergency response and for communicating the extent of the hazards to the appropriate off-site response organizations.

These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)).

5.36. Arrangements shall be made such that information on emergency conditions, assessments and protective actions and other response actions that have been recommended and have been taken is promptly made available, as appropriate, to all relevant response organizations and to the IAEA throughout the emergency.

5.37. Arrangements shall be made for actions to save human life or to prevent serious injury to be taken without any delay on the grounds of the possible presence of radioactive material (see paras 5.39 and 5.64). These arrangements shall include providing first responders in an emergency at an unforeseen location with information on the precautions to take in giving first aid or in transporting an individual with possible contamination.

5.38. For facilities in category I or II, arrangements shall be made for effectively making decisions on and taking urgent protective actions, early protective actions and other response actions[24] off the site in order to achieve the goals of emergency response, on the basis of a graded approach and in accordance with the protection strategy. The arrangements shall be made with account taken of the uncertainties in and limitations of the information available when protective actions and other response actions have to be taken to be effective, and shall include the following:

(a) The specification of off-site emergency planning zones and emergency planning distances[25] for which arrangements shall be made at the preparedness stage for taking protective actions and other response actions effectively. These emergency planning zones and emergency planning

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[24] Although defined under this overarching requirement, emergency planning zones and emergency planning distances are applicable for both urgent protective actions and early protective actions and other response actions. Within emergency planning zones, the main focus is on taking precautionary urgent protective actions, urgent protective actions and other response actions. However, within emergency planning distances, urgent decisions may be warranted, as a precaution, to prevent inadvertent ingestion and to restrict the consumption of food, milk and drinking water that could be directly contaminated following a significant release of radioactive material to the environment and then consumed.

[25] The off-site emergency planning zones and emergency planning distances may differ from those specified provided that, at the preparedness stage, such areas and distances are designated and arrangements are made to effectively take precautionary urgent protective actions, urgent protective actions and early protective actions and other response actions within these areas and distances in order to achieve the goals of emergency response.

distances shall be contiguous across national borders, where appropriate, and shall include:

(i) A precautionary action zone (PAZ), for facilities in category I, for which arrangements shall be made for taking urgent protective actions and other response actions, before any significant release[26] of radioactive material occurs, on the basis of conditions at the facility (i.e. conditions leading to the declaration of a general emergency; see para. 5.14), in order to avoid or to minimize severe deterministic effects.

(ii) An urgent protective action planning zone (UPZ), for facilities in category I or II, for which arrangements shall be made to initiate urgent protective actions and other response actions, if possible before any significant release of radioactive material occurs, on the basis of conditions at the facility (i.e. conditions leading to the declaration of a general emergency; see para. 5.14), and after a release occurs, on the basis of monitoring and assessment of the radiological situation off the site, in order to reduce the risk of stochastic effects.[27] Any such actions shall be taken in such a way as not to delay the implementation of precautionary urgent protective actions and other response actions within the precautionary action zone.

(iii) An extended planning distance (EPD) from the facility, for facilities in category I or II (beyond the urgent protective action planning zone), for which arrangements shall be made to conduct monitoring and assessment of the radiological situation off the site in order to identify areas, within a period of time that would allow the risk of stochastic effects in the areas to be effectively reduced by taking protective actions and other response actions within a day to a week or to a few weeks following a significant radioactive release.

(iv) An ingestion and commodities planning distance (ICPD) from the facility, for facilities in category I or II (beyond the extended planning distance), for which arrangements shall be made to take response

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[26] A significant release of radioactive material is a radioactive release that could lead to severe deterministic effects off the site and thus warrants taking protective actions or other response actions off the site.

[27] Taking actions within the urgent protective action planning zone in order to reduce the risk of stochastic effects would not mean that no severe deterministic effects could possibly be observed within the urgent protective action planning zone. However, any severe deterministic effects are most likely to occur within the precautionary action zone.

actions (1) for protecting the food chain and water supply[28] as well as for protecting commodities other than food from contamination following a significant radioactive release and (2) for protecting the public from the ingestion of food, milk and drinking water and from the use of commodities other than food with possible contamination following a significant radioactive release.

(b) Criteria, based on the emergency classification and on conditions at the facility and off the site (see paras 4.28(3), 4.28(4), 5.14 and 5.15), for initiating and for adjusting urgent protective actions and other response actions within the emergency planning zones and emergency planning distances, in accordance with the protection strategy.

(c) Authority and responsibility to provide sufficient and updated information to the notification point at any time to allow for an effective off-site emergency response.

5.39. Within the emergency planning zones and emergency planning distances, arrangements shall be made for taking appropriate protective actions and other response actions effectively, as necessary, promptly upon the notification of a nuclear or radiological emergency. These arrangements shall include arrangements for:

(a) Prompt exercise of authority and discharge of responsibility for making decisions to initiate protective actions and other response actions upon notification of an emergency (see para. 5.12);

(b) Warning the permanent population, transient population groups and special population groups or those responsible for them and warning special facilities;

(c) Taking urgent protective actions and other response actions such as evacuation, restrictions on the food chain and on water supply, prevention of inadvertent ingestion, restrictions on the consumption of food, milk and drinking water and on the use of commodities, decontamination of evacuees, control of access and traffic restrictions;

(d) Protection of emergency workers and helpers in an emergency.

The arrangements shall be coordinated with all jurisdictions (including, to the extent practicable, jurisdictions beyond national borders, where relevant) within any emergency planning zone or distance. These arrangements shall ensure

[28] 'Water supply' refers to water supplies that use rainwater or other untreated surface water.

that services necessary for ensuring public safety (e.g. rescue services and health services for the care of critically ill patients) are provided continuously throughout the emergency, including during the period when protective actions and other response actions are being taken.

5.40. Within emergency planning zones and emergency planning distances, arrangements shall be made for the timely monitoring and assessment of contamination, radioactive releases and exposures for the purpose of deciding on or adjusting the protective actions and other response actions that have to be taken or that are being taken. These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)).

5.41. The operating organization of a facility in category I, II or III shall make arrangements to ensure protection and safety for all persons on the site in a nuclear or radiological emergency. These shall include arrangements to do the following:

(a) To notify all persons on the site of an emergency on the site;
(b) For all persons on the site to take appropriate actions immediately upon notification of an emergency;
(c) To account for those persons on the site and to locate and recover those persons unaccounted for;
(d) To provide immediate first aid;
(e) To take urgent protective actions.

5.42. Arrangements as stated in para. 5.41 shall also include ensuring the provision, for all persons present in the facility and on the site, of:

(a) Suitable assembly points, provided with continuous radiation monitoring;
(b) A sufficient number of suitable escape routes;
(c) Suitable and reliable alarm systems and other means for warning and instructing all persons present under the full range of emergency conditions.

5.43. The operating organization of a facility in category I, II or III shall ensure that suitable, reliable and diverse means of communication are available at all times, under the full range of emergency conditions, for use in taking protective actions and other response actions on the site and for communication with off-site officials responsible for taking protective actions and other response actions off the site or within any emergency planning zones or emergency planning distances.

5.44. Operating personnel for activities in category IV, first responders in an emergency at an unforeseen location and those personnel at locations where there is a significant likelihood of encountering a dangerous source that is not under control (see para. 4.21) shall be provided with guidance and training on taking urgent protective actions and other response actions. This shall include guidance and training on the approximate radius of the inner cordoned off area in which urgent protective actions and other response actions would initially be taken and on the adjustment of this area on the basis of observed or assessed conditions on the site.

Requirement 10: Providing instructions, warnings and relevant information to the public for emergency preparedness and response

The government shall ensure that arrangements are in place to provide the public who are affected or are potentially affected by a nuclear or radiological emergency with information that is necessary for their protection, to warn them promptly and to instruct them on actions to be taken.

5.45. For facilities in category I or II and areas in category V, arrangements shall be made to provide the permanent population, transient population groups and special population groups or those responsible for them and special facilities within the emergency planning zones and emergency planning distances (see para. 5.38), before operation and throughout the lifetime of the facility, with information on the response to a nuclear or radiological emergency. This information shall include information on the potential for a nuclear or radiological emergency, on the nature of the hazards, on how people would be warned or notified, and on the actions to be taken in such an emergency. The information shall be provided in the languages mainly spoken by the population residing within the emergency planning zones and emergency planning distances. The effectiveness of these arrangements for public information shall be periodically assessed.

5.46. For facilities in category I or II and in areas in category V, arrangements shall be made to register those members of the public in special population groups and, as appropriate, those responsible for them, and to promptly issue them and the permanent population and transient population groups, as well as special facilities in the emergency planning zones and emergency planning distances, with a warning and with instructions to be followed upon declaration of a general emergency (see para. 5.14). This shall include providing instructions on the actions to be taken in the languages mainly spoken by the population

residing within these emergency planning zones and emergency planning distances (see para. 5.38).

5.47. For facilities in category III and category IV, arrangements shall be made to provide the public with information and instructions in order to identify and locate people who may have been affected by a nuclear or radiological emergency and who may need response actions such as decontamination, medical examination or health screening. These arrangements shall include arrangements for issuing a warning to the public and providing information in the event that a dangerous source could be in the public domain as a consequence of its loss or unauthorized removal.

5.48. Arrangements shall be made by response organizations in a State to promptly provide information and advice to its nationals and to those people with interests in other States[29] in the event of a nuclear or radiological emergency declared beyond national borders, with due account taken of the response actions recommended in the State in which the emergency occurs as well as in the State(s) affected by that emergency (see paras 5.73 and 6.14).

Requirement 11: Protecting emergency workers and helpers in an emergency

The government shall ensure that arrangements are in place to protect emergency workers and to protect helpers in a nuclear or radiological emergency.

5.49. Arrangements shall be made to ensure that emergency workers are, to the extent practicable, designated in advance and are fit for the intended duty. These arrangements shall include health surveillance for emergency workers for the purpose of assessing their initial fitness and continuing fitness for their intended duties (see also GSR Part 3 [8]).

5.50. Arrangements shall be made to register and to integrate into operations in an emergency response those emergency workers who were not designated as such in advance of a nuclear or radiological emergency and helpers in an emergency. This shall include designation of the response organization(s) responsible for ensuring protection of emergency workers and protection of helpers in an emergency.

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[29] Examples of people with interests in other States include people travelling, people working and/or living abroad, importers and exporters, and people working in companies operating abroad.

5.51. The operating organization and response organizations shall determine the anticipated hazardous conditions, both on the site and off the site, in which emergency workers might have to perform response functions in a nuclear or radiological emergency in accordance with the hazard assessment and the protection strategy.

5.52. The operating organization and response organizations shall ensure that arrangements are in place for the protection of emergency workers and protection of helpers in an emergency for the range of anticipated hazardous conditions in which they might have to perform response functions. These arrangements, as a minimum, shall include:

(a) Training those emergency workers designated as such in advance;
(b) Providing emergency workers not designated in advance and helpers in an emergency immediately before the conduct of their specified duties with instructions on how to perform the duties under emergency conditions ('just in time' training);
(c) Managing, controlling and recording the doses received;
(d) Provision of appropriate specialized protective equipment and monitoring equipment;
(e) Provision of iodine thyroid blocking, as appropriate, if exposure due to radioactive iodine is possible;
(f) Obtaining informed consent to perform specified duties, when appropriate;
(g) Medical examination, longer term medical actions and psychological counselling, as appropriate.

5.53. The operating organization and response organizations shall ensure that all practicable means are used to minimize exposures of emergency workers and helpers in an emergency in the response to a nuclear or radiological emergency (see para. I.2 of Appendix I), and to optimize their protection.

5.54. In a nuclear or radiological emergency, the relevant requirements for occupational exposure in planned exposure situations established in GSR Part 3 [8] shall be applied, on the basis of a graded approach, for emergency workers, except as required in para. 5.55.

5.55. The operating organization and response organizations shall ensure that no emergency worker is subject to an exposure in an emergency that could give rise to an effective dose in excess of 50 mSv other than:

(1) For the purposes of saving human life or preventing serious injury;
(2) When taking actions to prevent severe deterministic effects or actions to prevent the development of catastrophic conditions that could significantly affect people and the environment;
(3) When taking actions to avert a large collective dose.

5.56. For the exceptional circumstances of para. 5.55, national guidance values shall be established for restricting the exposures of emergency workers, in accordance with Appendix I.

5.57. The operating organization and response organizations shall ensure that emergency workers who undertake emergency response actions in which doses received might exceed an effective dose of 50 mSv do so voluntarily[30]; that they have been clearly and comprehensively informed in advance of associated health risks as well as of available protective measures; and that they are, to the extent possible, trained in the actions that they might be required to take. Emergency workers not designated as such in advance shall not be the first emergency workers chosen for taking actions that could result in their doses exceeding the guidance values of dose for lifesaving actions, as given in Appendix I. Helpers in an emergency shall not be allowed to take actions that could result in their receiving doses in excess of an effective dose of 50 mSv.

5.58. Arrangements shall be made to assess as soon as practicable the individual doses received in a response to a nuclear or radiological emergency by emergency workers and helpers in an emergency and, as appropriate, to restrict further exposures in the response to the emergency (see Appendix I).

5.59. Emergency workers and helpers in an emergency shall be given appropriate medical attention for doses received in a response to a nuclear or radiological emergency (see Appendix II) or at their request.

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[30] The voluntary basis for response actions by emergency workers is usually covered in the emergency arrangements.

5.60. Emergency workers who receive doses in a response to a nuclear or radiological emergency shall normally not be precluded from incurring further occupational exposure. However, qualified medical advice[31] shall be obtained before any further occupational exposure occurs if an emergency worker has received an effective dose exceeding 200 mSv, or at the request of the emergency worker.

5.61. Information on the doses received in the response to a nuclear or radiological emergency and information on any consequent health risks shall be communicated, as soon as practicable, to emergency workers and to helpers in an emergency.

Requirement 12: Managing the medical response in a nuclear or radiological emergency

The government shall ensure that arrangements are in place for the provision of appropriate medical screening and triage, medical treatment and longer term medical actions for those people who could be affected in a nuclear or radiological emergency.

5.62. On the presentation by an individual of clinical symptoms of radiation exposure or other indications associated with a possible nuclear or radiological emergency, the medical personnel or other responsible parties who identify the clinical symptoms or other indications shall notify the appropriate local or national officials and shall take response actions as appropriate.

5.63. Arrangements shall be made for medical personnel, both general practitioners and emergency medical staff, to be made aware of the clinical symptoms of radiation exposure, and of the appropriate notification procedures and other emergency response actions to be taken if a nuclear or radiological emergency arises or is suspected.

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[31] Such qualified medical advice is intended for assessing the continuing fitness of workers for their intended tasks involving occupational exposure in line with GSR Part 3 [8]. In line with para. 5.59 of this Safety Requirements publication, any emergency worker is to be given appropriate medical attention for doses received. To illustrate this, the generic criterion for dose that is received (100 mSv effective dose in a month), as provided in Table II.2 of Appendix II, will indicate that an emergency worker receiving such a dose needs to be registered and subjected to health screening and that the emergency worker will then need appropriate longer term medical follow-up in order to detect radiation induced health effects early and to treat them effectively.

5.64. Arrangements shall be made so that, in a nuclear or radiological emergency, individuals with possible contamination can promptly be given appropriate medical attention. These arrangements shall include ensuring that transport services are provided where needed and providing instructions[32] to medical personnel on the precautions to take.

5.65. For facilities in categories I, II and III, arrangements shall be made to manage an adequate number of any individuals with contamination or of any individuals who have been overexposed to radiation, including arrangements for first aid, the estimation of doses, medical transport and initial medical treatment in predesignated medical facilities.

5.66. For areas within emergency planning zones (see para. 5.38), arrangements shall be made for performing medical screening and triage and for assigning to a predesignated medical facility any individual exposed at levels exceeding the criteria in Table II.1 of Appendix II. These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)).

5.67. Arrangements shall be made to identify individuals with possible contamination and individuals who have possibly been sufficiently exposed for radiation induced health effects to result, and to provide them with appropriate medical attention, including longer term medical follow-up. These arrangements shall include:

(a) Guidelines for effective diagnosis and treatment;
(b) Designation of medical personnel trained in clinical management of radiation injuries;
(c) Designation of institutions for evaluating radiation exposure (external and internal), for providing specialized medical treatment and for longer term medical actions.

These arrangements shall also include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)) and arrangements for medical consultation on treatment following any exposure

[32] These instructions include advice that universal precautions in health care against infection (e.g. surgical masks and gloves) generally provide medical personnel with adequate protection when treating individuals with possible contamination.

that could result in severe deterministic effects (see Appendix II) with medical personnel experienced in dealing with such injuries.[33]

5.68. Arrangements shall be made for the identification of individuals who are in those population groups that are at risk of sustaining increases in the incidence of cancers as a result of radiation exposure in a nuclear or radiological emergency. Arrangements shall be made to take longer term medical actions to detect radiation induced health effects among such population groups in time to allow for their effective treatment. These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)).

Requirement 13: Communicating with the public throughout a nuclear or radiological emergency

The government shall ensure that arrangements are in place for communication with the public throughout a nuclear or radiological emergency.

5.69. Arrangements shall be made for providing useful, timely, true, clear and appropriate information to the public in a nuclear or radiological emergency, with account taken of the possibility that the usual means of communication might be damaged in the emergency or by its initiating event (e.g. by an earthquake or by flooding) or overburdened by demand for its use. These arrangements shall also include arrangements for keeping the international community informed, as appropriate. These arrangements shall take into account the need to protect sensitive information in circumstances where a nuclear or radiological emergency is initiated by a nuclear security event. Communication with the public in a nuclear or radiological emergency shall be carried out on the basis of a strategy to be developed at the preparedness stage as part of the protection strategy. Arrangements shall be made to adjust this strategy in the emergency response on the basis of prevailing conditions.

5.70. Arrangements shall be made to ensure that information provided to the public by response organizations, operating organizations, the regulatory body, international organizations and others in a nuclear or radiological emergency

[33] Such arrangements for medical consultation on treatment could include international assistance to be provided through or to be coordinated by the IAEA and by WHO; for example, under the Assistance Convention [13].

is coordinated and consistent, with due recognition of the evolutionary nature of an emergency.

5.71. Arrangements shall be made so that in a nuclear or radiological emergency information is provided to the public in plain and understandable language.

5.72. The government shall ensure that a system for putting radiological health hazards in perspective in a nuclear or radiological emergency is developed and implemented with the following aim:

— To support informed decision making concerning protective actions and other response actions to be taken;
— To help in ensuring that actions taken do more good than harm;
— To address public concerns regarding potential health effects.

In the development of such a system, due consideration shall be given to pregnant women and children as the individuals who are most vulnerable with regard to radiation exposure.

5.73. Arrangements shall be made to explain to the public any changes in the protective actions and other response actions recommended in the State and any differences from those recommended in other States (see paras 6.13–6.15).

5.74. Arrangements shall be made to identify and address, to the extent practicable, misconceptions, rumours and incorrect and misleading information that might be circulating widely in a nuclear or radiological emergency, in particular those that might result in actions being taken beyond those emergency response actions that are warranted[34] (see Requirement 16).

5.75. Arrangements shall be made to respond to enquiries from the public and from news media, both national and international, including enquiries received from or through the IAEA. These arrangements shall recognize the evolutionary

[34] Actions beyond those emergency response actions that are warranted include, but are not limited to: actions that interfere with prompt implementation of protective actions, such as self-evacuation both from within and from outside areas from which evacuation is ordered; actions that unnecessarily burden the health care system; actions that shun or otherwise discriminate against people or products from an area affected by a nuclear or radiological emergency; elective terminations of pregnancy that are not radiologically informed; and cancellations of commercial flights that are not radiologically informed.

nature of emergencies and the need to respond in a timely manner to enquiries even when the information requested is not yet available.

Requirement 14: Taking early protective actions and other response actions

The government shall ensure that arrangements are in place to take early protective actions and other response actions effectively in a nuclear or radiological emergency.

5.76. Within the extended planning distance (see para. 5.38), arrangements shall be made for effective relocation that may be required following a significant radioactive release and for the prevention of inadvertent ingestion, in accordance with the protection strategy (see Requirement 5). These arrangements shall include:

(a) Provision of instructions and advice to prevent inadvertent ingestion;
(b) Prompt monitoring and assessment;
(c) Use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4));
(d) The means for accomplishing relocation and for assisting those persons who have been relocated;
(e) Provisions to extend monitoring and assessment and actions beyond the extended planning distance if necessary.

5.77. For areas within the ingestion and commodities planning distance (see para. 5.38), arrangements shall be made for prompt protection in relation to, and for restriction of, non-essential local produce, forest products (e.g. wild berries, wild mushrooms), milk from grazing animals, drinking water supplies, animal feed and commodities with contamination or possibly with contamination following a significant radioactive release, in accordance with the protection strategy (see Requirement 5). These arrangements shall include:

(a) Provision of instructions and advice:
    (i) To protect the food chain, water supply and commodities from contamination;
    (ii) To prevent ingestion of food, milk and drinking water with contamination or possibly with contamination;
    (iii) To prevent use of commodities with contamination or possibly with contamination;
(b) Prompt monitoring, sampling and analysis.

(c) Use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)).
(d) The means to enforce the restrictions.
(e) Provisions to expand monitoring and assessment and actions beyond this distance if necessary.

5.78. Within the emergency planning zones and the inner cordoned off area, arrangements shall be made for monitoring the levels of contamination of people, vehicles and goods moving out of areas with contamination, in order to control the spread of contamination and, as applicable, for the purposes of decontamination in accordance with the protection strategy (see Requirement 5). These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)) and shall take into consideration that some vehicles and items potentially with contamination, as well as members of the public and emergency workers, might have left these areas before the establishment of contamination control points and boundaries.

5.79. Arrangements shall be made for access control and enforcing of restrictions for areas in which evacuations and relocations would be carried out within emergency planning zones, the extended planning distance and the inner cordoned off area, in accordance with the protection strategy (see Requirement 5). Returns to these areas for short periods of time shall be permitted if justified (e.g. to feed animals left behind) and provided that those individuals entering the area are:

(a) Subject to controls and to dose assessment while in the area;
(b) Instructed on how to protect themselves;
(c) Briefed on the associated health hazards.

5.80. Arrangements shall be made to test methods of decontamination before their general use and to assess their effectiveness in terms of dose reduction.

5.81. For a transnational emergency in category IV, arrangements shall be made for taking early protective actions and other response actions as appropriate for areas beyond category V, including promptly conducting monitoring and assessment of contamination (a) of food, milk and drinking water and, as appropriate, of commodities other than food, and (b) of vehicles and cargoes that are likely to have contamination, with the aim of mitigating the consequences of a nuclear or radiological emergency and reassurance of the public. These arrangements shall include the use of pre-established operational criteria in accordance with the protection strategy (see para. 4.28(4)).

5.82. Monitoring in response to a nuclear or radiological emergency shall be carried out on the basis of a strategy to be developed at the preparedness stage as part of the protection strategy. Arrangements shall be made to adjust the monitoring in the emergency response on the basis of prevailing conditions.

5.83. Arrangements shall be made to carry out retrospective assessment of exposure of members of the public in a nuclear or radiological emergency, and to make the results of these assessments publicly available. The assessments shall be based on the best available information, shall be put into perspective in terms of the associated health hazards (see para. 5.72) and shall be promptly updated in the light of information that would yield substantially more accurate results.

Requirement 15: Managing radioactive waste in an emergency

The government shall ensure that radioactive waste is managed safely and effectively in a nuclear or radiological emergency.

5.84. The national policy and strategy for radioactive waste management [19] shall apply for radioactive waste generated in a nuclear or radiological emergency, with account taken of paras 5.85 to 5.88.

5.85. The protection strategy (see Requirement 5) shall take into account radioactive waste that might arise from protective actions and other response actions that are to be taken.

5.86. Radioactive waste arising in a nuclear or radiological emergency, including radioactive waste arising from associated protective actions and other response actions taken, shall be identified, characterized and categorized in due time and shall be managed in a manner that does not compromise the protection strategy, with account taken of prevailing conditions as these evolve.

5.87. Arrangements shall be made for radioactive waste to be managed safely and effectively. These arrangements shall include:

(a) A plan to characterize waste, including in situ measurements and analysis of samples;
(b) Criteria for categorization of waste;
(c) Avoiding, to the extent possible, the mixing of waste of different categories;
(d) Minimizing the amount of material unduly declared as radioactive waste;

(e) A method for determining appropriate options for predisposal management of radioactive waste (including processing, storage and transport), with account taken of the interdependences between all steps as well as impacts on the anticipated end points (clearance, authorized discharge, reuse, recycling, disposal) [19, 20];
(f) A method of identifying appropriate storage options and sites;
(g) Consideration of non-radiological aspects of waste (e.g. chemical properties such as toxicity, and biological properties).

5.88. Consideration shall be given to the management of human remains and animal remains with contamination as a result of a nuclear or radiological emergency, with due account taken of religious practices and cultural practices.

Requirement 16: Mitigating non-radiological consequences of a nuclear or radiological emergency and of an emergency response

The government shall ensure that arrangements are in place for mitigation of non-radiological consequences of a nuclear or radiological emergency and of an emergency response.

5.89. Non-radiological consequences of a nuclear or radiological emergency and of an emergency response shall be taken into consideration in deciding on the protective actions and other response actions to be taken in the context of the protection strategy (see Requirement 5).

5.90. Arrangements shall be made for mitigating the non-radiological consequences of an emergency and those of an emergency response and for responding to public concern in a nuclear or radiological emergency. These arrangements shall include arrangements for providing the people affected with:

(a) Information on any associated health hazards and clear instructions on any actions to be taken (see Requirement 10 and Requirement 13);
(b) Medical and psychological counselling, as appropriate;
(c) Adequate social support, as appropriate.

5.91. Arrangements shall be made to mitigate the impacts on international trade of a nuclear or radiological emergency and associated protective actions and other response actions, with account taken of the generic criteria in Appendix II. These arrangements shall provide for issuing information to the public and interested parties (such as importing States) on controls put in place in relation to traded

commodities, including food, and on vehicles and cargoes being shipped, and on any revisions of the relevant national criteria.

5.92. Arrangements shall be put in place for any actions taken, beyond those emergency response actions that are warranted, by members of the public and by commercial, industrial, infrastructural or other governmental or non-governmental bodies to be, to the extent practicable, promptly identified and appropriately addressed. This shall include the designation of organization(s) with the responsibility for monitoring for, identifying and addressing such actions.

Requirement 17: Requesting, providing and receiving international assistance for emergency preparedness and response

The government shall ensure that adequate arrangements are in place to benefit from, and to contribute to the provision of, international assistance for preparedness and response for a nuclear or radiological emergency.

5.93. Governments and international organizations shall put in place and shall maintain arrangements to respond in a timely manner to a request made by a State, in accordance with established mechanisms and respective mandates, for assistance in preparedness and response for a nuclear or radiological emergency.

5.94. Arrangements shall be put in place and maintained for requesting and obtaining international assistance from States or international organizations and for providing assistance to States (either directly or through the IAEA) in preparedness and response for a nuclear or radiological emergency, on the basis of international instruments (e.g. the Assistance Convention [13]), bilateral agreements or other mechanisms. These arrangements shall take due account of compatibility requirements for the capabilities to be obtained from and to be rendered to different States so as to ensure the usefulness of these capabilities.

Requirement 18: Terminating a nuclear or radiological emergency

The government shall ensure that arrangements are in place and are implemented for the termination of a nuclear or radiological emergency, with account taken of the need for the resumption of social and economic activity.

5.95. Adjustment of protective actions and other response actions and of other arrangements that are aimed at enabling the termination of an emergency shall be made by a formal process that includes consultation of interested parties.

5.96. Arrangements for communication with the public in a nuclear or radiological emergency (see Requirement 13) shall include arrangements for communication on the reasons for any adjustment of protective actions and other response actions and other arrangements aimed at enabling the termination of the emergency. This shall include providing the public with information on the need for any continuing protective actions following termination of the emergency and on any necessary modifications to their personal behaviour. Arrangements shall be made, during this period, to closely monitor public opinion and the reaction in the news media in order to ensure that any concerns can be promptly addressed. These arrangements shall ensure that any information provided to the public puts health hazards in perspective (see para. 5.72).

5.97. The termination of a nuclear or radiological emergency shall be based on a formal decision that is made public and shall include prior consultation with interested parties, as appropriate.

5.98. Both radiological consequences and non-radiological consequences shall be considered in deciding on the termination of an emergency as well as in the justification and optimization of further protection strategies as necessary.

5.99. The transition to an existing exposure situation or to a planned exposure situation shall be made in a coordinated and orderly manner, by making any necessary transfer of responsibilities and with the increased involvement of relevant authorities and interested parties.

5.100. The government shall ensure that, as part of its emergency preparedness, arrangements are in place for the termination of a nuclear or radiological emergency. The arrangements shall take into account that the termination of an emergency might be at different times in different geographical areas. The planning process shall include as appropriate:

(a) The roles and functions of organizations;
(b) Methods of transferring information;
(c) Means for assessing radiological consequences and non-radiological consequences;
(d) Conditions, criteria and objectives to be met for enabling the termination of a nuclear or radiological emergency (see Appendix II);
(e) A review of the hazard assessment and of the emergency arrangements;
(f) Establishment of national guidelines for the termination of an emergency;

(g) Arrangements for continued communication with the public, and for monitoring of public opinion and the reaction in the news media;
(h) Arrangements for consultation of interested parties.

5.101. Once the emergency is terminated, all workers undertaking relevant work shall be subject to the relevant requirements for occupational exposure in planned exposure situations [8], and individual monitoring, environmental monitoring and health surveillance shall be conducted subject to the requirements for planned exposure situations or existing exposure situations, as appropriate [8].

Requirement 19: Analysing the nuclear or radiological emergency and the emergency response

The government shall ensure that the nuclear or radiological emergency and the emergency response are analysed in order to identify actions to be taken to avoid other emergencies and to improve emergency arrangements.

5.102. Arrangements shall be made to document, protect and preserve, in an emergency response, to the extent practicable, data and information important for an analysis of the nuclear or radiological emergency and the emergency response. Arrangements shall be made to undertake a timely and comprehensive analysis of the nuclear or radiological emergency and the emergency response with the involvement of interested parties. These arrangements shall give due consideration to the need for making contributions to relevant internationally coordinated analyses and for sharing the findings of the analysis with relevant response organizations. The analysis shall give due consideration to:

(a) The reconstruction of the circumstances of the emergency;
(b) The root causes of the emergency;
(c) Regulatory controls including regulations and regulatory oversight;
(d) General implications for safety, including the possible involvement of other sources or devices (including those in other States);
(e) General implications for nuclear security, as appropriate;
(f) Necessary improvements to emergency arrangements;
(g) Necessary improvements to regulatory control.

5.103. Arrangements shall be made to enable comprehensive interviews on the circumstances of the nuclear or radiological emergency to be conducted with those involved.

5.104. Arrangements shall be made to acquire (e.g. from the IAEA, from another State or from the manufacturer of relevant equipment) the expertise necessary to conduct an analysis of the circumstances of the nuclear or radiological emergency.

5.105. Arrangements shall be made to take actions promptly on the basis of an analysis to avoid other emergencies, including provision of information to other operating organizations, as relevant, or to other States, directly or through the IAEA.

6. REQUIREMENTS FOR INFRASTRUCTURE

GENERAL

6.1. This section establishes the requirements for infrastructural elements that are essential to providing the capability for fulfilling the requirements established in Section 5 in accordance with the hazard assessment and the protection strategy.

Requirement 20: Authorities for emergency preparedness and response

The government shall ensure that authorities for preparedness and response for a nuclear or radiological emergency are clearly established.

6.2. The authorities for developing, maintaining and regulating arrangements, both on the site and off the site, for preparedness and response for a nuclear or radiological emergency shall be established by means of acts, legal codes or statutes.

6.3. All of the functions specified in Section 5 shall be assigned to the appropriate operating organizations and to local, regional and national response organizations. The involvement of all these organizations in the performance of these functions, or in support of their performance, shall be documented.[35] The documentation shall specify their roles, functions, authorities and responsibilities in emergency preparedness and response and shall

[35] Typically, the involvement of operating organizations and local, regional and national response organizations is documented as part of the appropriate facility, local, regional and national emergency plans.

assent to the authorities, roles and responsibilities of other response organizations. Conflicting or potentially conflicting and overlapping roles and responsibilities shall be identified and conflicts shall be resolved at the preparedness stage through the national coordinating mechanism (see para. 4.10).

6.4. The authority and responsibility for making decisions on response actions to be taken on the site and off the site (see para. 5.7) and the authority and responsibility for communication with the public shall be clearly assigned for each phase of the response.

6.5. The emergency arrangements shall include clear assignment of responsibilities and authorities, and shall provide for coordination and for communication in all phases of the response. These arrangements shall include:

— Ensuring that for each response organization a position in the response hierarchy has the authority and responsibility to direct and to coordinate its response actions;
— Clearly assigning the authority and responsibility for the direction and coordination of the entire response (see para. 5.7) and for the prevention and resolution of conflicts between response organizations;
— Assigning to an on-site position the authority and responsibility for notifying the appropriate response organization(s) of an emergency and for taking immediate on-site actions;
— Assigning to an on-site position the responsibility for directing the entire on-site emergency response (see paras 5.2 and 5.7).

These arrangements shall be such as to ensure that those personnel with authority and responsibility to perform critical response functions[36] in an emergency response are not assigned any other responsibilities in an emergency that would interfere with the prompt performance of the specified functions.

6.6. The arrangements for delegation and/or transfer of authority shall be specified in the relevant emergency plans, together with arrangements for notifying all appropriate parties of the transfer.

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[36] Critical response functions are functions that must be performed promptly and correctly in order to classify, declare and notify an emergency, to activate an emergency response, to manage the response, to take mitigatory actions, to protect emergency workers and to take urgent protective actions on and off the site.

Requirement 21: Organization and staffing for emergency preparedness and response

The government shall ensure that overall organization for preparedness and response for a nuclear or radiological emergency is clearly specified and staffed with sufficient personnel who are qualified and are assessed for their fitness for their intended duties.

6.7. The organizational relationships for preparedness and response for a nuclear or radiological emergency and interfaces between all the response organizations shall be established.

6.8. The positions responsible within each operating organization and response organization for performance of the response functions specified in Section 5 shall be assigned in the emergency plans and procedures. The positions responsible in each operating organization, in each response organization and in the regulatory body for the performance of activities at the preparedness stage, in accordance with these requirements, shall be assigned as part of the routine organizational structures and shall be specified, as appropriate, in the emergency plans and procedures.

6.9. Personnel who are assigned to positions in all operating organizations and response organizations to perform the functions necessary to meet the requirements established in Section 5 shall be qualified and shall be assessed for their initial fitness and continuing fitness for their intended duties.

6.10. Appropriate numbers of suitably qualified personnel shall be available at all times (including during 24 hour a day operations) so that appropriate positions can be promptly staffed as necessary following the declaration and notification of a nuclear or radiological emergency. Appropriate numbers of suitably qualified personnel shall be available for the long term to staff the various positions necessary to take mitigatory actions, protective actions and other response actions.

6.11. For a site where multiple facilities in category I or II are collocated, an appropriate number of suitably qualified personnel shall be available to manage an emergency response at all facilities if each of the facilities is under emergency conditions simultaneously (see para. 5.4).

Requirement 22: Coordination of emergency preparedness and response

The government shall ensure that arrangements are in place for the coordination of preparedness and response for a nuclear or radiological emergency between the operating organization and authorities at the local, regional and national levels, and, where appropriate, at the international level.

6.12. Arrangements shall be developed, as appropriate, for the coordination of emergency preparedness and response and of protocols for operational interfaces between operating organizations and authorities at the local, regional and national levels, including those organizations and authorities responsible for the response to conventional emergencies and to nuclear security events (see paras 4.3, 4.10, 6.3 and Requirement 6). The arrangements shall be clearly documented and the documentation shall be made available to all relevant parties. Arrangements shall be put in place to ensure effective working relationships among these organizations, both at the preparedness stage and in an emergency.

6.13. When several different organizations of the State or of other States are expected to have or to develop tools, procedures or criteria for use in the response to an emergency, arrangements for coordination shall be put in place to improve the consistency of the assessments of the situation, including assessments of contamination, doses and radiation induced health effects and any other relevant assessments made in a nuclear or radiological emergency, so as not to give rise to confusion.

6.14. Arrangements shall be made to coordinate with other States in the event of a transnational emergency any protective actions and other response actions that are recommended to their citizens and to their embassies in order either to ensure that they are consistent with those recommended in other States, or to provide an opportunity for them to explain to the public the basis for any differences (see para. 5.73).

6.15. Arrangements shall be made to ensure that States with areas in category V are provided with appropriate information for developing their own preparedness to respond to a transboundary emergency and that appropriate coordination across national borders is in place. These arrangements shall include:

(a) Agreements and protocols to provide information necessary to develop a coordinated means for notification, classification schemes and criteria for taking and for adjusting protective actions and other response actions;

(b) Arrangements for communication with the public;
(c) Arrangements for the exchange of information between decision making authorities.

Requirement 23: Plans and procedures for emergency response

The government shall ensure that plans and procedures necessary for effective response to a nuclear or radiological emergency are established.

6.16. Plans, procedures and other arrangements for effective emergency response, including coordinating mechanisms, letters of agreement or legal instruments, shall be made for coordinating a national emergency response. The arrangements for a coordinated national emergency response:

— Shall specify the organization responsible for the development and maintenance of the arrangements;
— Shall describe the responsibilities of operating organizations and other response organizations;
— Shall describe the coordination effected between these arrangements and the arrangements for response to a conventional emergency and to a nuclear security event.

Consideration shall be given in these plans, procedures and other arrangements to the need to protect information that might be confidential.

6.17. Each response organization shall prepare an emergency plan or plans for coordinating and performing their assigned functions as specified in Section 5 and in accordance with the hazard assessment and the protection strategy. An emergency plan shall be developed at the national level that integrates all relevant plans for emergency response in a coordinated manner and consistently with an all-hazards approach. Emergency plans shall specify how responsibilities for managing operations in an emergency response are to be discharged on the site, off the site and across national borders, as appropriate. The emergency plans shall be coordinated with other plans and procedures that may be implemented in a nuclear or radiological emergency, to ensure that the simultaneous implementation of the plans would not reduce their effectiveness or cause conflicts. Such other plans and procedures include:

(a) Emergency plans for facilities in category I and for areas in category V;
(b) Security plans and contingency plans [9, 10];

(c) Procedures for the investigation of a nuclear security event, including identification, collection, packaging and transport of evidence contaminated with radionuclides, nuclear forensics and related activities [11];
(d) Evacuation plans;
(e) Plans for firefighting.

6.18. The appropriate responsible authorities shall ensure that:

(a) A 'concept of operations'[37] for emergency response is developed at the beginning of the preparedness stage.
(b) Emergency plans and procedures are prepared and, as appropriate, approved for any facility or activity, area or location that could give rise to an emergency warranting protective actions and other response actions.
(c) Response organizations and operating organizations, as appropriate, are involved in the preparation of emergency plans and procedures, as appropriate.
(d) Account is taken in the content, features and extent of emergency plans of the results of any hazard assessment and any lessons from operating experience and from past emergencies, including conventional emergencies (see paras 4.18–4.26).
(e) Emergency plans and procedures are periodically reviewed and updated (see paras 6.36 and 6.38).

6.19. The operating organization of a facility or for an activity in category I, II, III or IV shall prepare an emergency plan. This emergency plan shall be coordinated with those of all other bodies that have responsibilities in a nuclear or radiological emergency, including public authorities, and shall be submitted to the regulatory body for approval.

6.20. The operating organization and response organizations shall develop the necessary procedures and analytical tools to be able to perform the functions specified in Section 5 for the goals of emergency response to be achieved and for the emergency response to be effective.

[37] A concept of operations is a brief description of an ideal response to a postulated nuclear or radiological emergency, used to ensure that all those personnel and organizations involved in the development of a capability for emergency response share a common understanding.

6.21. Procedures and analytical tools shall be tested under simulated emergency conditions and shall be validated prior to initial use. Any arrangements for the use of analytical tools early in an emergency response for supporting decision making on protective actions and other response actions shall be made in due recognition of the limitations[38] of such analytical tools and in a way that would not reduce the effectiveness of response actions. These limitations shall be made clear to, and shall be recognized by, those responsible for decision making.

Requirement 24: Logistical support and facilities for emergency response

The government shall ensure that adequate logistical support and facilities are provided to enable emergency response functions to be performed effectively in a nuclear or radiological emergency.

6.22. Adequate tools, instruments, supplies, equipment, communication systems, facilities and documentation (such as documentation of procedures, checklists, manuals, telephone numbers and email addresses) shall be provided for performing the functions specified in Section 5. These items and facilities shall be selected or designed to be operational under the conditions (such as radiological conditions, working conditions and environmental conditions) that could be encountered in the emergency response, and to be compatible with other procedures and equipment for the response (e.g. compatible with the communication frequencies used by other response organizations), as appropriate. These support items shall be located or provided in a manner that allows their effective use under the emergency conditions postulated.

6.23. For facilities in categories I and II, as contingency measures, alternative supplies for taking on-site mitigatory actions, such as an alternative supply of water and an alternative electrical power supply, including any necessary equipment, shall be ensured. This equipment shall be located and maintained so that it can be functional and readily accessible when needed (see also Safety of Nuclear Power Plants: Design (SSR-2/1) [18]).

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[38] An example of such limitations is that the timing and magnitude of radioactive releases in an emergency at a nuclear power plant that would warrant taking precautionary urgent protective actions and urgent protective actions off the site before, or shortly after, a radioactive release may not be predictable. In addition, the radioactive release could occur over several days, resulting in complex deposition patterns off the site.

6.24. Emergency response facilities or locations to support an emergency response under the full range of postulated hazardous conditions shall be designated and shall be assigned the following functions, as appropriate:

(a) Receiving notifications and initiating the response;
(b) Coordination and direction of on-site response actions;
(c) Providing technical and operational support to those personnel performing tasks at a facility and those personnel responding off the site;
(d) Direction of off-site response actions and coordination with on-site response actions;
(e) Coordination of national response actions;
(f) Coordination of communication with the public;
(g) Coordination of monitoring, sampling and analysis;
(h) Managing those people who have been evacuated (including reception, registration, monitoring and decontamination, as well as provision for meeting their personal needs, including for housing, food and sanitation);
(i) Managing the storage of necessary resources;
(j) Providing individuals who have undergone exposure or contamination with appropriate medical attention including medical treatment.

6.25. For facilities in category I, emergency response facilities[39] separate from the control room and supplementary control room shall be provided so that:

(a) Technical support can be provided to the operating personnel in the control room in an emergency (from a technical support centre).
(b) Operational control by personnel performing tasks at or near the facility can be maintained (from an operational support centre).
(c) The on-site emergency response is managed (from an emergency centre).

These emergency response facilities shall operate as an integrated system in support of the emergency response, without conflicting with one another's functions, and shall provide reasonable assurance of being operable and habitable under a range of postulated hazardous conditions, including conditions not considered in the design.

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[39] Emergency response facilities may be collocated (i.e. these functions may be performed from a single emergency response facility or location) provided that it is ensured that they do not conflict with each other in performing their specified functions and provided that they are separated from the control rooms.

6.26. Arrangements shall be made for performing appropriate and reliable analyses of samples[40] and measurements of internal contamination for the purposes of emergency response and of health screening, as appropriate. Such arrangements shall include the designation of laboratories that would be operational under postulated emergency conditions.

6.27. Arrangements shall be made to obtain appropriate support from organizations responsible for providing support in conventional emergencies for logistics and communication, for social welfare and in other areas.

Requirement 25: Training, drills and exercises for emergency preparedness and response

The government shall ensure that personnel relevant for emergency response shall take part in regular training, drills and exercises to ensure that they are able to perform their assigned response functions effectively in a nuclear or radiological emergency.

6.28. The operating organization and response organizations shall identify the knowledge, skills and abilities necessary to perform the functions specified in Section 5. The operating organization and response organizations shall make arrangements for the selection of personnel and for training to ensure that the personnel selected have the requisite knowledge, skills and abilities to perform their assigned response functions. The arrangements shall include arrangements for continuing refresher training on an appropriate schedule and arrangements for ensuring that personnel assigned to positions with responsibilities in an emergency response undergo the specified training.

6.29. For facilities in category I, II or III, all personnel and all other persons on the site shall be instructed in the arrangements for them to be notified of an emergency and of their actions if notified of an emergency.

6.30. Exercise programmes shall be developed and implemented to ensure that all specified functions required to be performed for emergency response, all organizational interfaces for facilities in category I, II or III, and the national level programmes for category IV or V are tested at suitable intervals. These programmes shall include the participation in some exercises of, as appropriate

[40] Arrangements for analyses could include, for example, arrangements for performing analyses of environmental and biological samples as well as analyses of other samples taken from the facility for the purpose of assessing its operational status.

and feasible, all the organizations concerned, people who are potentially affected, and representatives of news media. The exercises shall be systematically evaluated (see para. 4.10(h)) and some exercises shall be evaluated by the regulatory body. Programmes shall be subject to review and revision in the light of experience gained (see paras 6.36 and 6.38).

6.31. The personnel responsible for critical response functions shall participate in drills and exercises on a regular basis so as to ensure their ability to take their actions effectively.

6.32. Officials off the site who are responsible for making decisions on protective actions and other response actions shall be trained and shall regularly participate in exercises. Officials off the site who are responsible for communication with the public in a nuclear or radiological emergency shall regularly participate in exercises.

6.33. The conduct of exercises shall be evaluated against pre-established objectives of emergency response to demonstrate that identification, notification, activation and response actions can be performed effectively to achieve the goals of emergency response (see para. 3.2).

Requirement 26: Quality management programme for emergency preparedness and response

The government shall ensure that a programme is established within an integrated management system to ensure the availability and reliability of all supplies, equipment, communication systems and facilities, plans, procedures and other arrangements necessary for effective response in a nuclear or radiological emergency.

6.34. The operating organization, as part of its management system (see Ref. [14]), and response organizations, as part of their emergency management system, shall establish a programme to ensure the availability and reliability of all supplies, equipment, communication systems and facilities, plans, procedures and other arrangements necessary to perform functions in a nuclear or radiological emergency as specified in Section 5 (see para. 6.22). The programme shall include arrangements for inventories, resupply, tests and calibrations, to ensure that these are continuously available and are functional for use in a nuclear or radiological emergency.

6.35. The programme shall also include periodic and independent appraisals against functions as specified in Section 5, including participation in international appraisals[41].

6.36. Arrangements shall be made to maintain, review and update emergency plans, procedures and other arrangements and to incorporate lessons from research, operating experience (such as in the response to emergencies) and emergency exercises.

6.37. The operating organization and response organizations shall establish and maintain adequate records in relation to both emergency arrangements and the response to a nuclear or radiological emergency, to include dose assessments, results of monitoring and inventory of radioactive waste managed, in order to allow for their review and evaluation. These records shall also provide for the identification of those persons requiring longer term medical actions, as necessary, and shall provide for the long term management of radioactive waste.

6.38. The operating organization and response organizations shall make arrangements to review and evaluate responses in actual events and in exercises, in order to record the areas in which improvements are necessary and to ensure that the necessary improvements are made (see Requirement 19).

6.39. Relevant international organizations shall review and update their applicable standards and guidelines and their relevant arrangements in emergency preparedness and response on the basis of research and lessons from the response to actual emergencies and in emergency exercises.

[41] Examples of international appraisals include those organized by the IAEA, such as Emergency Preparedness Review (EPREV) missions.

Appendix I

GUIDANCE VALUES FOR RESTRICTING EXPOSURE OF EMERGENCY WORKERS

I.1. This Appendix provides guidance values as a basis for operational guidance for restricting the exposure of emergency workers.

I.2. Table I.1 provides guidance values for restricting the exposure of emergency workers in an emergency response in terms of personal dose equivalent $H_p(10)$ from external exposure to strongly penetrating radiation. The values for $H_p(10)$ in Table I.1 assume that every effort has been made for protection against external exposure to weakly penetrating radiation and against exposure due to intakes or skin contamination (see para. 5.53).

I.3. The total effective dose and the relative biological effectiveness (RBE) weighted absorbed dose to a tissue or organ via all exposure pathways (i.e. both dose from external exposure and committed dose from intakes) need to be estimated as early as possible in a nuclear or radiological emergency. Table I.1 also provides guidance on the effective dose and the RBE weighted absorbed dose to a tissue or organ for consideration in restricting further exposure in the response to a nuclear or radiological emergency once these doses have been estimated.

I.4. Severe deterministic effects to a fetus could possibly occur following an equivalent dose to the fetus of greater than 100 mSv. Consequently, in the response to a nuclear or radiological emergency, female workers who are aware that they are pregnant or who might be pregnant need to be (1) informed of this risk and (2) excluded from taking actions that might result in an equivalent dose to the embryo and fetus exceeding 50 mSv for the full period of in utero development of the embryo and fetus.

TABLE I.1. GUIDANCE VALUES FOR RESTRICTING EXPOSURE OF EMERGENCY WORKERS

Tasks	Guidance value[a]		
	$H_p(10)$[b]	E[c]	AD_T[d]
	<500 mSv	<500 mSv	$<\frac{1}{2}AD_{T,\ \text{Table II.1}}$[e]
Lifesaving actions	This value may be exceeded — with due consideration of the generic criteria in Table II.1 of Appendix II — under circumstances in which the expected benefits to others clearly outweigh the emergency worker's own health risks, and the emergency worker volunteers to take the action and understands and accepts these health risks		
Actions to prevent severe deterministic effects and actions to prevent the development of catastrophic conditions that could significantly affect people and the environment	<500 mSv	<500 mSv	$<\frac{1}{2}AD_{T,\ \text{Table II.1}}$
Actions to avert a large collective dose	<100 mSv	$E < 100$ mSv	$<\frac{1}{10}AD_{T,\ \text{Table II.1}}$

[a] These values are set to be two to ten times lower than the generic criteria in Table II.1 of Appendix II and they apply for:

(a) The dose from external exposure to strongly penetrating radiation for $H_p(10)$. Doses from external exposure to weakly penetrating radiation and from intake or skin contamination need to be prevented by all possible means. If this is not feasible, the effective dose and the RBE weighted absorbed dose to a tissue or organ have to be limited to minimize the health risk to the individual in line with the risk associated with the guidance values given here.

(b) The total effective dose E and the RBE weighted absorbed dose to a tissue or organ AD_T via all exposure pathways (i.e. both dose from external exposure and committed dose from intakes) which are to be estimated as early as possible in order to enable any further exposure to be restricted as appropriate.

[b] Personal dose equivalent $H_p(d)$ where $d = 10$ mm.

[c] Effective dose.

[d] RBE weighted absorbed dose to a tissue or organ.

[e] Values of RBE weighted absorbed dose to a tissue or organ given in Table II.1 of Appendix II.

Appendix II

GENERIC CRITERIA FOR USE IN EMERGENCY PREPAREDNESS AND RESPONSE

II.1. This Appendix provides generic criteria:

(a) For doses for which protective actions and other response actions are expected to be undertaken under any circumstances in a nuclear or radiological emergency to avoid or to minimize severe deterministic effects;
(b) For doses for which protective actions and other response actions are expected to be taken, if they can be taken safely, in a nuclear or radiological emergency to reasonably reduce the risk of stochastic effects;
(c) For doses for which restriction of international trade is warranted in a nuclear or radiological emergency, with due consideration of non-radiological consequences;
(d) For doses for use as a target dose for the transition to an existing exposure situation.

Appendix II includes examples of associated protective actions and other response actions. These generic criteria and associated protective actions and other response actions shall be taken into account in the development of the protection strategy, including national generic criteria in accordance with Requirement 5. If protective actions in the context of the protection strategy are to be taken when doses are below the generic criteria given in this Appendix, careful consideration is necessary to ensure that such actions are justified (i.e. that they do more good than harm) and that they are optimized in accordance with Requirement 5.

GENERIC CRITERIA FOR DOSES RECEIVED WITHIN A SHORT PERIOD OF TIME FOR WHICH RESPONSE ACTIONS ARE EXPECTED TO BE TAKEN UNDER ANY CIRCUMSTANCES IN AN EMERGENCY RESPONSE

II.2. Table II.1 provides generic criteria for doses received within a short period of time for which protective actions and other response actions are expected to be taken under any circumstances in a nuclear or radiological emergency to avoid or to minimize severe deterministic effects.

TABLE II.1. GENERIC CRITERIA FOR DOSES RECEIVED WITHIN A SHORT PERIOD OF TIME FOR WHICH PROTECTIVE ACTIONS AND OTHER RESPONSE ACTIONS ARE EXPECTED TO BE TAKEN UNDER ANY CIRCUMSTANCES IN AN EMERGENCY TO AVOID OR TO MINIMIZE SEVERE DETERMINISTIC EFFECTS

Acute external exposure (<10 h)		
$AD_{\text{red marrow}}$[a]	1 Gy	If the dose is projected: — Take precautionary urgent protective actions immediately (even under difficult conditions) to keep doses below the generic criteria; — Provide public information and warnings; — Carry out urgent decontamination.
AD_{fetus}	0.1[b] Gy	
AD_{tissue}[c]	25 Gy at 0.5 cm	
AD_{skin}[d]	10 Gy to 100 cm^2	
Acute internal exposure due to an acute intake ($\Delta = 30$ d[e])		
$AD(\Delta)_{\text{red marrow}}$	0.2 Gy for radionuclides with atomic number $Z \geq 90$[f] 2 Gy for radionuclides with atomic number $Z \leq 89$[f]	If the dose has been received: — Perform immediate medical examination, medical consultation and indicated medical treatment; — Carry out contamination control; — Carry out immediate decorporation[g] (if applicable); — Conduct registration for longer term medical follow-up; — Provide comprehensive psychological counselling.
$AD(\Delta)_{\text{thyroid}}$	2 Gy	
$AD(\Delta)_{\text{lung}}$[h]	30 Gy	
$AD(\Delta)_{\text{colon}}$	20 Gy	
$AD(\Delta')_{\text{fetus}}$[i]	0.1[b] Gy	

[a] $AD_{\text{red marrow}}$ represents the average RBE weighted absorbed dose to internal tissues or organs (e.g. red marrow, lung, small intestine, gonads, thyroid) and to the lens of the eye from exposure in a uniform field of strongly penetrating radiation.

[b] At 0.1 Gy there would be only a very small probability of severe deterministic effects to the fetus and only during certain periods post-conception (e.g. between 8 and 15 weeks of in utero development), and only if the dose is received at high dose rates. During other periods post-conception and for lower dose rates, the fetus is less sensitive. There is a high probability of severe deterministic effects at 1 Gy. Therefore, 1 Gy is used as the generic criterion for doses to the fetus received within a short period of time: (i) in the hazard assessment (see para. 4.23), to identify facilities and activities, on-site areas, off-site areas and locations for which a nuclear or radiological emergency could warrant precautionary urgent protective actions to avoid or to minimize severe deterministic effects; (ii) for identifying situations in which exposure is dangerous to health; and (iii) for making arrangements (see para. 5.38) for applying decisions on urgent protective actions and other response actions to be taken off the site to avoid or to minimize the occurrence of severe deterministic effects (e.g. establishing a precautionary action zone).

[c] Dose delivered to 100 cm^2 at a depth of 0.5 cm under the body surface in tissue due to close contact with a radioactive source (e.g. source carried in the hand or pocket).

[d] The dose is to the 100 cm^2 dermis (skin structures at a depth of 40 mg/cm^2 (or 0.4 mm) below the surface).

[e] $AD(\Delta)$ is the RBE weighted absorbed dose delivered over a period of time Δ by the intake (I_{05}) that will result in a severe deterministic effect in 5% of exposed individuals. This dose is calculated as described in appendix I of Ref. [21].

[f] Different generic criteria are used to take account of the significant difference in RBE weighted absorbed dose from exposure at the intake threshold values specific for these two groups of radionuclides.

[g] Decorporation is the action of the biological processes, facilitated by chemical or biological agents, by means of which incorporated radionuclides are removed from the human body. The generic criterion for decorporation is based on the projected dose without decorporation.

[h] For the purposes of these generic criteria, 'lung' means the alveolar–interstitial region of the respiratory tract.

[i] For this particular case, 'Δ' refers to the period of in utero development of the embryo and fetus.

GENERIC CRITERIA FOR PROTECTIVE ACTIONS AND OTHER RESPONSE ACTIONS TO REDUCE THE RISK OF STOCHASTIC EFFECTS

II.3. Table II.2 provides generic criteria for taking protective actions and other response actions in a nuclear or radiological emergency to reduce the risk of stochastic effects.

TABLE II.2. GENERIC CRITERIA FOR PROTECTIVE ACTIONS AND OTHER RESPONSE ACTIONS IN AN EMERGENCY TO REDUCE THE RISK OF STOCHASTIC EFFECTS

Generic criteria		Examples of protective actions and other response actions[a]
Projected dose that exceeds the following generic criteria: Take urgent protective actions and other response actions		
$H_{thyroid}$	50 mSv[b] in the first 7 days	Iodine thyroid blocking[c]
E[d]	100 mSv in the first 7 days	Sheltering[e]; evacuation; prevention of inadvertent ingestion; restrictions on food, milk and drinking water[g] and restrictions on the food chain and water supply; restrictions on commodities other than food; contamination control; decontamination; registration; reassurance of the public
H_{fetus}[f]	100 mSv in the first 7 days	
Projected dose that exceeds the following generic criteria: Take early protective actions and other response actions		
E[d]	100 mSv in the first year	Temporary relocation; prevention of inadvertent ingestion; restrictions on food, milk and drinking water[g] and restrictions on the food chain and water supply; restrictions on commodities other than food; contamination control; decontamination; registration; reassurance of the public
H_{fetus}[f]	100 mSv for the full period of in utero development	

TABLE II.2. GENERIC CRITERIA FOR PROTECTIVE ACTIONS AND OTHER RESPONSE ACTIONS IN AN EMERGENCY TO REDUCE THE RISK OF STOCHASTIC EFFECTS (cont.)

Generic criteria		Examples of protective actions and other response actions[a]
Dose that has been received and that exceeds the following generic criteria: Take longer term medical actions to detect and to effectively treat radiation induced health effects		
E[d]	100 mSv in a month	Health screening based on equivalent doses to specific radiosensitive organs (as a basis for longer term medical follow-up)[h], registration, counselling
H_{fetus}[f]	100 mSv for the full period of in utero development	Counselling to allow informed decisions to be made in individual circumstances

[a] These examples are neither exhaustive nor grouped in a mutually exclusive way.

[b] The equivalent dose to the thyroid ($H_{thyroid}$) only due to exposure to radioiodine.

[c] This generic criterion applies only for administration of iodine thyroid blocking. For the thyroid, iodine thyroid blocking is an urgent protective action that is prescribed: (a) if exposure due to radioactive iodine is involved, (b) before or shortly after a release of radioactive iodine, and (c) within only a short period before or after the intake of radioactive iodine.

[d] Effective dose.

[e] As a less disruptive protective action, sheltering may be ordered at lower doses as long as justified and optimized in accordance with Requirement 5 with due consideration of the reference level in para. 4.28(2).

[f] H_{fetus} is the equivalent dose to the fetus, derived as the sum of the dose from external exposure and the maximum committed equivalent dose to any organ of the embryo or fetus from intake to the embryo or fetus for different chemical compounds and different times relative to conception.

[g] Restrictions on food, milk and drinking water using these generic criteria are to be applied before sampling and analysis of food, milk and drinking water are carried out. Such restrictions apply as long as replacements of food, milk and drinking water or other alternatives are available to ensure they would not result in severe malnutrition, dehydration or other severe health impacts.

[h] When results of the health screening indicate that the criteria in Table II.1 are exceeded, then appropriate medical attention on the basis of Appendix II (see Table II.1) is necessary.

GENERIC CRITERIA FOR FOOD, MILK AND DRINKING WATER AND OTHER COMMODITIES TO REDUCE THE RISK OF STOCHASTIC EFFECTS

II.4. Table II.3 provides generic criteria for taking protective actions and other response actions to reduce the risk of stochastic effects from the ingestion of food, milk and drinking water and from the use of other commodities in a nuclear or radiological emergency.

II.5. A value of 1/10 of the generic criteria given in Table II.2 for early protective actions and other response actions is established as generic criteria for restrictions on food, milk and drinking water and on other commodities to ensure that the dose via all exposure pathways, including ingestion, will not exceed the generic criteria given in Table II.2 for early protective actions and other response actions.

II.6. If restrictions on food, milk or drinking water would result in severe malnutrition or dehydration because replacements are not available, food, milk or drinking water with concentration levels of radionuclides that are projected to result in doses above the generic criteria given in Table II.3 may be consumed until replacements are available provided that this would not result in doses from all exposure pathways above the generic criteria given in Table II.2; otherwise, the people affected may be relocated.

TABLE II.3. GENERIC CRITERIA FOR FOOD, MILK AND DRINKING WATER AND OTHER COMMODITIES TO REDUCE THE RISK OF STOCHASTIC EFFECTS

Generic criteria		Examples of protective actions and other response actions
Projected dose from ingestion of food, milk and drinking water and from the use of other commodities that exceeds the following generic criteria: Take protective actions and other response actions		
E[a]	10 mSv in the first year	Restrict consumption, distribution and sale of non-essential[b] food, milk and drinking water[c] and restrict the use and distribution of other commodities. Replace essential food, milk and drinking water as soon as possible or relocate the people affected if replacements are not available. Estimate the doses of those who might have consumed food, milk and drinking water or used other commodities to determine whether this may have resulted in doses warranting medical attention in accordance with Table II.2.
H_{fetus}[d]	10 mSv for the full period of in utero development	

[a] Effective dose.

[b] Restricting essential food, milk or drinking water could result in dehydration, severe malnutrition or other severe health impacts; therefore, essential food, milk and drinking water is to be restricted only if alternatives are available.

[c] These criteria for taking actions on food, milk and drinking water are applied once the sampling and analysis of food, milk and drinking water is carried out. This would also provide a basis for discontinuing restrictions imposed on food, milk and drinking water as a precaution on the basis of the generic criteria in Table II.2.

[d] H_{fetus} is the equivalent dose to the fetus derived as the sum the dose from external exposure and the maximum committed equivalent dose to any organ from intake to the embryo or fetus for different chemical compounds and different times relative to conception.

GENERIC CRITERIA FOR VEHICLES, EQUIPMENT AND OTHER ITEMS TO REDUCE THE RISK OF STOCHASTIC EFFECTS

II.7. Table II.4 provides generic criteria for taking protective actions and other response actions to reduce the risk of stochastic effects arising from the use of vehicles, equipment and other items from an area affected by a nuclear or radiological emergency.

II.8. A value of 1/10 of the generic criteria given in Table II.2 for early protective actions and other response actions is established as generic criteria for vehicles, equipment and other items from an affected area, to ensure that the dose via all exposure pathways, including the use of such vehicles, equipment and other items, would not exceed the generic criteria given in Table II.2 for early actions for a member of the public.

II.9. Restricting the use of vehicles, equipment and other items from an affected area could interfere with taking urgent protective actions and other response actions or with providing services essential to public health or well-being (e.g. restricting the use of vehicles for transferring individuals requiring critical medical treatment or preventing a ship or an aircraft that has left an affected area from reaching its final destination). Such vehicles, equipment and other items whose use would give rise to a projected dose to their users above the generic criteria given in Table II.4 may be used until replacements are available, provided that:

(a) Their use will not result in doses from all exposure pathways that exceed the generic criteria given in Table II.2 for members of the public or the guidance values given in Appendix I for restricting the exposure of emergency workers, or the restriction set in para. 5.57 for exposures of helpers in an emergency.
(b) Actions are taken to manage and control the exposure of the user as an emergency worker, a helper in an emergency or a member of the public, as appropriate.

TABLE II.4. GENERIC CRITERIA FOR VEHICLES, EQUIPMENT AND OTHER ITEMS TO REDUCE THE RISK OF STOCHASTIC EFFECTS

Generic criteria		Examples of protective actions and other response actions
Projected dose from the use of vehicles, equipment or other items from an affected area that exceed the following generic criteria: Take protective actions and other response actions		
E[a]	10 mSv in the first year	Restrict non-essential[b] use. Use essential vehicles, equipment and other items from an affected area until replacements are available provided that: (a) their use will not result in doses from all exposure pathways exceeding the generic criteria given in Table II.2 for a member of the public or the guidance values given in Appendix I for restricting the exposure of emergency workers, or the restriction set in para. 5.57 for exposures of helpers in an emergency; and (b) actions are taken to control the dose to the user as an emergency worker, helper in an emergency or a member of the public, as appropriate. Estimate doses to those emergency workers, helpers in an emergency and members of the public who may have used a vehicle, equipment or other item from an affected area to determine whether this could have resulted in a dose warranting medical attention in accordance with Table II.2.
H_{fetus}[c]	10 mSv for the full period of in utero development	

[a] Effective dose.

[b] Restricting the use of vehicles, equipment and other items from an affected area could interfere with taking urgent protective actions and other response actions or with providing services essential to public health or well-being (e.g. restricting the use of vehicles for transferring individuals requiring critical medical treatment).

[c] H_{fetus} is the equivalent dose to the fetus derived as the sum of the dose from external exposure and the maximum committed equivalent dose to any organ from intake to the embryo or fetus for different chemical compounds and different times relative to conception.

GENERIC CRITERIA FOR FOOD AND OTHER COMMODITIES TRADED INTERNATIONALLY

II.10. Table II.5 provides generic criteria aimed at the effective implementation of response actions to reduce the non-radiological consequences of a nuclear or radiological emergency by providing a basis for the continuation or the resumption of international trade.

II.11. Values that exceed the generic criteria in Table II.5 may be acceptable under emergency (temporary) conditions.

II.12 The generic criteria for food traded internationally derive from the level used by the Joint FAO/WHO Codex Alimentarius Commission [22]. These generic criteria, and generic criteria for other commodities traded internationally that could contain radionuclides following a nuclear or radiological emergency, are established at 1/100 of the generic criteria given in Table II.2 for early protective actions and other response actions to ensure that doses to the public would be a small fraction of those for which actions are warranted to reduce the risk of stochastic effects.

II.13. For food traded internationally that could contain radionuclides following a nuclear or radiological emergency, the operational criteria (i.e. guideline levels) as published by the Joint FAO/WHO Codex Alimentarius Commission [22] may ultimately be used (see para. 5.23 of GSR Part 3 [8]).

II.14. If restricting trade in food and other commodities could result in severe health impacts or other detrimental effects in another State, then the food and other commodities that would give rise to a projected dose that exceeds the generic criteria in Table II.5 may be traded — if the trade is justified — until replacements are available, provided that:

(a) The trade is approved with the receiving State.
(b) The trade will not result in doses that exceed the generic criteria for the public given in Table II.2 and Table II.3.
(c) Actions are taken to manage and control exposures during shipping.
(d) Actions are taken to control the consumption of food and use of other commodities and to reduce the exposure of members of the public.

TABLE II.5. GENERIC CRITERIA FOR FOOD AND OTHER COMMODITIES TRADED INTERNATIONALLY

Generic criteria		Examples of other response actions
Projected dose from food and other commodities that exceed the generic criteria: Take response actions to restrict international trade.		
E[a]	1 mSv per year	Restrict non-essential[b] international trade. Trade essential food and other commodities until replacements are available if: (a) trade is approved with the receiving State; (b) trade will not result in doses to the public that exceed the generic criteria given in Table II.2 for all exposure pathways and in Table II.3 for the respective pathways; (c) actions are taken to manage and control the dose during shipping; and (d) actions are taken to control the consumption and use of food and other commodities and to reduce the exposure of members of the public.
H_{fetus}[c]	1 mSv for the full period of in utero development	

[a] Effective dose.

[b] Restricting the trade of essential commodities and food could result in severe health impacts or other detrimental conditions in another State.

[c] H_{fetus} is the equivalent dose to the fetus derived as the sum of the dose from external exposure and the maximum committed equivalent dose to any organ from intake to the embryo or fetus for different chemical compounds and different times relative to conception.

GENERIC CRITERIA FOR ENABLING A TRANSITION TO AN EXISTING EXPOSURE SITUATION

II.15. Generic criteria shall be established in terms of the projected dose for the implementation of protective actions and other actions aimed at enabling the termination of a nuclear or radiological emergency and the subsequent transition to an existing exposure situation with due consideration of, and verification of the fulfilment of, the conditions set out in para. II.16. These criteria are established as 1/5 of the generic criteria for the early protective actions and other response actions given in Table II.2[42] and are:

(a) An effective dose of 20 mSv per year;
(b) An equivalent dose to a fetus of 20 mSv for the full period of in utero development.

II.16. The decision to terminate the nuclear or radiological emergency and the subsequent transition to an existing exposure situation is to be taken after:

(a) Justified actions (see para. 4.29) have been taken to reach the generic criteria[43] for enabling the transition to an existing exposure situation and it has been confirmed that any further actions to reach these criteria would do more harm than good;
(b) Confirmation that the source of exposure is fully characterized for all members of the public living as normal in the area;
(c) The situation with regard to exposure has been understood and has remained stable;

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[42] Criteria established as 1/5 of the generic criteria for the early protective actions and other response actions given in Table II.2 are considered to be generically justified. This is of the order of the dose for which the government is required to establish an action plan to reduce activity concentrations for sources of exposure (e.g. ^{222}Rn) for existing exposure situations [8]. Being at the lower bound of the reference level for emergency exposure situations (see para. 4.28 (2)), this level is also consistent with the reference levels established in GSR Part 3 [8] for both emergency exposure situations and existing exposure situations.

[43] Actions taken (see para. 4.29) to reach the generic criteria in para. II.15 need to be justified and optimized in accordance with Requirement 5. However, it may not be feasible to reach these criteria for enabling the transition to an existing exposure situation. If it is not feasible or justified to reach these generic criteria, the transition may still be possible as long as the generic criteria for early protective actions and other response actions given in Table II.2 are not exceeded.

(d) Any restrictions on normal living conditions are limited and provisions are in place to confirm compliance with such restrictions;

(e) Confirmation that interested parties, including the public, have been consulted and are being kept informed about the basis for the adjustment of emergency response actions and for the transition, with the associated health hazards put into perspective (see para. 5.72).

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Annex

APPLICABILITY OF PARAGRAPHS IN THIS PUBLICATION BY EMERGENCY PREPAREDNESS CATEGORY

A–1. Table A–1 presents the applicability of paragraphs in this publication by emergency preparedness category.

TABLE A–1. APPLICABILITY OF PARAGRAPHS IN THIS PUBLICATION BY EMERGENCY PREPAREDNESS CATEGORY

<table>
<tr><th>Category</th><th colspan="5">Paragraphs applicable in this publication</th></tr>
<tr><td>I</td><td rowspan="5">1.1–1.17
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3.1–3.2
4.1–4.10, 4.18–4.20, 4.22–4.25, 4.27–4.31
5.1, 5.6–5.8, 5.11, 5.18–5.20, 5.22, 5.31, 5.36–5.37, 5.48–5.61, 5.64, 5.67–5.75, 5.78–5.80, 5.82–5.105
6.1–6.10, 6.12–6.14, 6.16–6.18, 6.20–6.22, 6.24, 6.26–6.28, 6.30–6.39</td><td rowspan="4">4.11–4.17, 4.26
5.3, 5.5, 5.14–5.17, 5.23–5.24, 5.26
6.19</td><td rowspan="2">5.2, 5.25, 5.27, 5.32–5.34, 5.41–5.43, 5.65
6.29</td><td>5.4, 5.9, 5.12, 5.21, 5.38–5.40, 5.45–5.46, 5.66, 5.76–5.77
6.11, 6.15, 6.23</td><td>6.25</td></tr>
<tr><td>II</td><td rowspan="3">5.10, 5.47</td><td rowspan="4"></td></tr>
<tr><td>III</td></tr>
<tr><td>IV</td><td>4.21
5.13, 5.28–5.30, 5.35, 5.44, 5.62–5.63, 5.81</td></tr>
<tr><td>V</td><td colspan="3">5.9, 5.12, 5.21, 5.39–5.40, 5.45–5.46, 5.66, 5.76–5.77
6.15</td></tr>
</table>

DEFINITIONS

The following definitions apply for the purposes of this safety standard. The symbol 'ⓘ' denotes an information note; this note is not part of the definition.

arrangements. See '(emergency) arrangements'.

authorization. The granting by a regulatory body or other governmental body of written permission for a person or organization (the operator) to conduct specified activities.

control. The function or power or (usually as controls) means of directing, regulating or restraining.

ⓘ It should be noted that the usual meaning of the English word control in safety related contexts is somewhat 'stronger' (more active) than that of its usual translations and other similar words in some other languages. For example, 'control' typically implies not only checking or monitoring something but also ensuring that corrective or enforcement measures are taken if the results of the checking or monitoring indicate such a need. This is in contrast, for example, to the more limited usage of the equivalent word in French and Spanish.

regulatory control. Any form of control or regulation applied to facilities and activities by a regulatory body for reasons relating to nuclear safety and radiation protection or to nuclear security.

ⓘ In the Nuclear Security Series[1] , the phrase 'out of regulatory control' is used to describe a situation in which nuclear material or other radioactive material is present without an appropriate authorization, either because controls have failed for some reason, or because they never existed.

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[1] EUROPEAN POLICE OFFICE, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL CIVIL AVIATION ORGANIZATION, INTERNATIONAL CRIMINAL POLICE ORGANIZATION–INTERPOL, UNITED NATIONS INTERREGIONAL CRIME AND JUSTICE RESEARCH INSTITUTE, UNITED NATIONS OFFICE ON DRUGS AND CRIME, WORLD CUSTOMS ORGANIZATION, Nuclear Security Recommendations on Nuclear and Other Radioactive Material out of Regulatory Control, IAEA Nuclear Security Series No. 15, IAEA, Vienna (2011).

dangerous source. See 'source'.

deterministic effect. A radiation induced health effect for which generally a threshold level of dose exists above which the severity of the effect is greater for a higher dose.

ⓘ Such an effect is described as a **severe deterministic effect** if it is fatal or life threatening or results in a permanent injury that reduces quality of life.

ⓘ The level of the threshold dose is characteristic of the particular health effect but may also depend, to a limited extent, on the exposed individual. Examples of deterministic effects include erythema, damage to the haemopoietic system and acute radiation syndrome (radiation sickness). Deterministic effects are also referred to as 'harmful tissue reactions'.

early protective actions. See 'protective actions'.

emergency. A non-routine situation or event that necessitates prompt action, primarily to mitigate a hazard or adverse consequences for human life, health, property or the environment.

ⓘ This includes nuclear and radiological emergencies and conventional emergencies such as fires, releases of hazardous chemicals, storms or earthquakes.

ⓘ This includes situations for which prompt action is warranted to mitigate the effects of a perceived hazard.

nuclear or radiological emergency. An emergency in which there is, or is perceived to be, a hazard due to:

(a) The energy resulting from a nuclear chain reaction or from the decay of the products of a chain reaction;
(b) Radiation exposure.

ⓘ When used in IAEA publications, the term radiation normally refers only to ionizing radiation. The IAEA has no statutory responsibilities in relation to non-ionizing radiation.

emergency action level (EAL). A specific, predetermined criterion for observable conditions used to detect, recognize and determine the emergency class.

(emergency) arrangements. The integrated set of infrastructural elements, put in place at the preparedness stage, that are necessary to provide the capability for performing a specified function or task required in response to a nuclear or radiological emergency.

ⓘ These elements may include: authorities and responsibilities, organization, coordination, personnel, plans, procedures, facilities, equipment or training.

emergency class. A set of conditions that warrant a similar immediate emergency response.

ⓘ This is the term used for communicating to the response organizations and to the public the level of response needed. The events that belong to a given emergency class are defined by criteria specific to the installation, source or activities, which, if exceeded, indicate classification at the prescribed level. For each emergency class, the initial actions of the response organizations are predefined.

emergency classification. The process whereby an authorized official classifies an emergency in order to declare the applicable emergency class.

ⓘ Upon declaration of the emergency class, the response organizations initiate the predefined emergency response actions for that emergency class.

emergency exposure situation. A situation of exposure that arises as a result of an accident, a malicious act or other unexpected event, and requires prompt action in order to avoid or to reduce adverse consequences.

ⓘ Exposure in an emergency can be reduced only by protective actions and other response actions.

emergency plan. A description of the objectives, policy and concept of operations for the response to an emergency and of the structure, authorities and responsibilities for a systematic, coordinated and effective response. The emergency plan serves as the basis for the development of other plans, procedures and checklists.

ⓘ A ***concept of operations*** is a brief description of an ideal response to a postulated nuclear or radiological emergency, used to ensure that all those personnel and organizations involved in the development of a capability for emergency response share a common understanding.

emergency planning distance. The extended planning distance (EPD) and the ingestion and commodities planning distance (ICPD).

emergency planning zone. The precautionary action zone (PAZ) and the urgent protective action planning zone (UPZ).

emergency preparedness. The capability to take actions that will effectively mitigate the consequences of an emergency for human life, health, property and the environment.

emergency procedures. A set of instructions describing in detail the actions to be taken by emergency workers in an emergency.

emergency response. The performance of actions to mitigate the consequences of an emergency for human life, health, property and the environment.

ⓘ The emergency response also provides a basis for the resumption of normal social and economic activity.

emergency (response) action. An action to be taken in response to a nuclear or radiological emergency to mitigate the consequences of an emergency for human life, health, property and the environment.

ⓘ Emergency response actions comprise protective actions and other response actions.

other response action. An emergency response action other than a protective action.

ⓘ The most common other response actions are: medical examination, consultation and treatment; registration and longer term medical follow-up; providing psychological counselling; and public information and other actions for mitigating non-radiological consequences and for public reassurance.

emergency response facility or location. A facility or location necessary for supporting an emergency response, for which specific functions are to be assigned at the preparedness stage, and which need to be usable under emergency conditions.

ⓘ There are two different types of emergency response facility or location: those established in advance (e.g. a technical support centre for a nuclear power plant) and those designated at the time of an emergency (e.g. a medical screening and triage area).

ⓘ For both types, advance preparations are necessary to ensure their operability under emergency conditions. Depending on the emergency preparedness category and on the nature of an emergency, an emergency response facility may be designated as an emergency response location.

emergency services. The local off-site response organizations that are generally available and that perform emergency response functions. These may include police, firefighters and rescue brigades, ambulance services and control teams for hazardous materials.

emergency worker. A person having specified duties as a worker in response to an emergency.

ⓘ Emergency workers may include workers employed, both directly and indirectly, by registrants and licensees, as well as personnel of response organizations, such as police officers, firefighters, medical personnel, and drivers and crews of vehicles used for evacuation.

ⓘ Emergency workers may or may not be designated as such in advance to an emergency. Emergency workers not designated as such in advance of an emergency are not necessarily workers prior to the emergency.

existing exposure situation. An existing exposure situation is a situation of exposure that already exists when a decision on the need for control needs to be taken.

ⓘ Existing exposure situations include exposure to natural background radiation that is amenable to control; exposure due to residual radioactive material that derives from past practices that were never subject to regulatory control; and exposure due to residual radioactive material deriving from a nuclear or radiological emergency after an emergency has been declared to be ended.

extended planning distance (EPD). Area around a facility for which emergency arrangements are made to conduct monitoring following the declaration of a general emergency and to identify areas warranting emergency response actions to be taken off the site within a period following a significant radioactive release that would allow the risk of stochastic effects among members of the public to be effectively reduced.

ⓘ The area within the extended planning distance serves for planning purposes and may not be the actual area in which monitoring is to be conducted to identify areas where early protective actions such as relocation are necessary. While efforts need to be made at the preparedness stage to prepare for taking effective early protective actions within this area, the actual area will be determined by the prevailing conditions in an emergency.

ⓘ As a precaution, some urgent protective actions may be warranted within the EPD to reduce the risk of stochastic effects among members of the public.

facilities and activities. A general term encompassing nuclear facilities, uses of all sources of ionizing radiation, all radioactive waste management activities, transport of radioactive material and any other practice or circumstances in which people may be subject to exposure to radiation from naturally occurring or artificial sources.

ⓘ Facilities includes: nuclear facilities; irradiation installations; some mining and raw material processing facilities such as uranium mines; radioactive waste management facilities; and any other places where radioactive material is produced, processed, used, handled, stored or disposed of — or where radiation generators are installed — on such a scale that consideration of protection and safety is required.

ⓘ Activities includes: the production, use, import and export of radiation sources for industrial, research and medical purposes; the transport of radioactive material; the decommissioning of facilities; radioactive waste management activities such as the discharge of effluents; and some aspects of the remediation of sites affected by residues from past activities.

ⓘ This term is intended to provide an alternative to the terminology of sources and practices (or interventions) to refer to general categories of situations. For example, a practice may involve many different facilities and/or activities, whereas the general definition (1) of source is too broad in some cases: a facility or activity might constitute a source, or might involve the use of many sources, depending upon the interpretation used.

ⓘ The term 'facilities and activities' is very general, and includes those for which little or no regulatory control may be necessary or achievable: the more specific terms authorized facility and authorized activity should be used to distinguish those facilities and activities for which any form of authorization has been given.

ⓘ In the Fundamental Safety Principles (Safety Fundamentals[2]), the term 'facilities and activities — existing and new — utilized for peaceful purposes' is abbreviated for convenience to facilities and activities as a general term encompassing any human activity that may cause people to be exposed to radiation risks arising from naturally occurring or artificial sources (see para. 1.9 of the Safety Fundamentals[2]).

first responders. The first members of an emergency service to respond at the site of an emergency.

generic criteria. Levels for the projected dose, or the dose that has been received, at which protective actions and other response actions are to be taken.

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[2] EUROPEAN ATOMIC ENERGY COMMUNITY, FOOD AND AGRICULTURE ORGANIZATION OF THE UNITED NATIONS, INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL LABOUR ORGANIZATION, INTERNATIONAL MARITIME ORGANIZATION, OECD NUCLEAR ENERGY AGENCY, PAN AMERICAN HEALTH ORGANIZATION, UNITED NATIONS ENVIRONMENT PROGRAMME, WORLD HEALTH ORGANIZATION, Fundamental Safety Principles, IAEA Safety Standards Series No. SF-1, IAEA, Vienna (2006).

ⓘ The term generic criteria as defined here relates to emergency preparedness and response only.

graded approach. (1) For a system of control, such as a regulatory system or a safety system, a process or method in which the stringency of the control measures and conditions to be applied is commensurate, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control.

(2) An application of safety requirements that is commensurate with the characteristics of the facilities and activities or the source and with the magnitude and likelihood of the exposures.

hazard assessment. Assessment of hazards associated with facilities, activities or sources within or beyond the borders of a State in order to identify:

(a) Those events and the associated areas for which protective actions and other response actions may be required within the State;
(b) Actions that would be effective in mitigating the consequences of such events.

helper in an emergency. Member of the public who willingly and voluntarily helps in the response to a nuclear or radiological emergency.

ⓘ Helpers in an emergency are protected and are aware that they could be exposed to radiation while helping in response to a nuclear or radiological emergency.

ingestion and commodities planning distance (ICPD). Area around a facility for which emergency arrangements are made to take effective emergency response actions following the declaration of a general emergency in order to reduce the risk of stochastic effects among members of the public and to mitigate non-radiological consequences as a result of the distribution, sale and consumption of food, milk and drinking water and the use of commodities other than food that may have contamination from a significant radioactive release.

ⓘ The area within the ingestion and commodities planning distance serves for planning purposes to prepare for emergency response actions to monitor and control commodities, including food, either for domestic

use or for international trade. The actual area will be determined on the basis of the prevailing conditions in an emergency.

ⓘ As a precaution, some urgent protective actions may be warranted within the ingestion and commodities planning distance to prevent the ingestion of food, milk or drinking water and to prevent the use of commodities that may have contamination following a significant radioactive release.

inner cordoned off area. An area established by first responders in an emergency around a potential radiation hazard, within which protective actions and other emergency response actions are taken to protect first responders and members of the public from possible exposure and contamination.

interested party. A person, company, etc. with a concern or interest in the activities and performance of an organization, business, system, etc.

ⓘ The term interested party is used in a broad sense to mean a person or group having an interest in the performance of an organization. Those who can influence events may effectively become interested parties — whether their 'interest' is regarded as 'genuine' or not — in the sense that their views need to be considered. Interested parties have typically included the following: customers, owners, operators, employees, suppliers, partners, trade unions; the regulated industry or professionals; scientific bodies; governmental agencies or regulatory bodies (local, regional and national) whose responsibilities may cover nuclear energy; the media; members of the public (individuals, community groups and interest groups); and other States, especially neighbouring States that have entered into agreements providing for an exchange of information concerning possible transboundary impacts, or States involved in the export or import of certain technologies or materials.

justification. The process of determining for an emergency exposure situation or an existing exposure situation whether a proposed protective action or remedial action is likely, overall, to be beneficial; i.e. whether the expected benefits to individuals and to society (including the reduction in radiation detriment) from introducing or continuing the protective action or remedial action outweigh the cost of such action and any harm or damage caused by the action.

management system. A set of interrelated or interacting elements (system) for establishing policies and objectives and enabling the objectives to be achieved in an efficient and effective manner.

ⓘ The component parts of the management system include the organizational structure, resources and organizational processes. Management is defined (in ISO 9000) as coordinated activities to direct and control an organization.

ⓘ The management system integrates all elements of an organization into one coherent system to enable all of the organization's objectives to be achieved. These elements include the organizational structure, resources and processes. Personnel, equipment and organizational culture as well as the documented policies and processes are parts of the management system. The organization's processes have to address the totality of the requirements on the organization as established in, for example, IAEA safety standards and other international codes and standards.

non-radiological consequences. Adverse psychological, societal or economic consequences of a nuclear or radiological emergency or of an emergency response affecting human life, health, property or the environment.

ⓘ The term non-radiological consequences as defined here relates to emergency preparedness and response only.

notification. (1) A report submitted promptly to a national or international authority providing details of an emergency or a possible emergency; for example, as required by the Convention on Early Notification of a Nuclear Accident[3].

(2) A set of actions taken upon detection of emergency conditions with the purpose of alerting all organizations with responsibility for emergency response in the event of such conditions.

notification point. A designated organization with which arrangements have been made to receive notification (meaning (2)) and to initiate promptly the predetermined actions to activate a part of the emergency response.

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[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Convention on Early Notification of a Nuclear Accident and Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, Legal Series No. 14, IAEA, Vienna (1987).

notifying State. The State that is responsible for notifying (see notification meaning (1)) potentially affected States and the IAEA of an event of actual, potential or perceived radiological significance for other States.

ⓘ This includes:

(1) The State Party that has jurisdiction or control over the facility or activity (including space objects) in accordance with Article 1 of the Convention on Early Notification of a Nuclear Accident[3];
(2) The State that initially detects or discovers evidence of a transnational emergency, for example by: detecting significant increases in atmospheric radiation levels of unknown origin; detecting contamination in transboundary shipments; discovering a dangerous source that may have originated in another State; or diagnosing clinical symptoms that may have resulted from exposure outside the State.

nuclear or radiological emergency. See 'emergency'.

nuclear security. The prevention and detection of, and response to, criminal or intentional unauthorized acts involving nuclear material, other radioactive material, associated facilities or associated activities.

nuclear security event. An event that has potential or actual implications for nuclear security that must be addressed.[4]

off-site (area). See 'site (area)'.

on-site (area). See 'site (area)'.

operating personnel. Individual workers engaged in the operation of an authorized facility or the conduct of an authorized activity.

operating organization. Any organization or person applying for authorization or authorized to operate an authorized facility or to conduct an authorized activity and responsible for its safety.

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[4] INTERNATIONAL ATOMIC ENERGY AGENCY, Objective and Essential Elements of a State's Nuclear Security Regime: Nuclear Security Fundamentals, IAEA Nuclear Security Series No. 20, IAEA, Vienna (2013).

ⓘ This includes, inter alia, private individuals, governmental bodies, consignors or carriers, licensees, hospitals and self-employed persons.

ⓘ 'Operator' includes either those who are directly in control of a facility or an activity during use of a source (such as radiographers or carriers) or, in the case of a source not under control (such as a lost or illicitly removed source or a re-entering satellite), those who were responsible for the source before control over it was lost.

operational criteria. Values of measurable quantities or observable conditions (i.e. observables) to be used in the response to a nuclear or radiological emergency in order to determine the need for appropriate protective actions and other response actions.

ⓘ Operational criteria used in emergency preparedness and response include operational intervention levels (OILs), emergency action levels (EALs), specific observable conditions (i.e. observables) and other indicators of conditions on the site.

ⓘ The operational criteria are sometimes referred to as triggers.

operational intervention level (OIL). A set level of a measurable quantity that corresponds to a generic criterion.

ⓘ Operational intervention levels are typically expressed in terms of dose rates or of activity of radioactive material released, time integrated air activity concentrations, ground or surface concentrations, or activity concentrations of radionuclides in environmental, food or water samples.

ⓘ An operational intervention level is used immediately and directly (without further assessment) to determine the appropriate protective actions on the basis of an environmental measurement.

optimization (of protection and safety). The process of determining what level of protection and safety would result in the magnitude of individual doses, the number of individuals (workers and members of the public) subject to exposure and the likelihood of exposure being as low as reasonably achievable, economic and social factors being taken into account (ALARA).

planned exposure situation. A planned exposure situation is a situation of exposure that arises from the planned operation of a source or from a planned activity that results in an exposure from a source.

ⓘ Since provision for protection and safety can be made before embarking on the activity concerned, associated exposures and their probabilities of occurrence can be restricted from the outset. The primary means of controlling exposure in planned exposure situations is by good design of installations, equipment and operating procedures. In planned exposure situations, a certain level of exposure is expected to occur.

precautionary action zone (PAZ). An area around a facility for which emergency arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avoid or to minimize severe deterministic effects off the site. Protective actions within this area are to be taken before or shortly after a release of radioactive material or an exposure, on the basis of prevailing conditions at the facility.

preparedness stage. The stage or phase at which arrangements for an effective emergency response are established prior to a nuclear or radiological emergency.

projected dose. The dose that would be expected to be received if planned protective actions were not taken.

protective action. An action for the purposes of avoiding or reducing doses that might otherwise be received in an emergency exposure situation or an existing exposure situation.

early protective action. A protective action in the event of a nuclear or radiological emergency that can be implemented within days to weeks and still be effective.

ⓘ The most common early protective actions are relocation and longer term restriction of the consumption of food potentially affected by contamination.

mitigatory action. Immediate action by the operator or other party:

(a) To reduce the potential for conditions to develop that would result in exposure or a release of radioactive material requiring emergency response actions on the site or off the site; or
(b) To mitigate source conditions that may result in exposure or a release of radioactive material requiring emergency response actions on the site or off the site.

urgent protective action. A protective action in the event of a nuclear or radiological emergency which must be taken promptly (usually within hours to a day) in order to be effective, and the effectiveness of which will be markedly reduced if it is delayed.

ⓘ Urgent protective actions include iodine thyroid blocking, evacuation, short term sheltering, actions to reduce inadvertent ingestion, decontamination of individuals and prevention of ingestion of food, milk or drinking water possibly with contamination.

ⓘ A precautionary urgent protective action is an urgent protective action taken before or shortly after a release of radioactive material, or an exposure, on the basis of the prevailing conditions to avoid or to minimize severe deterministic effects.

radiological assessor. A person or team who in the event of a nuclear or radiological emergency assists the operator or off-site response organizations by performing radiological surveys, performing dose assessments, controlling contamination, ensuring the radiation protection of emergency workers and formulating recommendations on protective actions and other response actions.

reference level. For an emergency exposure situation or an existing exposure situation, the level of dose, risk or activity concentration above which it is not appropriate to plan to allow exposures to occur and below which optimization of protection and safety would continue to be implemented.

ⓘ The value chosen for a reference level will depend upon the prevailing circumstances for the exposure under consideration.

regulatory body. An authority or a system of authorities designated by the government of a State as having legal authority for conducting the regulatory process, including issuing authorizations, and thereby regulating nuclear, radiation, radioactive waste and transport safety.

ⓘ The national competent authority for the regulation of radioactive material transport safety is included in this description, as is the regulatory body for protection and safety.

representative person. An individual receiving a dose that is representative of the doses to the more highly exposed individuals in the population.

ⓘ The International Commission on Radiological Protection's Publication 101[5] indicates that the dose to the representative person "is the equivalent of, and replaces, the mean dose in the 'critical group'", and provides guidance on assessing doses to the representative person. The concept of critical group remains valid.

residual dose. The dose expected to be incurred after protective actions have been terminated (or after a decision has been taken not to take protective actions).

ⓘ Residual dose applies for an existing exposure situation or an emergency exposure situation.

response organization. An organization designated or recognized by a State as being responsible for managing or implementing any aspect of an emergency response.

ⓘ This also includes those organizations or services necessary to support the management and/or conduct of an emergency response, such as meteorological services.

site area. A geographical area that contains an authorized facility, authorized activity or source, and within which the management of the authorized facility or authorized activity or first responders may directly initiate emergency response actions.

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[5] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Assessing Dose of the Representative Person for the Purpose of the Radiation Protection of the Public and the Optimisation of Radiological Protection: Broadening the Process, ICRP Publication 101, Elsevier, Oxford (2006).

ⓘ This is typically the area within the security perimeter fence or other designated property marker. It may also be the controlled area around a radiography source or an inner cordoned off area established by first responders around a suspected hazard.

on-site (area). (Area) within the site area.

off-site (area). (Area) outside the site area.

source. (1) Anything that may cause radiation exposure — such as by emitting ionizing radiation or by releasing radioactive substances or radioactive material — and can be treated as a single entity for purposes of protection and safety.

ⓘ For example, materials emitting radon are sources in the environment; a sterilization gamma irradiation unit is a source for the practice of irradiation preservation of food and sterilization of other products; an X ray unit may be a source for the practice of radiodiagnosis; a nuclear power plant is part of the practice of generating electricity by nuclear fission, and may be regarded as a source (e.g. with respect to discharges to the environment) or as a collection of sources (e.g. for occupational radiation protection purposes). A complex or multiple installation situated at one location or site may, as appropriate, be considered a single source for the purposes of application of safety standards.

(2) Radioactive material used as a source of radiation.

ⓘ Such as those sources used for medical applications or in industrial instruments. These are, of course, sources as defined in (1), but this usage in (2) is less general.

dangerous source. A source that could, if not under control, give rise to exposure sufficient to cause severe deterministic effects. This categorization is used for determining the need for emergency arrangements and is not to be confused with categorizations of sources for other purposes.

ⓘ The term dangerous source relates to dangerous quantities of radioactive material (D-values) as recommended in the IAEA publication on Dangerous Quantities of Radioactive Material.[6]

radioactive source. A source containing radioactive material that is used as a source of radiation.

special facility. A facility for which predetermined facility specific actions need to be taken if urgent protective actions are ordered in its locality in the event of a nuclear or radiological emergency.

ⓘ Examples include chemical plants that cannot be evacuated until certain actions have been taken to prevent fire or explosions and telecommunications centres that must be staffed in order to maintain telephone services.

special population group. Members of the public for whom special arrangements are necessary in order for effective protective actions to be taken in the event of a nuclear or radiological emergency. Examples include persons with disabilities, hospital patients and prisoners.

stochastic effect. A radiation induced health effect, the probability of occurrence of which is greater for a higher radiation dose and the severity of which (if it occurs) is independent of dose.

ⓘ Stochastic effects may be somatic effects or hereditary effects, and generally occur without a threshold level of dose. Examples include solid cancers and leukaemia.

transient population group. Those members of the public who are residing for a short period of time (days to weeks) in a location (such as a camping ground) that can be identified in advance. This does not include members of the public who may be travelling through an area.

transnational emergency. A nuclear or radiological emergency of actual, potential or perceived radiological significance for more than one State.

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[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Dangerous Quantities of Radioactive Material (D-values), Emergency Preparedness and Response Series, EPR-D-VALUES 2006, IAEA, Vienna (2006).

ⓘ This may include:

(1) A significant transboundary release of radioactive material (however, a transnational emergency does not necessarily imply a significant transboundary release of radioactive material);
(2) A general emergency at a facility or other event that could result in a significant transboundary release (atmospheric or aquatic) of radioactive material;
(3) Discovery of the loss or illicit removal of a dangerous source that has been transported across, or is suspected of having been transported across, a national border;
(4) An emergency resulting in significant disruption to international trade or travel;
(5) An emergency warranting the taking of protective actions for foreign nationals or embassies in the State in which it occurs;
(6) An emergency resulting in or potentially resulting in severe deterministic effects and involving a fault and/or problem (such as in equipment or software) that could have serious implications for safety internationally;
(7) An emergency resulting in or potentially resulting in great concern among the population of more than one State owing to the actual or perceived radiological hazard.

significant transboundary release: A release of radioactive material to the environment that may result in doses or levels of contamination beyond national borders from the release which exceed generic criteria for protective actions and other response actions, including food restrictions and restrictions on trade.

urgent protective action. See 'protective action'.

urgent protective action planning zone (UPZ). An area around a facility for which arrangements have been made to take urgent protective actions in the event of a nuclear or radiological emergency to avert doses off the site in accordance with international safety standards. Protective actions within this area are to be taken on the basis of environmental monitoring — or, as appropriate, prevailing conditions at the facility.

warning point. A designated organization to act as a point of contact that is staffed or able to be alerted at all times for promptly responding to, or initiating a response to, an incoming notification (meaning (1)) warning message, request for assistance or request for verification of a message, as appropriate, from the IAEA.

worker. Any person who works, whether full time, part time or temporarily, for an employer and who has recognized rights and duties in relation to occupational radiation protection.

ⓘ A self-employed person is regarded as having the duties of both an employer and a worker.

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Safety through international standards

"Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them."

Yukiya Amano
Director General

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978–92–0–105715–0
ISSN 1020–525X

Ontario

PROVINCIAL NUCLEAR EMERGENCY RESPONSE PLAN (PNERP):

A Strategic Response Framework for Nuclear & Radiological Emergency Management

2025

Prepared by Emergency Management Ontario
Ministry of Emergency Preparedness and Response

Ontario

Executive Council of Ontario
Order in Council

On the recommendation of the undersigned, the Lieutenant Governor of Ontario, by and with the advice and concurrence of the Executive Council of Ontario, orders that:

Conseil exécutif de l'Ontario
Décret

Sur la recommandation de la personne soussignée, le lieutenant-gouverneur de l'Ontario, sur l'avis et avec le consentement du Conseil exécutif de l'Ontario, décrète ce qui suit :

Whereas section 8 of the *Emergency Management and Civil Protection Act*, as amended, requires the Lieutenant Governor in Council to formulate an emergency plan respecting emergencies arising in connection with nuclear facilities;

Therefore the document entitled "Provincial Nuclear Emergency Response Plan (PNERP): A Strategic Response Framework for Nuclear & Radiological Emergency Management" be adopted as an emergency plan respecting emergencies arising in connection with nuclear facilities formulated under section 8 of the *Emergency Management and Civil Protection Act*, as amended.

And further the "Provincial Nuclear Emergency Response Plan (PNERP): A Strategic Response Framework for Nuclear & Radiological Emergency Management", may be revised from time to time by the minister responsible for the *Emergency Management and Civil Protection Act* limited to updates clerical and administrative in nature.

And further that Order in Council O.C. 2317/2017, dated December 13, 2017, be revoked effective the date of this Order in Council.

Attendu que l'article 8 de la *Loi sur la protection civile et la gestion des situations d'urgence*, dans sa version modifiée, exige que le lieutenant-gouverneur en conseil établisse un plan de mesures d'urgence relatif aux situations d'urgence liées aux installations nucléaires;

Par conséquent, le document intitulé « Plan provincial d'intervention en cas d'urgence nucléaire (PPIUN) : Cadre d'intervention stratégique pour la gestion des situations d'urgence nucléaire et radiologique » est adopté à titre de plan de mesures d'urgence relatif aux situations d'urgence liées aux installations nucléaires, établi en vertu de l'article 8 de la *Loi sur la protection civile et la gestion des situations d'urgence*, dans sa version modifiée.

De plus, le « Plan provincial d'intervention en cas d'urgence nucléaire (PPIUN) : Cadre d'intervention stratégique pour la gestion des situations d'urgence nucléaire et radiologique » peut être révisé selon les besoins par le ministre responsable de la *Loi sur la protection civile et la gestion des situations d'urgence* en se limitant à des mises à jour de nature rédactionnelle et administrative.

O.C. | Décret : 1020/2025 1

En outre, le décret 2317/2017 du 13 décembre 2017 est révoqué à compter de la date du présent décret.

Recommended: Minister of Emergency Preparedness and Response
Recommandé par : Ministre de la Protection civile et de l'Intervention en cas d'urgence

Concurred: Chair of Cabinet
Appuyé par : la présidence du Conseil des ministres

Approved and Ordered:
Approuvé et décrété le : JUL 31 2025

Administrator of the Government
L'administrateur du gouvernement

Forward and Synopsis

The Government of Ontario is responsible for leading and directing the preparations for, the response to, and recovery from nuclear and radiological emergencies that affect Ontario. The Provincial Nuclear Emergency Response Plan (PNERP) details how the province will respond to nuclear and radiological emergencies.

Ontario's nuclear facilities have operated since 1962, for over 63 years, without having any incidents that have posed a risk to public safety. Nuclear Facilities have produced over 3600 TWh of electricity with a combined total of over 4.7M reactor operating hours without incident. Ontario's nuclear generating facilities are world leaders when it comes to being the safest and cleanest form of electrical production.

The revised PNERP is based upon a new technological study that has enabled it to be flexible, agile, and applicable to all existing, planned e.g., Small Modular Reactors (SMRs), and new future nuclear reactor technologies. The revised PNERP supports the Government of Ontario's policy of growing its nuclear power production footprint anywhere in Ontario while ensuring public safety.

Ontario has adopted a SAFE, SAFER and SAFEST approach to nuclear emergency management. The PNERP's planning basis, has accounted for the safety record of Ontario's nuclear industry and the substantive improvements that have been implemented including the enhanced safety systems that have been introduced in a post Fukushima environment.

The PNERP and its existing Nuclear Emergency Management Program (NEMP) is built upon the industry's safe foundation to make Ontario SAFER.

The revised PNERP is an essential component to Ontario's setting of the industry standard and its ability to ensure Ontario is Safe, Practiced and Prepared in a nuclear and radiological context for now and into the future.

The PNERP:

- Pursuant to section 8 of the *Emergency Management and Civil Protection Act* (EMCPA) the Lieutenant Governor in Council shall formulate an emergency plan respecting emergencies arising in connection with nuclear facilities, and any provisions of an emergency plan of a municipality respecting such an emergency shall conform to the plan formulated by the Lieutenant Governor in Council and are subject to the approval of the Minister responsible for emergency management, and the Minister responsible for emergency management may make such alterations as he or she considers necessary for the purpose of co-

ordinating the plan with the plan formulated by the Lieutenant Governor in Council.

- The PNERP describes the general roles and responsibilities for the off-site response to a radiological or nuclear emergency in Ontario and includes seven Implementing Plans. **(Note**: the province is responsible for directing the response to all nuclear or radiological emergencies in/affecting Ontario**.)**
- The PNERP is operationally applied through its detailed Implementing Plans that are designed specifically for each major nuclear facility in or within the immediate vicinity of Ontario and or for all other types of radiological emergencies.

The Scientific basis for the revised PNERP:

- A new technological study and planning basis that is applicable to existing CANDU reactors, new large nuclear reactors, and Small Modular Reactors was developed.
- It was developed in collaboration with nuclear health physicists, nuclear engineers, and reactor safety experts from Ontario Power Generation (OPG), Bruce Power (BP), Chalk River Nuclear Laboratories (CNL), Emergency Management Ontario (EMO) and the Chief Nuclear Engineer from Calian Nuclear Ltd. (**Note**: the new technical study and planning basis have been peer reviewed and endorsed by this scientific cohort.)
- The existing 2017 PNERP's technical study and planning basis, was revised to account for post-Fukushima enhancements to safety systems and designs of operating nuclear facilities in the province as well as incorporated into lessons learned from continuous improvements, international best practices, and standards. **Note:** the flexibility/agility provided by the new technological study and planning basis for the revised PNERP supports an expansion of Ontario's nuclear footprint across the province.

The Duty to Consult regarding the PNERP

The province has a Duty to Consult regarding the PNERP. Emergency Management Ontario (EMO) has engaged and consulted with all First Nation and Indigenous communities in Ontario that are close to or have traditional territories that could be affected by licensed CNSC facilities within Ontario. (Note: the implementation of the PNERP during the unlikely event of a nuclear or radiological emergency could affect First Nation or Indigenous community rights etc., thus we continue to work together on a partnership basis regarding nuclear emergency management.)

"Over the next 25 years, demand for electricity in Ontario is projected to grow by at least 75 per cent, the equivalent of adding four and a half cities the size of Toronto.

Ontario's energy policy will determine the success of our province's economy, today and for a generation to come.

It's why for the last seven years your government has been hard at work restoring the promise of Ontario's energy advantage. Now, today, the province boasts one of the

cleanest grids in North America, with a steady and reliable supply of affordable electricity powering Ontario's growing economy.

Next, your government will invest unprecedented amounts in new energy production, transmission, and storage to transform our province into a global clean energy superpower, able to export clean energy and technology across the continent and beyond.

This includes the first small modular nuclear reactors in the G7 at Darlington nuclear site, alongside upgrades to existing nuclear facilities at the Darlington, Pickering, and Bruce nuclear generating stations to extend their lifespans and expand their generating capacity.

Your government is also working with interested municipalities and First Nations to dramatically expand Ontario's energy generating capacity, including exploring building the first new large-scale nuclear power plants in the province in more than 30 years in Bruce County and Port Hope" ***- Ontario's Speech from the Throne April 15, 2025 – Protect Ontario by Unleashing Our Economy***

Executive Summary

How Ontario responds in the very unlikely event of a nuclear or radiological emergency in this province is the substance of this document. Ontario's Strategic Response Plan Framework for Nuclear and Radiological Emergency Management is short titled: Provincial Nuclear Emergency Response Plan (PNERP). With increased electricity needs and new, reduced carbon footprint technology, there is a need to meet this moment through development of this framework and nuclear implementing plans that meet or exceed all national and international public safety standards.

This plan is the guiding document for the province to prepare its ministries and employees to respond to a radiological or nuclear emergency within Ontario's borders, and to consequences that arise due to a nuclear or radiological emergency outside of its borders. In either event, the province's response shall protect people, the environment and infrastructure by ensuring any necessary radiation exposure is mitigated to the extent possible, which includes taking all reasonable measures to minimize the possibility of radiation exposure health effects.

The plan includes a requirement to provide appropriate information to members of the public and maintain public education on nuclear and radiological hazards. This awareness includes sources of radiation present during daily activities as well as those hazards present in the event of an emergency. During an emergency, the province will provide timely and accurate information to the public regarding protective actions necessary to prevent exposure to harmful levels of radiation and ensure that any exposure is kept as low as reasonably achievable (ALARA).

Furthermore, the province cultivates the expertise of its employees by providing necessary equipment and maintaining high standards of training. The use of appropriate equipment and a high degree of training will help to ensure strong situational awareness in an emergency thereby enabling timely decision making by emergency managers at all levels of government.

This is a scientifically based document for which a new, evidence-based technical study was conducted that has and will form the planning basis of the site-specific implementing plans for utilities and the nuclear-designated municipalities in Ontario regarding existing CANDU reactors and all new future builds/technologies including, but not limited to Small Modular Reactors (SMRS) that are being and or will be introduced anywhere within Ontario. A standardized methodology has been developed based upon the work described that is technologically neutral and agile in its application.

The results from the application of the methodology will inform the planning basis and policy decisions that are made by Emergency Management Ontario (EMO) when setting zone sizes and preparedness requirements for emergency planning zones for specific technologies/sites. The details for each technology's site-specific emergency planning zones are outlined in the individual respective Site-Specific Implementing Plans.

The operational governance structure is based upon the Incident Management System (IMS). A Scientific Section for nuclear and radiological emergency response purposes, has been added to the IMS structure to provide expert scientific information and advice to the EMO Command Structure. Operational details are found in the Site-Specific Implementing Plans. EMO, on behalf of the province, will direct the response to all nuclear & radiological emergencies. EMO's operational command relationship with provincial ministries, has been detailed in the PNERP. Specific details regarding all aspects of social and media related communications are detailed in separate communications plans.

Based upon the application of the PNERP's methodology, eight designated/host municipalities have been identified. The roles and responsibilities of the eight designated/host communities are clearly specified in the PNERP and the applicable Site -Specific Implementing Plans so that they can follow EMO's direction in preparation for, response to and recovery from a nuclear emergency that affects their municipality. The operational relationship between EMO and the licensed nuclear facilities is specified to ensure timely notification, communications and liaison is maintained, and support provided for the PNERP's operational implementation. The Canadian federal government and identified federal departments have operational support roles as detailed in the PNERP.

First Nation and Indigenous communities have jurisdiction over their respective community's emergency management programs including nuclear response within their communities. The strategic government to government operational relationship between Ontario as represented by EMO with the First Nation and Indigenous communities that are, or have traditional territories, in proximity to a licensed nuclear facility, has been outlined to ensure notification, mutual operational support and communications:

Individual First Nation and Indigenous community's operational details are specified in their community's emergency plans and the applicable site-specific implementing plans.

A communications and liaison structure has been detailed for cross boarder operational coordination purposes with contiguous provinces and or states within the United States.

Finally, to ensure its ongoing operational capacity, the PNERP will be reviewed regularly and updated accordingly to ensure that it maintains its applicability in an evolving nuclear radiological environment for a safe, practiced, and prepared Ontario in a nuclear and radiological context.

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This is version 1.0 of the 2025 Provincial Nuclear Emergency Response Plan.
This version, once approved, will replace the 2017 PNERP Master Plan.

Administrative updates to the current version will be recorded in the following table.

Revision Number	Description of change	Date of Publication
1.0	Initial publication of 2025 version	2025-07-31

Publications Management

This publication is subject to review and administrative amendments in collaboration with ministries and consultation with key interested parties. This process is the responsibility of Emergency Management Ontario. Interested parties are encouraged to review and evaluate this plan as they use it and to submit comments and suggestions where applicable.

Copies of this Provincial Nuclear Emergency Response Plan are distributed according to the Distribution List - Annex V. Organizations that keep hard or digital copies of this document as a resource for their emergency operations are responsible for ensuring that they have the most up-to-date version of the document.

For more information on Ontario's nuclear emergency management program please visit Ontario Emergency Preparedness: Nuclear Incident.

Please submit your comments and suggestions on the Provincial Nuclear Emergency Response Plan or, to request it in a different format, please Contact us.

Website:
Nuclear Incident (English)
Incident Nucléaire (French)

Acronyms and Abbreviations

AAZ	Automatic Action Zone
ADM	Assistant Deputy Minister
AHJ	Authority Having Jurisdiction
ALARA	As Low As Reasonably Achievable
BDBA	Beyond Design Basis Accident
CBRNE	Chemical, Biological, Radiological, Nuclear, and Explosives
CCEM	Cabinet Committee on Emergency Management
CEM	Commissioner of Emergency Management
CFIA	Canadian Food Inspection Agency
CMOH	Chief Medical Officer of Health
CNSC	Canadian Nuclear Safety Commission
ConOps	Concept of Operations
CPZ	Contingency Planning Zone
CSA	Canadian Standards Association
DBA	Design Basis Accident
DMSC–EM	Deputy Ministers Steering Committee on Emergency Management
DO	Duty Officer
DPZ	Detailed Planning Zone
DRD	Direct Reading Dosimeters
EB	Emergency Bulletin
EDU	Ministry of Education

EIC	Emergency Information Centre
EMCPA	Emergency Management and Civil Protection Act
EMO	Emergency Management Ontario
ENERGY	Ministry of Energy and Mines
EOC	Emergency Operations Center
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
ER	Emergency Response
ERAMG	Environmental Radiation and Assurance Monitoring Group
ERO	Emergency Response Organization
ETE	Evacuation Time Estimate
EWC	Emergency Worker Centre
FEMA	Federal Emergency Management Agency
FERP	Federal Emergency Response Plan
FNEP	Federal Nuclear Emergency Plan
GC	Generic Criteria
GOC	Government Operations Centre
HazMat	Hazardous Materials
HC	Health Canada
HPPA	Health Protection and Promotion Act
HSEM	Health System Emergency Management
IAEA	International Atomic Energy Agency
IAFNER	Ministry of Indigenous Affairs and First Nation Economic Reconciliation
IAP	Incident Action Plan
IMS	Incident Management System
IP	Implementing Plans
IPZ	Ingestion Planning Zone
ITB	Iodine Thyroid Blocking
J-EIC	Joint Emergency Information Centre
KI	Potassium Iodide
LGIC	Lieutenant Governor in Council
LO	Liaison Officer
MCCSS	Ministry of Children, Community and Social Services
MDU	Monitoring and Decontamination Units
MECP	Ministry of the Environment, Conservation and Parks
MEOC	Ministry Emergency Operations Centre
MERP	Ministry's Emergency Response Plan
MEPR	Ministry of Emergency Preparedness and Response
MLITSD	Ministry of Labour, Immigration, Training and Skills Development
MMAH	Ministry of Municipal Affairs and Housing
MND	Ministry of Northern Economic Development and Growth
MNR	Ministry of Natural Resources
MOH	Ministry of Health
MTO	Ministry of Transportation
NAADS	National Alert Aggregation and Dissemination System

NEAT	Nuclear Emergency Assessment Team
NEMAC	Northern Emergency Management Assistance Compact
NEMCC	Nuclear Emergency Management Coordinating Committee
NER	Nuclear Emergency Response
NGO	Non-Government Organization
NGS	Nuclear Generating Station
NOTAM	Notice to Air Missions (AKA Notice to Airmen)
NAVWARN	Navigational Warning
NPP	Nuclear Power Plant
NRCan	Natural Resources Canada
NRSRP	Nuclear, Radiological and Scientific Response Programs
NSCA	Nuclear Safety and Control Act
OACP	Ontario Association of Chiefs of Police
OEEP	Ontario Electricity Emergency Plan
OFI	Opportunity for Improvement
OHSA	Occupational Health and Safety Act
OIL	Operational Intervention Level
OMAFA	Ontario Ministry of Agriculture, Food and Agribusiness
OPP	Ontario Provincial Police
ORSP	Ontario Reactor Surveillance Program
OSST	Offsite Survey Team
PAG	Protective Action Guidelines
PCEIO	Provincial Chief Emergency Information Officer
PEIP	Provincial Emergency Information Plan
PEOC	Provincial Emergency Operations Centre
PIMS	Public Information Management Section
PNEPF	Provincial Nuclear Emergency Preparedness Framework
PNERP	Provincial Nuclear Emergency Response Plan
PNROP	Provincial Nuclear Recovery Operations Plan
PNOORF	Provincial Notification of Offsite Response Form
PPE	Personal Protective Equipment
PS	Public Safety Canada
RASCAL	Radiological Assessment System for Consequence Analysis
REGDOC	Regulatory Document (a CNSC publication)
RHRP	Radiation Health Response Plan
RMO	Recovery Management Organization
SI	International System of Units
SOLGEN	Ministry of the Solicitor General
TAG	Technical Assessment Group
TBS	Treasury Board Secretariat
TLD	Thermoluminescent Dosimeter
USAR	Urban Search and Rescue
USNRC	United States Nuclear Regulatory Commission
URI	Unified RASCAL Interface
UTCC	Unified Transportation Coordination Centre
UTMP	Unified Transportation Management Plan

1

1. INTRODUCTION

1.1 Introduction

The legislative authority for emergency management (including planning and response) in the province of Ontario (hereinafter referred to as “the province”) is provided for by the *Emergency Management and Civil Protection Act* (EMCPA). Pursuant to Section 8 of the EMCPA, the Lieutenant Governor in Council (LGIC) shall formulate an emergency plan respecting emergencies arising in connection with nuclear facilities. Pursuant to Section 6 of the EMCPA, the Minister responsible for emergency management is responsible for developing the plan to respond to nuclear emergencies through Emergency Management Ontario (EMO).

During a nuclear emergency, the province is primarily responsible for mitigating the offsite effects and coordinating the offsite response. Provincial responsibilities shall be executed by supporting and coordinating the efforts of organizations with nuclear emergency responsibilities as set out in this Provincial Nuclear Emergency Response Plan (PNERP). The PNERP is the foundational document in the response pillar of Ontario’s nuclear emergency program (Figure 1). The other two pillars are the preparedness and the recovery pillars.

The nuclear emergency response terminology used in this document is fully defined in the Provincial Nuclear Emergency Preparedness Framework (PNEPF). This PNERP is operationally focused and is intended for audiences that are already familiar with concepts such as event categorization, provincial offsite response levels, response phases, etc.

This diagram is for visual reference only. See the text below this figure for the full description.

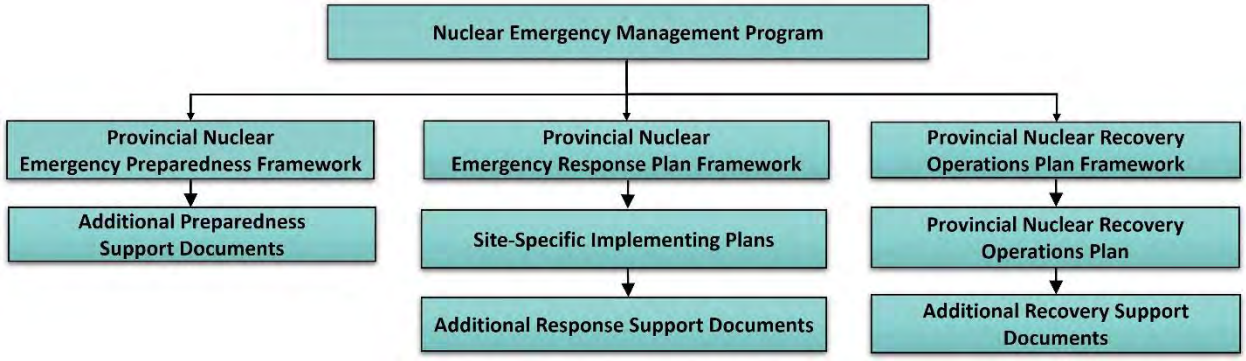

Figure 1: Nuclear Emergency Program Document Hierarchy

The diagram shows the structure of the Nuclear Emergency Management Program in Ontario, presented in a top-down approach. The program branches into three main

components: the Provincial Nuclear Emergency Preparedness Framework, the Provincial Nuclear Emergency Response Plan and Implementing Plans, and the Provincial Nuclear Recovery Operations Plan. Each of these components is linked to a corresponding set of documents: Additional Preparedness Support Documents, Site-Specific Implementing Plans, Additional Response Support Documents, Provincial Nuclear Recovery Operations Plan and Additional Recovery Support Documents, respectively.

1.2 Provincial Objective

The province, in the unlikely event of a nuclear emergency, will act to minimize the impact on public health and safety, property and the environment. To that end, the province will issue operational directives and, in the event of a declared provincial emergency, emergency orders, where warranted and appropriate in alignment with the requirements and criteria outlined in the EMCPA, and as further detailed in this Plan.

1.3 Nuclear Emergency Response Policy

The province has jurisdiction over public health and safety, property, and the environment within its borders. The province's nuclear emergency response policy (Section 2) applies to the province's capability to manage a nuclear or radiological emergency where supplementary response or assistance by the province is necessary to support the immediate onsite response and to manage offsite consequences of the accident. The policy also applies to support for the province's nuclear and radiological emergency response capacity generation and capability maintenance. The province's policy supports nuclear and radiological emergency management with First Nation and Indigenous Communities and their traditional territories.

1.4 Goal of the PNERP

The goal of this PNERP is to provide the basis upon which offsite nuclear emergency response would be undertaken to achieve the province's aim.

To achieve this goal, the province has adopted guiding principles (Annex A) to direct operational planning and ensure any protective actions taken are justified.

1.5 Scope

This Plan addresses the response to a nuclear emergency occurring at a Nuclear Generating Station (NGS) within Ontario or at the Fermi 2 Nuclear Power Plant (NPP) in Michigan. Along with the contents herein, specifics of response to other nuclear emergencies occurring outside of Ontario and directly impacting the province are covered in the Transborder Implementing Plan (IP). Non-nuclear radiological emergencies are addressed in the Implementing Plan for Other Radiological Emergencies.

The PNERP is designed to align with national and international guidance and best practices including the Canadian Nuclear Safety Commission (CNSC), the International Atomic Energy Agency (IAEA), the United States Nuclear Regulatory Commission (USNRC), and the Federal Emergency Management Agency (FEMA), and the CSA N1600 – General requirements for nuclear emergency management programs.

1.6 Supporting Plans and Procedures

Nuclear emergency plans formulated by ministries, municipalities, operators of reactor facilities and other CNSC licensed facilities, and other agencies and organizations should conform to this PNERP, to contribute to the achievement of the aim detailed in Section 1.2.

Document interdependencies are visualized in Figure 1: Nuclear Emergency Program Document Hierarchy. Specifically, the PNERP interfaces and has interdependencies with the following elements (Annex B):

a. NGS-specific Implementing Plans;
b. Major Organization Plans (provincial ministries and municipalities);
c. Provincial Nuclear Emergency Preparedness Plan;
d. Provincial Nuclear Emergency Preparedness Support Plans;
e. Provincial Nuclear Recovery Operations Plan;
f. Provincial Nuclear Recovery Operations Support Plans; and
g. Procedures associated with the above (not in Annex).

1.7 Introduction of New Technologies

PNERP will be reviewed regularly and updated accordingly to ensure that it maintains its applicability in an evolving nuclear radiological environment for a safe, practiced, and prepared Ontario in a nuclear and radiological context. The implementation of new technologies will be tracked and any impact on future editions will be assessed and implemented as required in future revisions.

2. ONTARIO'S NUCLEAR EMERGENCY RESPONSE POLICY

2.1 General

To ensure Ontarians are safe, practiced, and prepared, the province, through Emergency Management Ontario (EMO) as part of the Ministry of Emergency Preparedness and Response, maintains the PNERP. To this end, the requirements of the EMCPA, the guidance provided in CSA N1600 – General requirements for nuclear emergency management programs, and international best practices (including that of the IAEA) have been used to inform the policy for nuclear emergency response.

2.2 Policy Principles

The following principles underlie the province's nuclear emergency response policy and guides all offsite nuclear and radiological emergency management within the province:

a. The province, through its ministries, agencies, boards and commissions, has responsibility for the health, safety, and welfare of all inhabitants of the province, and the protection of property and the environment.
b. As the Authority Having Jurisdiction (AHJ), the province shall direct, coordinate and/or support other organizations in offsite response to manage the offsite consequences of a nuclear emergency and may, where warranted and appropriate, issue operational directives as action taken under the emergency plan where an emergency exists but a provincial emergency has not yet been declared and, in the event of a declared provincial emergency, issue emergency orders under the EMCPA.
c. In the event of a radiological emergency, other than one occurring at a nuclear reactor facility, the province's role could vary from providing support to a local municipality, to direction regarding the conduct of response and recovery operations when the situation is beyond the capability of the municipality to address safely.
d. Even though reactor facilities are designed and operated according to stringent safety standards, emergency preparedness and response must operate on the basis that mechanical failure, human error, extreme natural events, or hostile action could lead to nuclear emergencies.
e. The province intends to maintain a capability to respond to Design Basis Accidents (DBA) solely through internal resources and without federal assistance (DBA are accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits).
f. Plans administered under this policy will be able to deal effectively with a broad range of possible emergencies, including severe reactor facility accidents.
g. Timely and accurate information will be provided to the public and media.

2.3 Policy Objective

This policy aims to meet EMO's nuclear emergency response mandate by delineating responsibility for management of the province's nuclear emergency management program in order to enable a robust and versatile response in the unlikely event of a nuclear or radiological emergency. Under this policy, the province shall safeguard the health, safety, welfare, and property of the people in Ontario and ensure the protection of the environment.

The Province Shall:

a. Be prepared to respond to a radiological or nuclear incident in Ontario.
b. Protect people, the environment and infrastructure from increased levels of radiation exposure to the extent possible.
c. Provide appropriate information, and proactively engage education regarding nuclear and radiological hazards.
d. In the event of an emergency, provide timely and accurate information to the public regarding protective actions required.
e. Ensure that those charged with the planning for, response to, and recovery from a radiological or nuclear emergency are properly trained and equipped to do so, ensuring appropriate capability and capacity.
f. Ensure it is positioned to understand and adapt to new nuclear technologies as it relates to offsite preparedness, response, and recovery.

The province's policy includes supporting impacted municipalities to mitigate local emergencies such that the response of local first responders is safe and effective.

2.4 Enabling Authority, Roles, and Responsibilities

The enabling authority for emergency response in Ontario is the EMCPA, which identifies the roles and responsibilities of provincial ministries and municipalities in preparing for and responding to emergencies that fall within their jurisdiction. Thus, the EMCPA authorizes and directs the province to take the actions necessary to meet their responsibilities under the EMCPA. Pursuant to Section 6 of the EMCPA, the Minister responsible for emergency management is assigned responsibility for nuclear and radiological emergencies while the development of a provincial emergency response plan for terrorism and civil disorder is assigned to SOLGEN. The primary responsibility for managing a terrorism and civil disorder emergency rests with the local police service of jurisdiction for the community/municipality in which the event occurs, supported by the associated municipal government. In the event of a radiological or nuclear emergency caused by terrorism or civil disorder, the MEPR, through EMO, shall act in support of SOLGEN's overall management of the incident, with EMO primarily focused on consequence management, beyond their capability and/or capacity, inclusive of any pre-existing mutual aid agreements.

A list of identified ministries and municipalities as specified in this PNERP, shall ensure effective fulfillment of their respective preparedness and response roles. The federal government also has some roles and responsibilities when a nuclear emergency happens in Canada, or when Canada is affected by a nuclear emergency abroad. See Annex D for Federal Legislative Authority. Additional emergency response organization responsibilities for nuclear emergencies are described in:

a. federal departments – Annex E
b. provincial ministries – Annex F
c. reactor facilities – Annex G
d. designated municipalities – Annex I
e. designated host municipalities – Annex J

The hierarchy of authorities related to nuclear emergency response is illustrated in Figure 2: Hierarchy Authority.

This diagram is for visual reference only. See the text below this figure for the full description.

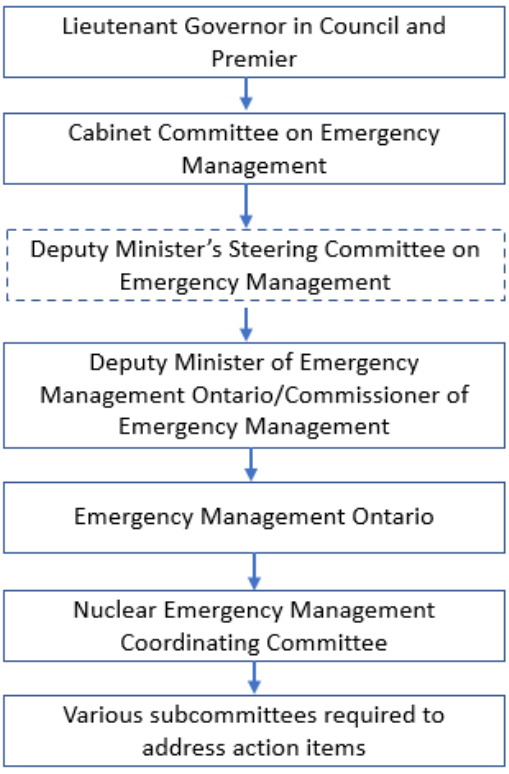

Figure 2: Hierarchy Authority

The flowchart presents a top-down hierarchical structure of authorities involved in a nuclear emergency response in Ontario. The Lieutenant Governor in Council and the Premier have the highest authority, followed by the Cabinet Committee on Emergency

Management. Below this is the Deputy Minister's Steering Committee on Emergency Management. The next level is the Deputy Minister of Emergency Management Ontario, who also serves as the Commissioner of Emergency Management. This is followed by Emergency Management Ontario, then the Nuclear Emergency Management Coordinating Committee. At the base of the hierarchy are various subcommittees responsible for addressing specific action items. Downward-pointing arrows between each level illustrate the flow of authority throughout the structure.

2.4.1 Lieutenant Governor in Council and Premier

The LGIC and the Premier of Ontario provide overall direction to the management of the emergency response.

2.4.2 2.4.2 Cabinet Committee on Emergency Management / Minister of Emergency Preparedness and Response

Appointed by the LGIC, the Cabinet Committee on Emergency Management (CCEM) provides strategic advice to Cabinet on the government's efforts to prevent, mitigate, prepare for, respond to, and recover from emergency situations.

2.4.3 Commissioner of Emergency Management and Chair of Deputy Minister's Steering Committee on Emergency Management

The Commissioner/Chair will administer all programs, plans, and procedures identified in the PNERP including those considered subordinate to the PNERP and will chair the Deputy Minister's Steering Committee on Emergency Management (DMSC) which provides whole-of-government coordination, alignment, and oversight for preparedness, response and recovery operations as needed.

2.4.4 Emergency Management Ontario

EMO will lead provincial ministries in preparing for, responding to, mitigating, and recovering from the offsite radiological consequences of a nuclear emergency with the goal of protecting the health, safety, welfare, and property of the people of Ontario and the environment. In the event of a radiological emergency, other than one occurring at a facility with site-specific response plans, EMO's role may extend to directing the response in addition to the aforementioned radiological emergency functions.

EMO will:

a. Maintain the PNERP by continually evaluating the PNERP and updating as needed.
b. Conduct preparedness activities including:
    i. Training and acquisition / maintenance of required survey and sampling equipment.
    ii. Participation in drills and exercises to consolidate knowledge and training.

iii. Through drills and exercises, identify planning gaps or areas for improvement to ensure that the province is safe, practiced, and prepared.

c. Oversee, design, and provide public education and awareness program parameters.

2.4.5 Nuclear Emergency Management Coordinating Committee

The Nuclear Emergency Management Coordinating Committee (NEMCC) shall convene for semi-annual meetings. Provincial representation at this committee will include (but is not limited to) Directors that have a responsibility for any facet within the PNERP. These meetings are intended to monitor the status of the PNERP and identify gaps and areas for improvement. The product of these meetings shall be action plans to address issues at the lowest level possible. NEMCC membership also includes representatives from the designated and host municipalities, First Nation, and Indigenous communities in close proximity to licensed Canadian Nuclear Safety Commission (CNSC) facilities, federal departments, nuclear facilities, and contiguous states of the United States.

2.4.6 Subcommittees and Working Groups

As required, specific subcommittees and/or working groups should be utilized to enable the PNERP, with the intent of growing and maintaining provincial capabilities and capacities across all ministries that have been assigned responsibilities within the PNERP. It is intended that these subcommittees and working groups will be used as a vehicle to harmonize efforts, identify areas of concern, and develop recommendations to ensure the requirements of the PNERP are satisfied.

2.4.7 Provincial Employees

Provincial employees shall be provided training opportunities commensurate with their role as identified within the PNERP such that they can develop the skills necessary to fulfill their role in daily preparedness operations as well as during an emergency. Employees must maintain the necessary training qualifications as described within the PNERP and its subordinate documents, such as the Nuclear Training Program. The level of training required shall be commensurate with the employee’s role in the program. Further, all employees engaged with the PNERP shall strive to enhance the program by identifying non-compliance/conformance and areas for improvement.

3. FOUNDATIONS OF THE PNERP

3.1 Acts, Regulations, and Orders

This section outlines the provincial legislation establishing the PNERP. Further description of provincial legislation is presented in Annex B.

3.1.1 Emergency Management and Civil Protection Act

The *Emergency Management and Civil Protection Act* establishes the province's legal framework for managing emergencies. The legislation and its regulations do this by setting out various authorities and responsibilities accorded to provincial ministries, municipalities, and specific individuals.

3.2 First Nation and Indigenous Communities

Responsibility for public safety within First Nation communities' rests with the Chief and Council, whereas in Indigenous communities, it resides with their respective leadership. Communities that have Radiological or Nuclear (RN) hazards identified as part of their Hazard and Risk profile are encouraged and supported to have community emergency plans that reflect these hazards/risks as appropriate for the protection of community members.

While there are no First Nation or Indigenous communities within the Detailed Planning Zones (DPZ) of any of Ontario's reactor facilities, there are communities and traditional territories that are within the Ingestion Planning Zone (IPZ) for reactor facilities. A DPZ is a pre-designated area immediately surrounding a reactor facility where pre-planned protective actions are implemented as needed. An IPZ is a pre-designated area identified to monitor and ensure the safety of the food, drinking water and or wild hunts/harvests and to restrict/eliminate the consumption and or distribution of potentially contaminated consumables including for example grazing animals, milk, and animal feed. First Nation Chiefs/Councils and Indigenous leadership should consider in their plans the need to undertake ingestion control measures (refer to the PNERP site specific implementing plans for Pickering NGS, Darlington NGS, Bruce NGS, Fermi 2 NPP, and for a Transborder Nuclear Emergency).

Some First Nation and/or Indigenous communities may be situated near other licensed CNSC facilities. In such cases, the communities are advised to consider relevant response measures in the event of a radiological emergency at the CNSC licensed facilities. This would include details of relevant notifications to and from the involved organizations.

Given the broad range of other radiological incidents possible (e.g., transportation accidents), First Nation and Indigenous communities should consider identifying

incidents as a potential risk within their emergency plans (refer to the PNERP Implementing Plan for Other Radiological Emergencies).

3.3 International Agreements and Arrangements

Ontario has several agreements in place for the purposes of emergency response mutual assistance (e.g., Northern Emergency Management Assistance Compact (NEMAC) Agreement) as well as more specific nuclear notification agreements. The latter are detailed in the PNERP Implementing Plan for the Fermi 2 NPP and for a Transborder Nuclear Emergency. Health Canada (HC) and the United States Department of Energy developed a statement of intent supporting joint Canada-USA nuclear emergency preparedness and response capabilities with the objective to identify areas where coordination and cooperation, including information sharing and mutual assistance would be beneficial to nuclear emergency management programs and capabilities.

Canada is a signatory of the IAEA's Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency (1986), which sets out an international framework for co-operation among countries and with the IAEA to facilitate prompt assistance and support in the event of nuclear accidents or radiological emergencies. Requests to the IAEA for international assistance would be undertaken by CNSC, as the Competent Authority-Abroad, and Global Affairs Canada.

Canada is also a signatory of the IAEA's Convention on Early Notification of a Nuclear Accident (1986). The Convention establishes a notification system for nuclear accidents with potential for international, trans-boundary releases that could be of radiological safety significance.

International notification to the IAEA of a nuclear accident in Canada would be undertaken by the CNSC as the Competent Authority-Domestic.

4. GOVERNANCE

4.1 General

The Policy (Section 2) prescribes provincial senior management's commitment to the program; detailing roles and responsibilities and assigning accountability and authority to participants of the program (see Figure 2: Hierarchy Authority).

4.1.1 PNERP Oversight

Oversight of the PNERP is the responsibility of the Minister of Emergency Preparedness and Response, who directs the actions of the Commissioner of Emergency Management (CEM). A detailed explanation of the enabling authority, roles, and responsibilities for maintaining the PNERP is detailed in Section 2.

4.1.2 PNERP Support

It is expected that provincial senior management will support the PNERP in all aspects by:

a. Promoting the requirements of the PNERP within their organizations.
b. Accepting management roles with PNERP oversight responsibilities within their organizations.

4.1.3 Administration

The approval authority for the PNERP is the Lieutenant Governor in Council. The Minister responsible for emergency management with the support of the CEM, and EMO, leads the development, implementation, evaluation, maintenance, and continual improvement of the PNERP.

4.1.4 Resource Allocation

Senior management of ministries with PNERP responsibilities shall ensure that their ministry has the resources to develop, implement, evaluate, and meet their responsibilities as identified within the PNERP. If those responsibilities cannot be met (e.g., lack of personnel, insufficient resources, lack of training, etc.) senior managers shall report their deficiencies to their superiors for resolution. If the deficiencies persist, the CEM shall be notified for discussion/resolution as needed with the appropriate Deputy Minister.

4.1.5 Program Coordination

Each ministry shall appoint at least one primary and at least one alternate point of contact to coordinate PNERP actions with other ministries. As the lead organization EMO may have more than one point of contact to coordinate and track its PNERP related actions with other ministries.

4.1.6 PNERP Review Committee

4.1.6.1 Establishment

Ministries which have a role as identified in the PNERP shall establish a ministerial review committee and conduct bi-annual meetings to assess the state of their emergency preparedness (see Figure 3). Part of that meeting shall address their responsibilities within the PNERP specifically. Minutes of those meetings shall be shared with the CEM, with outstanding issues and proposed solutions to be reported to the CEM for resolution.

In addition to the DMSC-EM, a NEMCC shall be established. The NEMCC will be comprised of key federal, provincial, and municipal representatives, as well as other invited organizations (e.g., First Nation and Indigenous communities, United States member states, and CNSC licensed facilities). The PNERP review committee includes the NEMCC and others who have the required expertise, knowledge of the organization, and the capability to identify resources from all key functional areas within the organization. Various committees as shown in Figure 3 have a role in reviewing the PNERP.

This diagram is for visual reference only. See the text below this figure for the full description.

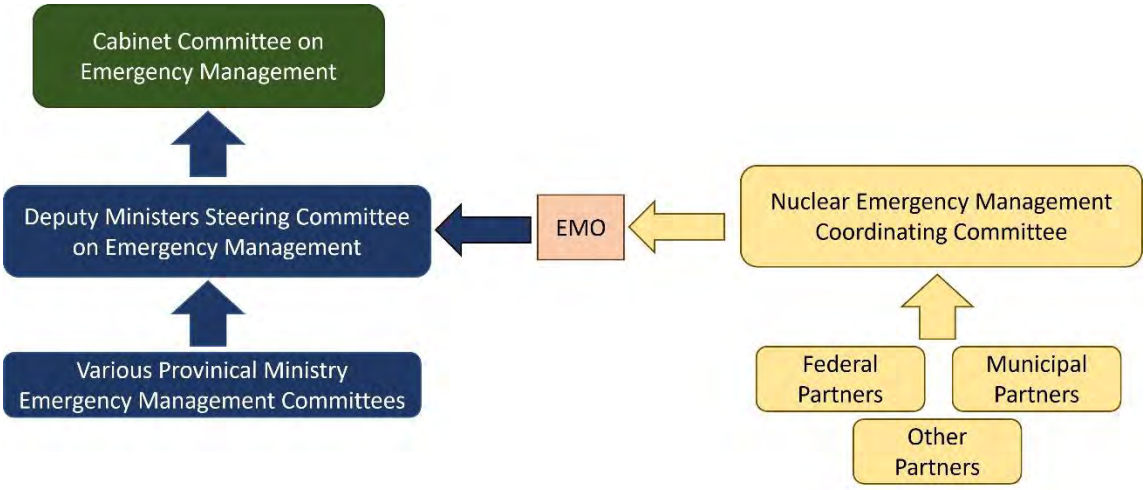

Figure 3: Committee Relationships

The diagram illustrates the committee relationships within Ontario's nuclear emergency management structure. At the top is the Cabinet Committee on Emergency Management, followed by the Deputy Ministers Steering Committee on Emergency Management, and ending with the Various Provincial Ministry Emergency Management Committees. Emergency Management Ontario is positioned alongside this structure and connects to the Nuclear Emergency Management Coordinating Committee, which includes Federal Partners, Municipal Partners, and Other Partners. Arrows indicate the relationships among these committees.

4.1.6.2 Strategic Input

NEMCC, sub-committees and working groups should provide strategic input and direction to their organizations to ensure the coordination, development, implementation, evaluation, maintenance, and continual improvement of the PNERP. Minutes shall be maintained for each meeting.

5. PLAN, VALIDATION, AND COMPLIANCE

5.1 Review and Evaluation Process

This document is subject to review and administrative amendments in collaboration with ministries and consultation with key interested parties. This process is the responsibility of Emergency Management Ontario. Interested parties are encouraged to review and evaluate this plan as they use it and to submit comments and suggestions where applicable.

During training or exercises, the PNERP and its supporting elements may be evaluated. Resulting lessons-learned should identify gaps or opportunities for improvement (OFI) as well as best practices.

Evaluation of PNERP related documents should not only assess whether they are complete, but whether they meet the overall goals of the program and international best practices. Gaps and OFIs should be addressed through a defined issue resolution process including the development of an after-action review (AAR) detailing recommendation on how the gap or OFI shall be resolved. The process should assign responsibility to an OPI and prescribe a timeline for resolution. Best practices should also be assessed for their potential to be incorporated into other aspects of the PNERP or within other response organizations.

5.2 Assessment and Benchmarking

Internal assessments of the PNERP are accomplished through internal reviews, audits, and lessons-learned from exercises.

External assessments of the PNERP are accomplished through external audits (e.g., Auditor General) or though assessment by an international body (e.g., IAEA).

PNERP benchmarking is accomplished through comparison with other response organizations or via lessons learned from real-world nuclear emergency responses.

5.3 Audits

Internal audits, external audits, and non-conformance reports will also identify gaps and OFIs. Provincial senior management shall ensure that issue resolution is monitored and that the OPIs adhere to prescribed timelines, effectively communicating progress and any potential for delay. Lead agencies, as identified in the PNERP, shall coordinate issue resolution with any additional agencies involved.

5.4 Change Validation

After-action plans and audits that result in updates to the PNERP shall be documented and trained during subsequent PNERP management courses. New policies, plans,

procedures, and other supporting documents shall be tested during training and ongoing exercises. An internal EMO exercise (e.g., a tabletop exercise) shall be conducted specifically to evaluate the effectiveness of changes to the relevant documents. Evaluation of the exercise should aim at testing and evaluating the changes and making further updates as required to ensure the changes are valid and meeting their intended purpose. The new or revised policies, plans, procedures, and other supporting documents should not be tested for the first time in a full-scale exercise with external organizations playing.

5.5 Records Management

Records management within the PNERP shall comply with the Corporate Policy on Recordkeeping, Access and Privacy, the Freedom of Information and Protection of Privacy Act, the Municipal Freedom of Information and Protection of Privacy Act as well as the Personal Health Information Protection Act. Other organizations should comply with their applicable records management and privacy legislation.

Documentation associated with the management of the PNERP during the preparedness phase shall be maintained by the province for a period of ten (10) years (e.g., NEMCC meeting minutes, training records, personnel trained, exercises conducted, exercise evaluation reports, audit reports, AARs, etc.).

Records associated with an operational response to a nuclear or radiological emergency shall be kept indefinitely by the province (e.g., time of release for emergency bulletins, time of release for Incident Management System (IMS) documents, time and actioning of immediate actions and protective measures, logs for Provincial Emergency Operations Centre (PEOC) Section Heads, reports from the NGS to EMO regarding plant conditions, dose tracking of response personnel and dose management decisions, etc.).

Records associated with recovery operations shall be maintained indefinitely by the province (e.g., minutes of NEMCC meetings, dose rates in areas being mitigated and exposure records for mitigation effort workers, recovery plans, relocation plans (if required), etc.).

6. NUCLEAR EMERGENCY RESPONSE

6.1 General

Protecting the public during a nuclear emergency is a complex operation that requires the efforts of multiple emergency response organizations. One of EMO's goals is to set the conditions to allow a transition to recovery operations as quickly as possible. This plan details the operational functions that enable transition to recovery operations to be as effective as possible.

6.2 Response Activities

The response to a nuclear emergency includes the following activities:

a. Rapidly identify that a potential emergency is developing
b. Rapidly notify offsite response organizations to the developing situation
c. Activate response organizations to a level commensurate to the developing nuclear emergency, staffed with well-trained individuals in all critical positions
d. Share information from the NGS such that response organizations can take measures to protect the public, as indicated and appropriate
e. Take protective actions and measures to protect the public (Section 6.3)
f. Coordinate response activities between response organizations
g. Ensure the continuity of critical functions
h. Communicate with the public in a timely manner to maintain public awareness and the public's trust in the ongoing response efforts
i. Gather radiological information to develop a complete picture of radiological contamination in the environment following a release (even a filtered release following a DBA)
j. Manage the medical response, including efforts to minimize the psychosocial impacts of the emergency
k. Ensure emergency responders are aware of the radiological risk and take actions to keep their effective doses ALARA
l. Identify when a nuclear emergency can transition from response to recovery operations
m. Establish a recovery organization to manage the long-term recovery efforts.

6.3 Protective Actions

6.3.1 Guiding Principles for Protective Action Decision-Making

Based on the Planning Basis (Annex K), protective actions to mitigate nuclear and radiological emergencies include both precautionary measures (Annex L.1) and protective measures (Annex L.2).

Protective actions are complementary to each other and may be applied in combination as a protective action response strategy appropriate to the situation, while considering their respective efficacies and limitations.

Protective actions should be applied to prevent increased public exposure to radiation. In practice, this may not always be justified as protective actions entail risks and costs, including psychosocial, economic, and health risks to vulnerable populations (e.g., evacuation could result in early or premature deaths). It is therefore necessary to justify and optimize the application of protective actions in order to minimize the total risk or detriment involved (Annex A.2).

In a nuclear emergency, if a protective measure is warranted at any boundary of a response sector in the DPZ (see Annex M) it should be applied to the whole response sector. If a protective measure is warranted in a sector of the Automatic Action Zone (AAZ) and DPZs, it should be applied to the entire ring of sectors in order to ensure protection when winds are shifting.

For areas beyond the DPZ, modeling may be used to identify areas where surveys and sampling are required to assess the need for protective actions. It is unlikely for evacuation to be required as a plume exposure control measure beyond the DPZ.

6.3.2 Responsibilities for Protective Action Decision Making

The CEM, as operational lead for the offsite response (Section 7.6.2.5.1), has the authority for protective action decision-making where an emergency exists but has not yet been declared to exist.

After a provincial emergency has been declared, certain protective action decisions may be issued by the Lieutenant Governor in Council (LGIC) via provincial emergency order subject to the criteria and conditions outlined in the EMCPA. During a declared provincial emergency, the authority to issue emergency orders under subsection 7.0.2(4) may be delegated from the LGIC to a Minister of the Crown or the CEM. The CEM shall be responsible for communicating the protective actions and the areas they apply to.

Detailed information regarding direction and implementation of precautionary and protective measures will be developed and communicated as follows:

a. During the preparedness stage through the Nuclear Public Education and Awareness Program Framework by the applicable responsibilities as defined in the PNERP
b. During the emergency, through public direction (Section 8.7.2) and through the emergency public information operation (Section 8.7.2.2).

Detailed descriptions of Protective Actions can be found in Annex L.

6.4 Emergency Planning Zones

Emergency Planning Zones (EPZ) are areas in which implementation of operational and protective actions are, or might be, required during a nuclear emergency, in order to protect public health, safety, and the environment. The four planning zones include:

a. Automatic Action Zone (AAZ)
b. Detailed Planning Zone (DPZ)
c. Contingency Planning Zone (CPZ)
d. Ingestion Planning Zone (IPZ)

Ongoing operational response will generally occur within the EPZs based on the action to be taken within the zone. See Annex M for more information on planning zones.

EPZs are determined by the technical methodology and planning basis and are site and technology specific. EPZ maps are included in the site-specific implementing plans.

7. NUCLEAR EMERGENCY RESPONSE ORGANIZATION

7.1 Response Organization Basis

Ontario uses the Incident Management System 2.0 (IMS) framework for nuclear and radiological emergency response with a standardized organizational structure, functions, processes, and terminology. The core functions for a response include the following IMS 2.0 sections:

a. Command
b. Operations
c. Planning
d. Logistics
e. Scientific/Technical
f. Public Information Management
g. Finance and Administration.

The nuclear response organizational structure is illustrated in Figure 4. External interfaces between EMO and other provincial ministries, municipal, and federal Emergency Operations Centres (EOCs) at the command level will be conducted using coordination calls via web-based conferencing systems (e.g., MS Teams, Zoom, etc.) or other means, as required to maintain communication (e.g., telephone-based conference calls). Direct contact between EOC sections may be used to coordinate planning and maximize response efforts (e.g., PEOC Planning and Ministry of Health (MOH) Planning).

This diagram is for visual reference only. See the text below this figure for the full description.

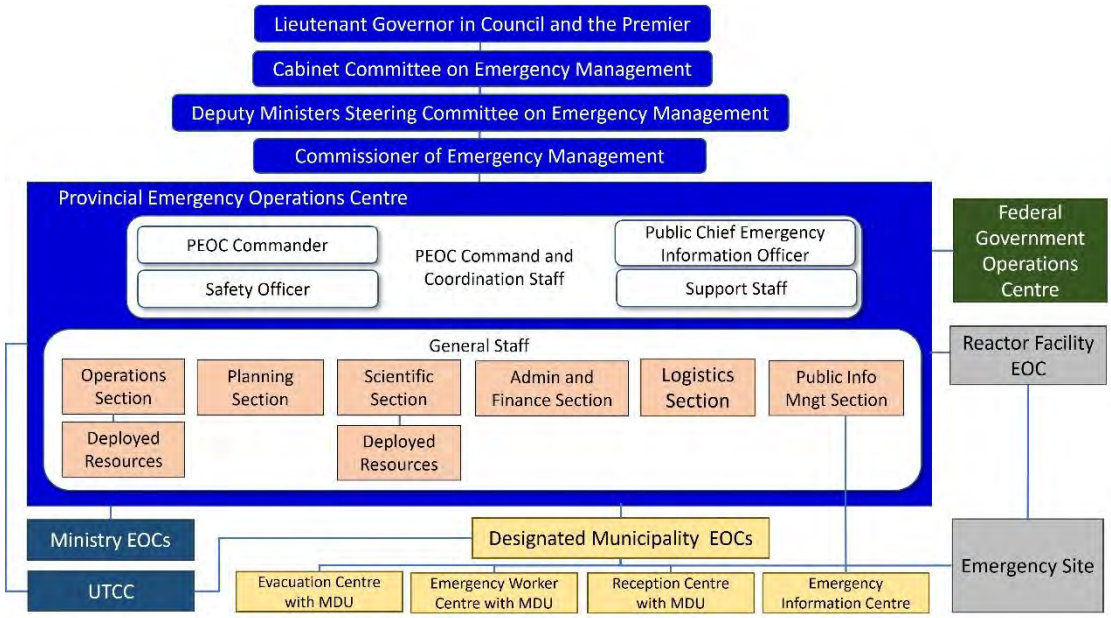

Figure 4: Provincial Nuclear Emergency Response Organizational Structure

The diagram illustrates Ontario's Nuclear Emergency Response Organizational Structure. The Lieutenant Governor in Council and the Premier have the highest authority, followed by the Cabinet Committee on Emergency Management. This is succeeded by the Deputy Ministers Steering Committee on Emergency Management, which provides oversight to the Commissioner of Emergency Management. Below this, the Provincial Emergency Operations Centre (PEOC) serves as the central coordination hub, with PEOC Command and Coordination Staff comprising of the PEOC Commander, Safety Officer, Public Chief Emergency Information Officer, and Support Staff. Supporting this team is the General Staff, organized into six functional sections: Operations, Planning, Scientific, Administration and Finance, Logistics, and Public Information Management, with deployed resources assigned to the Operations and Scientific Sections. The PEOC maintains direct links to Ministry Emergency Operations Centres (MEOC) and the Unified Transportation Coordination Centre (UTCC), as well as to Designated Municipality EOCs, which includes Evacuation Centres, Emergency Worker Centres, and Reception Centres, each supported by Monitoring and Decontamination Units (MDUs). Emergency Information Centres are also part of the Designated Municipality EOC structure. Additionally, the PEOC connects to external jurisdictions, including the Federal Government Operations Centre and the Reactor Facility EOC, which is linked to the Emergency Site. Arrows throughout the diagram represent the flow of authority and coordination across all levels and jurisdictions.

7.2 Concept of Operations

The PEOC provides overall coordination of the provincial response and is led by the CEM. The PEOC provides timely support, information, and analysis to the CEM or delegate (e.g., the PEOC Commander) to co-ordinate the provincial emergency

response and assists communities in responding to nuclear emergencies by providing protective action direction, advice, assistance, and support in coordinating the provision of additional resources. The PEOC may deploy personnel to assist in coordinating the emergency response.

The emergency response effort for the province of Ontario is led by the CEM and the organizations supporting the CEM's efforts (see Figure 4). A nuclear emergency consists of several phases:

a. Urgent Response Phase
b. Early Response Phase
c. Transition Phase
d. Recovery Phase (see Annex N).

The task of the CEM or delegate (e.g., the PEOC Commander) is to direct the offsite response of the Provincial Emergency Response Organization (ERO) to protect the health, safety, and welfare of all inhabitants of the province, and the protection of property and the environment. The following assumptions are made with respect to emergency phases:

a. The Urgent Response Phase consists of the pre-release and the release phases.
b. The Early Response Phase begins when once the nuclear release is brought under control (though not necessarily contained).
c. From the beginning of the release, where an emergency exposure situation exists, the PEOC can contextualize the radiological situation by characterizing the contamination in the environment through survey and sampling efforts. This situation is characterized by a state of situational uncertainty, with potentially high dose rates, an apparent lack of control, and an unknown level of health risks to the local population.
d. An emergency exposure situation continues to exist during the Transition Phase, and protective action decision-making remains complex as other protective actions are considered (i.e., permanent relocation, temporary relocation, lifting food restrictions).
e. Recovery operations will be managed by a Recovery Management Organization (RMO), and the PEOC may return to its normal daily operational tempo.

7.3 The PEOC's Role

The PEOC is the central coordinating hub of the provincial ERO, with other components connecting to it to ensure a coordinated emergency response operation.

7.4 Preparing the PEOC for Nuclear Emergency Response

Several PNERP documents are foundational in preparing the PEOC for NER. These include the:

a. Provincial Nuclear Emergency Preparedness Framework

b. Staffing Support Plan
c. Core Competency Report
d. Nuclear Training Plan
e. Nuclear Exercise Plan
f. PNERP, its Implementing Plans, and associated procedures

7.5 Strategic Command During an Emergency

7.5.1 Lieutenant Governor in Council and the Premier

The LGIC and the Premier of Ontario provide strategic direction to the CEM and Minister responsible for emergency management through the CCEM.

7.5.2 Cabinet Committee on Emergency Management

Appointed by the LGIC, the CCEM provides an advisory role to Cabinet. The membership of the CCEM includes the:

a. Premier and President of the Executive Council
b. Minister responsible for emergency management (Chair)
c. President of the Treasury Board (Vice-Chair)
d. Solicitor General
e. Attorney General
f. Minister of Energy and Mines
g. Minister of Finance
h. Minister of Health
i. Indigenous Affairs and First Nations Economic Reconciliation
j. Minister of Municipal Affairs and Housing
k. Minister of Natural Resources
l. Minister of Public and Business Service Delivery and Procurement

In a nuclear or radiological emergency, additional Ministers may be invited to participate in the CCEM (based on the provincial organizations listed in Annex F).

7.5.3 Deputy Ministers Steering Committee on Emergency Management

The Deputy Ministers Steering Committee on Emergency Management (DMSC-EM) supports the Commissioner/Deputy Minister in operationalizing their respective ministry resources to support the provincial response.

7.5.4 Commissioner of Emergency Management

The CEM is responsible for:

a. Maintaining ongoing communication with ministries and partners involved in the emergency response

b. During an emergency or pending emergency, the CEM shall serve as the direct link between the EMO and DMSC-EM
c. The CEM shall ensure that strategic and operational information and decisions are relayed between EMO, DMSC-EM and CCEM in a timely and effective manner.

7.6 PEOC Sections

Each IMS section and sub-group has identified roles and responsibilities. Detailed checklists and procedures have been developed for immediate actions and ongoing requirements of each section and are located in separate documentation.

Responsibilities of each IMS section are included in Section 7.6.

7.6.1 Command

7.6.1.1 Commissioner of Emergency Management

When a nuclear or radiological emergency occurs, the Commissioner of Emergency Management (CEM):

a. Is primarily responsible for leading the offsite response, on behalf of the province, by supporting and coordinating the emergency response.
b. Except for any emergency order-making powers that may be first delegated to the CEM in accordance with the EMCPA, may delegate responsibilities to others (e.g., the PEOC Commander), as required, to ensure continuity of operations, and the delegate(s) shall be deemed to represent the CEM when assuming responsibilities so assigned in this plan. All CEM responsibilities, as detailed in this PNERP, shall be assumed by a qualified delegate wherever the CEM has so delegated the responsibilities of this position.
c. May issue Operational Directives prior to a provincial emergency declaration or liaise with ministry authorities where appropriate (see Section 7.7.2)
d. Provide advice, assistance, and recommendations to CCEM and the LGIC
e. Shall, where time permits, consult with affected municipalities, First Nation and Indigenous communities on any protective actions that are being considered.

7.6.1.2 Command Staff

Led by the CEM, the role of the PEOC Commander and the Command Staff is to:

a. Approve the Incident Action Plan (IAP)
b. Identify and resolve response issues
c. Identify unresolved issues to be addressed by the strategic leadership team described in Section 7.5
d. Implement direction provided by the CEM
e. Issue operational directives and guidance including emergency bulletins (EBs)

f. Liaise with the command function of other EOCs.

Command Staff include a Safety Officer who is responsible for monitoring, tracking, and ensuring the health and safety of all PEOC personnel, the Provincial Chief Emergency Information Officer (PCEIO), and support staff including the Emergency Bulletin Officer. Liaison Officers (LO) serve as the link between the CEM and other organizations involved in emergency response outside the provincial ERO (e.g., contiguous states).

7.6.2 General Staff

Each Section Chief shall attend regular meetings to provide information and advice to PEOC Command and obtain the necessary information to implement Command decisions.

Additional staff may include technical advisors (e.g., the Chief Medical Officer of Health (CMOH) or delegate, where public health issues require attention).

7.6.2.1 Operations Section

The role of the PEOC Operations Section is to implement the IAP and provide operational input to the Commander.

Led by the Operations Section Chief, the Operations Section is comprised of representation from the following organizations:

a. Provincial ministries as appropriate for the level and stage of the emergency
b. Reactor facilities
c. Federal departments including Health Canada, Public Safety Canada, Department of National Defence, and Canadian Nuclear Safety Commission
d. State representatives from the U.S., as appropriate
e. Non-governmental organizations
f. Others as requested.

Functions performed by Operations Section staff include:

a. Identifying and coordinating the operational requirements of the response
b. Monitoring and coordinating deployed provincial resources
c. Providing operational input to the decision-making process through the Operations Section Chief
d. Implementing operational decisions by issuing advice or directions, as appropriate.

7.6.2.2 Planning Section

The Planning Section, led by the Planning Section Chief, coordinates the development of the IAP, oversees all incident-related non-radiological data gathering and analysis

regarding incident operations and assigned resources, and executes the Protective Action Response Planning Procedure.

The Planning Section is responsible for sharing information between all members of the PEOC, as required.

The Planning Section may include representation and/or input from the following organizations, as appropriate:

a. Provincial ministries as appropriate for the level and stage of the emergency
b. Deployed provincial resources
c. Unified Transportation Coordination Centre (UTCC)
d. Municipal planning team from designated municipalities, designated host municipalities, and support municipalities
e. Federal departments and agencies
f. Others as required.

7.6.2.3 Logistics Section

Under the direction of the Logistics Section Chief, staff arrange for and co-ordinate all material, personnel services, equipment, and resources required to manage and resolve the emergency, including the coordination of all logistical support for staff deployed by the PEOC.

7.6.2.4 Finance & Administration Section

Under the direction of the Finance and Administration Section Chief, staff perform administrative, financial, and staffing duties specific to the emergency. This may include the capture of incident-related costs, maintenance and scheduling of support personnel, maintenance of appropriate support records, and administering procurement contracts as necessary.

7.6.2.5 Scientific Section

The PEOC Scientific Section is led by the Scientific Section Chief, who reports to the PEOC Commander. The Scientific Section Chief is supported by a core team that includes an Admin Assistant, Scribe, and Deputy Scientific Section Chief. The Scientific Section organizational structure is illustrated in Figure 5.

This diagram is for visual reference only. See the text below this figure for the full description.

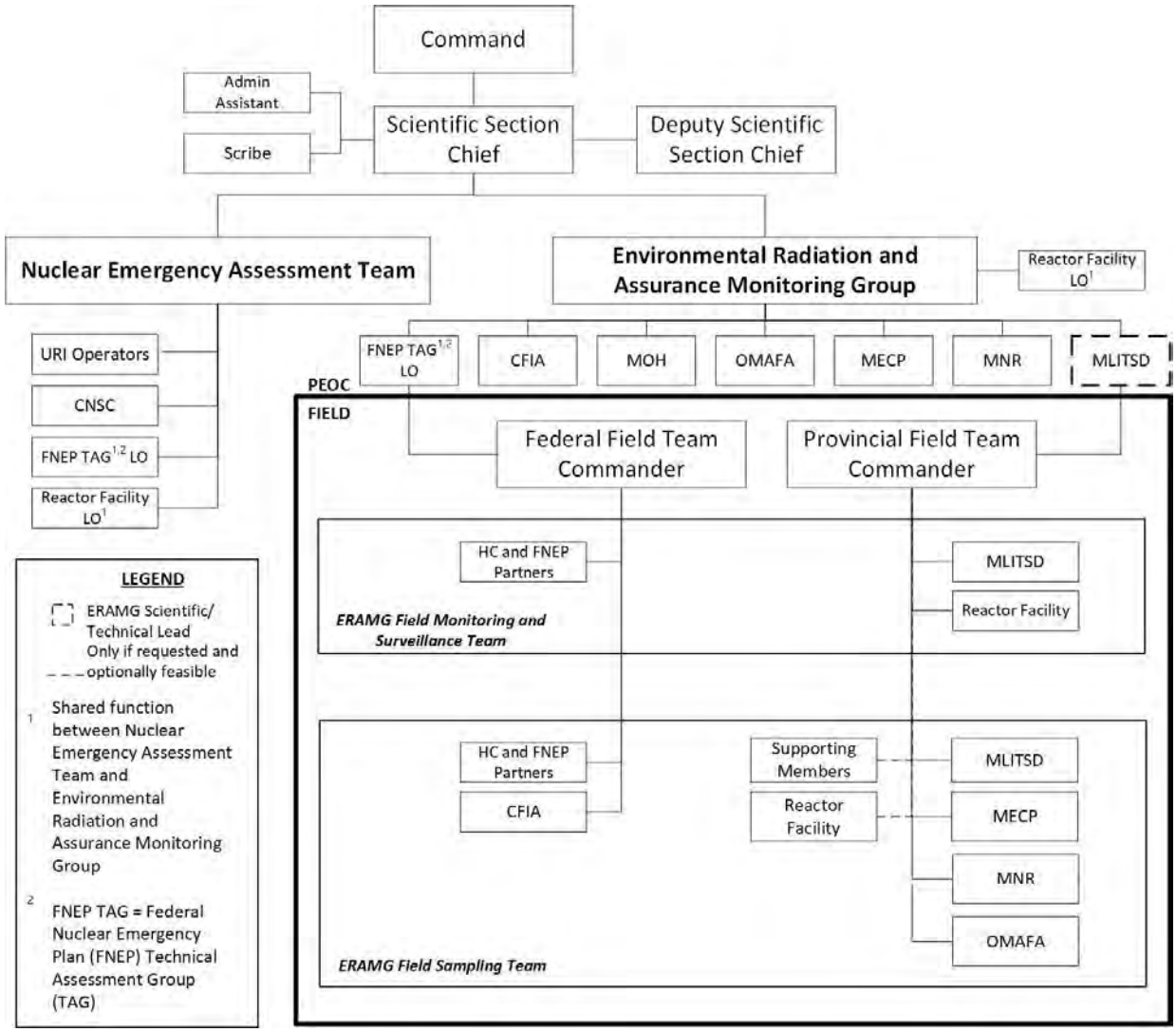

Figure 5: Scientific Section Organizational Structure

The diagram illustrates the Scientific Section Organizational Structure within Ontario's Nuclear Emergency Response Organization. At the top is Command, followed by the core positions of the Scientific Section, which includes the Scientific Section Chief and Deputy Scientific Section Chief, supported by an Administrative Assistant and a Scribe. The Scientific Section Chief reports directly to Command. The structure then branches into two primary groups. On the left is the Nuclear Emergency Assessment Team (NEAT), which includes Unified RASCAL Interface (URI) Operators, the Canadian Nuclear Safety Commission (CNSC), the Federal Nuclear Emergency Plan (FNEP) Technical Assessment Group (TAG) Liaison Officer, and the Reactor Facility Liaison Officer. On the right is the Environmental Radiation and Assurance Monitoring Group (ERAMG), which is divided into two levels. The PEOC-based ERAMG includes the FNEP TAG Liaison Officer, representatives from the CFIA, MOH, OMAFA, MECP, MNR, MLITSD, and the Reactor Facility Liaison Officer. The field-based ERAMG is further

divided into two operational teams: the ERAMG Field Monitoring and Surveillance Team and the ERAMG Field Sampling Team. These field operations are organized by jurisdiction and function to reflect the distinct Occupational Health and Safety requirements for federal and provincial staff. Coordination of these operations is managed through two command roles. The Federal Field Team Commander reports to the FNEP TAG Liaison Officer in the PEOC and oversees Health Canada and FNEP partners in both the Field Monitoring and Surveillance Team and the Field Sampling Team, which also includes CFIA. The Provincial Field Team Commander reports to the MLITSD Liaison Officer in the PEOC and oversees MLITSD and Reactor Facility members in the Field Monitoring and Surveillance Team, as well as MLITSD, MECP, MNR, OMAFA, Reactor Facility, and supporting members in the Field Sampling Team. A legend in the diagram clarifies the role of the ERAMG Scientific/Technical Lead and identifies positions with shared functions between NEAT and ERAMG.

The Scientific Section is responsible for:

a. Conducting air dispersion (plume) modelling
b. Conducting dose assessments and modelling
c. Planning and conducting radiation monitoring, surveillance, and sampling activities in areas surrounding the emergency site and throughout the province
d. Analysing and evaluating the collected monitoring, surveillance, and sampling data.

The results obtained from the actions above are used to formulate recommendations on the adoption of precautionary and protective measures as well as longer term measures or the removal of such measures.

The Scientific Section utilizes data from a variety of sources to inform recommendations on protective measures. Sources include:

a. Ongoing monitoring and surveillance
b. Field-based monitoring and surveillance
c. Field-based sampling
d. Station parameters
e. Source terms
f. Meteorological data and forecasts
g. Dose projections and modelling
h. Plume modelling

The rest of the Scientific Section is divided into two groups that each report to the Scientific Section Chief:

a. The Nuclear Emergency Assessment Team (NEAT)
b. The Environmental Radiation and Assurance Monitoring Group (the ERAMG).

The ERAMG is divided into a component that sits in the PEOC, and a larger component that performs tasks in the field.

7.6.2.5.1 Environmental Radiation and Assurance Monitoring Group

The ERAMG includes representatives from the following organizations:

a. Ministry of Labour, Immigration, Training and Skills Development (MLITSD)
b. Ministry of the Environment, Conservation and Parks (MECP)
c. Ontario Ministry of Agriculture, Food and Agribusiness (OMAFA)
d. Ministry of Health (MOH)
e. Ministry of Natural Resources (MNR)
f. Health Canada, as the lead for the Federal Nuclear Emergency Plan (FNEP) Technical Assessment Group (TAG)
g. The Canadian Food Inspection Agency (CFIA) Ontario Regional Office
h. Reactor facility operators:
    i. Bruce NGS
    ii. Pickering NGS
    iii. Darlington NGS
i. Ad hoc (e.g., Academia or private industry) and Supporting Members (e.g., Dairy Farmers of Ontario).

While Health Canada and the CFIA are the only federal organizations specifically named as members of the ERAMG, other FNEP member departments may be involved in accordance with the FNEP.

The ERAMG is tasked with gathering and analyzing radiological data during a nuclear emergency and is comprised of:

a. A PEOC-based team
b. Field-based monitoring and surveillance teams
c. Field-based sampling teams.

The field-based component of the ERAMG is a large organization that is divided into federal and provincial teams, each of which is led by a Field Team Commander who reports into the PEOC via the PEOC based ERAMG representative(s). Each of the federal and provincial teams have a surveillance and monitoring component, as well as a sampling component.

The federal field-based surveillance and monitoring component is made up of staff from Health Canada and other partners under the FNEP. The federal field-based sampling component is made up of staff from Health Canada, CFIA, and other partners under the FNEP.

7.6.2.5.2 Nuclear Emergency Assessment Team

The NEAT provides technical inputs to the pre-release decision making process by:

a. Calculating projected offsite radiological consequences based on meteorological data, field monitoring data, and source term estimates
b. Performing technical assessments of the developing situation
c. Making recommendations on protective measures when warranted
d. Assigning a safety status to response sectors
e. Providing technical assistance to the ERAMG.

The NEAT's membership includes:

a. URI Operators, who run a software code called Unified RASCAL Interface (URI), where RASCAL is an abbreviation of Radiological Assessment System for Consequence Analysis for Radiological Emergencies
b. A Liaison Officer from the Canadian Nuclear Safety Commission
c. A Liaison Officer from the Federal Nuclear Emergency Plan Technical Assessment Group (FNEP TAG, HC representative or delegate). The FNEP TAG and Reactor Facility Liaison Officers are shared between the NEAT and the ERAMG
d. A Liaison Officer from the reactor facility that is having the emergency.

7.6.2.6 Public Information Management Section

The Provincial Emergency Information Plan (PEIP) details how the public information function shall be carried out, including provisions regarding the following:

a. The PCEIO is responsible for the implementation of the PEIP and leads the Public Information Management Section (PIMS)
b. The coordination of emergency information amongst individual ministry communications branches
c. The flow of information between PEOC Command and General Staff and the PIMS to ensure that Command-identified public information issues are incorporated into the emergency public information messaging
d. A two-way information flow to ensure that the PEOC is made aware of any communications issues that may affect the overall response
e. The dispatch of provincial emergency public information Liaison Officers to local emergency information centre(s) as soon as the need for assistance arises and to ensure an appropriate level of coordination and provision of emergency public information
f. The functions of the PIMS include:
    i. Issuing news releases and other public information products to the media on behalf of the province that describes the nature of the emergency and the measures that the province is taking to manage it
    ii. Coordinating news conferences on behalf of the province and provide supportive documents for provincial spokesperson(s)

iii. Monitoring media, social media, and the public's perception of, and reaction to, the situation and keeping the PEOC Commander and local Joint Emergency Information Centre (J-EIC) informed
iv. Identifying misinformation and countering it with verified and credible information
v. Providing key messages and information to activated call centres.

7.7 Other Operations Centres

7.7.1 Unified Transportation Coordination Centre (UTCC)

The Ministry of Transportation (MTO) is responsible for coordinating the development, maintenance, and implementation of the UTCC, which may be a physical or virtual centre. Within the UTCC, the operations co-leads are the Ontario Provincial Police (OPP) and local police service of jurisdiction.

During a nuclear emergency, the UTCC takes direction from EMO for all operations related to the emergency itself. The UTCC is expected to act independently for resolving issues that arise from nuclear emergency related operations (e.g., an evacuation could cause traffic flow issues far from the affected NGS).

Plans, procedures, roles, and responsibilities for the set up, operation, and decommissioning of the UTCC shall be prepared in advance by UTCC members. Reporting and notification structures under which the UTCC shall operate shall be prepared in advance by UTCC members.

Further guidance on site-specific functions and responsibilities of the UTCC shall be detailed in the PNERP Implementing Plans and site-specific Unified Transportation Management Plan (UTMP).

The MTO is responsible for coordinating the development, maintenance, and implementation of all-hazards evacuation transportation management methodology and site-specific UTMPs for nuclear emergency response and recovery. The UTCC shall be the entity responsible for the implementation of the UTMPs for nuclear emergency response and recovery.

Each site-specific UTMP, shall:

a. Identify decision-making authorities
b. Identify the roles and responsibilities of each participating organization
c. Identify notification, communication, and reporting structures and procedures
d. Identify strategies and mechanisms that may be utilized to manage transportation aspects of evacuation and the transportation impacts in surrounding areas
e. Ensure timely input to public direction and emergency public information processes.

Evacuation Time Estimate (ETE) studies shall be carried out by the operators of the Pickering, Darlington, and Bruce Power NGSs in collaboration with the province, , in order to inform the UTMPs developed for those sites.

7.7.2 Ministry Emergency Operations Centres

In accordance with the requirements of the PNERP, designated ministries shall set up Ministry Emergency Operations Centres (MEOCs) to carry out their respective responsibilities and to direct and co-ordinate provincial ministry actions (including their regional or area offices supporting the affected areas) according to the requirements of this Plan and the directions of EMO.

The Ministry of Health Emergency Operations Centre (MOH EOC) is responsible for managing the ministry response and coordinating health and public health sector actions according to the requirements of the Radiation Health Response Plan (RHRP) and in coordination with the PEOC.

The MOH EOC coordinates a network of health and public health accountabilities exercised out of departmental EOCs or co-located with local and municipal EOCs, as appropriate.

Key health and public health sector partners include:

a. Public Health Ontario
b. Ontario Health/Home and Community Care Support Services
c. Hospitals
d. Local public health units
e. Paramedic services
f. Regional health authorities
g. Government of Canada, Health Portfolio EOC

7.7.3 Federal Government

7.7.3.1 Government Operations Centre

The Government Operations Centre (GOC) coordinates federal activities at the national whole-of-government level, including coordination of activities in support of the PEOC, when required.

Representation includes subject matter experts and LOs from federal/provincial/ territorial government departments and non-governmental organizations.

The GOC may include federal department representatives, based on the requirements of the response, who serve as the link between the GOC and their home institution. They provide knowledge of their home institution including roles, responsibilities, mandates, and plans.

7.7.3.2 FNEP TAG

Led by Health Canada, the FNEP TAG coordinates federal scientific and technical response capabilities and activities in support of the provincial and whole-of-government response. FNEP TAG representation includes subject matter experts from federal departments with nuclear emergency functions under the FNEP.

FNEP TAG Liaison Officers are deployed to the PEOC Scientific Section to serve as the link between the PEOC and the FNEP TAG. They facilitate requests for information and assistance from the province to the FNEP TAG, provide the PEOC with FNEP TAG products (e.g., plume and dose projection models, environmental monitoring data, advice relating to protective action recommendations) and clarify and resolve any technical questions related to FNEP TAG products.

7.7.3.3 Other Department-Specific Operations Centres

Other department-specific operations centres support their departmental roles and mandates and contribute to the integrated Government of Canada response through the GOC.

Some provincial ministries may have established working arrangements with their federal counterparts through these departmental operations centres in the event of an emergency.

7.7.3.4 The Federal Coordination Group and Public Safety Canada – Ontario Region

Coordination of the federal government emergency response activities in Ontario is managed by the Public Safety Canada Regional Office through the Federal Coordination Group.

This is distinguished from the GOC, as the Federal Coordination Group focuses on federal activities within Ontario, while the GOC focuses on the whole federal government.

The Federal Coordination Group is chaired by Public Safety Canada's (PS) Ontario Regional Director and is supported by the PS Ontario Regional Office.

The purpose of the Federal Coordination Group is to:

a. Facilitate federal coordination
b. Provide the space and capacity for executive federal officials to assess an incident
c. Plan the overall strategy of federal actions in response and recovery
d. Receive strategic guidance and direction from senior/executive level decision makers via the GOC.

The Federal Coordination Group is normally the single point of contact between the PEOC and federal government operations during a major response within Ontario.

If, in a rapidly evolving situation requiring urgent coordination, the PEOC directly contacts the GOC, the PEOC shall also ensure that the Public Safety Regional Director/Office is kept informed of those communications.

The PS Ontario Regional Office will, when required, send a federal LO to the PEOC.

7.7.4 Reactor Facility

In the event of a nuclear emergency involving an Ontario NGS, the reactor facility shall make the necessary provisions to conduct their offsite responsibilities through their Emergency Operations Facility. Reactor facility responsibilities shall include:

a. Provision of personnel to offsite provincial and municipal operations centres (e.g., reception centres)
b. Provision of the necessary information and data to the PEOC as described in the applicable PNERP Implementing Plan
c. The provision of equipment and personnel for offsite activities such as field monitoring, emergency worker safety with respect to sector safety status (as part of the NEAT), and personal monitoring (as part of EWC and MDUs).

For reception centres, EWCs and MDUs related to an emergency at the Fermi 2 NPP, EMO supported by resources from nuclear facilities in Ontario and Designated/Host municipalities, will support these centres.
Further details of Reactor Facility responsibilities are found in Annex G.

7.7.5 Community or Municipal Emergency Operations Centres

The municipal emergency response is under the direction and coordination of the Municipal Emergency Control Group at its Emergency Operations Centre (EOC), which in turn receives information, support, and direction from EMO/PEOC.

Municipal EOCs should include representatives of appropriate municipal departments and local boards/regional organizations such as boards of health, social services departments, police services, fire services, paramedic services, public information services, and hospitals, or any other group the EOC determines advice and assistance is required from. Such groups shall provide staff as required for the various other emergency centres to be established.

EMO may deploy Liaison Officers (LOs) to municipal EOCs to act as a link, and to convey information, direction and access to resources.

Further guidance on the function and responsibilities of these centres is provided in the site-specific implementing plans.

7.8 Other Community/Municipal Emergency Facilities

7.8.1 Emergency Facilities

The structure of the municipal organization for undertaking emergency operations, including additional offsite emergency facilities, shall be defined in the municipal emergency response plans for incident site communities and designated/host Municipalities.

Additional offsite emergency facilities include:

a. Reception Centre(s)
b. Evacuation Centre(s)
c. Emergency Worker Centre(s)
d. Emergency Information Centre(s).

7.8.2 Designated/Host/Support Municipalities

Designated municipalities and designated host municipalities for each reactor facility are listed in Annex O. These municipalities are either in the vicinity of a reactor facility or act as a designated/host community for evacuees and as such must address the responsibilities presented in Annex O in their emergency management plans.

Nuclear emergency management plans for designated and host municipalities shall include provisions for the reception, care, and shelter of people (Section 8.10.5) and animals (Section 8.10.6) evacuated from their homes. Further, if the nature of the emergency is such that evacuees may have been exposed to a radioactive plume, these municipalities' plans must also include provisions for accommodating the monitoring and decontamination function (see Annex O).

Support municipalities may be specified by an Emergency Order (Section 8.6.3) and may be responsible for providing support and assistance to designated municipalities.

8. NUCLEAR EMERGENCY RESPONSE FUNCTIONS

8.1 Overview

The emergency response functions identified in this section are consistent with the functional requirements identified in the IAEA's Preparedness and Response for a Nuclear or Radiological Emergency and CSA N1600 – General requirements for nuclear emergency management programs.

8.2 Detection

The responsibility for the detection of an issue that may result in a nuclear emergency rests primarily with the operators of Ontario's NGSs. NGSs in Ontario have robust procedures in place that will provide indicators of a problem that may deteriorate into a nuclear emergency. Hence, a nuclear emergency should not happen without a significant amount of warning (i.e., days or longer). However, the NGS operator is not responsible for offsite response, and thus, does not maintain detection capabilities in the communities and environment outside of its fence line.

The province is responsible for offsite response and shall operate and maintain a detection network capable of detecting a release from a nuclear reactor operating within Ontario, or from a nearby NGS for which emergency planning zones (EPZ) extend into the province. The network shall be capable of continuously monitoring the environment for gamma radiation, and automatically informing the Scientific Section Duty Scientist if elevated readings are detected. The density of the detector network shall be such that it is unlikely for a release to go undetected.

Upon detection of elevated readings where no notification of a problem has been received from an NGS, these readings shall be verified before further response actions are triggered. Verification may be achieved when multiple detectors on the network detect an increase in radiation levels or when an independent detector used in the same vicinity as the initiating detector confirms the presence of radiation above normal background for unknown reasons. Once verified, the province will investigate and if warranted, take the appropriate actions to protect the public as detailed herein.

8.3 Categorization

Categorizing a potential nuclear emergency is solely the responsibility of the NGS operator. Categorizations include:

a. Reportable Event
b. Abnormal Incident
c. Onsite Emergency (notification of Onsite Emergency will include whether a release is expected within 12 hours or is more that 12 hours away)
d. General Emergency.

A detailed list of accident characteristics associated with each categorization is available in Annex P. Following diagnosis of the event onsite, the reactor facility Shift Manager categorizes the event and, within 15 minutes (CNSC Regulatory Document 2.10.1, section 2.2.2(5)(b)), notifies the province, the CNSC and designated municipalities. For each categorization, the PEOC adopts an activation level (Table 1).

Table 1: NGS Categorizations vs. Offsite Response Level

Categorization (Ontario)	Categorization (US)	Offsite Response Level
Reportable Event	Unusual Event	Routine Monitoring (Section 8.4.3)
Abnormal Incident	Alert	Enhanced Monitoring (Section 8.4.4)
Onsite Emergency (notification that no release is expected within 12 hours)	Site Area Emergency (all instances)	Partial Activation (Section 8.4.4)
Onsite Emergency (notification that a release is expected within 12 hours or is ongoing)	General Emergency	Full Activation (Section 8.4.5)
General Emergency		

Within 15 minutes of ending the confirmatory notification call with the reactor facility, the PEOC notifies designated municipalities and the OPP of the provincial offsite response level declared (i.e., the recipients of the Provincial Notification of Offsite Response Form (PNOORF)). At Partial and Full Activation, MTO is also informed.

8.4 Notification and Activation

8.4.1 Initial Notification

Initial notification for a nuclear emergency is defined as the notification made by the reactor facility to designated offsite authorities whenever an event occurs or conditions arise, which require such notification under the prescribed criteria. The initial notification period starts with the first notification and end after four hours, when the Scientific Section is established and receiving adequate information to assess the emergency at the incident NGS.

Canadian reactor facilities are required to notify the CNSC within 15 minutes of activation of their emergency response organization (CNSC Regulatory Document 2.10.1, section 2.2.2).

8.4.1.1 Purpose

The purpose of the initial notification and response system is to:

a. Inform offsite authorities of the fact that a notifiable event or situation has occurred at the reactor facility
b. Provide an indication, through the notification category system, to all relevant partners as to the appropriate initial offsite response in the initial stage of an emergency.

8.4.1.2 Application

These provisions apply to notification from reactor facilities located in Ontario, or those in contiguous American states. The specific initial notification criteria and provincial offsite response level for each reactor facility is described in the applicable site-specific implementing plan.

8.4.1.3 Internal Notification

Each organization or agency required to respond to a nuclear emergency should have an internal notification system to inform all concerned staff of the imminence or occurrence of an emergency under this plan, and of the appropriate response to the notification.

8.4.1.4 External Notification

Organizations or agencies which might be affected by a nuclear emergency under this plan, or which may be required to assist in response, should be notified at an appropriate stage by their links in the ERO. The responsibility for making such notification shall be described in the relevant Implementing Plan.

8.4.1.5 System Outline

In accordance with federal law and regulations under which they operate (i.e., Nuclear Safety and Control Act and CNSC's Regulatory Document (REGDOC) 2.10.1, or applicable US regulations), the reactor facility's nuclear emergency response plan shall include provisions to:

a. Notify offsite authorities of an accidental release or the imminence of an accidental release
b. Describe how these events are determined and categorized
c. Describe the immediate notification process.

A site-specific initial notification and response system has been developed, in consultation with each Ontario nuclear operator, to facilitate a timely emergency notification categorization which corresponds to initial provincial and municipal response actions.

The category notification form from the reactor facility shall include the notification categorization (see Table 1). Where more than one criterion is applicable, the highest

category triggered shall be reported in the notification. The criterion used to determine the categorization is contained within the Implementing Plan for each NGS.

Once notified of an initial categorization or recategorization, the PEOC Duty Officer (DO) will inform the PEOC Commander of the categorization. The PEOC Commander shall inform the CEM or designate and a decision will be made by the CEM or designate (e.g. the PEOC Commander) on the appropriate provincial offsite response level to be adopted. This level should normally be the one linked to the category notification form received from the NGS (as indicated in Table 1) unless another level is judged to be more appropriate. The provincial offsite response level must be transmitted to designated municipalities (Annex O) and other PNOORF recipients within 15 minutes, and then communicated to rest of provincial ERO via IMS Form 201. Site-specific provisions for each provincial offsite response level can be found in the relevant Implementing Plan. Contiguous states and provinces shall only be notified of an Abnormal Incident notification or higher (8.4.4 below).

If, at any time during the emergency, the assessment of the onsite situation changes to warrant a different category than the last one communicated, the reactor facility shall issue an update within 15 minutes of re-categorization to the designated provincial and municipal contact points.

The initial notification period of the emergency ends once ongoing reporting by the reactor facility EOC to the Scientific Section of the PEOC is established. Thereafter, any change in plant situation will be communicated by the reactor facility EOC to the PEOC Duty Officer and the PEOC Scientific Section as part of the regular reporting, as agreed to under the Reactor Facility responsibilities stated in the Annex G.

Following any change in categorization communicated by the reactor facility in this initial stage, the CEM or designate (e.g. PEOC Commander) will normally consult with the Scientific Section Chief or Duty Scientist on the technical situation to determine any need for change to the provincial offsite response level. Otherwise, the CEM or designate (e.g. PEOC Commander) will reference Table 1 to decide on the provincial offsite response level.

The CEM or designate (e.g., PEOC Commander) can, at any time, direct that a change be made to increase the provincial offsite response level.

Once appropriate, the CEM or designate (e.g. PEOC Commander) may terminate the offsite response and notify all relevant partners of the termination.

To ensure a timely and appropriate nuclear notification system, EMO/PEOC shall maintain 24/7 contact point and alternate backup and detailed notification procedures.

Upon receipt of a category notification form from a reactor facility:

a. The CEM or designate (e.g. PEOC Commander) shall confirm the provincial offsite response level to be adopted
b. The PEOC DO shall draft, and the PEOC Commander will authorise the issuance of, a Provincial Notification of Offsite Response Form (PNOORF) and an IMS Form 201 for applicable EROs
c. EROs should, upon receipt of the PNOORF or an IMS 201 or 209, adopt the appropriate, corresponding response level prescribed in their respective plans prepared in alignment with this PNERP
d. Following this initial provincial offsite response level decision, EMO may upgrade/downgrade the response level in support of the operational needs of the situation.

The PEOC, under direction of the CEM or delegate (e.g. PEOC Commander), may adopt a provincial offsite response level that is different from the default provincial offsite response level illustrated in Table 1 (as appropriate) and may notify the ERO on the basis of the chosen offsite response level.

8.4.2 Activation Criteria

When a Partial (8.4.4) or Full Activation (8.4.5) response is adopted, the PEOC shall use the provincial public alerting system (8.7.2.1), which includes the Alert Ready system, to issue an EB informing the affected public of the following:

a. That a problem exists
b. The area affected
c. To stay tuned to the media for further information, and
d. Precautionary and protective measures being directed.

Protective measures may be rescinded using priority communication methods. EBs will not be used to rescind protective measures. Rescinding of protective measures is further addressed in the Provincial Nuclear Recovery Operations Plan (PNROP).

8.4.3 Routine Monitoring

The province, through the PEOC Duty Officer, maintains a state of Routine Monitoring. No additional staff are required to support Routine Monitoring as an offsite response level.

8.4.4 Activation to Enhanced Monitoring or Partial Activation

Upgrading the PEOC's offsite response level will make additional resources available to the CEM or designate (e.g. PEOC Commander) to assess the ongoing situation developing at the NGS. At Enhanced Monitoring, staff provide support to the PEOC Commander as required, and are put on notice that they may be recalled at any time. At this time, the Duty Scientist assumes the role of the Chief of the Scientific Section and recommends to the PEOC Commander whether the remainder of the Scientific Section

should be placed on standby or activated. EMO is responsible for managing and maintaining a system to recall PEOC staff in the event a recall is required (as of publication, Send Word Now).

At Partial Activation, PEOC staff are notified of the activation level and will assess and respond to the emergency (see Figure 66) as the situation dictates. Recalled personnel will report to the PEOC or Alternate PEOC as directed and will be accounted for upon their arrival. Rotation of personnel will be noted in the appropriate log for each section, and planning for rotation should begin once the section is activated. Depending on the situation, virtual attendance in the PEOC may be authorized by the CEM or designate (e.g. PEOC Commander).

8.4.5 Full Activation

At full activation, the PEOC shall be fully staffed 24/7 and organized as seen in Figure 6. Provincial actions normally taken when Full Activation is ordered are listed in Section 8.5.4. It should be noted, however, that these actions may need to be adapted at the time to suit the specific situation/local conditions.

This diagram is for visual reference only. See the text below this figure for the full description.

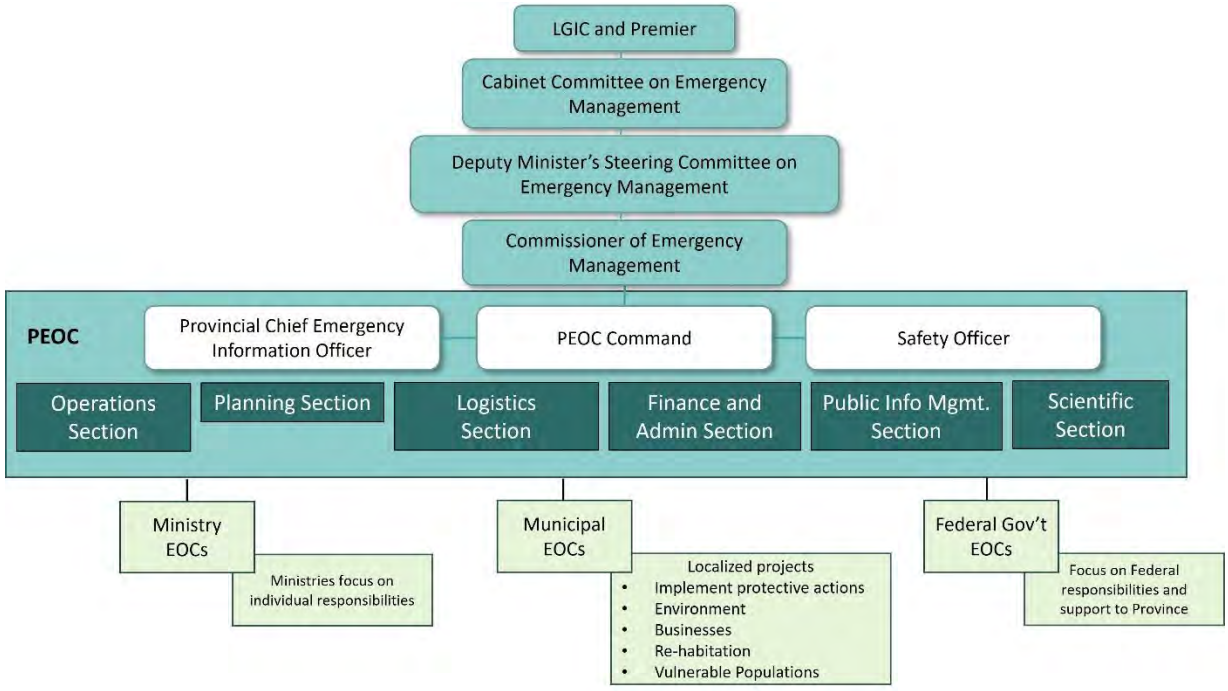

Figure 6: PEOC at Partial and Full Activation

The diagram illustrates the Provincial Emergency Operations Centre (PEOC) at Partial and Full Activation. The Lieutenant Governor in Council and the Premier have the highest authority, followed by the Cabinet Committee on Emergency Management. Directly beneath them is the Deputy Minister's Steering Committee on Emergency Management, which connects to the Commissioner of Emergency Management. The Commissioner links to the PEOC, where Command and Coordination staff include PEOC Command, the Provincial Chief Emergency Information Officer, and a Safety Officer. Below this group are six functional sections: Operations, Planning, Logistics, Finance and Administration, Public Information Management, and Scientific. These sections are connected to Ministry Emergency Operations Centres (EOCs), Municipal EOCs, and Federal Government EOCs.

8.4.6 Activation of ERAMG and NEAT

At Enhanced Monitoring, the NEAT will be fully operation and the ERAMG will be on standby. At PEOC Partial or Full Activation, both the ERAMG and the NEAT will be fully operational.

8.4.7 Activation of the UTCC

Activation of the UTCC begins with MTO Duty Officer’s receipt of the PNOORF indicating a provincial offsite response level of Partial or Full Activation, which will trigger activation of the UTCC by MTO. The MTO Duty Officer will be notified of a potential emergency when the PEOC activates to Enhanced Monitoring, which ensures the UTCC is prepared for a potential escalation of the provincial offsite response level. Following receipt of the PNOORF, the MTO Duty Officer will connect with the appropriate UTCC leads based on the location of the incident to share any initial intelligence, and to confirm the following details:

a. anticipated UTCC membership
b. member roles (including identification of the Operations Lead and the Planning Lead for the UTCC)
c. UTCC meeting time and format (physical/virtual)
d. anticipated operational cycle.

8.4.8 Activation of other EROs

Other ministries will be notified through the release of IMS Forms 201 and 209, which is released by the PEOC Commander at Routine Monitoring or higher. Upon receipt of either form, those EROs will assess the information presented and activate their EOCs as appropriate.

8.4.9 Emergency Organizations’ Situational Awareness

The PEOC functions to support and coordinate the efforts of the entire ERO, comprised of organizations with nuclear emergency responsibilities.

The PEOC shall support the situational awareness of other responding organizations to ensure the timely coordination of response efforts and to minimize duplication of effort when and where possible. Information may be exchanged through the PNOORF, IMS forms, web-based applications (e.g., MS Teams meetings), conference calls, group emails, and dispatched LOs and Field Officers.

The PEOC coordinates the virtual and physical deployment of provincial staff to designated municipalities and host municipalities (see Annex O) emergency operations and information centres, as appropriate and as available, to provide support and advice through liaison with the PEOC.

Event coordination briefings with the provincial ERO, chaired by the PEOC Commander, shall be conducted at least once per operational cycle to:

a. update on the current situation
b. update on major tasks undertaken by each organization
c. discuss new and ongoing risks
d. identify potential conflicts
e. identify new support needs

The PEOC shall outline the response coordination strategy adopted by the ERO for the operational period.

The CEM or delegate (e.g. PEOC Commander) is responsible for ensuring the continuing practice of justification and optimization of protective actions, using the tools developed in the PNEPF and, in consultation with the designated municipalities (Annex I) and appropriate members of the ERO (see Section 5 of the Provincial Emergency Response Plan).

8.5 Initial Response

8.5.1 Routine Monitoring

At Routine Monitoring, the PEOC Duty Officer, the Duty Scientist, and the PEOC Commander are required to monitor the situation until it has been resolved or upgraded to a higher activation level.

8.5.2 Enhanced Monitoring

Enhanced Monitoring is normally adopted in response to an Abnormal Incident (or Alert for US facilities only) notification from a reactor facility. This response level requires some coordination, planning, and/or monitoring activities, without the need to activate the full response organization. As such, staffing of the PEOC may include:

a. IMS Sections shall be stood-up, according to the needs of the PEOC during that operational period(the Duty Scientist or representatives of the Scientific Section would be critical)
b. Ministries and other organizations that are directly involved in the response, deploying LOs to the PEOC
c. Deployment of provincial staff to liaise with communities, depending on the needs of the situation
d. The PIMS, or components of it, at the direction of the PCEIO.

All Routine Monitoring activities continue to be carried out at Enhanced Monitoring, as well as a combination of the following:

a. Contingency planning for protective action response, where an escalation of the situation may be expected
b. Situational assessment by the Duty Scientist based on information and data from the reactor facility in regular transmittals and direct outreach
c. Consideration of the need to pre-position resources (e.g., ERAMG Field Teams)
d. Coordination with designated municipalities, ministries, or other key response partners as applicable
e. Distribution of event-specific information products, as detailed in the PEOC Response Procedures.

At Enhanced Monitoring, the PEOC Commander shall:

a. Set an operational period for the provincial ERO
b. Initiate the development of an IAP, as detailed in the PEOC Response Procedures
c. Utilize the PEOC Duty Team (consisting of operations staff, scientific staff, reactor facility operator representative(s), PIMS staff, and others as required) to monitor the situation and notify the ERO of changes to the provincial response or planned course of action
d. Begin conducting regular event coordination briefings, as detailed in the PEOC Response Procedures.

8.5.3 Partial Activation

At Partial Activation, PEOC and municipal EOCs shall be fully staffed 24/7. Provincial actions at Partial Activation include:

a. Ministry EOCs and the UTCC to be staffed to the level appropriate for the situation, allowing monitoring and assessment of the situation on a continuous basis and implementation of associated plans, as considered appropriate
b. Upon receipt of station parameters from the reactor facility, and prior to a release, dose modelling and potential plume dispersion will be used to determine

the potential impact of a release. The URI dose modelling tool is provided to the Scientific Section by the NGS Operators

c. Within a few hours of notification of an Abnormal Incident or higher, the Scientific and other Sections of the PEOC are staffed, as appropriate to the response level, and an operational/business cycle for information gathering and decision-making is established to facilitate the response
d. In the event that protective measures become necessary, the following EMO actions will be assessed and decided upon by the CEM or designate (e.g. PEOC Commander):

    i. The need to upgrade to Full Activation response
    ii. The need for, and timing of, precautionary measures, operational actions, and/or urgent response actions and applicable recommendations to provincial executive authorities; these are based on Scientific Section technical assessment / recommendations and operational situation input / recommendations of the other PEOC Section Chiefs
    iii. The need for, and timing of, EBs to be issued to the affected public
    iv. The need for emergency public information issued through news releases and media briefings
    v. The need for the establishment of a J-EIC (by MEPR Communications Branch) to facilitate media briefings. These would be held by a provincial spokesperson together with representatives from other ministries, federal departments, the reactor facility, designated municipalities and other organizations as necessary
e. This system of operational and technical awareness, decision-making and emergency public information sharing is repeated, according to the PEOC operational/business cycle, until the emergency response is terminated.

Actions taken at the municipal level include:

a. Municipal ERO staff directed to begin taking response actions, or be on standby to take response actions, as appropriate
b. Issue notification placing municipal departments on standby
c. Emergency Information Centres (EICs) are to be established
d. Other emergency centres (e.g., Emergency Worker Centres, Evacuation Centres, Reception Centres) readied to become operational without undue delay.

8.5.4 Full Activation

At Full Activation, all of the actions required at Partial Activation will be completed or continued as necessary. Organizations and emergency centres that are stood-up at Partial Activation are now fully operational.

In the early hours of a nuclear emergency, the full technical assessment and decision-making capabilities will require time to stand up. In the highly unlikely event that the initial notification indicates an ongoing or imminent release, and where protective

actions must be implemented to protect public health and safety prior to a full technical assessment being completed, urgent response actions have been pre-defined for the Commissioner of Emergency Management or designate (e.g. PEOC Commander). Urgent response actions are implemented to protect public health and safety until a more fulsome technical assessment can be undertaken, which may result in revision to protective actions. Urgent response actions include:

a. Public alerting initiated via provincial (Alert Ready) and municipal alerting systems
b. The CEM or designate (e.g. PEOC Commander) will normally:
    i. Direct the implementation of protective measures in the AAZ and/or the DPZ
    ii. Direct precautionary measures in the AAZ and DPZ
    iii. Issue an emergency bulletin advising the affected public of the above protective measures and where to get more information.

Following the implementation of the urgent response actions above or, for an escalating emergency that has reached Full Activation, the CEM or designate (e.g. PEOC Commander) determines the need for additional protective measures:

a. Through an assessment of potential venting strategies (see Section 8.6.4) based on NGS plant parameters
b. Through the Scientific Section's assessment of field survey results from ERAMG
c. If so required, evacuations will be initiated for those sectors of the DPZ, closest to the reactor facility, in all directions, prior to the radioactive release. The entire DPZ may be directed to evacuate if NEAT's analysis determines the need (Note: federal agencies (e.g., Transport Canada, the Canadian Coast Guard, etc.) are notified about protective measures through the GOC)
d. Where there is a risk of contamination of food, water, milk, or commodities, the public may be advised to:

    i. Remove animals from pasture and place grazing animals on stored (covered) feed
    ii. Restrict the consumption of some sources of drinking water until further assessments are performed
    iii. Restrict consumption and distribution of local produce, foods that are fished, foraged or hunted, milk from grazing animals, rainwater, and animal feed
    iv. Restrict distribution of commodities until further assessments are performed.

Actions undertaken at the municipal level include:

a. Local Reception and Evacuation Centres are fully operational at pre-determined facilities outside of the DPZ; reception Centres can provide registration and

inquiry, health, and psychosocial services, as well as connections to accommodation for those who require it

b. Local Monitoring and Decontamination Units (MDU) are fully operational and located at fixed sites, including co-location with Reception Centres or, may be set up as mobile units for deployment where needed. These offsite facilities provide monitoring and decontamination facilities
c. Local Emergency Worker Centres (EWC) are fully operational and established at pre-determined facilities outside the DPZ and are used as an entry control point for all offsite municipal, provincial, and federal emergency workers accessing the affected area; appropriately trained staff resource the EWC to provide monitoring and decontamination services.

Once the radioactive emission (if any) has ended, the ERAMG:

a. Dispatches field-based survey and sampling teams who, using vehicle, ground, and aerial surveys, report on areas of deposition within and beyond the DPZ
b. Assesses the results of surveys against pre-determined Operational Intervention Levels (OILs), to identify areas where additional protective actions or further investigation may be warranted, or where protective measures should be rescinded
c. Dispatches field-based sampling teams once the boundaries of hot spots and other areas of contamination have been determined; these teams conduct sampling of milk, water, crops, produce and food and foodstuffs that are fished, foraged, or hunted to support the determination of the continued need for ingestion control measures. Foodstuff that may fall outside of any other category, but could still serve as a route of radiological exposure, include honey, maple syrup, herbs used for smudging, etc.

8.5.5 Nuclear Emergency Response Phases

Nuclear emergency response phases are used to situate response organizations in the timeline of an emergency as it relates to a potential release of nuclear material from the incident NGS. There are four phases to a nuclear emergency; the Urgent Response Phase, the Early Response Phase, the Transition Phase and the Recovery Phase (see Table 2).

Detailed information on these phases can be found in Annex N. The Recovery Phase is detailed in greater detail in the Provincial Nuclear Recovery Operations Plan.

Table 2: Nuclear Emergency Phase vs. Release Phase

Nuclear Emergency Phase	Release Phase
Urgent Response Phase	Pre-release and release. No radiological situation pre-release. An unknown radiological situation as the release is ongoing.

Early Response Phase	Release and post release. An emergency exposure situation.
Transition Phase	No further release possible.
Recovery	No further release possible. An existing exposure situation for local residents. A planned exposure situation for recovery workers.

8.6 Ongoing Response

Ongoing response requirements will be communicated to the EROs using updated PNOORF (designated recipients) and IMS Form 201 or 209 and coordination calls (e.g., MS Teams) as detailed in Table 3.

Table 3: Initial PEOC and Municipal Response Actions at each provincial offsite response level

Provincial Offsite Response Level (Associated Categorization)	PEOC Actions	Municipal Response	Emergency Information / Emergency Bulletins / Public Alerting
Routine Monitoring (Reportable/ Unusual Event)	1. PEOC issues PNOORF informing municipality (and others per form) of provincial offsite response level. 2. Members of provincial ERO notified, as appropriate. 3. PEOC monitors event. 4. Duty Scientist consulted. 5. Communications Branch staff consulted, as required.	1. Emergency response (ER) staff remain in touch with the PEOC.	1. No EBs or Public Alerting will be issued at this Response Level. 2. News releases may be required if the public learns about, and is concerned about, the reportable event (this shouldn't happen).
Enhanced Monitoring (Abnormal Incident/Alert)	1. PEOC issues PNOORF informing municipality (and others per form) of provincial offsite response level.	1. ER staff monitor event, preferably from municipal EOC.	1. If appropriate, MEPR Communications Branch/PEOC PIMS coordinates issuance of news release(s).

Provincial Offsite Response Level (Associated Categorization)	PEOC Actions	Municipal Response	Emergency Information / Emergency Bulletins / Public Alerting
	2. Members of Provincial ERO notified, as appropriate. 3. PEOC monitors event. 4. Scientific Section may be fully or partially activated depending on situation. 5. EB Officer activated. PIMS is not normally activated at this level, but Communications Branch consulted as required.		
Partial Activation (Onsite/Site Area Emergency (US facilities only) with no ongoing or imminent release)	1. PEOC issues PNOORF informing municipality (and others per form) of provincial offsite response level. 2. Provincial EROs notified. 3. All PEOC IMS Sections fully activated and staffed. 4. Scientific Section is activated. 5. PIMS activated. 6. Ministry EOCs activated 7. UTCC set up and staffed as appropriate. 8. Precautionary measures directed as appropriate.	1. Issue notification to municipal EROs as appropriate. 2. Municipal EOC, EIC fully staffed and operational. 3. Other emergency centres readied to become operational without undue delay.	1. EB issued (unless deemed inappropriate). 2. If appropriate, the PIMS issues a news release to supplement the EB, and/or amplifies the EB on social media

Provincial Offsite Response Level (Associated Categorization)	PEOC Actions	Municipal Response	Emergency Information / Emergency Bulletins / Public Alerting
Full Activation (Onsite/Site Area Emergency with ongoing or imminent release or a General Emergency)	1. PEOC issues PNOORF informing municipality (and others per form) of provincial offsite response level. 2. Members of provincial ERO notified. 3. All PEOC IMS Sections fully activated and staffed. 4. Scientific Section is activated. 5. PIMS activated. 6. Ministry EOCs set up and staffed as appropriate. 7. UTCC set up and staffed as appropriate. 8. Protective (including precautionary) measures directed, as appropriate.	1. Issue notification activating municipal ER organization. 2. Municipal EOC, EIC, and other offsite centres activated (e.g., reception centres), fully staffed and operational. 3. Implement protective measures, as directed by PEOC.	1. PEOC directs initiation of public alerting. 2. Municipality initiates public alerting. 3. PEOC issues EB. 4. If appropriate, the PIMS issues a news release to supplement the EB, and/or amplifies the EB on social media.

8.6.1 Precautionary Measures

Precautionary measures (Annex L) are not considered urgent response actions (8.6.2), since the timing of their implementation is at the discretion of the CEM or designate (e.g. PEOC Commander), depending on how the nuclear emergency is unfolding. Precautionary measures should be implemented during the Urgent Response Phase either in advance of, or in combination with, urgent response actions to facilitate their implementation. Unlike protective measures (see Section 8.6.6), precautionary measures are not associated with a numerical intervention level.

The CEM or designate (e.g. PEOC Commander) will direct precautionary measures as appropriate, and where time permits, in consultation with applicable responsible ministries and the affected designated municipalities. Designated and host

Municipalities, First Nation and Indigenous communities, and implicated ministries will be informed of precautionary measures by the PEOC.

Precautionary measures shall be directed via EBs (Section 8.7.2.2) issued by the CEM or designate (e.g. PEOC Commander) to the affected public as well as through normal public communications channels.

Precautionary measures include (but are not limited to):

a. closing of beaches, recreation areas, etc.
b. closing of workplaces and schools
c. suspension of non-critical patient admissions in hospitals
d. entry control.

8.6.2 Implementation of Urgent Response Actions

Urgent response actions are implemented once a specific categorization is made by the NGS. Urgent response actions are taken in pre-designated zones, with each zone created for a specific purpose.

8.6.2.1 Actions at Onsite Emergency (no anticipated release within 12 hours)

If the release is not anticipated within 12 hours, but the NGS has categorized an Onsite Emergency, the PEOC should adopt Partial Activation (for details, see Section 8.4.4), and shall initiate the appropriate internal and external notifications, including the municipal contact points and host communities. In addition, consideration shall be given to issuing an EB (Section 6.4), a news release or both. Finally, Ministry EOCs and the UTCC are to be established and appropriately staffed.

8.6.2.2 Actions at Onsite Emergency (release anticipated within 12 hours)

If a release is anticipated within 12 hours, the PEOC shall activate to Full Activation and shall initiate the appropriate internal and external notifications (Section 8.4.1.3 and Section 8.4.1.4 respectively), including the host community.

Once the release is expected to occur within 12 hours, the PEOC shall:

a. Issue the appropriate emergency bulletin (Section 8.7.2.2)
b. Issue operational directives implementing the following urgent response actions, unless there are good reasons for modifying this response, for:

    i. Sheltering (Annex L.2.2) in the AAZ
    ii. Suspension of road and rail traffic through the AAZ
    iii. Clearance of all boaters in appropriate lake sectors impacted

c. Issue further EBs, as appropriate

d. Order PIMS staff to issue news releases, as appropriate
e. Order that the UTCC and Ministry EOCs be established
f. Assess venting requirements (see Section 8.6.4)
g. Assess the situation for further action.

8.6.2.3 Actions at General Emergency

Once a General Emergency categorization is received from the incident site NGS, the CEM or designate (e.g. PEOC Commander) shall implement the following actions, unless alternative actions are justified by operational or logistical constraints:

a. Notify and ensure that the municipal contacts have activated the public alerting system (Section 8.7.2.1)
b. Issue the appropriate EB (Section 8.7.2.2)
c. Issue operational directives implementing the following urgent response actions (assuming conditions allow):

    i. Evacuation of the AAZ and appropriate Lake Sectors
    ii. Suspension of road, rail and air traffic throughout the AAZ
    iii. Precautionary measures in the DPZ

d. Assess venting requirements (see Section 8.6.4)
e. If emission is ongoing, or if evacuations will not be completed prior to emission, issue operational directives implementing the protective measures for:

    i. Evacuees caught in a release (e.g., those that self-evacuated when they were asked to shelter-in-place) shall report for contamination monitoring or, if not possible, to evacuate to a destination beyond the DPZ and undertake self-decontamination
    ii. For those in the AAZ sheltering-in-place during a release (i.e., those that cannot evacuate), ingestion of potassium iodide (KI) pills (Annex L.2.4)
    iii. Sheltering (Annex L.2.2) in the rest of the DPZ. Otherwise, take this action 4 hours (or, at a time deemed appropriate) before the expected time of commencement of the release.

f. Adopt Full Activation (8.4.5), and initiate the appropriate internal and external notifications (Sections 8.4.1.3 and 8.4.1.4 respectively), including the host community
g. Issue further EBs, as appropriate.
h. Order PIMS staff to issue news releases, as appropriate.
i. Order that Ministry EOCs and UTCC be established (if not done at an earlier stage).
j. Assess the situation for further action.

8.6.2.4 Expanding on Initial Actions

After urgent response actions are taken, the Scientific Section shall assess whether expansion of specific zones is necessary based on projected offsite doses as compared to generic criteria (GC). While a release is ongoing and after it has ended, the Scientific Section shall assess whether expansion of specific zones is necessary based on the field measurements and OILs. The Scientific Section Chief shall make recommendations to the CEM through the PEOC Commander to expand the area for evacuation, expand the area for shelter-in-place, expand the area for ingestion control or a combination thereof as the radiological situation dictates. See Annex Q for more details.

8.6.2.5 Deviation Process

During a provincial declaration of emergency, the LGIC or Premier may authorize operational or protective measures which deviate from the PNERP through emergency orders, subject to the conditions and criteria under the EMCPA. Additionally, the LGIC or Premier may authorize the necessary resources (human, physical, informational, and financial) required to support the proposed deviation.

If deviations are required, then all participating EOCs shall be informed through an updated PNOORF or IMS Forms 201 and 209, as well as during interagency conference calls/web meetings.

8.6.3 Declaring a provincial emergency

8.6.3.1 Authority

The EMCPA sets out provisions for the declaration of a provincial emergency throughout Ontario or in any part of Ontario. Subject to the criteria for declaring an emergency as set out in the EMCPA (see 8.6.3.2 below), the LGIC has the authority by order to declare that an emergency exists throughout Ontario or in any part of Ontario. If in the Premier's opinion the urgency of the situation requires that an order be made immediately, the Premier may also by order declare that an emergency exists.

During a provincial declaration of emergency, emergency orders may be made under sections 7.0.2 and 7.1 of the EMCPA subject to certain criteria, limitations, conditions and other legal considerations.

Additionally, pursuant to subsection 7.0.3(2), if a provincial declaration of emergency order is made under section 7.0.1 of the EMCPA and the emergency area or any part of it is within the jurisdiction of a municipality, the Premier, where he or she considers it necessary, may by order made under this section,

(a) direct and control the administration, facilities and equipment of the municipality in the emergency area, and, without restricting the generality of the foregoing, the exercise by the municipality of its powers and duties in the emergency area, whether under an emergency plan or otherwise, is subject to the direction and control of the Premier; and

(b) require any municipality to provide such assistance as he or she considers necessary to an emergency area or any part of the emergency area that is not within the jurisdiction of the municipality and direct and control the provision of such assistance.

Pursuant to the EMCPA a municipality is authorized to exercise a municipal power in response to an order of the Premier or his or her delegate made under subsection (2) without a by-law.

8.6.3.2 Criteria for a provincial declaration of emergency

Pursuant to 7.0.1(3) of the EMCPA, the following criteria must be satisfied in order to declare a provincial emergency:

1. There is an emergency that requires immediate action to prevent, reduce or mitigate a danger of major proportions that could result in serious harm to persons or substantial damage to property.

2. One of the following circumstances exists:

    i. The resources normally available to a ministry of the Government of Ontario or an agency, board or commission or other branch of the government, including existing legislation, cannot be relied upon without the risk of serious delay.
    ii. The resources referred to in subparagraph I may be insufficiently effective to address the emergency.
    iii. It is not possible, without the risk of serious delay, to ascertain whether the resources referred to in subparagraph I can be relied upon.

8.6.3.3 Termination of a Provincial Emergency Declaration

A provincial emergency declared under EMCPA section 7.0.1 is terminated at the end of the 14th day following its declaration unless the LGIC by order declares it to be terminated at an earlier date. The LGIC may by order extend an emergency before it is terminated for one further period of no more than 14 days. As often as required, the Legislative Assembly may, by resolution and on the recommendation of the Premier, extend the period of an emergency for additional periods of no more than 28 days.

In accordance with subsection 7.0.1(2) of the EMCPA, a provincial declaration of emergency made by the Premier under subsection 7.0.1(1) of the EMCPA is terminated after 72 hours, unless the provincial declaration of emergency is confirmed by order of the LGIC before it terminates.

8.6.3.4 Municipal Declarations

Pursuant to EMCPA Subsection 4(1), the Head of Council of a municipality may declare that an emergency exists in the municipality or in any part thereof and may take such action and make such orders as he or she considers necessary and are not contrary to law to implement the emergency plan of the municipality and to protect property and the health, safety and welfare of the inhabitants of the emergency area.

a. The Head of Council should consider making such a declaration whenever the municipal nuclear emergency plan is activated.
b. When a radiological emergency occurs that requires activation of the local emergency plan, the Head of Council should consider declaring an emergency.

Pursuant to EMCPA Subsection 4(3), heads of council shall ensure that the Minister responsible for emergency management is notified forthwith whenever a municipal emergency is declared or terminated.

Pursuant to subsection 4(4) of the EMCPA, the Premier may at any time declare that an emergency has terminated.

Pursuant to EMCPA Subsection 4(2), the Head of Council or the council of a municipality may at any time declare that an emergency has terminated.

Considerations for municipalities declaring an emergency could include:

a. Whenever the municipal nuclear emergency plan is activated.
b. When a radiological emergency occurs that requires activation of the local emergency plan.

8.6.4 Venting Containment Strategy

8.6.4.1 General

Should the reactor design require a venting containment strategy, the impacted NGS may need to vent its containment structure to maintain the integrity of the overall containment system. The requirement to vent is typically known well in advance of the intended venting time and this is especially the case for a DBA. Information associated with venting from containment hold-up systems, in an accident situation, can help to inform protective action decision-making.

The Pickering, Bruce and Darlington NGSs are equipped with sub-atmospheric containment systems designed to hold, for periods of time, radioactive material released from failed nuclear fuel in an accident.

8.6.4.2 Aim of the Venting Strategy

The ability to control (within certain limits) the venting of radioactivity from containment systems within a venting window could be useful to allow prior implementation of protective actions before the release takes place.

Information about characteristics of the venting strategy, including wind direction, duration, whether the release is expected to be continuous or intermittent, and whether the release could be interrupted to allow additional protective actions can be used to support protective action decision-making.

Any decision to use a venting strategy (nominal or alternate), should be taken only after consultation among the province, CNSC, the reactor facility, and affected municipalities and First Nation and Indigenous communities which are in, or have traditional territories in, the EPZs or IPZ. In situations where venting must be performed in an urgent manner to protect the structural integrity of containment, the venting consultation may be delayed until after venting has commenced but must be conducted as soon as practicable.

8.6.4.3 Nominal Venting Strategy

In most CANDU reactor accident scenarios, released nuclear materials from the reactor building will be drawn into the vacuum building. The nominal venting strategy is to commence venting through an emergency venting system shortly before the vacuum building repressurizes to atmospheric pressure, and to continue venting at a rate just sufficient to keep containment sub-atmospheric. Controlled venting may be employed for a prolonged period (days, weeks, or months) until the internal sources of airborne contamination are removed, have decayed, or are otherwise contained.

8.6.4.4 Other Venting Strategies

Other venting strategies which do not activate the vacuum building containment system are possible, as well:

a. Radioactivity may be confined to other parts of the containment system, and vented through other exhaust systems, such as the contaminated exhaust stack, which contains equipment capable of removing most of the radioactive materials from the exhaust.
b. As part of post-Fukushima improvements, some Ontario reactors have had passive containment filtered venting systems installed. These systems provide an additional backup to the filtered air discharge system that can run during a station power outage.

8.6.4.5 Containment Venting Responsibilities

Reactor facility emergency response plans include:

a. Provision for a designated person with the authority to initiate and suspend venting.
b. Provision for consultation with the CEM or designate (e.g. PEOC Commander), (CNSC, and affected municipalities and First Nation and Indigenous communities) prior to undertaking any venting activity unless venting must be performed in an urgent manner to protect the structural integrity of containment.
c. A requirement to provide time estimates, to EMO, indicating when venting will be required.
d. A requirement to inform the CEM or designate (e.g. PEOC Commander), as early as possible, if venting must be performed in an urgent manner to protect the structural integrity of containment.

The CEM or designate (e.g. PEOC Commander), as operational lead for the offsite response, should initiate the following measures as appropriate to the venting situation:

a. If time allows, consult with the appropriate relevant partners, including the applicable designated municipalities, and First Nation and Indigenous communities, on the proposed protective measures in light of the venting strategy to be initiated.
b. In coordination with the GOC, Transport Canada and NAV CANADA, notify and restrict air traffic in the vicinity of the station before venting is initiated.
c. In coordination with the police of jurisdiction, and where applicable, the Canadian Coast Guard, notify and restrict marine traffic in the adjacent lake/river sectors before venting is initiated.
d. Confirm through the federal LOs in the PEOC that the federal government has initiated consultations with the affected U.S. jurisdictions.
e. Issue the appropriate EBs to the affected population.
f. Following the decision-making authorities outlined in Section 7.5, direct or recommend the appropriate protective measures in any populated area prior to venting being carried out.

The nuclear emergency response plans and procedures of organizations with responsibilities related to venting should identify how venting decisions are established, documented, approved, and communicated.

8.6.5 Liquid Emissions

Where a liquid emission has occurred at a reactor facility in conjunction with an event that meets the notification category system as detailed in the site-specific implementing plans (Pickering, Darlington, Bruce Power), it shall be dealt with under the applicable site-specific implementing plan.

8.6.6 Protective Measures

Protective measures are designed to protect individuals from unnecessary exposure to radiation during a nuclear emergency, thereby limiting any potential health effects. Some of these measures are applied as urgent response actions in the AAZ and the DPZ, once the emergency is categorized as an Onsite Emergency (with an imminent release) or a General Emergency. Protective measures taken at different phases of the emergency have different names and decision-making processes. These include:

a. Urgent Response Actions: Taken based on the categorization from the NGS Urgent Response Actions include sheltering, evacuation, KI pill ingestion, ingestion control, protective clothing, respiratory protection, staying indoors and self decontamination.
b. Protective Actions: Taken during or after a release, based on radiological surveys from the field and compared against OILs; Protective Actions include sheltering-in-place, evacuation, KI pill ingestion, ingestion control, protective clothing, respiratory protection, and decontamination.
c. Long-Term Protective Actions: Taken after all releases have ended and there is an existing exposure situation; these actions include temporary relocation, resettlement, and ingestion control.

Precautionary measures can be taken at any time at the discretion of the CEM or designate (e.g. PEOC Commander). Precautionary measures include closing beaches and recreational areas, closing workspaces and schools, suspension of non-critical patient admission to hospitals, and entry controls.

Table 4 provides a summary of protective measures for each response phase.

The guiding principles of the PNERP (Annex A, para A.1 g), dictates that protective measures shall be implemented to prevent acute (deterministic) effects and should be implemented to prevent or reduce chronic (stochastic) effects to the public. Flexibility must be built into all plans and procedures to allow for the optimization and justification of protective measures in response to the changing nature of the emergency and its consequences for public health and safety.

Table 4: Summary of Protective Measures by Offsite Emergency Phase

Urgent Response Phase (Annex N.1)		
Precautionary Measures	**Urgent Response Actions (Implementation Criteria)**	**Ingestion Control Measures**
▪ Closing of beaches, recreation areas, etc. ▪ Closing of workplaces and	▪ Sheltering-in-place (as appropriate; when projections available - Generic Criteria, Annex Q) ▪ Iodine thyroid blocking (ITB) (as appropriate; when projections	▪ Milk control ▪ Water control ▪ Pasture control ▪ Produce and crop control ▪ Livestock control

schools ▪ Suspension of non-critical patient admissions in hospitals ▪ Entry (access) control	available - Generic Criteria) ▪ Evacuation (as appropriate; when projections available - Generic Criteria) ▪ Protective Clothing (as applicable) ▪ Respiratory Protection (as applicable) ▪ Self-Decontamination (as applicable) ▪ Staying indoors (as far as practicable)	▪ Wild-harvested foods control ▪ Food control ▪ Land control

Early Response Phase (Annex N.2)

Precautionary Measures	Protective Actions	Ingestion Control Measures(Based on OILs)
▪ Closing of beaches, recreation areas, etc. ▪ Closing of workplaces and schools ▪ Suspension of non-critical patient admissions in hospitals ▪ Entry control	▪ Evacuation (OILs) ▪ Relocation (OILs) ▪ Protective Clothing (as applicable) ▪ Respiratory Protection (as applicable) ▪ Staying indoors (as far as practicable) ▪ Self-Decontamination (as applicable)	▪ Milk control ▪ Water control ▪ Pasture control ▪ Produce and crop control ▪ Livestock control ▪ Wild-harvested foods control ▪ Food control ▪ Land control

Transition Phase (Annex N.3) and Recovery Phase (Annex N.4)

Long Term Protective Actions	Ingestion Control Measures (New Restrictions Based on OILs)
▪ Temporary Relocation ▪ Resettlement ▪ Rescinded as appropriate.	▪ Milk control ▪ Water control ▪ Pasture control ▪ Produce and crop control ▪ Livestock control ▪ Wild-harvested foods control ▪ Food control ▪ Land control ▪ Environmental decontamination ▪ Rescinded as appropriate

Once the PEOC Scientific Section activates the ERAMG field teams, and data is received and analyzed, these protective measures can be adjusted or new ones directed based on operational, technical (e.g., monitoring data applied against OILs, (Annex Q.2), and public policy considerations.

Detailed information regarding direction and implementation of precautionary and protective measures will be developed and communicated as follows:

a. During the preparedness stage through public awareness and education by the applicable responsibilities as defined in the Provincial Nuclear Emergency Preparedness Framework;
b. During the emergency, through public direction (Section 8.7).

8.6.6.1 Revising Protective Measures

Protective measures implemented as urgent protective actions may need to be revised as the incident evolves as a result of the accident being more or less severe than anticipated, or because of inherent limitations to modeling. After the release has ended, protective measures shall be revised based upon the measured level of radiological contamination in the environment, and the resulting dose rate to individuals in the local area. Revising protective measures (Annex L) shall require the input and analysis of the Scientific Section and the justification for any changes shall be based on the GC and OILs as described in Annex Q. Changes to protective measures shall be documented and approved by the CEM or designate (e.g. PEOC Commander) or the LGIC (as needed in the context of an emergency order during a provincial declaration of emergency).

Protective measures should be adjusted or rescinded when radiological surveys and/or sampling indicate that the action is no longer required in a specific location, or if the burden on an individual for continued implementation realistically exceeds their capacity to comply (e.g. individuals may be limited in how long they can remain sheltered-in-place).

8.7 Emergency Public Communications

8.7.1.1 General

The emergency public information function during an emergency shall be carried out in accordance with the PEIP.

The PEIP describes the means by which prompt and co-ordinated information from the Ontario government is disseminated to the public, media, Members of the Provincial Parliament, other levels of government, Ontario ministries, emergency response organizations, and when appropriate, private sector organizations.

The PEIP shall be activated by the PCEIO to support a Partial or a Full Activation response. The CEM or designate (e.g. PEOC Commander) may consider releasing emergency public information upon adoption of an Enhanced Monitoring response, in which case the PCEIO shall be so directed.

The PCEIO may dispatch provincial emergency information liaison officers to the local Emergency Information Centre as soon as the need for assistance arises or, when a Partial or Full Activation response is implemented. This will ensure that the information being issued locally remains consistent and co-ordinated with that being released by other jurisdictions.

Multiple jurisdictions (federal, provincial, municipal, and non-government interests such as licensees) are involved in the response and, as such, every effort shall be made to ensure that the information being developed is consistent in content and issued in a co-ordinated manner. In this regard, consideration shall be given to establishing a Joint Information Centre, co-ordinated by the PCEIO, whenever the province adopts a Partial or Full Activation response.

8.7.2 Communication requirements during a nuclear emergency

The aim of emergency public communications is to directly communicate to the affected public, direction and guidance regarding protective measures they should take in order to ensure their safety and welfare.

Public direction shall be provided through the co-ordinated release of EBs (Section 8.7.2.2) issued from the PEOC, and broadcast through the media and all other mechanisms normally available to provincial authorities.

Every effort shall be made to:

a. Consult with relevant partners as to the bulletin content if time and circumstance permit
b. Issue EBs in a timely manner to ensure that the affected public has the most up-to-date information as to the actions they may need to take.

Depending on the nature and progress of the emergency, some EBs may need to be prepared or revised during the emergency. Modifying an EB will be conducted by the Emergency Bulletin Officer (a position within the PEOC Command Staff), reviewed by the PCEIO, and approved for release by the CEM or designate (e.g. PEOC Commander).

8.7.2.1 Public Alerting

Public alerting informs the affected population, by means of an appropriate signal, that a nuclear emergency is occurring or is about to occur. For any nuclear emergency where protective measures are required, the PEOC shall use the National Public Alert System

(Alert Ready). Alert Ready delivers critical information to Canadians through television, radio, and Long-Term Evolution (LTE)-connected and compatible wireless devices. In a nuclear emergency, municipal and provincial public alerting shall be initiated on the direction of the CEM or designate (e.g. PEOC Commander) as follows:

a. Whenever notification of a General Emergency has been received from the reactor facility
b. Whenever a notification of an Onsite Emergency (emission ongoing or imminent) is received from the reactor facility
c. As a possible action for notification of an Onsite Emergency (no emission occurring)
d. In all other circumstances as required by the situation.

Public alerting systems, in place at both municipal and provincial levels, shall:

a. Have the capability to complete the alert within 15 minutes of initiation of the system
b. Have the capability to initiate a public alert within 30 minutes of receiving notification of the applicable notification category from the nuclear facility. The time of initiation is defined as the moment the first message is sent out from the alerting system. For the sirens and Alert Ready, this goes to everyone at once. For other systems such as a landline auto-dialer, it would be when the first call is made
c. Alert practically 100 % of the population in the affected area(s). The term “practically 100%” means that the public alert can be heard or received by everyone in the alerting area unless exceptional circumstances provide an impediment.

Some alerting systems have the capability to include audio or text information (e.g., Alert Ready system), while others cannot (sirens). As such, those that can, shall provide information on the protective measures required to be undertaken and the population so affected.

The site-specific implementing plans shall include details on site-specific public alerting requirements or specifications, as appropriate.

Where possible, to ensure that the population receives coordinated and accurate information on what protective measures to take once they have been alerted of an emergency, the timing of the public alerting signals and public direction EBs shall be coordinated with emergency public information being issued. Due to the urgent nature of public alerting and public direction EBs, they may need to be issued prior to more detailed public information being issued.

Authorities having jurisdiction for public alerting shall ensure that an evaluation of any new public alerting system is completed to verify that the requirements under this

PNERP have been met. Further, regular integrated testing of existing public alerting systems shall be undertaken.

8.7.2.2 Emergency Bulletins

EBs are one component of the province's public alerting process and are a decisive form of communications that gets widespread public attention. EBs are required when an emergency results in immediate danger to the impacted area (e.g., an imminent tornado).

EBs contain pre-scripted and pre-approved content used to notify the public of specific situations occurring at an NGS. There are three types of information contained within an EB:

a. Section 1 contains information for public release (e.g., on websites, in print, through social media, etc.) as well as (if practical) a graphic image detailing the emergency planning zones surrounding the NGS site
b. Section 2 contains information that broadcasters must share with the public (e.g., content that interrupts the current signal and is played over the audio on a radio program, or as a ticker on a TV/Cable broadcast)
c. Section 3 contains concise information that can be posted to the National Alert Aggregation and Dissemination System (NAADS), one component of which is Alert Ready. The use of Alert Ready provides the public with information they need to act on immediately to avoid putting their health at risk.

Pre-scripted EBs are drafted by the Nuclear, Radiological and Scientific Response Program Branch (NRSRP) of EMO, in consultation with MEPR Communications Branch and other relevant contributing partners. Pre-scripted EBs are pre-approved by the CEM for immediate release in an emergency. Additionally, EBs that reference KI tablets will also be approved by Ontario's CMOH. EBs that are developed in situ during an emergency are drafted by the Emergency Bulletin Officer residing in the PEOC Command Section, in consultation with the PCEIO, the PEOC Scientific Section and the PIMS and are approved for release by the CEM or designate (e.g. PEOC Commander). EBs are not included in this document due to the level of control maintained over their authorization and use.

EBs are released in accordance with the PEOC's Emergency Bulletin Procedure.

8.7.2.3 Media Communications

Media communications during a nuclear emergency are guided by the PEIP. In addition to EBs, ongoing communications with the media will be required to meet the information demands from the public, and to maintain public confidence in the province's efforts to respond to the nuclear emergency. Press releases and social media post will play a significant role in meeting the demands of the public.

8.7.3 National Public Alert System (Alert Ready)

Alert Ready is an automated web-based notification system that allows the user to broadcast a text message to cellphones within specific geo-locations (i.e., a municipality). Alert Ready is part of the NAADS. The use of Alert Ready provides the public with information they need to act on immediately to avoid putting their health at risk (e.g., a tornado warning).

EMO maintains a 24/7 capability in the PEOC to operate the Alert Ready system. This is done through the PEOC Duty Officer, who is physically present at the PEOC at all times. EMO/PEOC is the authority in Ontario for issuing alerts through Alert Ready for nuclear and radiological emergencies.

8.8 Protection of Emergency Workers

8.8.1 General

An emergency worker is any person performing emergency services in support of emergency response. These workers may be federally or provincially regulated, and as such, the relevant workplace health and safety legislation applies to them. This includes people on the periphery of the response, like personnel staffing the EWC.

The Scientific Section shall periodically (as defined in Scientific Section Procedures) recommend to the CEM, through the PEOC Commander, the safety status colour code for all sectors in which emergency workers may be required to operate. This information shall then be communicated to the ERO by the PEOC Commander, per PEOC procedure.

For those working in the field after a release, sector safety status (Annex R) shall be determined as follows:

a. Initially based on notification category, as described in the relevant Implementing Plan
b. If the release is imminent or ongoing, the assignment of safety status should be done immediately
c. If the release is delayed, the assignment of safety status should be done approximately two hours prior to the release
d. After a release, periodically by the PEOC as described in the relevant implementing plan.

8.8.2 Reactor Facility Responsibility

Reactor facilities shall support EWCs as described in 8.8.3, Annex R, and according to applicable provisions in their emergency plans and procedures.

8.8.3 Emergency Worker Centres

In the event of a nuclear emergency, reactor facilities (except Fermi 2) shall provide the equipment and trained staff to perform monitoring and decontamination activities (pursuant to federal licensing requirements to provide offsite assistance). In Ontario, these activities are provided through EWCs. During a nuclear emergency, EWCs shall be established as an operations centre, with the capability to monitor and decontaminate emergency workers and to maintain their radiation records.

At an EWC, emergency workers shall be provided, as necessary, with:

a. Personal dosimeters and other radiation monitoring devices as appropriate
b. Personal Protective Equipment (PPE)
c. Information and training on safe working precautions.

During a nuclear emergency there may be a need for emergency workers to enter affected areas prior to the establishment of EWCs (e.g., police coordinating an evacuation). Whenever possible, dosimeters should be issued, respiratory protection should be worn and workers going off shift should be checked for contamination. If contamination is found, self decontamination should be conducted until the contamination is removed.

In the event of a nuclear emergency at Fermi 2 in Michigan, EMO shall co-ordinate with relevant partners to establish arrangements for emergency worker monitoring and decontamination for the Town of Amherstburg.

8.8.4 Permissible Dose

The GC for emergency workers (Annex R) sets the overall permissible dose at 50 mSv for the duration of the response. Additionally, Annex R contains:

a. guidance for ensuring emergency worker safety
b. precautions associated with each sector safety status
c. permissible dose for non-licensee emergency workers.

Qualified radiation protection personnel shall be available to provide advice and guidance to employers, supervisors and workers registered at the EWC on how to keep their doses ALARA. This should include, at a minimum, advice on PPE, dosimetry, administrative controls, and any job-specific guidance (i.e., stay time, back-out dose rate, etc.). In addition, accumulated worker doses shall be tracked to ensure that no individual exceeds their permissible dose.

8.8.5 OHSA Responsibilities

The Minister of Labour, Immigration, Training and Skills Development (MLITSD) has the responsibility for administering the Occupational Health and Safety Act (OHSA).

Inspectors are appointed under the Act to enforce the Act, including employer obligations for worker safety during nuclear emergencies.

EROs, whose responsibilities are set out in the OHSA:

a. Should ensure the provision of equipment and training to their personnel, as appropriate, to prevent contamination and enable them to respond to nuclear and radiological emergencies prior to the establishment of EWCs. Equipment should include personal dosimeters, other radiation monitoring devices, and PPE, as required
b. May access the necessary equipment and training through mutual aid agreements or through the reactor facility, as necessary.

In the event of a conflict between the EMCPA or an order made under subsection 7.0.2 (4) of the EMCPA, and the *Occupational Health and Safety Act* or a regulation made under it, the *Occupational Health and Safety Act* or the regulation made under it prevails (EMCPA, s.7.2 (8)).

During a nuclear emergency:

a. Designated municipalities shall include provisions in their municipal plans for the establishment of EWCs. Locations for these centres should ideally be able to accommodate the co-location of an ERAMG Field Command Post for field monitoring purposes
b. Reactor facilities shall support EWCs as described in Annex G, and according to applicable provisions in their emergency plans and procedures.

8.8.6 Consultants/Contractors

Any private industry organizations engaged to assist with response efforts will fall under the relevant jurisdiction's health and safety legislative and regulatory requirements for personal protection of their workers. In most cases, this will be the provincial Occupational Health and Safety Act, however federally regulated workplaces will be subject to the Canada Labour Code. Any company hiring consultants or field workers to assist with gathering radiological survey or sampling data shall develop a workplace safety plan, including radiological safety, as appropriate under the OHSA prior to deployment in the field. The plan will detail how worker doses will be managed, and how dose information will be provided to the MOH for tracking in the individual's medical records.

8.9 Radiological Assessments of the Environment

The extent of radiological impacts to the environment is characterised through radiological surveys, which provide near-instantaneous results, and radiological sampling , which requires laboratory analysis of samples. Protective actions may initially be based on pre-release plume modeling, or during rapidly deteriorating situations,

automatic actions. Following release, protective actions can be revised once the radiological situation has been characterized, based on survey results and followed up with environmental radiological sampling (Section 8.6.6).

Knowing the level of radiological contamination in the environment provides:

a. An understanding of the radiological distribution in the environment including short and long-lived isotopes, which leads to effective protective actions
b. Control of the response through the ability to prioritize the mitigation of high dose rate areas
c. An understanding of the potential health risks to individuals tasked with mitigating the contamination in the environment.

Survey and sampling methodologies are further discussed in the ERAMG Plan.

8.10 Additional Protective Measures

8.10.1 General

During the response phase of a nuclear emergency, protecting the public consists of ordering specific actions to be taken at various times (Annex L). Actions initially include urgent protective actions based on NGS categorization. Subsequent protective actions are based on GCs and OILs. As the emergency progresses, additional protective measures may be required to minimize the potential effective dose received by the public; that is, personal monitoring and decontamination.

The CEM or designate (e.g. PEOC Commander) shall, in consultation with the MOH and affected designated municipalities, consider the need for personal monitoring and decontamination, and direct implementation of these measures as appropriate. Medical management of individuals who were contaminated, or received an effective dose during evacuation, may be necessary based on their potential exposure.

8.10.2 Personal Monitoring and Decontamination

8.10.2.1 General

Individuals who have evacuated during or after a release should be monitored for contamination, and decontaminated (Annex L.2.6) as required. Evacuees shall be directed to a reception centre (Section 8.10.3) in host communities where they can be processed through an MDU (see Section 8.10.4). See Annex S for more information.

Reactor facilities in Ontario shall resource and operate mobile MDUs and those co-located with Reception Centres according to applicable provisions in their emergency plans and procedures.

Emergency plans of host municipalities and their nearby reactor facility shall:

a. Be co-ordinated to ensure compatibility and mutual support.
b. Ensure that a personal monitoring and decontamination system meets the requirements specified in Annex S
c. Ensure that adequate resources are in place to monitor and decontaminate the affected population, as far as practical.
d. Ensure that resources are in place to undertake monitoring and decontamination of the facility housing the MDU.

8.10.2.2 Designated Host Municipality

Designated host municipalities shall be responsible for arranging the necessary space and facilities for the accommodation of an MDU. Designated host municipalities may include provisions in their municipal plans for the establishment of Reception Centres and Evacuation Centres which have MDUs co-located.

Where MDUs are located within Reception Centres, municipal emergency plans for nuclear emergencies shall include provisions to ensure:

a. That the Reception Centre procedures and the MDU procedures are co-ordinated to ensure effective and expeditious processing of evacuees
b. The municipal appointment of a Manager of the Reception Centre who shall, during an emergency, have overall responsibility for the efficient functioning of the Reception Centre.

8.10.3 Reception Centres

A Reception Centre is generally the first destination for evacuees. Evacuee monitoring and decontamination may be accomplished either in a Reception Centre that receives evacuees immediately upon leaving the emergency area or in a facility set up separately from the Reception Centre. Reception Centres are organized to perform many of the following functions:

a. registration and inquiry
b. allocation to Evacuation Centres
c. first aid
d. monitoring and decontamination.

In the event of a nuclear emergency, reactor facilities in Ontario shall provide the equipment and trained staff to perform monitoring and decontamination activities (pursuant to federal licensing requirements to provide offsite assistance).

For those individuals unable or unwilling to attend a reception centre, they will be advised to undertake self-decontamination (instructions to be provided).

8.10.4 Medical Management

8.10.4.1 General

First and foremost, medical management shall prioritize lifesaving above all else, regardless of the risk of cross-contamination from a contaminated individual. Medical management, as detailed in the MOH RHRP, shall ensure that monitored and decontaminated individuals (i.e., those that arrive at an MDU) are medically tracked (including those with no contamination) so that their effective dose can be determined post-emergency and appropriate steps taken to monitor potential developing health issues. See Annex S for more information.

8.10.4.2 Responsibility

The CMOH will consider the need for medical management and direct implementation of these measures as appropriate.

Medical personnel trained in the clinical management of radiation injuries shall be designated in the preparedness phase, to the extent possible, and deployed to the impacted area as needed to meet the needs of the local population.

8.10.5 Reception and Care of Evacuees

8.10.5.1 General

Reception Centres is generally the first destination for evacuees who require assistance following evacuation from their homes. After monitoring and decontamination, evacuees may require assistance with lodging and unforeseen expenses.

8.10.5.2 Responsibility

In a nuclear emergency, emergency social service needs (e.g., reception, registration and inquiry, shelter, food, clothing and personal services) for members of the public who have been evacuated from their homes, shall be assessed and provided by:

a. Host municipalities designated under this plan
b. Municipalities directed to provide support or assistance under a provincial emergency declaration (Section 8.6.3).

8.10.6 Protection and Care of Animals

8.10.6.1 General

Any emergency that affects humans may affect their animals whether these are raised for foodstuff production, kept as companion or service animals, or for other purposes, such as in zoos. Pursuant to EMCPA Section 7.0.2, provincial emergency orders can include evacuation of animals under a declared provincial emergency. Regardless of a declaration of provincial emergency, evacuating pets will be a priority for some members of the public and should be accounted for in planning.

8.10.6.2 Responsibility

Provisions for the protection and care of animals should be included in public awareness and education materials in advance of a nuclear emergency. Municipal emergency response plans for nuclear emergencies shall make provisions for the protection and care of all animals, including those left behind during an evacuation. See Section Annex U for more information.

8.10.7 Selection of Evacuees for Monitoring

Evacuees should not be directed to report to an MDU for monitoring and decontamination if it is estimated that the evacuees can exit the affected area before a release occurs. Evacuees may be directed to report to an MDU for monitoring and decontamination if they are unable to exit the affected area before a release occurs:

a. In the case of a filtered release, evacuees from affected sectors in the AAZ and DPZ inner ring should be directed to report to an MDU. Evacuees from affected sectors in the DPZ outer ring should be instructed how to conduct basic self-decontamination of their bodies, equipment and vehicles after evacuating.
b. In the case of an unfiltered release, evacuees from all affected sectors should be directed to report to an MDU, if available. Otherwise, evacuees should be instructed how to conduct basic self-decontamination as per a) above. Follow-up monitoring should be provided in accordance with the RHRP.

8.11 Transportation Management

8.11.1 General

The MTO develops and maintains plans to manage evacuations, and the transportation impacts in the surrounding areas during a nuclear or radiological emergency response.

A site-specific UTMP shall be developed for each of the following areas:

a. Pickering NGS
b. Darlington NGS
c. Bruce NGS
d. Fermi 2 Nuclear Power Plant.

A UTCC, either physically or virtually, shall be resourced to provide overall coordination of the transportation management function and implementation of the UTMPs.

Where UTCC activation is required, the MTO Duty Officer will initiate notification of UTCC member organizations through email and follow-up telephone call to confirm receipt.

The UTCC shall include representatives from MTO, OPP, designated municipalities, designated host municipalities, local police, road and transit authorities, applicable provincial ministries, Metrolinx and others as required. Table 5 lists the UTCC site location based on the incident site.

Table 5: UTCC Location Based on Incident NGS

Incident NGS	UTCC Location	Responsible Organization
Pickering NGS and Darlington NGS	MTO Central Region COMPASS Transportation Management Centre (CRCTMC) 97 Arrow Road, North York, ON M9M 2L4	MTO Traveller Communications Office
Bruce Power NGS	South Bruce OPP Detachment, Kincardine Office 700 Kincardine Avenue, Kincardine, ON, N2Z 0B1	Ontario Provincial Police
Fermi 2 in Michigan	Emergency Worker Centre in the Libro Credit Union Centre 3295 Meloche Rd, Amherstburg, ON N9V 2Y8	Town of Amherstburg, Amherstburg Fire Department

8.11.2 Special Cases

Motorists travelling through the affected area and exposed to a release may be directed (through appropriate broadcasts and social media) to report for monitoring.

Marine craft in the affected area shall be treated on the same basis as evacuees. When required, marine craft may be directed to report to an appropriate harbour or landing for monitoring and decontamination (if required).

The CEM or designate (e.g. PEOC Commander) shall advise the federal GOC of any change in the provincial offsite response level with known or potential impacts to marine, air, and rail travel. Navigation Canada will release a Notice to Air Missions (AKA Notice to Airmen – NOTAM) and the Canadian Coast Guard will issue a Navigational Warning (NAVWARN) in the area surrounding the NGS as communicated to the GOC from the PEOC. Likewise, the Transport Canada, may direct any train passing through

the affected area and exposed to a release to stop at an appropriate station to enable monitoring and decontamination of passengers, as required.

8.12 Response Continuity and Sustainment

The requirement for continuity and sustainment is dependent of the offsite response level. When the PEOC is at Routine or Enhanced Monitoring, operations are conducted with minimal staff. At Partial and Full Activation, the PEOC is manned 24/7, thus requiring shift rotations and a multi-fold number of trained staff to fulfill critical functions. Identifying personnel within the pool of provincial employees who could take on a role within the PEOC during a nuclear emergency should be a priority once the PEOC has any indication that Partial or Full Activation will likely happen. Although identification and training of provincial employees should be accomplished in the preparedness phase, the realities of retirement, changes in employment status, lapsed training, etc. will affect the staffing of some positions. In these instances, just-in-time training may be required to prepare individuals for their nuclear emergency response duties.

As operational lead, whenever a nuclear emergency occurs the CEM may delegate responsibilities, as required, to ensure continuity of operations and the delegate(s) shall be deemed to represent the CEM when assuming responsibilities so assigned in this plan. This would not include any emergency order-making powers that may be first delegated to the CEM in accordance with the EMCPA.

Sustainment includes ensuring the staff on-watch are fed and stay hydrated for the duration of their shift. Many individuals will not take the time to step away from a hectic emergency response situation, however, it is vital to the ongoing operations that individuals do not exhaust themselves. The Logistics Section should be tasked with sustainment planning to ensure the ongoing availability of the limited staff available for operations.

During the transition-to-recovery, the operational tempo may slow down and allow operations to be suspended (or at least minimally staffed) at night. Once the RMO is stood-up and recovery operations are transferred, the PEOC can return to normal, Routine Monitoring operations.

8.13 Transition Phase

Transitioning to recovery is the first step in transitioning from response operations to recovery operations. As the RMO stands-up (see Figure 7), the PEOC is transitioning away from nuclear emergency response towards other issues that may be impacting the province. Additionally, the PEOC may be reverting to Partial Activation, Enhanced Monitoring, or Routine Monitoring.

The ultimate goal of the transition is to remove the PEOC from the ongoing nuclear incident and turning over responsibility for managing the ongoing remediation to other organizations within the provincial government, who have the mandate to manage the

emergency's impact on various sectors of the province's environment and economic base. During the Transition Phase, the partially activated RMO (along with the appropriate ministry), should focus on:

a. Establishing a recovery management plan based on the PNROP;
b. Strategies for restoration of normal socio-economic activities;
c. Monitoring contaminated areas and assessing potential doses to public and workers (through ERAMG members or contracted assistance);
d. Assessing medium- and long-term health hazards that may result in temporary relocation or permanent resettlement; and
e. Maintaining dose registries for emergency workers in cooperation with the MOH.

This diagram is for visual reference only. See the text below this figure for the full description.

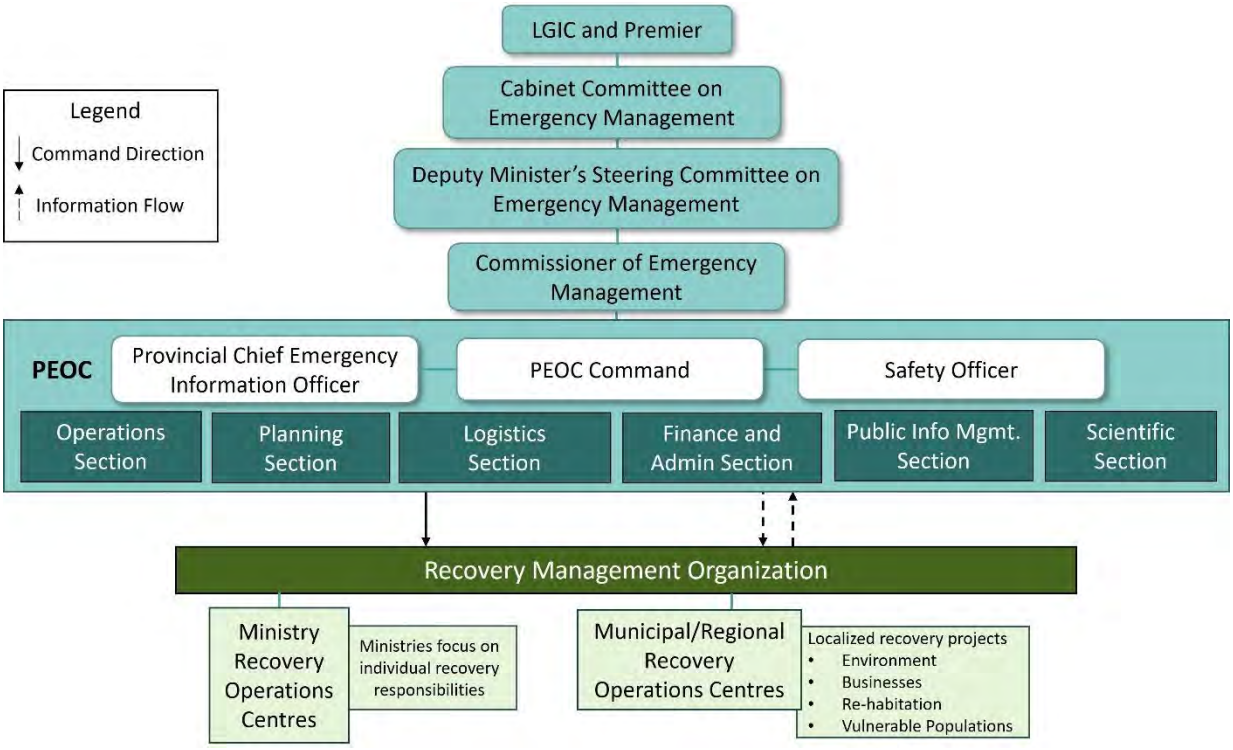

Figure 7: Transition Phase Organizational Structure

The diagram outlines the organizational structure for the province's transition from emergency response to recovery. The Lieutenant Governor in Council and the Premier have the highest authority, followed by the Cabinet Committee on Emergency Management. Directly beneath them is the Deputy Minister's Steering Committee on Emergency Management, which connects to the Commissioner of Emergency Management. The Commissioner connects to the Provincial Emergency Operations Centre (PEOC), which includes PEOC Command, the Provincial Chief Emergency Information Officer, and a Safety Officer. Below this group are six functional sections:

Operations, Planning, Logistics, Finance and Administration, Public Information Management, and Scientific. These sections are connected to the Recovery Management Organization (RMO), which in turn links to Ministry Recovery Operations Centres and Municipal and Regional Recovery Operations Centres. Ministry Recovery Operations Centres are ministries that focus on individual recovery responsibilities, while Municipal and Regional Recovery Operations Centres oversee localized recovery projects. Solid lines represent command direction, while dotted lines indicate information flow.

8.14 Termination

The CEM or designate (e.g. PEOC Commander) may downgrade the provincial offsite response level during the Early Response Phase or when the response enters the Recovery Phase. Considerations for downgrading the provincial offsite response level include:

a. There is no chance for an additional release (as communicated from the incident NGS)
b. The radiological picture is known, and long-term protective actions have been implemented
c. Other EROs are starting Transition Phase actions
d. The public is well informed regarding the emergency and the planned remediation efforts.

Termination of the emergency can be considered once the RMO is fully stood-up and assumes responsibility for managing recovery operations. Further information on recovery operations can be found in the PNROP. Once the emergency is over, the PEOC should revert to Routine Monitoring for other potential emergencies within the province.

ANNEX A. GUIDING PRINCIPLES

A.1 GUIDING PRINCIPLES

The following principles underlie the PNERP and guide all offsite nuclear emergency management in the province:

a. The province, through its ministries, agencies, boards and commissions, has the primary responsibility for the health, safety and welfare of all inhabitants of the province, and the protection of property and the environment.
b. The province shall support and coordinate the response to the offsite consequences of a nuclear emergency and may, where warranted and appropriate, issue operational directives and, in the event of a declared provincial emergency, emergency orders, in accordance with the EMCPA.
c. In the event of a radiological emergency, other than one occurring at a reactor facility, the province's role may vary from providing support to coordinating the response.
d. Even though reactor facilities are designed and operated according to stringent safety standards, emergency preparedness and response must operate on the basis that mechanical failure, human error, extreme natural events, or hostile action can lead to nuclear emergencies.
e. This PNERP details the response for managing the consequences resulting from a reactor facility accident and, as such, does not contemplate its root cause. However, all plans under the PNERP should be able to deal effectively with a broad range of possible emergencies, including severe reactor facility accidents.
f. The protection of the public's health, safety, welfare, property and the protection of the environment must be balanced with other important considerations to ensure that the response actions result in more benefit than harm.
g. Protective measures shall be implemented to prevent acute (deterministic) effects and should be implemented to prevent or reduce chronic (stochastic) effects on the public.
h. Flexibility must be built into all plans and procedures to allow for the optimization of protective and operational measures in response to the changing nature of the emergency and its consequences for public health and safety.
i. Protective actions should be applied to keep the dose to the public ALARA, accounting for the social and economic risks and costs of such avoidance.
j. As far as is practicable, preparedness activities should be undertaken in advance to enable a rapid, effective, and efficient response to a nuclear or radiological emergency.
k. Preparedness activities should include a program of public awareness and education for people who might be affected, to inform them of emergency plans, how they should prepare for an emergency and what they should expect to do in an emergency.
l. Public awareness and education preparedness activities should further ensure ongoing and meaningful consultation with vulnerable populations in consideration of their unique needs.

m. As far as is practicable, operational measures (especially alerting and notification systems) and protective measures should be implemented to avoid significant radiation exposure.
n. A policy of truth and transparency should be followed in providing information to the public and media prior to, during and after a nuclear or radiological emergency.

A.2 PRINCIPLES OF JUSTIFICATION AND OPTIMIZATION

The principle of justification ensures that, in an emergency, protective action decisions result in greater benefit than harm.

Optimization of the protective actions ensures that doses are kept ALARA, while minimizing, as far as possible, health, safety, psychosocial, economic and environmental impacts.

In the planning and preparedness stage, these principles were applied when detailing the following:

a. Appropriate precautionary, protective and ingestion control measures and considerations for their implementation (Annex L)
b. GC and OILs at which to apply protective and ingestion control measures (Annex Q)
c. Initial, pre-determined, protective measures based on the initial notification categories transmitted by the reactor facility (Section 8.5).

The CEM is responsible for ensuring the continuing practice of justification and optimization of protective actions, using the tools developed in the preparedness phase and, in consultation with the designated municipalities and appropriate members of the emergency response organization.

As the emergency progresses through the Urgent and Early Response Phases and ultimately to the Recovery Phase, the time available for recovery planning decreases and with it, the opportunity for public input into the protective measures strategy and the justification and optimization thereof. Recovery Phase planning should be initiated during preparedness and should include a system, as detailed in the PNROP, to prioritize and facilitate the involvement of the affected public and other relevant partner organizations in developing the initial Recovery Phase protection strategy and updating the strategy based on the prevailing conditions. Planning during preparedness will support justified and optimized protective action decision-making during recovery.

A.3 JUSTIFICATION AND OPTIMIZATION OF PROTECTIVE ACTION STRATEGIES

The objective of protective action decision-making is to reduce radiation doses while minimizing, as far as possible, societal, economic, and environmental disruptions.

When applied to protective action decision-making, the principle of justification requires that decisions will result in greater benefit than harm for the affected population.
Upon receipt of an initial notification from the reactor facility of an ongoing or imminent emission, protective measures should be directed by the CEM or designate (e.g. PEOC Commander) subject to operational situation considerations, based on a conservative estimate of the situation.

Urgent response actions (which may include sheltering-in-place, evacuation, and ITB) are pre-defined because time or data may not be available to carry out a comprehensive assessment of imminent risk and are detailed in the applicable implementing plan.

The basis for the justification process rests in calculating projected doses or measuring actual dose rates against GC or OILs respectively. This analysis is detailed in the Scientific Section procedures.

A technical assessment of actual environmental radiation monitoring (surveillance and sampling) results applied against OILs (Annex Q.2) should inform the decision-making process to determine the need for both ingestion control and exposure control measures.

Optimization principles must then be applied to the justified protective action decision to ensure that it is implemented in such a way as to minimize psychosocial, economic, and other disruptions. This application is detailed in the Planning Section procedures.

Justification and optimization reassessment of the protective action strategy should be undertaken regularly as the emergency progresses and as more information becomes available.

Where the effectiveness or safety of a protective action might be impacted by another emergency happening at the same time, the CEM or designate (e.g. PEOC Commander) shall consult with the ministry responsible for that type of emergency to ensure the proposed actions are justified.

Authorities having jurisdiction should develop decision-making guidance and tools for the justification and optimization of protective action strategies during all phases of the emergency.

ANNEX B. PNERP SUPPORT PLANS

B.1 IMPLEMENTING PLANS

The six PNERP Implementing Plans apply the principles, concepts and policies contained in the PNERP, to provide detailed guidance and direction for dealing with a specific nuclear or radiological emergency. The six plans include:

a. Four site-specific Implementing Plans are available to manage the consequences of Nuclear Emergencies at Pickering NGS, Darlington NGS, Bruce NGS, and the Fermi 2 NPP in Monroe County, Michigan, USA
b. An Implementing Plan for a Transborder Nuclear Emergency that addresses nuclear emergencies occurring outside Ontario, other than at Fermi 2 NPP, that could affect the province
c. An Implementing Plan for Other Radiological Emergencies.

In the case of any apparent difference between this PNERP and the relevant implementing plan, the relevant provision(s) of the more recently approved plan should be applied.

B.2 MAJOR ORGANIZATION PLANS

B.2.1 Ministry Plans

Provincial ministries, agencies, boards, and commissions shall develop their own plans and procedures to fulfil the responsibilities as outlined in the Annex F.

B.2.2 Municipal Plans

Pursuant to EMCPA, Section 8, provisions of an emergency plan of a municipality respecting emergencies arising in connection with nuclear facilities shall conform to this PNERP and shall address the responsibilities outlined throughout this document.

Municipalities in close proximity to, or with CNSC licensed facilities within their boundaries, should include in their emergency response plans the measures they may need to take to respond to a nuclear emergency (i.e., the ability to enact the protective actions and measures ordered by the province). This would include details on the relevant notifications to and from the involved organizations.

All municipal nuclear or radiological emergency response plans shall provide for the development of plans and procedures involving local boards (as defined in the EMCPA) and police services operating in the area to provide necessary support and assistance required by such plans, or that may be needed in an emergency.

B.2.3 Reactor Facility Plans

Reactor facilities shall have emergency plans to fulfil their onsite responsibilities as well as to discharge offsite responsibilities in accordance with the Nuclear Safety and Control Act, its associated regulations, and with the responsibilities outlined in Annex G.

B.2.4 CNSC Licensed Facilities Plans

CNSC licensed facilities (nuclear establishments) have plans and procedures for the control of radioactive material and for the notification of offsite authorities in the event of an accident. These plans are in accordance with the Nuclear Safety and Control Act and its associated regulations, as well as with the responsibilities outlined in Annex H.

B.2.5 Federal Emergency Response Plan

Under the Emergency Management Act, the Minister of Public Safety is responsible for coordinating the Government of Canada's response to an emergency. The Federal Emergency Response Plan (FERP) is the Government of Canada's all-hazards response plan.

The FERP outlines the processes and mechanisms to facilitate an integrated Government of Canada response to an emergency.

B.2.6 Federal Nuclear Emergency Plan

The Federal Nuclear Emergency Plan (FNEP), prepared by the Minister of Health, is an annex to the FERP and describes the roles, responsibilities, and capabilities of the federal government to support the scientific and technical response to nuclear or radiological emergencies.

The FNEP contains provincial annexes which outline the provision of scientific and technical support from the federal government to provinces during a nuclear emergency.

The FNEP's Ontario Annex functions as a pre-arrangement between the province of Ontario and the federal government for such support, under Section 6.3 of the federal Emergency Management Act. This arrangement replaces the need for further requests for response assistance. However, requests for support beyond those specified in the FNEP Ontario Annex need to be made pursuant to the normal request for federal assistance process (see Section 7.7.3).

B.2.7 First Nation and Indigenous Communities

Individual First Nation and Indigenous communities are the first line of response in the event of an emergency within their community and are advised to create and implement emergency management plans in order to increase community preparedness.

While there are no First Nation or Indigenous communities within the Detailed Planning Zones (DPZ) of any of Ontario's reactor facilities, First Nation and Indigenous communities that are, or have traditional territories, within the Ingestion Planning Zone (IPZ) for reactor facilities should consider in their plans, the need to undertake ingestion control measures (refer to the site-specific implementing plans for the Pickering NGS, Darlington NGS, Bruce NGS, Fermi 2 NPP, and for a Transborder Nuclear Emergency).

First Nation and Indigenous communities that are, or have traditional territories, in close proximity to other types of CNSC-licensed facilities should include in their emergency response plans, the measures they may need to take to respond to a radiological emergency. This would include details on the relevant notifications to and from the involved organizations.

B.2.8 International Agreements and Arrangements

B.2.8.1 Northern Emergency Management Assistance Compact

Ontario participates in several agreements for the purposes of emergency response mutual assistance (e.g., Northern Emergency Management Assistance Compact (NEMAC) Agreement, a memorandum of agreement, entered into by the Central and Prairie regions of the United States with select Canadian provinces (including Ontario) intended to facilitate cross border emergency management assistance through mutual aid) as well as more specific nuclear notification agreements. The latter are detailed in the PNERP Implementing Plan for the Fermi 2 Nuclear Power Plant and Implementing Plan for a Transborder Nuclear Emergency.

B.2.8.2 Statement of Intent

Health Canada and the US Department of Energy developed a statement of intent supporting joint Canada-U.S. nuclear emergency preparedness and response capabilities with the objective to identify areas where coordination and cooperation, including information sharing and mutual assistance would be beneficial to nuclear emergency management programs and capabilities.

B.2.8.3 IAEA's Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency

Canada is a signatory of the IAEA's Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency, which sets out an international framework for co-operation among countries and with the IAEA to facilitate prompt assistance and support in the event of nuclear accidents or radiological emergencies. Requests to the IAEA for international assistance would be undertaken by Health Canada, as the Competent Authority-Abroad, and Global Affairs Canada.

ANNEX C. PROVINCIAL LEGISLATIVE AUTHORITY

C.1 PROVINCIAL ROLES AND RESPONSIBILITIES

The provincial government has jurisdiction over public health and safety, property, and the environment within its borders. In the event of a radiological or nuclear emergency, the province is primarily responsible for mitigating the offsite consequences of the emergency, by conducting the offsite response to those emergencies as detailed in the PNERP. This annex is supplemental to provincial legislation found in Section 3.1.

C.2 EMERGENCY MANAGEMENT AND CIVIL PROTECTION ACT R.S.O 1990, CHAPTER E.9

The legislative authority for emergency management (including planning and response) in Ontario is the Emergency Management and Civil Protection Act (EMCPA).

One of the actions available to the province under the EMCPA is a provincial declaration of emergency. A provincial declaration of emergency can be made by the Lieutenant Governor in Council (LGIC) or the Premier for an emergency that exists throughout Ontario or in any part of Ontario. The provincial declaration of emergency allows the province to immediately make emergency orders to address the emergency quickly and effectively.

C.2.1 Emergency Orders

Once a provincial declaration of emergency has been made, the LGIC has authority to make emergency orders and may delegate these powers to a Minister or to the CEM. All emergency orders are subject to the Canadian Charter of Rights and Freedoms.

A Minister to whom powers have been delegated may in turn delegate any of their powers to the Commissioner of Emergency Management (CEM).

In accordance with the EMCPA (s. 7.0.2(2)):
During a declared provincial emergency, the Lieutenant Governor in Council may make orders that the Lieutenant Governor in Council believes are necessary and essential in the circumstances to prevent, reduce or mitigate serious harm to persons or substantial damage to property, if in the opinion of the Lieutenant Governor in Council it is reasonable to believe that,

(a) the harm or damage will be alleviated by an order; and
(b) making an order is a reasonable alternative to other measures that might be taken to address the emergency.

2006, c. 13, s. 1 (4).”. s. 7.0.2(3)(1) goes on to say that actions authorized by emergency orders "shall be exercised in a manner which, consistent with the objectives of the order, limits their intrusiveness."

C.2.2 Reporting Requirements

Pursuant to Section 7.0.6 of the EMCPA, during a provincial emergency, the Premier or a Minister as delegated is required to regularly report to the public with respect to the emergency. Pursuant to subsection 7.0.10(1) of the EMCPA, the Premier is required to submit a report in respect of the declared emergency to the Assembly within 120 days following the termination of the declared emergency. If the Assembly is not in session at that time, the Premier is required to submit a report within seven (7) days of the Assembly reconvening.

C.2.3 Liability for Action

Pursuant to subsection 11(1) of the EMCPA, ministers of the Crown, Crown employees, members of municipal councils, municipal employees, public servants, or any other individual acting pursuant to this Act or an order made under this Act for any act are protected from personal liability for doing any act done in good faith under the Act or pursuant to an Order made under the Act.

C.2.4 Authority

The responsibilities, functions, and tasks outlined in the PNERP, and its implementing plans shall carry the following authority in accordance with the EMCPA Section 9:

An emergency plan formulated under section 3, 6, 6.0.1 or 8 shall,

a. in the case of a municipality, authorize employees of the municipality or, in the case of a plan formulated under section 6 or 8, authorize public servants to take action under the emergency plan where an emergency exists but has not yet been declared to exist;
b. specify procedures to be taken for the safety or evacuation of persons in an emergency area;
c. in the case of a municipality, designate one or more members of council who may exercise the powers and perform the duties of the head of council under this Act or the emergency plan during the absence of the head of council or during his or her inability to act;
d. establish committees and designate employees to be responsible for reviewing the emergency plan, training employees in their functions and implementing the emergency plan during an emergency;
e. provide for obtaining and distributing materials, equipment and supplies during an emergency;
e. provide for any other matter required by the standards for emergency plans set under section 14; and
f. provide for such other matters as are considered necessary or advisable for the implementation of the emergency plan during an emergency. R.S.O. 1990, c. E.9, s. 9; 2002, c. 14, s. 13; 2006, c. 35, Sched. C, s. 32 (5); 2022, c. 11, Sched. 1, s. 7.

C.3 ONTARIO DRINKING WATER QUALITY STANDARDS (O. REG. 169/03)

Provincial standards for water quality are set out in the Ontario Drinking Water Quality Standards (O. Reg. 169/03), a regulation under the Safe Drinking Water Act, 2002, S.O. 2002, c. 32.

C.4 OCCUPATIONAL HEALTH AND SAFETY ACT, R.S.O. 1990, C. O.1

In the event of a conflict between an Emergency Order made under the EMCPA and the Occupational Health and Safety Act or a regulation made under it, OHSA or the regulation made under it prevails (subsection 7.2 (8) of the EMCPA).

C.5 HEALTH PROTECTION AND PROMOTION ACT, R.S.O. 1990, C. H.7

The Health Protection and Promotion Act (HPPA) includes provisions for the Minister of Health, Chief Medical Officer of Health, and local Medical Officers of Health to issue orders and directives. It further authorizes the Minister to publish public health standards for the provision of mandatory health programs and services, and every board of health shall comply with them.

C.6 RESPONSIBILITIES

Where the PNERP assigns responsibilities to the province, those responsibilities are to be carried out by the Ontario provincial government as a whole, as coordinated by Emergency Management Ontario (EMO) through the Provincial Emergency Operations Centre (PEOC). The CEM is responsible for leading response and recovery coordination.

The responsibilities, functions, and tasks outlined in the PNERP, and its implementing plans shall carry the following authorities:

a. In the case of those assigned to an organization, it should be the responsibility of the operational or administrative head of the organization to ensure their implementation.
b. In the case of those assigned to a position, implementation shall also be the responsibility of any substitute, alternate, or the person next in line of authority if the permanent incumbent of that position is absent or otherwise unable to take the necessary action.

ANNEX D. FEDERAL LEGISLATIVE AUTHORITY

D.1 FEDERAL ROLES AND RESPONSIBILITIES

The federal government maintains exclusive jurisdiction over the regulation of nuclear energy in Canada.

The province has exclusive jurisdiction for matters of property and civil rights in Ontario and has jurisdiction over matters that affect the public health, safety, and environment of Ontario.

D.2 EMERGENCIES ACT, R.S.C. 1985, C.22 (4TH SUPP.)

Pursuant to Section 6 of the Emergencies Act, the federal Governor in Council may declare a public welfare emergency, which includes an emergency caused by a real or imminent accident or pollution that results in or may result in a danger to life or property, social disruption or a breakdown in the flow of essential goods, services or resources, so serious as to be a national emergency.

Pursuant to Section 14 of the Emergencies Act, before the Governor in Council issues, continues or amends a declaration of a public welfare emergency, the Lieutenant Governor in Council of each province in which the direct effects of the emergency occur shall be consulted with respect to the proposed action. However, where the direct effects of the emergency are confined to, or occur principally in one province, the Governor in Council may only issue a declaration of public welfare emergency when the "Lieutenant Governor in Council of the province" has indicated to the federal Governor in Council that the emergency exceeds the capacity or the authority of the province to deal with it.

Pursuant to Section 8 of the Emergencies Act, while a declaration of a public welfare emergency is in effect, the Governor in Council may make orders or regulations "that the Governor in Council believes, on reasonable grounds, are necessary for dealing with the emergency."

D.3 EMERGENCY MANAGEMENT ACT, S.C. 2007, C.15

The Emergency Management Act provides for the coordination of emergency management activities including the development and implementation of emergency management plans. Federal authorities also coordinate or support the provision of assistance to a province in respect of a provincial emergency. Assistance could include financial assistance, including where the emergency has been declared to be of concern to the federal government and the province has requested assistance.

D.4 NUCLEAR SAFETY AND CONTROL ACT, R.S.C. 1997, C.9

The NSCA establishes the CNSC, which is a federal entity responsible for matters including the regulation of certain activities related to nuclear energy, including the construction and operation of reactor facilities, and response to emergencies with radiological or nuclear aspects.

The CNSC regulates all activities, actions and decisions regarding possession, handling, transport, or storage of licensed radioactive material.

The CNSC is authorized to make any order in an emergency that it considers necessary to protect the environment or the health and safety of persons or to maintain national security and compliance with Canada's international obligations (see Section 47 (1) of the NSCA).

D.5 NUCLEAR SAFETY AND CONTROL ACT, CLASS I NUCLEAR FACILITIES REGULATIONS (SOR/2000-204)

Under the Nuclear Safety and Control Act, Class I Nuclear Facilities Regulations, licensed nuclear facilities (e.g., reactor facilities) face various requirements, including being required to demonstrate proposed measures to prevent or mitigate the effects of accidental releases of nuclear substances and hazardous substances on the environment, the health and safety of persons and the maintenance of national security, including measures to:

a. assist offsite authorities in planning and preparing to limit effects of an accidental release.
b. notify offsite authorities of an accidental release or the imminence of an accidental release.
c. report information to offsite authorities during and after an accidental release.
d. assist offsite authorities in dealing with the effects of an accidental release.
e. test the implementation of the measures to prevent or mitigate the effects of an accidental release.

D.6 NUCLEAR LIABILITY AND COMPENSATION ACT S.C. 2015, C.4, S.120

The federal Nuclear Liability and Compensation Act S.C. 2015, c.4, s.120 (NLCA), administered by Natural Resources Canada (NRCan), governs civil liability and compensation for damage in Canada in case of a nuclear incident. This legislation provides for various matters, including the following:

a. Operators of nuclear power plants designated under the NLCA are liable for up to one billion dollars for damage resulting from a nuclear incident and must have

insurance from an insurer approved under the NLCA with terms and conditions approved by the federal Minister of Natural Resources

b. Compensable damage includes bodily injury or death, property damage, psychological trauma of a person who suffered bodily injury caused by a nuclear incident, certain environmental damage, economic loss incurred as a result of a person's bodily injury or damage to their property and that is caused by a nuclear incident, or psychological trauma that results from that bodily injury, certain lost wages, and certain costs related to preventive measures. Actions or claims in relation to bodily injury or death must be brought within 30 years, while any other actions or claims must be brought within 10 years.

Liaison between the federal government, the Nuclear Insurance Association of Canada (NIAC) and the Ontario Ministry of Municipal Affairs and Housing (MMAH) shall be undertaken to ensure that any financial assistance provided by the province does not duplicate assistance provided under the federal legislation.

D.7 TRANSPORTATION OF DANGEROUS GOODS ACT (S.C.1992, C.34)

The Transportation of Dangerous Goods Act (S.C.1992, C.34), administered by Transport Canada, governs the transportation of dangerous goods (including nuclear substances, within the meaning of the NSCA, that are radioactive). It provides for various matters, including:

a. certain safety and security requirements and standards.
b. requirements for emergency response assistance plans, which outline what is to be done to respond to an actual or anticipated release of dangerous goods in the course of their handling or transporting that endangers, or could endanger, public safety.

ANNEX E. FEDERAL DEPARTMENT RESPONSIBILITES

E.1 OVERVIEW

Under the Emergency Management Act, the Minister of Public Safety is responsible for coordinating the Government of Canada's response to an emergency. The Federal Emergency Response Plan (FERP) is the Government of Canada's all-hazards response plan. The FERP outlines the processes and mechanisms to facilitate an integrated Government of Canada o response to an emergency.

The federal response to emergencies is housed within the Government Operations Centre (GOC), which is managed by Public Safety Canada (PS) and is the single point of contact in emergencies. The GOC supports provincial and local authorities, and coordinates with federal government departments, non-governmental organizations, the private sector, and allied governments.

The National Emergency Response System (NERS) forms the basis for a federal response. The NERS:

a. Provides for harmonization of joint federal, provincial, and territorial response to emergencies.
b. Supports and facilitates procurement and logistics coordination between all levels of government, the private sector, non-governmental organizations, and international consulting partners.
c. Applies to federal support at the request of a province or territory and can also be used when provinces or territories support federal response to an emergency under federal jurisdiction.

Health Canada administers the Federal Nuclear Emergency Plan (FNEP), which can be activated when:

a. A nuclear emergency requires coordination between several Canadian provinces or territories.
b. Federal response activities for multi-departmental offsite response must be managed and coordinated for a radiological or nuclear emergency.

The FNEP is an annex to the FERP and describes the roles, responsibilities, and capabilities of the federal government to support the scientific and technical response to radiological or nuclear emergencies.

The FNEP contains provincial annexes which outline the provision of scientific and technical support from the federal government to provinces during a nuclear emergency.

The FNEP's Ontario Annex functions as a pre-arrangement between the province of Ontario and the federal government for such support, under Section 6.3 of the

Emergency Management Act. This arrangement alleviates the requirement for certain requests for response assistance. However, requests for support beyond those specified in the FNEP Ontario Annex must be made pursuant to the normal request for federal assistance process.

In the event of a radiological or nuclear emergency, the CNSC monitors and evaluates the onsite response of the licensee. In the case of an event with no identified licensee, the CNSC, in compliance with the NSCA and Regulations:

a. Oversee and regulate the response activities of the licensees responding to the event.
b. Ensure the health, safety, and security of licensee, the response staff, the public and the environment.
c. Ensure compliance with Canada's international obligations.

In either case, the CNSC implements their CNSC Nuclear Emergency Response Plan.

In accordance with the FNEP and with the NERS, in the event of a nuclear emergency, the federal government (through PS) will liaise with the provinces and territories as well as with neighbouring countries and the international community.

NRCan administers the NLCA, which governs the compensation regime in the event of a nuclear emergency.

The federal government further coordinates Canada's response (through Global Affairs Canada) should Canadians be affected by a nuclear emergency that occurs in a foreign country.

The following departments have agreed to assume the identified responsibilities within the PNERP.

E.2 HEALTH CANADA

The following sections detail the responsibilities for Health Canada within the PNERP.

E.2.1 Planning and Preparedness

Under the PNERP, Health Canada will:

a. Execute the applicable preparedness responsibilities described in the Provincial Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency.
b. Ensure that federal and provincial nuclear emergency management activities are supported and coordinated in conjunction with the PNERP and FNEP and its Ontario Annex

c. Work with the province and other supporting organizations to put in place appropriate safety measures to protect the public and emergency workers from immediate and delayed health effects that may result from a radiological or nuclear event, and to mitigate the impacts of such an event on property and the environment.
d. On an annual basis, provide the province with most current data from their Fixed Point and other radiological monitoring networks.
e. Assist the Ministry of Labour, Immigration, Training and Skills Development (MLITSD) in maintaining an environmental radiation database.
f. Develop departmental procedures for carrying out the collection, delivery and radio- analysis of samples, and other assigned actions, as required under the Environmental Radiation and Assurance Monitoring Group (ERAMG) Plan
g. Ensure that HC / FNEP Technical Assessment Group (TAG) staff working in the PEOC have an overall knowledge of their emergency plans and PNERP.
h. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

E.2.2 Provision of Personnel

Under the PNERP, Health Canada will:

a. Provide suitable Health Canada personnel for staffing various positions in the province's Emergency Response Organization (ERO) including the PEOC Operations Section and qualified personnel for the ERAMG-PEOC Team and ERAMG-Field Team
b. Provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

E.2.3 Response

Under the PNERP, Health Canada will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. In the event of a nuclear emergency requiring a coordinated Government of Canada response, the Minister of Emergency Preparedness Canada shall be responsible for overall federal coordination unless otherwise specified. The federal technical support and coordination would be led by Health Canada as per the FNEP and the Ontario Annex
c. During a radiological or nuclear emergency, ensure coordination and communication between the Scientific Section and the FNEP TAG, including the provision of offsite assessments to support situational awareness.

d. The FNEP TAG members of the PEOC shall provide scientific and technical information and advice to the PEOC Scientific Section to assist in performing its functions.
e. Coordinate the release and content of emergency information for public release with the PEOC Public Information Management Section
f. As lead of the FNEP TAG, provide offsite atmospheric dispersion modelling and dose assessments to the PEOC Scientific Section to support protective action decision-making.

E.2.4 Training and Exercises

Under the PNERP, Health Canada will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of HC staff.
b. Participate in training and exercises held by the province according to a mutually agreed schedule.
c. Arrange appropriate training of FNEP TAG staff involved in the Scientific Section of the PEOC.

E.2.5 Study and Research

Under the PNERP, Health Canada will:

a. Make available to the province the results of studies and research, which affect offsite safety. Within available resources, participate in such studies and research undertaken by the province.
b. Provide the province with, or assist the province in obtaining information, studies, and research, having a bearing on offsite safety, from international agencies, other countries, and other provinces of Canada.

E.3 CANADIAN NUCLEAR SAFETY COMMISSION

The following responsibilities have been agreed to pursuant to the Memorandum of Understanding (MOU) maintained between CNSC and EMO, in accordance with their respective mandates under the federal NSCA and the provincial EMCPA with respect to nuclear safety regulation, public safety, and protection of the environment in the province of Ontario.

The CNSC adheres to the following principles regarding nuclear emergency management:

a. The top priorities in managing a nuclear emergency are health, safety, security, and the environment.

b. Nuclear emergencies are managed in accordance with Section 9 of the NSCA, which outlines the mandate of the Commission.
c. A risk-informed approach is used.
d. Activities are coordinated with relevant consulting partners.

E.3.1 Planning and Preparedness

Under the PNERP, CNSC will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency.
b. The CNSC maintains a Nuclear Emergency Response Plan, along with supporting procedures and guidelines, to address the CNSC's response to a nuclear emergency.
c. Pursuant to MOU article 3, the CNSC and EMO:

    i. Consult on the development and implementation of Commission regulations as they affect nuclear safety, regulation, public safety, and the protection of the environment within Ontario as well as nuclear events and issues occurring outside, particularly as they affect Class I nuclear facilities.
    ii. Consult on the evolution and implementation of the PNERP as well as the Commission's Nuclear Emergency Plan
    iii. Facilitate contact with other federal, provincial, municipal, or non-governmental organizations.
    iv. Facilitate contact with foreign governments and international organizations recognizing that EMO maintains contact with contiguous U.S. states regarding nuclear emergency preparedness and response.
    v. Pursuant to MOU article 4, provide EMO with annual reports of radiological environmental monitoring.
    vi. CNSC staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP.
    vii. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

E.3.2 Provision of Personnel

Under the PNERP, CNSC will:

a. Pursuant to MOU article 7:
    i. Provide staff for the PEOC Operations Section to liaise with the CNSC's Emergency Operations Centre (EOC) and technical personnel to work in the PEOC scientific section during an emergency.

ii. Exchange current contact numbers for duty personnel and operations centres.

b. Provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

E.3.3 Response

Under the PNERP, CNSC will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. Maintain the capability to receive notice of actual or potential nuclear emergencies and ensures that its Nuclear Emergency Response Plan can be activated at any time.
c. Pursuant to MOU article 7, provide EMO with timely notification of actual or potential nuclear, radiological, and other related emergencies.
d. The CNSC's EOC is activated in the event of a nuclear emergency.
e. The CNSC activates its EOC and assembles staff who have a role to play in order to provide assurance that appropriate actions are taken by the licensee and response organizations, to limit the risk to health, safety, security of the public and the environment.
f. For nuclear emergencies involving licensed facilities and substances, the CNSC:

    i. Performs regulatory oversight of the licensee's activities (monitoring the response actions, evaluation of protective action recommendations, and, when appropriate, direction in the form of directives and orders)
    ii. Performs an independent assessment of the Onsite conditions and potential offsite consequences, to provide or confirm the licensee's recommendations concerning any protective measures that may be needed.

g. The CNSC plays a supporting role to the response under the FNEP. This includes (but is not limited to) providing technical assistance and support to the lead organization, in accordance with CNSC's authorities and responsibilities.
h. The CNSC provides support to the whole-of-government response for nuclear emergencies involving non-licensees, such as foreign emergencies and malevolent acts.
i. During a nuclear emergency, the CNSC co-ordinates its activities with consulting partners and, as appropriate, shares information to enable informed and efficient decision-making. Where necessary, the Nuclear Emergency Organization (NEO) advises the CNSC in making timely regulatory decisions related to the emergency.
j. Coordinate the release and content of emergency public information for public release with the PEOC Public Information Management Section.

E.3.4 Training and Exercises

Under the PNERP, CNSC will:

a. The CNSC may participate in training and exercises held by the province.
b. Pursuant to MOU article 6, the CNSC and EMO:

    i. Exchange information on training, drill, and exercise schedules at least once per year.
    ii. Identify mutual training opportunities.
    iii. Support each other during exercises.
    iv. Arrange appropriate training of CNSC staff involved in the scientific section of the PEOC.

E.3.5 Study and Research

Under the PNERP, CNSC will:

a. Make available to the province the results of studies and research, which affect offsite safety. Within available resources, participate in such studies and research undertaken by the province, if invited to do so by the province
b. Provide the province with, or assist the province in obtaining information, studies and research having bearing on offsite safety, from international agencies, other countries, and other provinces of Canada
c. Share studies published by CNSC staff once published.

E.4 PUBLIC SAFETY CANADA

The following sections detail the responsibilities for Public Safety Canada (PS) within the PNERP.

E.4.1 Planning and Preparedness

Under the PNERP, PS will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency.
b. Confirm that federal and provincial nuclear emergency management activities are supported and coordinated in conjunction with the PNERP and FNEP
c. Confirm that threats, alerts, and advisories are issued to the PEOC to communicate information about potential, imminent, or actual threats that endanger public health and safety.
d. Public Safety Canada staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP.

e. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

E.4.2 Provision of Personnel

Under the PNERP, PS will:

a. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.
b. Provide suitable Public Safety personnel for staffing various positions in the province's emergency management organization including the PEOC Operations Section.

E.4.3 Response

Under the PNERP, PS will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. Confirm that the GOC is notified, to facilitate the coordination of federal activities relating to areas of federal jurisdiction. This may include liaising with any other potentially affected province, with United States, any other country, and international agencies.
c. Notify the province, as soon as possible, of any report received of an occurrence, which has resulted, or has the potential to result, in the receipt by any person offsite (in Ontario) of a dose of ionizing radiation in excess of prescribed regulatory limits.
d. Coordinate the release and content of emergency public information for public release with the PEOC Public Information Management Section.

E.4.4 Training and Exercises

Under the PNERP, PS will participate in training and exercises held by the province.

E.4.5 Study and Research

Under the PNERP, PS will provide the province with, or assist the province in obtaining information, studies and research, having a bearing on offsite safety, from international agencies, other countries, and other provinces of Canada.

E.5 CANADIAN FOOD INSPECTION AGENCY REGIONAL OFFICE

The following sections detail the responsibilities the Canadian Food Inspection Agency (CFIA) Ontario Regional Office has agreed to within the PNERP.

E.5.1 Planning and Preparedness

Under the PNERP, CFIA will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency.
b. Assist in the development and implementation of a public awareness and education program for farmers, and federally regulated food processing facilities and abattoirs prior to a nuclear emergency.
c. Prepare plans, federally regulated food processing facilities and abattoirs together with the Ontario Ministry of Agriculture, Food, and Agribusiness (OMAFA), to provide information and advice to farmers, and federally regulated food processing facilities in the DPZ of each reactor facility for the initial stages of an emergency. This includes the preparation of advisories covering different situations.
d. Participate in the preparation of plans and procedures for Environmental Radiation and Assurance Monitoring for nuclear emergencies.
e. Plan and prepare ingestion control measures in the DPZ of each reactor facility as a protective measure to minimize the radiation hazard.
f. Maintain an information database relating to federally regulated food processing facilities and abattoirs for use in nuclear emergency planning and management. The nuclear data shall be organized to allow access to information for the DPZ and each sub zone of the IPZ around each reactor facility.
g. Departmental staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP.
h. Assist the Ministry of Labour, Immigration, Training and Skills Development (MLITSD) in maintaining an environmental radiation database.
i. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

E.5.2 Provision of Personnel

Under the PNERP, CFIA will:

a. Provide suitable personnel for staffing various positions in the province’s emergency management organization, including qualified personnel for the ERAMG-PEOC Team and ERAMG-Field Team

b. Provide a suitable representative to participate on inter-organizational NEMCC set up under the PNERP to ensure alignment and address inter-organizational issues.

E.5.3 Response

Under the PNERP, CFIA will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. Act as the main federal department through which food ingestion control operations shall be conducted by the PEOC at the first practicable occasion during a nuclear emergency.
c. Execute a sampling program and other assigned actions required under the ERAMG Operations Manual at the first practicable occasion during a nuclear emergency.
d. Liaise with Agriculture and Agri-Food Canada and OMAFA at the first practicable occasion during a nuclear emergency to assist in securing agricultural commodities in affected areas.
e. Provide advice and guidance to farmers and federally regulated food processing facilities and abattoirs on the humane transport of animals during evacuations or other displacements.
f. Provide advice related to the import and export of potentially contaminated food products and bottled water in the affected zone, assist with the facilitation of food embargoes through appropriate channels.
g. Channel emergency public information for public release through the PEOC Public Information Management Section.

E.5.4 Training and Exercises

Under the PNERP, CFIA will execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of departmental staff.

E.5.5 Study and Research

Under the PNERP, CFIA will participate in studies and inter-comparisons in the area of environmental radiation monitoring and analysis of the impact of radiation on the food chain.

ANNEX F. PROVINCIAL MINISTRY RESPONSIBILITES

F.1 MINISTRY OF THE ATTORNEY GENERAL

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.1.1 Planning and Preparedness

In support of the PNERP, the Ministry of the Attorney General will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a radiological or nuclear emergency
b. Ensure that the administration of Ontario's public affairs is in accordance with the law
c. Superintend all government legislative matters
d. Ensure the administration of the courts in partnership with the constitutionally independent judiciary and superintend all matters connected with judicial offices
e. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
f. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.1.2 Provision of Personnel

In support of the PNERP, the Ministry of the Attorney General will:

a. Support the Attorney General's mandate and role as Chief Law Officer of the Crown.
b. Conduct all provincial Crown prosecutions.
c. Conduct and regulate all litigation for and against the Crown or any provincial ministry or agency of government in respect of any subject within the authority or jurisdiction of the Legislature
d. Provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.1.3 Response

In support of the PNERP, the Ministry of the Attorney General will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency

b. Advise government upon all matters of law referred to it, including the constitutionality and legality of emergency response issues
c. Represent the personal and property rights and obligations of children in the civil justice system
d. Provide court-based assistance services to the most vulnerable victims and witnesses of crime
e. Provide guardianship services to vulnerable and incapable adults
f. Provide legislative drafting services to ministers of the Crown, members of the legislature and applicants for private bills and drafting services for regulations.
g. Co-ordinate the response to legal issues that arise.
h. Channel emergency information for public release through the PEOC Public Information Management Section.

F.1.4 Training and Exercises

In support of the PNERP, Ministry of the Attorney General will execute the training and exercise responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.

F.2 MINISTRY OF AGRICULTURE, FOOD AND AGRIBUSINESS

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.2.1 Planning and Preparedness

In support of the PNERP, OMAFA will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Assist in the development and implementation of a public awareness and education program for farmers and food processors prior to a nuclear emergency
c. Together with the CFIA, prepare information and advice for farmers and food processors in the DPZ of each reactor facility for the initial stages of an emergency, including the preparation of advisories covering different situations
d. Prepare and plan ingestion control measures in the DPZ of each reactor facility to minimize the radiation hazard, including arrangements for clearing DPZ dairy farm milk storages at the first practicable opportunity during a nuclear emergency
e. Participate in the preparation of plans and procedures for Environmental Radiation and Assurance Monitoring for nuclear or radiological emergencies.

f. Prepare operating procedures for the Ministry Action Group and making necessary organizational and administrative arrangements to enable it to execute its functions
g. Maintain an information database relating to agricultural and food facilities, producers, marketing organizations, etc. for use in nuclear emergency planning and management. The nuclear data shall be organized to allow access to information for the DPZ and each sub zone of the IPZ around each reactor facility
h. Ministry staff working in the PEOC and on field teams should have an overall knowledge of their emergency plans and the PNERP
i. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups
j. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

F.2.2 Provision of Personnel

In support of the PNERP, OMAFA will:

a. Provide suitable ministry personnel for staffing various positions in the provincial ERO, including the Operations section and qualified personnel for the ERAMG-PEOC Team and ERAMG-Field Team
b. Assist the Ministry of Labour, Immigration, Training and Skills Development (MLITSD) in maintaining an environmental radiation database
c. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees and working groups, set up under the PNERP to ensure alignment and address inter-organizational issues.

F.2.3 Response

In support of the PNERP, OMAFA will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Act as the main ministry through which food (excluding water) ingestion control operations shall be conducted by the PEOC at the first practicable occasion during a nuclear emergency
c. Establish a Ministry Action Group to direct and co-ordinate ministry actions under the direction of the PEOC at the first practicable occasion during a nuclear emergency
d. Execute a sampling program and other assigned actions required under the ERAMG Operations Manual at the first practicable occasion during a nuclear emergency

e. Liaise with Agriculture and Agri-Food Canada and CFIA at the first practicable occasion during a nuclear emergency to assist in securing agricultural commodities, such as animal feed for affected areas
f. Channel emergency information for public release through the PEOC Public Information Management Section.

F.2.4 Training and Exercises

In support of the PNERP, OMAFA will:

a. Execute the training and exercise responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff
b. Arrange appropriate training of OMAFA staff involved in the Scientific Section of the PEOC.

F.3 MINISTRY OF CHILDREN, COMMUNITY AND SOCIAL SERVICES

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.3.1 Planning and Preparedness

In support of the PNERP, MCCSS will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Ensure that any 24/7 institutions operated by the ministry and located within the Detailed Planning Zone of designated reactor facilities have plans for implementing the various protective measures
c. Liaise with non-governmental emergency social service organizations on their role in a nuclear emergency
d. Prepare operating procedures for the Ministry Action Group and make the necessary organizational and administrative arrangements for the establishment of the Ministry Action Group to enable it to execute its functions
e. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
f. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups
g. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

F.3.2 Provision of Personnel

In support of the PNERP, MCCSS will:

a. Provide ministry representatives upon request, to assist in relevant operations in the PEOC
b. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.3.3 Response

In support of the PNERP, MCCSS will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Activate the Ministry Emergency Operations Centre (MEOC) and its associated processes, including activating the Ministry Action Group to direct and co-ordinate provincial ministry action, and engaging in the appropriate notification and communication protocols
c. Liaise, as required, between EMO and non-governmental emergency social service organizations, including the Canadian Red Cross Society, in the delivery of emergency social services
d. Assist affected municipalities in the delivery of emergency social services, including emergency shelter, food and clothing, registration and inquiry and personal services when their capacity is exceeded, and a provincial response is required
e. Channel emergency information for public release through the PEOC Public Information Management Section (PIMS).

F.3.4 Training and Exercises

In support of the PNERP, MCCSS shall execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.

F.4 MINISTRY OF ENERGY AND MINES

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.4.1 Planning and Preparedness

In support of the PNERP, ENERGY will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a radiological or nuclear emergency
b. Assist EMO in ensuring that the corporate head offices of designated reactor facilities execute their responsibilities under this program
c. Ensure that the Ontario Electricity Emergency Plan (OEEP) is consistent with the PNERP. This shall be done through the Independent Electricity Systems Operator, which shall co-ordinate the preparation and implementation of electricity emergency plans, to mitigate the impact of a nuclear emergency on the reliability of the bulk electricity system
d. Develop operating procedures for the Ministry Action Group aligned with the PEOC's notification protocol to ensure that ENERGY is kept fully informed of all aspects of the provincial response to a nuclear emergency, including the actions being taken by the operator of the reactor facility, the CNSC, EMO, and others to ensure public health and safety
e. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
f. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.4.2 Provision of Personnel

In support of the PNERP, ENERGY will:

a. Provide suitable ministry representatives to serve in the Operations Section of the PEOC. If requested, also provide suitable ministry personnel for staffing positions in the province's ERO
b. Provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups, set up under the PNERP to ensure alignment and address inter-organizational issues.

F.4.3 Response

In support of the PNERP, ENERGY will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Establish a Ministry Action Group to direct and co-ordinate ministry actions consistent with those of the EMO/PEOC, to provide advice to EMO/PEOC, and to keep the Minister of Energy and Mines informed of emergency public information on a regular and timely basis

c. Maintain liaison with the Independent Electricity Systems Operator as required to address matters affecting electricity supply
d. In a nuclear emergency, respond to EMO request for information specific to its (ENERGY's) responsibilities under Section 6 of the EMCPA.
e. Channel emergency information for public release through the PEOC PIMS.

F.4.4 Training and Exercises

In support of the PNERP, ENERGY will:

a. Participate in nuclear emergency training and exercises that simulate impacts on electricity supply
b. Ministry staff nominated to various elements of the emergency management organization shall participate in the associated training
c. Execute the training and exercise responsibilities described in the PNERP prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.

F.5 MINISTRY OF THE ENVIRONMENT, CONSERVATION AND PARKS

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.5.1 Planning and Preparedness

Under the PNERP, the MECP will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Prepare and maintain an Annex to the Ministry's Emergency Response Plan to guide ministry actions under PNERP including the activation of a Ministry Action Group to direct ministry response activities
c. Develop ministry procedures for carrying out the collection and delivery of samples, and other assigned actions, as required under the ERAMG Operations Manual
d. Assist the Ministry of Labour, Immigration, Training and Skills Development (MLITSD) in maintaining an environmental radiation database
e. Participate in the preparation of plans and procedures for the ERAMG for nuclear emergencies
f. Establish procedures to notify and evacuate provincial parks that lie within the DPZs of the designated reactor facilities
g. Establish procedures to notify conservation authorities that lie within the DPZs of the designated reactor facilities

h. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
i. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups
j. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

F.5.2 Provision of Personnel

Under the PNERP, the MECP will:

a. Provide suitable ministry personnel for staffing various positions in the province's emergency management organization, including the Operations Section and qualified personnel for the ERAMG-PEOC Team and ERAMG-Field Team
b. Provide suitable ministry personnel for staffing the waste management planning working group, if requested by the CEM or designate (e.g. PEOC Commander)
c. Provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.5.3 Response

Under the PNERP, the MECP will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Act with the MOH as the main ministry through which water control measures shall be conducted by EMO/PEOC at the first practicable occasion during a nuclear emergency
c. Activate a Ministry Action Group when the PNERP is activated. Under overall direction from the PEOC, the Ministry Action Group shall direct and co-ordinate the ministry's response to ensure the ministry is able to:

    i. Provide hydrological support to the PEOC
    ii. Identify municipal and non-municipal drinking water systems regulated under O. Reg. 170/03 (Drinking Water Systems), a regulation under the Safe Water Drinking Act, by sub-zones in any affected zone as required
    iii. Provide support, either directly or through the PEOC, to local Medical Officers of Health regarding the implementation of drinking water precautionary and protective measures
    iv. Support drinking water systems as needed during the recovery phase

d. The MECP Ministry Action Group shall also keep the Minister of Environment, Conservation and Parks informed of actions taken

e. Carry out the sampling program and other assigned activities required under the ERAMG Operations Manual and as directed by the ERAMG
f. Channel emergency information for public release through the PEOC PIMS.
g. Carry out precautionary actions as detailed in the PNERP when so ordered by the CEM or designate (e.g. PEOC Commander)
h. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

F.5.4 Training and Exercises

Under the PNERP, the MECP will execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.

F.5.5 Study and Research

Under the PNERP, the MECP will assist in studies and research as appropriate to nuclear emergencies, especially in the fields of meteorological and radionuclide dispersion forecasting.

F.6 MINISTRY OF HEALTH

The Minister, together with the agencies, boards, and commissions operating to assist the MOH, have the following responsibilities for leading and coordinating the health system preparation for, and response to nuclear emergencies consistent with provincial statutes and the responsibilities assigned by Order in Council, OIC 1739/2022, under Section 6 of the EMCPA.

These activities are detailed in the MOH Emergency Response Plan and the RHRP.

F.6.1 Planning and Preparedness

Under the PNERP, the MOH will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Lead the preparations necessary to ensure the effective implementation of the RHRP
c. Participate in provincial nuclear emergency response planning as led by EMO and in coordination with provincial ministries, designated municipalities, federal authorities, facility and installation operators, and other consulting partners
d. Assist the Ministry of Labour, Immigration, Training and Skills Development (MLITSD) in maintaining an environmental radiation database

e. Participate in the preparation of plans and procedures for PEOC Sections, including the ERAMG
f. Participate in the development and maintenance of a Provincial Emergency Information Plan (PEIP) and associated procedures to ensure the co-ordination of emergency information among provincial ministries and in coordination with other consulting partners
g. Develop and maintain the RHRP per the PNERP and ministry mandate as defined by provincial statute, and in coordination with health system partners
h. Promote awareness and understanding of the RHRP to health consulting partners and municipalities
i. Brief ministry executives and program areas on the ministry emergency management program, including the RHRP.
j. Monitor, co-ordinate, and assist in the development and maintenance of nuclear emergency response plans and programs of specified partners and consulting partners (e.g., local medical officers of health, designated municipalities, designated hospitals, paramedic services, etc.), including arrangements for implementing precautionary and protective measures for health institutions and the public.
k. Ensure arrangements in the Ministry's Emergency Response Plan (MERP), Continuity of Operations Plan (COOP), and MEOC procedures can be implemented to lead the ministry response and coordinate the health system response to a radiological or nuclear emergency
l. Provide guidance and advice to health consulting partners and local organizations on matters related to iodine thyroid blocking (ITB) with stable potassium iodine (KI) (detailed in the RHRP's Potassium Iodide Guidelines Annex)
m. Procure, in advance, adequate quantities of KI pills for use by local authorities of:

    i. The Fermi 2 DPZ and IPZ
    ii. Populations in the IPZs as identified in the Transborder Emergencies Implementing Plan

n. Provide support to local authorities for all reactor facilities' IPZ populations to ensure that the ITB related requirements are completed
o. Liaise with EMO to develop a strategy to distribute KI pills to IPZ populations, as required, during a nuclear emergency
p. Provide guidelines for handling contaminated persons as well as those exposed to high levels of radiation as outlined in the RHRP
q. Develop guidelines for population monitoring and the creation of a dose registry, should they be required
r. Provision of public awareness and education related to radiation health per the PNERP and RHRP
s. Assist in the development and implementation of a public awareness and education program, and coordinate MOH activities within it
t. Assist in studies and research related to radiation health and the psychosocial aspects of nuclear emergencies

u. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.6.2 Provision of Personnel

Under the PNERP, the MOH will:

a. Provide suitable ministry personnel for staffing various positions in the province's emergency management organization, including the PEOC Operations and Scientific Sections
b. Ensure ministry staff working in the PEOC have an overall knowledge of their emergency plans (including the RHRP) and the PNERP
c. Provide a suitable representative to participate on inter-organizational NEMCC, sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.6.3 Response

Under the PNERP, the MOH will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Act as the lead ministry through which health and public health sector accountabilities shall be conducted during a nuclear emergency, in coordination with EMO
c. Ensure continued delivery of the ministry's mandate and support executive decision-making through provision of timely emergency information options, and advice
d. Activate the MEOC to direct and co-ordinate MOH and health system actions per executive direction and ministry plans (MERP, COOP, and RHRP), and consistent with the general direction of the PEOC
e. Complete the appropriate ministry and health system notification and communication protocols for a nuclear emergency
f. Provide advice and support in the development and issuing of emergency orders or draft and issue Chief Medical Officer of Health (CMOH) / Minister of Health directive (under other ministry statutes) for health-related emergency actions (i.e., CMOH orders for KI consumption, suspension of non-critical hospital admissions)
g. Support the province's CMOH in deciding, in coordination with the CEM and the local Medical Officers of Health, the need for and timing to direct ITB as a protective action
h. Provide advice, either directly or through the PEOC, to regional and local authorities regarding the implementation of the CMOH directive for ITB using KI and other precautionary and protective measures

i. Monitor and assist affected health system partners in the implementation of emergency response plan actions, as appropriate (e.g., transportation and treatment of causalities, DPZ health institution evacuations)
j. Liaise with and provide radiation health or public health advice to the Scientific Section/ Coordination and Command Section and other nuclear emergency management consulting partners
k. During operations, oversee or implement the required arrangements for follow-up medical monitoring and dose registry, as well as care and rehabilitation for those with radiation exposure
l. Develop health-related emergency information in coordination with the PEOC PIMS and other ministry emergency information functions
m. Support MOH spokespersons in the delivery of emergency information in coordination with the PEOC PIMS, Premier's Office, and Cabinet Office as required
n. Identify small drinking water systems regulated under O. Reg. 319/08 (Small Drinking Water Systems), a regulation under the Health Protection and Promotion Act, by sub-zones in any affected area as required, in collaboration with the MECP
o. Provide support, either directly or through the PEOC, to local Medical Officers of Health regarding the implementation of food and drinking water protective measures (i.e., ingestion control protective actions).

F.6.4 Training and Exercises

Under the PNERP, the MOH will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff
b. Provide guidance regarding training for health consulting partners and care settings as outlined in the RHRP
c. Arrange appropriate training of MOH staff involved in the PEOC Scientific Section.

F.7 MINISTRY OF LABOUR, IMMIGRATION, TRAINING AND SKILLS DEVELOPMENT

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.7.1 Planning and Preparedness

Under the PNERP, the MLITSD will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Prepare and maintain plans for completing the responsibilities of the ERAMG in the PNERP
c. Prepare and maintain the appropriate notification lists, operating procedures, and technical manuals for the groups indicated above
d. Operate the Ontario Reactor Surveillance Program (ORSP) to monitor the environment around nuclear generating stations in order to measure routine, unplanned, unannounced, and emergency emissions from nuclear plants and assess their potential impact on the public
e. Maintain and operate the provincial radio-analysis laboratory
f. Maintain an environmental radiation database
g. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
h. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.7.2 Provision of Personnel

Under the PNERP, the MLITSD will:

a. Provide suitable ministry personnel for staffing various positions in the province's emergency management organization including the Operations section and qualified personnel for the ERAMG-PEOC Team and ERAMG-Field Team
b. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.7.3 Response

Under the PNERP, the MLITSD will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Ensure that provincial employers meet their obligations under OHSA during a nuclear emergency
c. Provide or arrange the required radio-analysis to support the PNERP
d. Carry out the radiation monitoring activities required under the PNERP
e. If requested, provide technical assistance to the emergency response organization responding to a radiological or nuclear emergency not being mitigated under the PNERP
f. Operate the provincial radio-analysis laboratory to provide environmental sampling results following a release to the environment

g. Monitor radioactivity in the environment around all reactor facilities in Ontario and notify the PEOC of any abnormal (above ambient background) results
h. Establish a Ministry Action Group to direct and co-ordinate ministry actions under the direction of the CEM or designate (e.g. PEOC Commander) at the first practicable occasion during a nuclear emergency
i. Channel emergency information for public release through the PEOC PIMS
j. Perform dose management for provincial members of the ERAMG and ensure that field doses are accurately recorded and entered into a dose registry system.

F.7.4 Training and Exercises

Under the PNERP, the MLITSD will:

a. Execute the training and exercise responsibilities described in the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.
b. Provide administrative support for radiation monitoring during planning, training, or emergency operations.
c. Inspect Emergency Worker Centres for compliance with OHSA.
d. Inspect radiation safety training programs (for compliance with OHSA) provided to first responders and emergency workers, as appropriate.
e. Inspect designated nuclear response hospitals for compliance with the OHSA and regulations for healthcare and residential facilities with attention to worker protection and training under the RHRP.
f. Arrange appropriate training of MLITSD staff involved in the PEOC Scientific Section of the PEOC.

F.7.5 Infrastructure

Under the PNERP, the MLITSD will:

a. Provide or arrange laboratory facilities for radio analysis of samples of air, water, soil, herbage, milk, foodstuffs, etc.
b. Equip, maintain and operate an adequate network of fixed radiological monitoring sites in the Ontario portions of IPZs.

F.7.6 Study and Research

Under the PNERP, the MLITSD will participate in studies and inter-comparisons in the area of radiation monitoring and analysis.

F.8 MINISTRY OF MUNICIPAL AFFAIRS AND HOUSING

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.8.1 Planning and Preparedness

In support of the PNERP, the MMAH will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a radiological or nuclear emergency.
b. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.
c. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the Public Education and Awareness Strategy.

F.8.2 Provision of Extraordinary Financial Assistance

In support of the PNERP, the MMAH will:

a. Throughout the emergency and recovery period, the ministry shall co-ordinate extraordinary provincial expenditures associated with the province's response to the emergency
b. If directed by the LGIC, the MMAH shall:

    i. Lead development and implementation of disaster financial assistance programs, in consultation with other ministries as required
    ii. Liaise with the federal government and the NIAC to ensure that any financial assistance provided by Ontario does not duplicate assistance provided under the federal legislation

c. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP

F.8.3 Provision of Personnel

In support of the PNERP, the MMAH will:

a. Provide suitable ministry representatives to serve in the Operations Section of the PEOC. If requested, provide additional suitable ministry personnel for staffing positions in the provincial ERO.

b. Provide a suitable representative to participate on inter-organizational NEMCC subcommittees or working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.8.4 Response

In support of the PNERP, the MMAH will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. If called upon to do so, the MMAH shall support the PEOC by providing:

    i. Advice and assistance regarding the provision of long-term emergency shelter;
    ii. Financial assistance to designated municipalities, host municipalities and supporting municipalities for their eligible response, emergency shelter and long-term emergency shelter costs, as directed by the government; and
    iii. Financial assistance to impacted residents for eligible costs not covered by the NLCA, as directed by the government.

c. Establish a Ministry Action Group to direct and co-ordinate ministry actions under the direction of the CEM or designate (e.g. PEOC Commander) at the first practicable occasion during a nuclear emergency.
d. Channel emergency information for public release through the PEOC PIMS.

F.8.5 Training and Exercises

In support of the PNERP, the MMAH will:

a. Execute the training and exercise responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff
b. Participate in nuclear emergency training and exercises.

F.9 MINISTRY OF NATURAL RESOURCES

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.9.1 Planning and Preparedness

In support of the PNERP, the MNR will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Provide maps and topographical data as required
c. Provide subject matter expertise related to wildlife and fish populations, water control structures, and overland water flows
d. Participate in the preparation of plans and procedures for Environmental Radiation and Assurance Monitoring for nuclear emergencies
e. Ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
f. Participate, as required, in a NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups
g. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the Public Education and Awareness Strategy.

F.9.2 Provision of Personnel

In support of the PNERP, the MNR will:

a. Establish a Ministry Action Group to direct and co-ordinate ministry actions under the direction of the CEM or designate (e.g. PEOC Commander) at the first practicable occasion during a nuclear emergency
b. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues
c. Provide suitable MNR representatives to serve in the Operations and Scientific Sections of the PEOC.

F.9.3 Response

In support of the PNERP, the MNR will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Execute the provincial ministry emergency response plan
c. Provide aircraft, telecommunications, and other resources, if required by the PEOC Commander
d. Channel emergency information for public release through the PEOC PIMS.

F.9.4 Training and Exercises

In support of the PNERP, the MNR will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.
b. Arrange appropriate training of MNR staff involved in the scientific section of the PEOC.

F.10 MINISTRY OF NORTHERN ECONOMIC DEVELOPMENT AND GROWTH

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.10.1 Planning and Preparedness

In support of the PNERP, the MND will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a radiological or nuclear emergency
b. Participate, as required, in a NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups
c. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the Public Education and Awareness Strategy.

F.10.2 Provision of Personnel

In support of the PNERP, the MND will:

a. Establish a Ministry Action Group to direct and co-ordinate ministry actions under the direction of the CEM or designate (e.g. PEOC Commander) at the first practicable occasion during a nuclear emergency
b. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.10.3 Response

In support of the PNERP, the MND will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Execute the provincial ministry emergency response plan, which could include, among other things, the actions outlined below

c. Support emergency response operations in Northern Ontario, whether coordinated by the PEOC or another provincial ministry that has been assigned lead responsibility
d. Channel emergency information for public release through the PEOC PIMS
e. Assist in communicating information and government messages to northern communities
f. Assist with communications between local emergency response units, the PEOC and other ministries, as appropriate
g. Provide personnel, equipment, and material to support emergency response operations in Northern Ontario, as required and available.

F.10.4 Training and Exercises

In support of the PNERP, the MND will execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.

F.11 MINISTRY OF TRANSPORTATION

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.11.1 Planning and Preparedness

In support of the PNERP, the MTO will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a radiological or nuclear emergency
b. Coordinate the development of the transportation management function of evacuation plans for nuclear emergencies with participation of the NEMCC Transportation Management Sub-Committee which includes Ministry of Transportation (MTO), Ontario Provincial Police (OPP), Metrolinx, designated municipalities and designated host municipalities (including local police, road, and transit authorities), applicable provincial ministries and others, as required
c. Lead the development of a Unified Transportation Coordination Centre (UTCC) to manage transportation aspects of evacuation planning and to ensure the operationalization of Unified Transportation Management Plans (UTMPs). Site-specific UTMPs shall be developed for each area covered by a site-specific Implementing Plan
d. Establish operating procedures related to the transportation management function and provide them to the Ministry Action Group to ensure their operational readiness
e. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups

f. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the Public Education and Awareness Strategy.

Note: Ministry staff working in the PEOC should have an overall knowledge of their emergency plans, the PNERP, and UTCC plans and procedures.

F.11.2 Provision of Personnel

In support of the PNERP, the MTO will:

a. Ensure suitable personnel for staffing various positions in the provincial ERO
b. Provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.11.3 Response

In support of the PNERP, the MTO will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. Convene the Ministry Action Group to direct and co-ordinate ministry actions under the direction of the CEM or designate (e.g. PEOC Commander) at the first practicable occasion during a nuclear emergency.
c. Provide assistance and resources for response, as directed by the CEM or designate (e.g. PEOC Commander), and as required by the UTMPs.
d. Inform the PEOC immediately upon the closing of any provincial roads which might be earmarked as evacuation routes and suggest alternatives as per the transportation management function methodologies and plans.
e. Correspond with the PEOC and other transportation organizations, as needed, to employ an “all routes out” strategy by arranging for route clearance, road maintenance, diversions, etc. on provincial roads.
f. Work with the OPP and other supporting organizations, to ensure the operationalization of the UTMPs.
g. Manage the transportation function during recovery, if required.
h. Channel emergency information for public release through the PEOC PIMS.

F.11.4 Training and Exercises

In support of the PNERP, the MTO will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff.

b. Co-ordinate and participate in nuclear evacuation transportation management training and exercises, with transportation consulting partners across the province, to ensure the efficiency and effectiveness of the UTCC and UTMPs.

F.11.5 Infrastructure

In support of the PNERP, the MTO will accord priority to maintaining and keeping open the provincial highways (if possible), earmarked for evacuation routes and diversion routes in the UTMPs.

F.11.6 Study and Research

In support of the PNERP, the MTO will lead the development and maintenance of site-specific UTMPs for reactor facilities in Ontario under the advisement of the NEMCC Transportation Management Sub-Committee.

F.12 MINISTRY OF THE SOLICTOR GENERAL

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.12.1 Planning and Preparedness

Under the PNERP, the SOLGEN will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a radiological or nuclear emergency
b. Maintain appropriate plans and preparations to execute its operational role in a nuclear emergency, including procedures for the OPP General Headquarters EOC, other OPP EOCs, and participation in the development of UTMPs
c. Maintain appropriate plans and preparedness for the emergent deployment of OPP “emergency workers” prior to and after the establishment of an Emergency Workers Centre (EWC), including dosimetry, thyroid blocking, decontamination, and other personal protective equipment, as necessary
d. Ensure the plans and process are developed internally with OPP Physician(s) to ensure that member health and safety is safeguarded to the greatest possible standard, and (on behalf of the Commissioner of the OPP) provide direction to members on radiological exposure, thyroid blocking and the utilization of other personal protective equipment and safeguards to help ensure member safety
e. Ensure that it maintains a provincial Rapid Mobilization Strategy to support sustained emergent operations, on a provincial level that also addresses second/ third order effects and the potential for multiple sites

f. Through partnerships at the Ontario Association of Chiefs of Police (OACP) under the Emergency Management Sub-Committee; establish and chair a Nuclear Emergency Working Group to:

    i. Facilitate the coordination between all Ontario Police Services of Jurisdiction (and external partner agencies) on a framework and plans to consolidate integrated police resource allocations
    ii. Assist in ensuring public safety in a radiological/nuclear emergency

g. Provide OPP representation on the provincial NEMCC, Transportation/ Evacuation Sub-Committee, and other NEMCC Sub-Committees and Working Groups as required
h. Maintain plans and procedures (integrated with the Solicitor General) to ensure the co- ordination of consistent emergency information and public messaging and direction on road closures, diversion routes, area orders, 911 services, and other related emergency messaging
i. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.12.2 Provision of Personnel

Under the PNERP, the SOLGEN will provide administrative support and a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.12.3 Response

Under the PNERP, SOLGEN will:

a. On behalf of the ministry or the province, ensure the provision of trained and knowledgeable OPP representation and applicable resources are provided within the:

    i. Ministry of the Solicitor General (Ministry Action Group)
    ii. Unified Traffic Coordination Centre (UTCC)
    iii. Emergency Worker Centre (EWC) (if required)
    iv. Joint Information Centre (J-EIC) (if required)

b. On behalf of the ministry or the province, the Office of the Fire Marshal shall:

    i. Coordinate disaster firefighting and suppression resources available through the Provincial Mutual Aid Plan under the Fire Protection and Prevention Act

ii. Coordinate the provision of provincial chemical, biological, radiological, nuclear and explosives (CBRNE) / hazardous materials (HazMat) teams
iii. Coordinate the provision of provincial urban search and rescue teams (USAR)
iv. Coordinate provision of CBRNE resources in the province.

F.12.4 Training and Exercises

Under the PNERP, SOLGEN will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff
b. Develop and maintain a coordinated internal training program to meet OPP Nuclear/ Radiological Plan(s) objectives and deliverables
c. Ensure participation by all required staff in municipal, provincial, and centre specific nuclear emergency training and exercises
d. Ensure OPP staff deployed for a radiological or nuclear emergency response have an overall knowledge of their emergency plans and PNERP.

F.13 MINISTRY OF EMERGENCY PREPAREDNESS AND RESPONSE

The Minister, together with the agencies, boards and commissions operating to assist the Ministry, have the following responsibilities consistent with the planning responsibilities assigned by Order in Council under Section 6 of the EMCPA.

F.13.1 Planning and Preparedness

In support of the PNERP, MEPR will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a radiological or nuclear emergency.
b. Undertake an annual internal review of the PNERP, PNEPP, and PNROP, and conduct a documented review with appropriate nuclear partners.
c. Administer the PNERP and oversee its implementation in collaboration with provincial ministries and in coordination with nuclear emergency management partners.
d. Monitor, co-ordinate, and assist in the development of the nuclear emergency response programs of provincial ministries (and agencies, boards, and commissions), nuclear facilities, and designated and host municipalities.
e. Issue, or arrange for the issuance, of procedures and other documents needed to ensure effective implementation of the PNERP.
f. Ensure the establishment of an adequate alerting, notification, and response system for nuclear emergencies.
g. Liaise with MOH to develop and maintain health-related nuclear emergency response strategies, including those related to KI pill distribution and ingestion,

suspension of non-urgent admissions to affected hospitals, evacuation of health centres, population monitoring and dose registry, and medical management of causalities.

h. Liaise with MOH to develop and maintain a strategy for the distribution of KI pills as necessary during an emergency.
i. Ensure that the infrastructure required to implement the PNERP is available and is kept in operational readiness.
j. Monitor and assess the operational readiness and effectiveness of all elements of the emergency response organization, including those of municipalities, provincial ministries and agencies, nuclear facilities, and facility operators, and make recommendations for improvement, where necessary and appropriate
k. Provide guidance and coordinate efforts with reactor facilities, designated municipalities, and other consulting partners as appropriate for the public awareness and education program for populations likely to be affected by a nuclear emergency.
l. Ensure the continuity of government services in coordination with ministry continuity of operations plans (as required under the EMCPA)
m. Ensure the divisions and branches of Emergency Management Ontario (EMO) co-ordinate and make plans and preparations for implementing any protective measure ordered during a nuclear emergency in any of its facilities or operations situated inside the Detailed Planning Zone of a reactor facility and for those that may be affected by a radiological emergency.
n. Coordinate with provincial ministry representatives to ensure essential functions in the emergency management organization are fulfilled.
o. Ensure ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP.
p. Equip and maintain the PEOC in a state of continuous readiness, in order to meet the 15 minutes notification requirement.
q. Develop and maintain continuity of operations arrangements for the PEOC, including the provision of an alternate service delivery location that can meet the technical and notification needs of the PNERP.
r. Develop and maintain the PNERP supporting documentation for Emergency Information Management and Recovery.
s. Facilitate, maintain and participate in a Public Education and Awareness Program.
t. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.13.2 Provision of Personnel

In support of the PNERP, the MEPR will provide administrative support and a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.13.3 Response

In support of the PNERP, the MEPR will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency.
b. Ensure that the PEOC co-ordinates and supports the response to a radiological or nuclear emergency in collaboration with ministry Emergency Operations Centres and in coordination with other members of the nuclear emergency response organization.
c. Provide recommendations to the government regarding declaration and termination of a provincial emergency.
d. Co-ordinate with ministries through the PEOC to ensure continuity of provincial government services.
e. Provide information through the CEM to the Deputy Ministers Steering Committee on Emergency Management and the Cabinet Committee on Emergency Management, in coordination with provincial ministries, regarding the provincial activities related to emergency response and continuity of government services during an emergency.
f. Channel emergency information for public release through the PEOC Public Information Management Section and coordinate with ministry emergency information functions and spokespersons as appropriate.
g. Through the PEOC, provide logistical arrangements and support for ERAMG field teams.

F.13.4 Training and Exercises

In support of the PNERP, the MEPR will:

a. Execute the training and exercise responsibilities described in the PNERP prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff
b. Develop and maintain a PEOC training program to ensure that EMO staff working within the PEOC have an overall knowledge of their emergency plans, the PNERP, and their roles and responsibilities during a radiological or nuclear event
c. Ensure participation by all required staff in nuclear emergency training and exercises.

F.13.5 Study and Research

In support of the PNERP, the MEPR will:

a. Arrange, through the NEMCC structure, for studies and research to be carried out in the area of nuclear emergency effects, planning, management, and response
b. Remain abreast of the state of the art and science in this area.

F.14 COMMUNICATIONS BRANCH

F.14.1 Planning and Preparedness

On behalf of MEPR and the province, the Communications Branch shall:

a. Develop and maintain a Provincial Emergency Information Plan.
b. Develop plans and procedures to ensure the co-ordination of emergency information from ministries and other consulting partners
c. Assist in the development of the emergency information program of provincial ministries (and agencies, boards, and commissions), reactor facilities and designated and host municipalities
d. Develop plans and procedures to prepare staff to assist any municipality that may have to respond to a radiological emergency to co-ordinate emergency information
e. Ensure ministry staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
f. Participate, as required, in an NEMCC sub-committee related to long-term housing and the associated economic, educational, health, psychosocial, and other issues associated with the relocation of large population groups.

F.14.2 Provision of Personnel

In support of the PNERP the Communications Branch will provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

F.14.3 Response

In support of the PNERP the Communications Branch will:

a. Direct and support emergency information activities in the PEOC during a radiological or nuclear emergency
b. Ensure that all provincial ministry emergency information is co-ordinated during an emergency response
c. Ensure the coordination of emergency information being released by all jurisdictional levels involved in the emergency response
d. Assist municipalities in their emergency information operations during an emergency response to help ensure that all emergency information is being handled in a timely, consistent, and accurate manner

e. Establish a Public Information Management Section within the PEOC to direct and co-ordinate emergency information at the first practicable occasion following a nuclear emergency.

F.14.4 Training and Exercises

In support of the PNERP the Communications Branch will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a radiological or nuclear emergency to ensure appropriate training of ministry staff
b. Develop and maintain a PEOC training program to ensure that Communications Branch staff working within the PEOC have an overall knowledge of their emergency plans and the PNERP
c. Ensure participation by all required staff in nuclear emergency training and exercises.

F.15 NON-ORDER IN COUNCIL MINISTRIES

This section applies to those Ministers (together with the leadership of the agencies, boards, and commissions operating to assist them) whose mandates lie outside the scope of the OIC. In accordance with EMPCA 5.1:

In accordance with section 5.1 of the EMCPA, every Minister of the Crown presiding over a ministry of the government of Ontario and every agency, board, commission and other branch of government designated by the Lieutenant Governor in Council shall develop and implement an emergency management program consisting of:

a. an emergency plan as required by section 6 of the EMCPA
b. training programs and exercises for public servants and other persons with respect to the provision of necessary services and the procedures to be followed in emergency response and recovery activities
c. public education on risks to public safety and on public preparedness for emergencies
d. any other element required by the standards for emergency management programs set under section 14 of the EMCPA.

While not specifically identified within the OIC, non-OIC ministries should still prepare for the impact of a nuclear emergency on their operations. Non-OIC ministries include:

a. Ministry of Education
b. Ministry of Finance
c. Ministry of Francophone Affairs
d. Ministry of Indigenous Affairs and First Nations Economic Reconciliation
e. Ministry of Infrastructure
f. Ministry of Intergovernmental Affairs

g. Ministry of Long-Term Care
h. Ministry of Tourism, Culture and Gaming

Ministry of Sport

To the extent that the circumstance in a nuclear emergency may permit, ministries without responsibilities outlined in Annex F1-14 of the PNERP should endeavor to conduct the following operations endeavour to prepare for the impact of a nuclear emergency on their operations.

F.15.1 Planning and Preparedness

In support of the PNERP, non-OIC ministries will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a nuclear emergency
b. Ensure ministry staff with emergency response assignments to the PEOC should have knowledge of their emergency responsibilities, consistent with the PNERP and with the ministry's mandate.

F.15.2 Provision of Personnel

In support of the PNERP, non-OIC ministries will:

a. As required, ensure suitable personnel are available for staffing positions in the PEOC, as consistent with their ministry mandate in an emergency
b. As required, provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups.

F.15.3 Response

In support of the PNERP, non-OIC ministries will:

a. Execute the applicable responsibilities described in the PNERP, and consistent with the ministry's mandate, at the first practicable occasion during a nuclear emergency
b. Convene a Ministry Action Group to direct and co-ordinate ministry actions under the direction of the PEOC at the first practicable occasion during a nuclear emergency
c. Provide assistance and resources as required for emergency response and consistent with the ministry's mandate.
d. Channel emergency public information, as it pertains to the ministry's mandate, for public release through the PEOC PIMS.

F.15.4 Training and Exercises

In support of the PNERP, non-OIC ministries will participate in training and exercises described in the Nuclear Training Program and the Nuclear Exercise Program to ensure appropriate training of ministry staff.

ANNEX G. ONTARIO REACTOR FACILITY RESPONSIBILITIES

Pursuant to the NSCA and Section 6 of the Regulations for Class 1 Nuclear Facilities, and based upon agreements made with the province, reactor facilities have the following responsibilities:

G.1 PLANNING AND PREPAREDNESS

In support the PNERP, Ontario's Reactor Facilities will:

a. Execute the applicable preparedness responsibilities described in the PNERP prior to a nuclear emergency.
b. Set up and maintain the organization, equipment, and procedures necessary to fulfil their functions and responsibilities under the PNERP
c. In accordance with CNSC Regulatory Document 2.10.1 and with CSA N1600, assist the province in carrying out reviews of the PNERP's planning basis through the provision of applicable risk assessment and related studies and by participating in associated working groups.
d. Assist the province and the designated municipalities in their planning and preparedness for a nuclear emergency.
e. Provide resources for and assist the designated municipalities to set up and maintain a KI pill distribution program pursuant to the PNERP.
f. Assist in the development and implementation of the public awareness and education program.
g. On an annual basis, provide the province with the most current data from their Radiological Environmental Monitoring Program
h. Ensure reactor facility staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP.
i. Develop procedures for carrying out the collection and delivery of samples, and other assigned actions, as required under the ERAMG Operations Manual
j. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the in Public Education and Awareness Strategy.

G.2 PROVISION OF PERSONNEL

In support the PNERP, Ontario's Reactor Facilities will:

a. Provide suitable personnel for staffing various positions in the provincial ERO (i.e., a technical liaison for the Scientific Section and a corporate liaison officer for Operations), plus qualified personnel for the Nuclear Emergency Assessment Team and ERAMG-PEOC Team
b. Provide suitable personnel from the Offsite Survey Team (OSST) to join the ERAMG-Field Team once facility monitoring, surveillance, and sampling tasks have been completed

c. Provide a suitable representative to participate on inter-organizational NEMCC sub- committees or working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

G.3 RESPONSE

In support the PNERP, Ontario's Reactor Facilities will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Send regular reports and plant data to the PEOC Scientific Section on the status of the emergency situation during the emergency response.
c. Provide personnel and resources for offsite personal monitoring as well as field monitoring services as specified in the PNERP and in the site-specific Implementing Plan and procedures
d. Provide field-based radiation monitoring service to the Environmental Radiation and Assurance Monitoring Group once all facility-assigned tasks have been completed
e. Pickering, Bruce, and Darlington: Carry out post-accident venting of containment according to the guidance contained in the PNERP
f. Assist the province and the designated municipalities mitigating the emergency.
g. Coordinate the release and content of emergency information for public release with the PEOC Public Information Management Section.

G.4 TRAINING AND EXERCISES

In support the PNERP, Ontario's Reactor Facilities will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a nuclear emergency to ensure appropriate training of reactor facility staff
b. Ensure that its personnel required to perform any of the tasks within its responsibilities are suitably trained
c. Assist the province and designated municipalities in the development and acquisition of training, as appropriate
d. Implement and participate in nuclear emergency training and exercises.
e. Where appropriate, arrange, or participate in emergency response exercises with the local public safety authorities or municipal response organization
f. Arrange appropriate training of reactor facility staff involved in the PEOC Scientific Section.

G.5 INFRASTRUCTURE

In support the PNERP, Ontario's Reactor Facilities will:

a. Provide and maintain reliable telecommunication links for the facility and specified offsite centres
b. Provide resources for and assist the designated municipalities to set up and maintain a public alerting system pursuant to the PNERP
c. Provide and maintain the equipment and facilities required to execute their responsibilities under the PNERP and its implementing documents.

G.6 STUDY AND RESEARCH

In support the PNERP, Ontario's Reactor Facilities will:

a. Carry out Evacuation Time Estimate (ETE) studies to inform the UTMPs developed for each Ontario reactor facility site
b. Carry out and sponsor studies on risk abatement, risk assessment, and the enhancement of Onsite and offsite safety, as agreed
c. Assist the province in carrying out studies to enhance public safety during nuclear emergencies.

ANNEX H. CNSC LICENSED FACILITIES RESPONSIBILITES

CNSC licensed facilities (nuclear establishments) have the following responsibilities pursuant to the NSCA and its regulations and based upon agreements made with the province.

H.1 PLANNING AND PREPAREDNESS

In support of the PNERP, nuclear establishments will:

a. Execute the applicable preparedness responsibilities described in the Nuclear Emergency Preparedness Framework prior to a nuclear emergency
b. Consult with the province as necessary, to ensure emergency plans and procedures are established for executing their responsibilities under the PNERP and this program, and meeting license requirements
c. Carry out all necessary measures onsite to contain and nullify the hazard.

H.2 PROVISION OF PERSONNEL

In support of the PNERP, nuclear establishments will:

a. Establish liaison and make arrangements with the local public safety authorities for notifying them, when necessary
b. Liaise with the Scientific Section of the PEOC at the first practicable opportunity to discuss and exchange scientific/technical information related to an onsite incident
c. For all other incidents (radiological or nuclear), and at the request of the PEOC Scientific Section, provide scientific/technical advice and guidance related to nuclear reactor accident prognosis or progression, offsite health impacts and health physics assessments.

H.3 RESPONSE

In support of the PNERP, nuclear establishments will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Notify the province, affected municipalities, and CNSC whenever there is an actual or potential hazard to public health, property or the environment from radiation or a radioactive material originating from or belonging to the establishment
c. Assist the public safety authorities in containing and neutralizing the hazard offsite
d. Coordinate the release and content of emergency information for public release with the PEOC Public Information Management Section.

H.4 TRAINING AND EXERCISES

In support of the PNERP, nuclear establishments will execute the applicable training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a nuclear emergency to ensure appropriate training of facility staff.

ANNEX I. DESIGNATED MUNICIPALITY RESPONSIBILITIES

Municipalities designated pursuant to Section 3(4) of the EMCPA as municipalities in nuclear DPZs have the following responsibilities.

I.1 PLANNING AND PREPAREDNESS

In support of the PNERP, designated municipalities will:

a. Execute the applicable preparedness responsibilities described in the Provincial Nuclear Emergency Preparedness Framework prior to a nuclear emergency
b. Prepare a municipal plan for mitigating nuclear emergencies, based on and in conformity with the PNERP. This municipal plan shall include:

    i. Establishment of a municipal contact point to receive and act upon an initial notification from the reactor facility and from the province on a 24/7 basis
    ii. Establishment of detailed arrangements and procedures for implementing precautionary and protective measures in accordance with the PNERP
    iii. Planning data including demographic data, institutional data, resource inventory, etc.
    iv. Details regarding a public alerting system meeting the requirements of public notification
    v. Details regarding a nuclear public awareness and education program;
    vi. Details the provision of emergency public information
    vii. Arrangements to receive and accommodate evacuees, including liaison arrangements with other host municipalities, as appropriate

c. Ensure municipal staff working in the PEOC have an overall knowledge of their emergency plans and the PNERP.
d. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the Public Education and Awareness Strategy.

I.2 PROVISION OF PERSONNEL

In support of the PNERP, designated municipalities will provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the PNERP to ensure alignment and address inter-organizational issues.

I.3 RESPONSE

In support of the PNERP, designated municipalities will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Implement the municipal emergency plan for nuclear emergencies (prepared pursuant to the PNERP and the EMCPA)
c. Carry out the required emergency response under the guidance and support of the province prior to a declaration of a provincial emergency
d. Implement the directions of the province following an emergency declaration, and pursuant to any orders which may be made by the province (Section 7.0.2 of the EMCPA)
e. Coordinate the release and content of emergency information for public release with the PEOC Public Information Management Section.

I.4 TRAINING AND EXERCISES

In support of the PNERP, designated municipalities will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a nuclear emergency to ensure appropriate training of municipal staff
b. Ensure that all municipal personnel assigned any functions under emergency plans for nuclear emergencies are suitably trained for their tasks
c. Implement and participate in nuclear emergency training and exercises.

I.5 INFRASTRUCTURE

In support of the PNERP, designated municipalities will ensure availability of the essential facilities, emergency centres, resources, and equipment required by municipal agencies to mitigate a nuclear emergency.

ANNEX J. DESIGNATED HOST MUNICIPALITY RESPONSIBILITES

Municipalities designated pursuant to Section 3(4) of the EMCPA as municipalities acting as host municipalities, have the following responsibilities.

J.1 PLANNING AND PREPAREDNESS

In support of the PNERP, host municipalities will:

a. Execute the applicable preparedness responsibilities described in the Provincial Nuclear Emergency Preparedness Framework prior to a nuclear emergency
b. Prepare a municipal plan for mitigating nuclear emergencies in conjunction with the lead designated municipalities, which includes:

    i. The municipal emergency plan shall be based upon and in conformity with the PNERP
    ii. Arrangements to receive and accommodate evacuees from the designated municipalities
    iii. Arrangements to host the ERAMG Field Team Command Post ideally co-located with an EWC
    iv. Coordination of reception plans and procedures with the reactor facility's monitoring & decontamination arrangements
    v. Establishment of a municipal contact point, which can receive and act upon an initial notification from the provincial contact point on a 24/7 basis
    vi. Liaison arrangements with the designated municipality (in reactor facility Detailed Planning Zones) officials and with the PEOC to ensure appropriate communication during an emergency
    vii. Detailed arrangements with various municipal departments, including social services, public health, police, fire, paramedic services and volunteer agencies which would be involved in staffing and security arrangements for the Reception and Evacuation Centres
    viii. Arrangements for the provision of emergency information on Reception and Evacuation Centre issues
    ix. This municipal emergency plan shall be based upon the PNERP and shall conform to it

c. Municipal staff working in the PEOC should have an overall knowledge of their emergency plans and the PNERP
d. Participate in Public Education and Awareness Program by maintaining respective strategies and products in accordance with the Public Education and Awareness Strategy.

J.2 PROVISION OF PERSONNEL

In support of the PNERP, host municipalities will provide a suitable representative to participate on inter-organizational NEMCC sub-committees and working groups set up under the Nuclear Emergency Preparedness Framework to ensure alignment and address inter-organizational issues.

J.3 RESPONSE

In support of the PNERP, host municipalities will:

a. Execute the applicable responsibilities described in the PNERP at the first practicable occasion during a nuclear emergency
b. Implement the municipal emergency plan for nuclear emergencies (prepared pursuant to the PNERP and the EMCPA)
c. Carry out the required emergency response under the guidance and support of the province prior to a declaration of a provincial emergency
d. Implement the directions of the province following an emergency declaration, and pursuant to any orders which may be made by the province (Section 7.0.2 of the EMCPA)
e. Coordinate the content and release of emergency information for public release with the PEOC Public Information Management Section.

J.4 TRAINING AND EXERCISES

In support of the PNERP, host municipalities will:

a. Execute the training and exercise responsibilities described in the Nuclear Training Program and the Nuclear Exercise Program prior to a nuclear emergency to ensure appropriate training of municipal staff
b. Ensure that all municipal personnel assigned any functions under emergency plans for nuclear emergencies are suitably trained for their tasks
c. Implement and participate in nuclear emergency training and exercises.

J.5 INFRASTRUCTURE

In support of the PNERP, host municipalities will ensure availability of the essential facilities, emergency centres, resources and equipment required by municipal agencies to mitigate a nuclear emergency.

ANNEX K. PLANNING BASIS

K.1 THE HAZARD

This PNERP details the response to an ionizing radiation hazard, resulting from an accident, malfunction, or loss of control at a reactor facility (i.e., nuclear emergency). The public may be harmed by radiation released from such incidents via the following exposure pathways:

a. External exposure resulting from:

    i. Radiation emitted by airborne radioactive materials in a plume ("cloudshine")
    ii. Radiation emitted by radioactive materials that have deposited on the ground ("groundshine")
    iii. Radiation emitted by radioactive materials that have deposited on clothing or skin ("external contamination").

b. Internal exposure resulting from radioactive materials that entered the body ("internal contamination") by:

    i. Inhalation of radioactive airborne materials
    ii. Skin absorption or wound absorption of radioactive material
    iii. Ingestion of drinking water, plant or animal products that have radioactive materials on or in them.

Health effects from exposure to ionizing radiation can be grouped into two main categories:

a. Stochastic effects are those resulting from chronic low doses of radiation. They present primarily in the form of increased risk of various latent cancer types.
b. Deterministic effects are those resulting from high doses of radiation and include cell death or tissue reactions.

The likelihood of stochastic and deterministic effects is reduced when protective actions are promptly implemented.

Nuclear and radiological hazards may be measured in both Imperial and System International (SI) units. A unit conversion table is provided in Annex R.

Other potential effects of the hazard, including those on mental health, psychological, psychosocial, environment, education, culture, ethics, political governance, traditional ways of knowing/living, etc. must also be considered in response to decision-making.

K.2 PLANNING BASIS FOR NUCLEAR EMERGENCIES

The planning basis involves the identification of hazards that the nuclear emergency management program must address based on the impact on health and safety, property and the environment. In a nuclear emergency, an ionizing radiation hazard could arise from an accident or event at a reactor facility, whereas a radiological emergency is not location dependent.

Elements of the planning basis include:

a. the radiological hazard(s)
b. the basis for protective action decision-making, i.e., GC and operational intervention levels (Annex Q)
c. the potential effects on public health and safety (i.e., potential radiation doses resulting from a) above)
d. the geographical extent of consequences.

K.3 REACTOR FACILITY ACCIDENTS

Nuclear emergency management requires a planning basis which considers both design basis accidents (DBAs) and beyond design basis accidents (BDBAs), including severe accidents and multi-unit scenarios where applicable.

While the planning basis should include a wide range of accidents, the amount of detailed planning should decrease as the probability of the accidents' occurrence decreases. For this reason, the planning basis for managing a nuclear emergency must strike an appropriate balance.

Reactor facility safety analyses and risk assessments shall be used to inform the planning basis.

This PNERP has been prepared in conformity with national and international standards and guidance for nuclear emergency management and, as such, it:

a. Provides detailed planning and preparedness to mitigate the effects of DBAs for which safety systems have been specifically designed to ensure that radiological releases are kept within authorized limits
b. Provides additional tools and mechanisms to mitigate the effects of BDBAs, including severe accidents, which are more unlikely than Design Basis Accidents.

K.3.1 Design Basis Accidents

The DBA release provides the main platform for detailed planning and is generally characterized by one or more of the following:

a. An accident that a nuclear facility has been specifically designed to withstand

b. Station containment systems would function normally, allowing radiation to decay prior to a controlled release
c. Sufficient time would be available to alert the public and implement protective measures prior to a release
d. The main radiological hazard to people would be external exposure to, and inhalation of, radioactive material
e. Filter systems would function to remove almost all of the radioiodines and particulate radionuclides. As a result, the plume would be mostly comprised of inert noble gases, which would dissipate and not pose a contamination hazard.
f. Radiation doses to the public would likely be below the GC as defined in Annex Q
g. Environmental contamination would be limited to very low levels
h. The area affected would likely be contained to the Detailed Planning Zone.
i. Low-level radioactive releases to the environment could occur on and off for some time (e.g., days or weeks).

Planning and preparedness shall be conducted in Ontario to mitigate the consequences of releases from DBAs and shall further provide the basis for an expanded response, as required, to mitigate the effects of a Beyond Design Basis Accident (BDBA).

K.3.2 Beyond Design Basis Accidents

The BDBA is an exceptionally low probability event that is more severe than a DBA and is generally characterized by one or more of the following:

a. A single unit or multi-unit accident (i.e., an accident involving more than one reactor) that exceeds the conditions a facility has been designed to withstand
b. Station containment system impairment leading to:

    i. Significantly reduced hold up time of radioactivity in containment resulting in reduced decay of radioactive materials;
    ii. Little time available to alert the public and implement protective measures;
    iii. An early release of radioactivity; or
    iv. An uncontrolled and unfiltered release of radioactivity

c. The main radiological hazard to people would be external exposure to, and inhalation or ingestion of, released radioactive material
d. Filter systems may fail, leading to the release of a mix of particulates, radioiodines, and noble gasses
e. Released radioiodines could be inhaled or ingested leading to internal exposure of the thyroid gland
f. Radiation doses could potentially be high and could potentially exceed the GC as defined in Annex Q
g. Environmental contamination could be quantitatively significant in both extent and duration

h. The area affected could extend beyond the Detailed Planning Zone into the Contingency and Ingestion Planning Zones
i. Radioactive releases to the environment could occur on and off for some time (e.g., days or weeks).

BDBAs which go unmitigated may evolve into a more severe accident involving fuel degradation in the reactor core.

The response to BDBAs is facilitated by the measures already in place to respond to DBAs and the ability to expand their function.

Additional planning and preparedness measures shall be undertaken to facilitate the implementation of the following response actions, as appropriate, to mitigate the much less probable, but possibly more severe, offsite effects of BDBAs:

a. Initiation of public alerting to direct protective actions, including sheltering and evacuation
b. Priority evacuations for those closest to the hazard
c. Radiation monitoring and, if necessary, decontamination of persons.
d. Urgent Response Phase Iodine Thyroid Blocking (ITB)
e. Timely dispatch of aerial and ground monitoring teams to determine areas of contamination
f. Medical assessment, treatment and counselling as required.

ANNEX L. PROTECTIVE ACTIONS

L.1 PRECAUTIONARY MEASURES

Precautionary measures should be implemented during the Urgent Response Phase either in advance of, or in combination with, exposure control protective measures to facilitate their implementation. Unlike protective measures, precautionary measures are not associated with a numerical intervention level.

The CEM or designate (e.g. PEOC Commander) will direct precautionary measures as appropriate, and where time permits, in consultation with applicable responsible ministries and the affected designated municipalities.

Precautionary measures shall be directed to the public via EBs (Section 8.7.2.2), or other public communications channels (as the situation dictates) and issued by the PEOC Commander. Municipalities will be informed via PNOORF update. The provincial ERO shall be informed in advance via IMS Form 201 or 209, as time permits. Precautionary measures include (but are not limited to):

a. closing of beaches, recreation areas, etc.
b. closing of workplaces and schools
c. suspension of non-critical patient admissions in hospitals
d. entry control.

L.2 PROTECTIVE MEASURES

L.2.1 Evacuation

An evacuation is:

a. A directed exposure control measure for the controlled displacement of the population from an area, which has been or might become contaminated by radioactive substances to avoid exposure
b. Normally less than a period of two weeks. If an evacuation is expected to last longer than two weeks, temporary relocation should be considered (see Annex L.2.5)
c. Normally undertaken during the Urgent or Early Response Phase to avoid or reduce short-term exposure to the plume or deposited radiation.

L.2.1.1 Responsibility for Evacuation

Responsibility for directing evacuation rests with the CEM or designate (e.g. PEOC Commander) or the LGIC, following the decision-making authorities outlined in Section 7.5.

In circumstances where the directive is not being issued as a protective measure following an initial notification from the reactor facility, the CEM or designate (e.g. PEOC Commander) shall consult with the following, as time permits, to determine the applicability of this measure:

a. Scientific Section Chief regarding emission timing, intervention levels, and the appropriateness of evacuation as a protective measure
b. affected designated municipalities
c. CMOH to determine the need to direct ITB as a supplemental protective measure.

The CEM or designate (e.g. PEOC Commander) shall ensure that all protective measures are issued to the public via emergency bulletin (Section 8.7.2.2) and that municipalities (notified through the PNOORF) are affecting the protective measures to the best of their abilities. Upon receipt of a PNOORF, or the issuance of an EB, the municipalities support the implementation of the protective action, where and when they can (e.g., UTCC and local police identify best evacuation options and direct traffic for an efficient evacuation).

L.2.1.2 Considerations for Evacuation

Evacuation should be directed in areas where the projected or actual dose is expected to equal or exceed the evacuation GC (pre-release) or OILs (post-release) (Annex L.2.5).

Evacuation is most effective at limiting exposures when completed prior to the radioactive release.

When evacuation is implemented during a radioactive release, radiation exposure could be increased. Therefore, if the release duration is known to be short, sheltering-in-place may be directed, followed by evacuation after the release. However, if the release is known to be prolonged, evacuation in the plume should be assessed in comparison to sheltering-in-place in order to determine the most effective protective measure to minimize the effective dose to the impacted population.

Evacuation can provide protection against unnecessary exposure to radiological contamination in the environment after a release, when sheltering-in-place is not a viable option.

Evacuation of the affected population should be considered for those areas where sheltering-in-place is expected to be required for more than 48 hours.

In the offsite area immediately adjacent to the source of radiation, evacuation may be the best protective action unless:

a. Persons are unable to safely evacuate (e.g., persons with health conditions that would make evacuation more dangerous than sheltering-in-place); or
b. Conditions exist which make the evacuation hazardous (e.g., inclement weather, road or transportation issues, consequences resulting from another hazard); in such cases, sheltering-in-place may be more appropriate followed by evacuation once the release has ceased and it is possible to safely evacuate.

L.2.2 Sheltering-In-Place

L.2.2.1 General

Sheltering-in-place is normally directed in the Urgent Response Phase of a nuclear emergency response. A directed sheltering-in-place order is a short-term exposure control measure for taking immediate refuge in an enclosed structure, with windows and doors closed and ventilation systems shut down, for protection from an airborne plume, deposited radioactive material, or both.

Sheltering-In-Place may be appropriate as an interim measure in areas where:

a. A release is expected to occur before evacuations can be safely completed.
b. The release is expected to be of short duration, not exceeding 48 hrs.
c. Sheltering-In-Place may be a preferable alternative to evacuation where circumstances prevent a safe and efficient evacuation or for maintenance of essential services:

    i. Severe weather conditions
    ii. Environmental or other hazards
    iii. Transportation infrastructure issues
    iv. For vulnerable populations (e.g., hospitals, long-term care homes, etc.) for whom evacuation may pose greater risks than that of the hazard itself; or
    v. For essential service staffing requirements (e.g., critical infrastructure).

d. For situations where sheltering-in-place is directed, the PEOC Scientific Section Chief should prioritize rapid surveys, undertaken by its Environmental Radiation and Assurance Monitoring Group (ERAMG). The analysis of this information will inform the Scientific Section's recommendations to either lift the shelter-in-place directive/order or conversely, to direct follow-up evacuations.
e. The CMOH should consider the need to direct ITB together with sheltering-in-place, in order to provide maximum protection. This additional layer of protection may not be necessary where the PEOC Scientific Section advises that radioiodine definitively does not pose a hazard (based on analysis of reactor facility plant data).

L.2.2.2 Responsibility

Responsibility for directing sheltering-in-place rests with the CEM or designate (e.g. PEOC Commander) or the LGIC, following the decision-making authorities outlined in Section 7.5.

In circumstances where the directive is not being issued as a protective measure following an initial notification from the reactor facility, the CEM or designate (e.g. PEOC Commander) shall consult with the following, as time permits, to determine the applicability of this measure:

a. Scientific Section Chief regarding emission timing, intervention levels, and the appropriateness of sheltering-in-place as a protective measure
b. affected designated municipalities
c. CMOH to determine the need to direct ITB as a supplemental protective measure.

The CEM or designate (e.g. PEOC Commander) shall ensure that all protective measures are issued to the public via emergency bulletin (Section 8.7.2.2) and that municipalities (notified through the PNOORF) are affecting the protective measures to the best of their abilities.

EBs shall contain sufficient guidance regarding the efficacy of sheltering structures (per L.2.2 above).

L.2.3 Ingestion Control

L.2.3.1 General

The following protective measures for ingestion control, defined in Annex W, include:

a. milk control
b. water control
c. pasture control
d. produce and crop control
e. livestock control
f. wild harvested foods
g. food and foodstuff control
h. land control

Strategies for deciding on and implementing ingestion control measures during a nuclear or radiological emergency include the following:

a. As a precaution, the direction of ingestion control measures during the Urgent Response Phase within the AAZ, the DPZ, and potentially the CPZ

b. The initiation of background environmental radiation and assurance monitoring during the Urgent Response Phase if time permits
c. The initiation of environmental radiation and assurance monitoring for the entire Ingestion Planning Zone (IPZ), at the end of the Urgent Response Phase (or beginning of the Early Response Phase), in accordance with priorities set by the ERAMG
d. The continuation of environmental radiation and assurance monitoring during the Early Response Phase and through to the Recovery Phase to guide recovery decision-making
e. Recommendations for implementation of additional ingestion control long-term protective actions shall be based on analysis of field monitoring data referenced against the OILs detailed in Annex L.2.5.

L.2.3.2 Responsibility

Recommendations on the application, revision, and rescinding of ingestion control measures shall be made by the PEOC Scientific Section Chief to the CEM or designate (e.g. PEOC Commander). These recommendations shall be based on an analysis of monitoring results.

The CEM or designate (e.g. PEOC Commander) shall consult on the recommended measures with affected municipalities, where time permits.

Ingestion control measures are communicated to the municipalities via the PNOORF, and to the public via the EBs or other communications methods issued by the PEOC Commander during the Urgent and Early Response Phases (Annex N) and, as deemed appropriate for the situation during the Recovery Phase.

Designated municipalities shall include the necessary provisions in their municipal nuclear emergency plans in order to ensure the effective implementation of the ingestion control measures for which they have associated responsibilities.

All other municipalities located in the IPZ should ensure that they have the necessary plans and procedures in place to ensure the effective implementation of the ingestion control measures for which they have associated responsibilities (e.g., shutting intake valves of affected drinking water systems).

L.2.4 Iodine Thyroid Blocking

L.2.4.1 General

Iodine Thyroid Blocking (ITB) is a protective measure normally undertaken in the Urgent Response Phase of a nuclear emergency. The ITB protective measure involves the ingestion of potassium iodide (KI) pills to prevent the thyroid gland’s uptake of radioactive iodine, one of the radioisotopes which may be present in a radioactive plume.

ITB is most effective if ingested either prior to, or soon after a radioactive emission.

Detailed ITB planning and preparedness shall be undertaken, in advance of a nuclear emergency, to ensure that the population that may be affected by the release of radioiodine has access to the pills in a timely manner.

L.2.4.2 Guidance for its use a protective measure

Because this measure protects only the thyroid gland, and only from one type of radioactive material (radioiodine), ITB should be used in combination with sheltering-in-place or evacuation. This combination provides more comprehensive, whole-body protection.

The Scientific Section determines whether ITB is required based on exceedance of the thyroid GC as determined through dose modeling. The resultant recommendation is sent to the CMOH, via the CEM, for authorization of consumption. If the CMOH authorizes the consumption of KI, the authorization is returned to the CEM or designate (e.g. PEOC Commander) who will issue the consumption order within the identified area. Guidance for its use as a protective measure includes:

a. Potassium Iodide (KI) pill ingestion should be directed 2-6 hours prior to, or immediately following exposure to an emission to be optimally effective
b. Because this measure protects only the thyroid gland, and only from one type of radioactive element (radioiodine), ITB should be used in combination with sheltering-in-place or evacuation. This combination provides more comprehensive, whole-body protection
c. If evacuation is not feasible, KI should be directed to be ingested using the manufacturer's daily recommended dose, until the risk of significant radioiodine exposure (through inhalation) no longer exists. Note: certain populations (e.g., pregnant and breastfeeding women, and infants <1-month-old) should only take one dose of KI.

Further guidance is provided in the Radiation Health Response Plan and on the Ministry of Health's website.

L.2.4.3 Preparedness

The preparedness requirements set out in this PNERP are consistent with the KI pill provisions found in CNSC regulations. These preparedness requirements do not preclude any organization, or person, from acquiring KI pills for protection in the event of a nuclear emergency. Emergency Social Services planning should consider the need for baby formula supplies for breastfeeding mothers where ITB has been implemented.

L.2.4.4 Response Considerations

Once the radioactive release has ended, provincial direction for ITB should cease unless otherwise justified.

L.2.4.5 Decision-Making

During the response to the initial notification from the reactor facility that an emission is ongoing or imminent, the CEM or designate (e.g. PEOC Commander) shall direct protective actions detailed in the appropriate site-specific implementing plan. This includes the consumption of KI pills, when authorized by the CMOH.

Subsequent ITB protective action decision-making by the CMOH will be based on the Scientific Section Chief's technical assessment and recommendation as communicated through the CEM, and implementation will be directed in consideration of the following:

a. The extent to which the affected public is in possession of KI pills
b. The means and timing to access KI pills, for those who do not possess a supply
c. Information provided through consultation with the local Medical Officer(s) of Health
d. Based on the above, the CMOH shall direct ITB implementation, as appropriate, in coordination with the PEOC.

L.2.4.6 Provincial Responsibilities

EMO and MOH, together with assistance from reactor facilities as applicable, shall develop plans, in coordination with designated municipalities within all Ontario DPZs and CPZs to ensure the availability of KI pills during an emergency. The applicable implementing plans shall include details to guide planning.

The MOH shall procure, in advance, adequate quantities of KI pills, for the Fermi 2 DPZ and CPZ.

L.2.4.7 Designated Municipality Responsibilities

Designated Municipality nuclear emergency plans shall detail provisions for undertaking the following:

a. Pre-distribution of KI pills for DPZ institutions (including schools, daycare centres, seniors' residences, long-term care facilities, hospitals, correctional facilities etc.)
b. Pre-distribution of KI pills for emergency centres designated under this PNERP (EWCs, Reception and Evacuation Centres, and MDUs)
c. Pre-distribution of KI pills for households and businesses in the DPZ.
d. Facilitating the availability of KI pills for populations within that area of the Contingency Planning Zones lying within their borders, wishing to possess a supply
e. Detailed planning and preparedness to ensure that pills can be made available to those that may require them, during a nuclear emergency response. The four

site-specific nuclear facilities (Pickering, Darlington, Bruce Power, and Fermi 2) and the Transborder PNERP Implementing Plans will provide applicable guidance for the development of emergency distribution processes and plans.

L.2.4.8 Reactor Facility Responsibilities

In accordance with CNSC regulations and pursuant to their responsibilities to assist offsite authorities under the Regulations of Class I Facilities (Nuclear Safety and Control Act) and operating licence requirements, reactor facilities are responsible for providing the necessary resources and support to provincial and municipal authorities to ensure that the ITB related requirements of this PNERP and of municipal emergency plans are completed.

L.2.5 Temporary Relocation

L.2.5.1 General

Temporary relocation is a protective measure undertaken post-release, during the intermediate response phase, based on actual measured contamination levels. It is the displacement of people from their homes for a period beyond two weeks and up to one year to avoid chronic exposure to radiation, usually from ground contamination. Beyond one-year, permanent resettlement must be considered. Temporary relocation (i.e., relocation with a duration between two weeks and one year) requires dedicated planning and preparedness. This will be led by a multi-ministry and multi-jurisdictional planning group managing issues including housing, social services, education, etc.

Temporary relocation can be directed as a subsequent measure to evacuation or sheltering-in-place or, as a separate measure.

The need for temporary relocation is determined following analysis of environmental radiation monitoring results and assessment against OILs (Annex L.2.5).

EMO shall consider a variety of factors (optimization) before recommending temporary relocation as the potential impacts of this action may not be justified in areas where the OIL for temporary relocation is minimally exceeded. The CEM on the technical advice of the Scientific Section and the operational advice of the other PEOC sections, will make a recommendation to government on the appropriateness of temporary relocation.

L.2.5.2 Responsibility

In collaboration with relevant ministry officials and following consultation with the affected designated municipalities and designated host municipalities, the CEM shall make recommendations regarding the need for temporary relocation.

Temporary relocation shall be directed through public communications issued by the CEM in coordination with the PEOC Provincial Information Management Section (PIMS).

Relocation planning involves a significant interdisciplinary approach, involving at minimum:

a. Transportation management (e.g., Ministry of Transportation)
b. Reception and evacuation centres (e.g., designated municipalities)
c. Long-term housing (e.g., multi-ministry and multi-jurisdictional planning group)
d. Access to health services and public health issues – coordinated by the Ministry of Health.

L.2.6 Decontamination

L.2.6.1 General

In nuclear and radiological emergencies, personal monitoring determines the presence of contamination on any individual who may be contaminated after a radiological release to the environment. Decontamination serves to reduce or remove contamination. Personal monitoring and decontamination are used to mitigate the exposure effects of external contamination.

Facilities and provisions for personal monitoring and decontamination shall be established and resourced with the specialized equipment and expertise required to undertake this function.

In nuclear and radiological emergencies, population monitoring and medical management (Annex S) refers to the process undertaken by the MOH to assess long term health impacts, following the personal monitoring and decontamination that is undertaken at MDUs.

With the exception of Fermi 2, personal monitoring and decontamination are not expected to be needed for a transborder emergency.

L.2.6.2 Responsibility

The operators of the Bruce, Darlington, and Pickering NGS have responsibility for the radiation monitoring and decontamination activities at emergency centres established for the public and for emergency workers in a nuclear emergency.

For a nuclear emergency at the Fermi 2 NPP, EMO is responsible for coordinating, in advance, arrangements for monitoring and decontamination for evacuees and for emergency workers.

Designated Municipalities shall ensure that their municipal emergency plans include provisions for the establishment, administration and operation of the non-radiological components of centres housing the monitoring and decontamination facilities for the public and for emergency workers.

L.2.6.3 Designated Communities

Reactor facilities in Ontario are responsible for providing resources, qualified staff and their training in order to meet their responsibility for the monitoring and decontamination of evacuees and emergency workers (see section above) in their designated community.

EMO shall co-ordinate with appropriate relevant partners to establish arrangements for monitoring and decontamination of both emergency workers and evacuees as outlined in Sections 8.8 and 8.10 for the Town of Amherstburg.

ANNEX M. EMERGENCY PLANNING ZONES

M.1 INTRODUCTION

The emergency planning zones (EPZ) define the areas beyond the boundary of a reactor facility, in which implementation of operational and protective actions are, or might be, required during a nuclear emergency in order to protect public health, safety and the environment.

Emergency Planning Zones shall be delineated in the appropriate site-specific implementing plan as follows:

a. With consideration for municipal boundaries, First Nation and Indigenous communities and traditional territories, existing roads and geographic features
b. To facilitate the effective implementation of operational and protective measures
c. To provide a distinction between the types and extent of protective actions that may be required to protect against the radioactive plume and the resulting contamination, as they increase in distance from the reactor facility.

Planning and preparedness activities are undertaken to facilitate the implementation of external exposure and ingestion control protective measures in the emergency planning zones during a response. Some protective measures, while requiring operational arrangements to facilitate their implementation, do not require detailed planning. For example:

a. Sheltering can be implemented through existing public alerting mechanisms, advising the population to stay indoors
b. ITB requires an effective distribution program in advance of an emergency.

In contrast, detailed plans and procedures are required to implement evacuations including those pertaining to the following:

a. Transportation and traffic
b. Monitoring and decontamination
c. Short-term accommodation
d. Long-term accommodation
e. Provision of health services, economic and trade disruptions, alternate education arrangements, etc.

As such, the DPZ for the CANDU stations have been delineated to accommodate the projected effective dose at which the evacuation generic criterion is reached. Projected doses for sheltering and ITB were not used in the DPZ determination.

Emergency Planning Zones have been defined in this PNERP as:

a. Automatic Action Zone (AAZ)

b. Detailed Planning Zone (DPZ)
c. Contingency Planning Zone (CPZ)
d. Ingestion Planning Zone (IPZ)

While each of the above zones are distinct and do not overlap, when ingestion control measures are initiated for the IPZ, they should be implemented for all other zones noted above.

This diagram is for visual reference only. See the text below this figure for the full description.

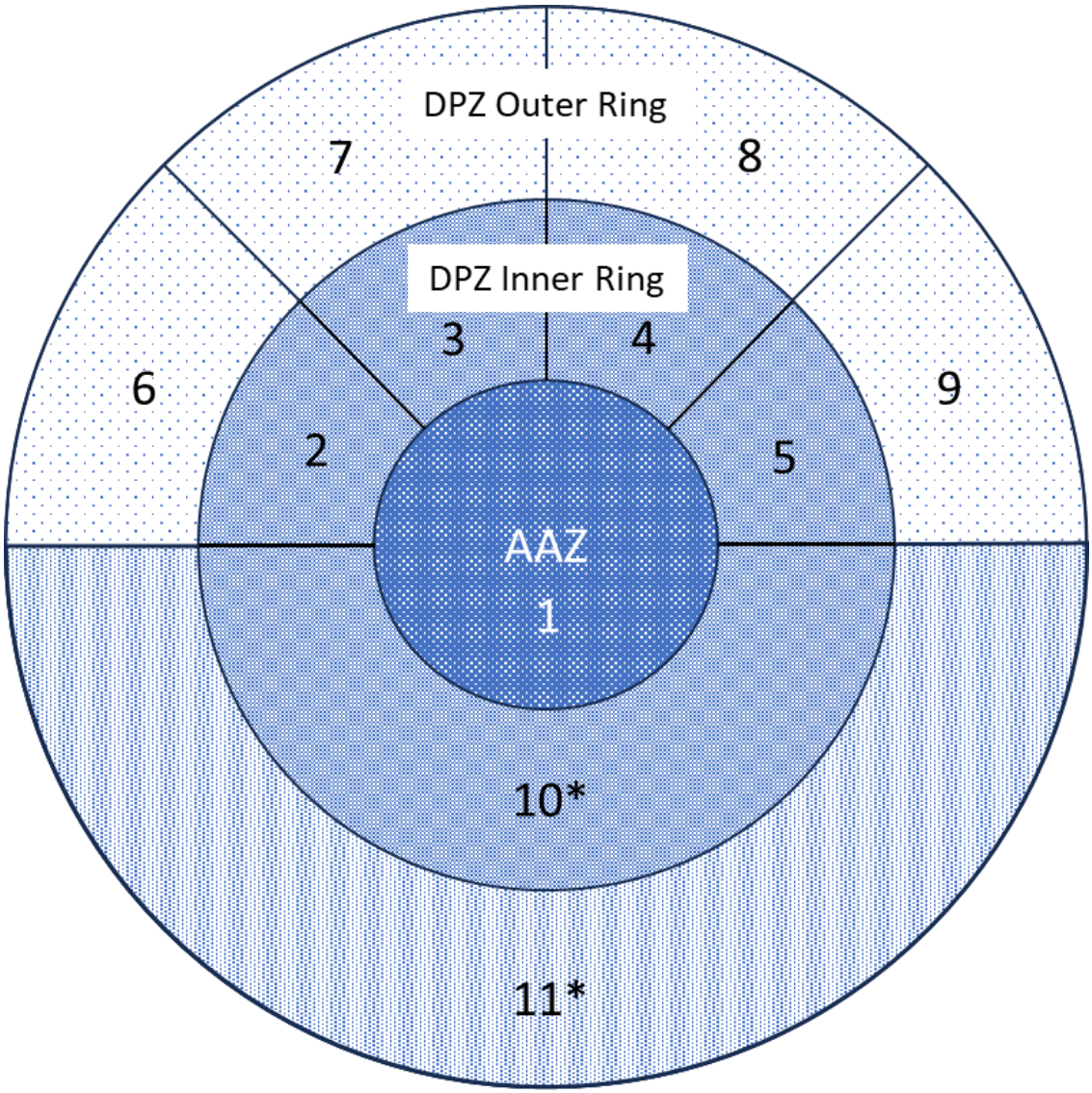

Figure M – 1: Detailed Planning Zone and Response Sectors
(Diagrammatic - Not to Scale, * = Lake/River Sectors)

The diagram presents a circular, multi-layered structure that organizes emergency planning zones into defined response sectors. At the center is the Automatic Action

Zone (AAZ 1), representing the innermost emergency planning zone. The Detailed Planning Zone (DPZ) encompasses AAZ 1. The DPZ Inner Ring is divided into five response sectors labeled 2, 3, 4, 5, and 10*. The DPZ Outer Ring encompasses the inner ring and AAZ 1 and is comprised of five additional response sectors labeled 6, 7, 8, 9, and 11*.

This diagram is for visual reference only. See the text below this figure for the full description.

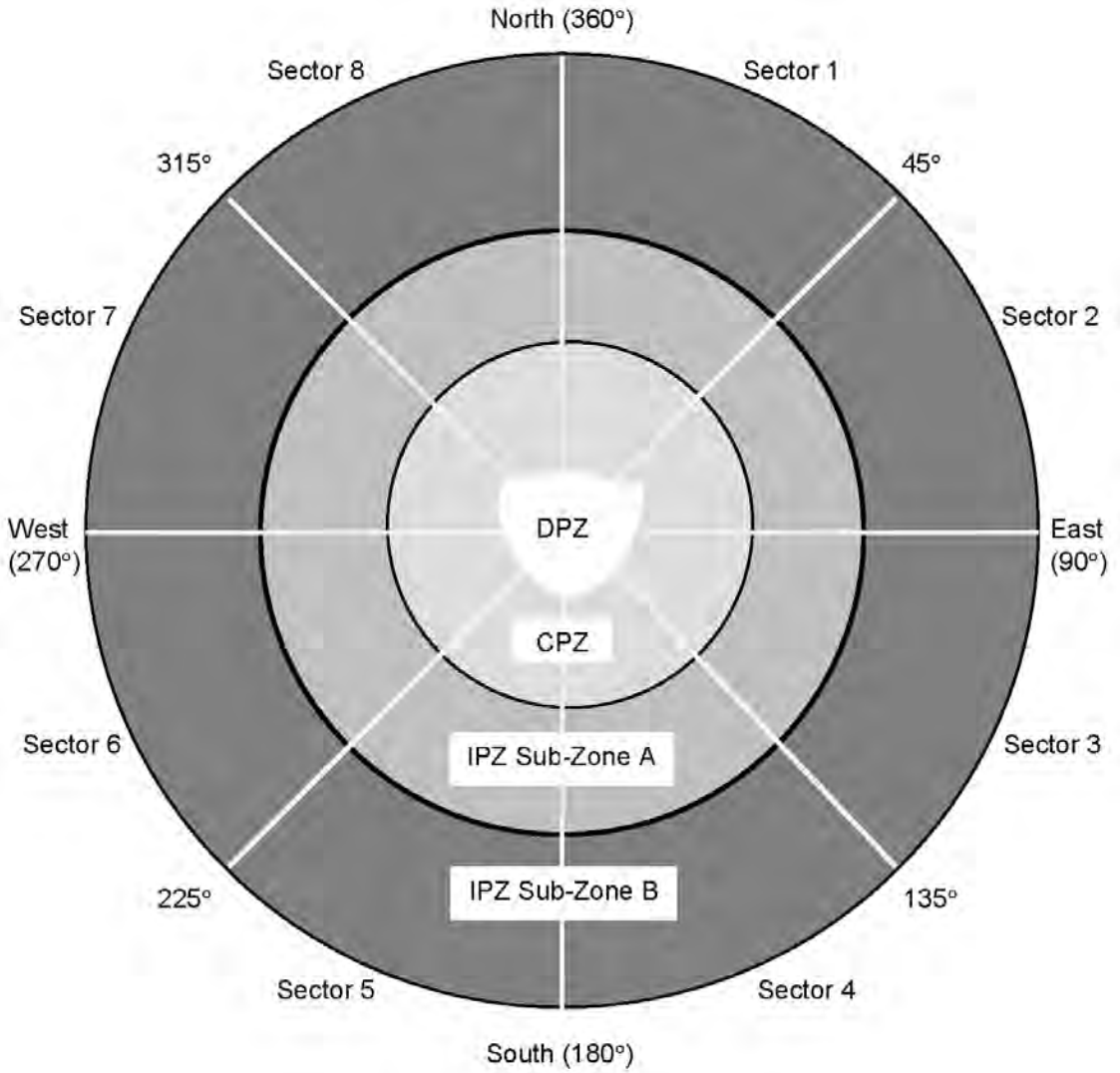

Figure M – 2: Ingestion Planning Zones

LEGEND

Detailed Planning Zone
Contingency Planning Zone
Ingestion Planning Zone (IPZ) Sub-Zone A

Ingestion Planning Zone (IPZ) Sub-Zone B

The diagram presents a circular, multi-layered structure that illustrates the relationship between the Detailed Planning Zone (DPZ), the Contingency Planning Zone (CPZ), and the Ingestion Planning Zone (IPZ). At the center is the DPZ, surrounded by the CPZ, forming the next ring. Beyond the CPZ are two additional concentric zones: IPZ Sub Zone A (inner) and IPZ Sub Zone B (outer), each extending outward from the CPZ. The entire circular layout is divided into eight sectors, labeled Sector 1 through Sector 8, aligned with the compass directions North (360 degrees), East (90 degrees), South (180 degrees), and West (270 degrees), as well as the intermediate bearings at 45, 135, 225, and 315 degrees.

M.2 AUTOMATIC ACTION ZONE (AAZ)

A pre-designated area immediately surrounding a reactor facility where pre-planned protective actions would be implemented on the basis of reactor facility conditions, with the aim of preventing or reducing the occurrence of severe deterministic effects.

Additional detailed planning and preparedness shall be undertaken for the AAZ to prevent or reduce deterministic effects for this zone, including the implementation of urgent protective actions (e.g., evacuation, sheltering-in-place, and ITB) during General Emergencies and some Onsite Emergencies.

M.3 DETAILED PLANNING ZONE (DPZ)

A pre-designated area surrounding a reactor facility where pre-planned protective actions are implemented as needed on the basis of reactor facility conditions, dose modelling, and environmental monitoring, with the aim of preventing or reducing the occurrence of stochastic effects (latent cancers).

Detailed planning and preparedness shall be conducted for the DPZ to ensure that protective measures can be implemented and that the associated needs and requirements of the affected public can be met.

The DPZ around a reactor facility shall be divided into response sectors. All emergency response measures, both operational and protective, shall be planned and implemented in terms of these sectors.

The desirable pattern of response sectors in a DPZ is illustrated in Figure M – 1. Response sectors lie within two rings around the reactor facility: an inner ring (which lies beyond the AAZ) and an outer ring. Within each ring, it is desirable to have as few sectors as possible while maintaining the need for flexibility and practicability in the application of the operational response strategy.

The actual demarcation of the response sector boundaries shall be such that, as far as practical, they lie along clearly recognizable features, such as roads, waterways and

railway tracks. Other factors to be taken into account shall be municipal boundaries, population densities and availability of appropriate evacuation routes.

M.4 THE CONTINGENCY PLANNING ZONE (CPZ)

A pre-designated area surrounding a reactor facility, beyond the DPZ (Figure M – 2), where contingency planning and arrangements are made in advance, so that during a nuclear emergency, protective actions can be extended beyond the DPZ, as required, to reduce the potential for exposure.

Plans and arrangements for this CPZ include considerations for:

a. Division into sub-zones
b. Population estimates for each sub-zone
c. Development of mechanisms, processes and procedures to provide for environmental radiation monitoring and data analysis by the PEOC Scientific Section
d. Familiarization sessions with impacted municipalities, as required
e. Identification of existing response centres that fall within the CPZ and development of a list of possible alternates located outside the CPZ
f. ITB arrangements as applicable and as agreed to by relevant partner consultation
g. Public awareness and education requirements consistent with Ingestion Planning Zone requirements
h. No requirement for designation of additional emergency response centres (e.g., Emergency Operations Centres (EOCs), the Emergency Information Centre (EIC), reception and evacuation centres, personal monitoring and decontamination facilities, etc.) beyond those designated for the DPZ
i. No additional public alerting and communications requirements beyond those established for the DPZ.

M.5 INGESTION PLANNING ZONE (IPZ)

The IPZ is a pre-designated area surrounding a reactor facility where plans or arrangements are made to:

a. Protect the food chain
b. Protect drinking water supplies
c. Restrict consumption and distribution of potentially contaminated produce, wild harvested foods (e.g., mushrooms and game animals) milk from grazing animals, rainwater, and animal feed
d. Restrict distribution of non-food commodities until further assessments are performed.

The IPZ is divided into concentric sub-zones in order to facilitate implementation of protective measures:

a. Ingestion Control Sub-Zones A and B are each divided into eight sub-zones
b. Ingestion control measures are further discussed in Section L.2.3.

The divisions described above are illustrated in Figure M – 2. Actual zones and response sectors for each reactor facility shall be described in the applicable site-specific Implementing Plan.

M.6 EMERGENCY PLANNING ZONE DISTANCES

EPZ distances for reactor facilities are established based on a number of scientific factors including the design and number of reactors on site, and for Fermi 2, considerations for cross-border coordination. EPZ distances are determined by the application of the methodology and the details of which are found in the site-specific implementing plans. EPZ distances may vary from site to site. The radii of the EPZ surrounding each of the reactor facilities, as measured from the venting or release stacks, are detailed in the relevant site-specific implementing plans.

ANNEX N. NUCLEAR EMERGENCY PHASES

Due to the nature of nuclear reactor emergencies, where the hazard and its location are known and, the extent of the hazard is normally verifiable, protective action decision-making can be categorized according to the following phases of the emergency:

a. Urgent Response Phase
b. Early Response Phase
c. Transition Phase
d. Recovery (See the PNROP for more details)

N.1 URGENT RESPONSE PHASE

The Urgent Response Phase starts with an initial notification from the reactor facility of a General or Onsite Emergency notification category, which may require urgent protective and operational actions to deal with the immediate effects of radiation.

Upon receipt of an initial notification from the reactor facility of an ongoing or imminent emission, protective measures should be implemented (subject to operational situation considerations), based on a conservative estimate of the situation because time or data may not be available to carry out a comprehensive assessment of imminent risk. These protective measures shall be detailed in the applicable site-specific implementing plan and may include sheltering-in-place, evacuation, and ITB.

In situations where a release is not imminent or ongoing, decision-making should commence as soon as data is available (i.e., status of affected reactor(s)' control, cooling, and containment) and well before exposure is expected to occur. This is accomplished with an assessment of projected doses, compared against the GC (Annex Q), to determine the need for protective measures (Annex L).

Ingestion control measures shall be directed as appropriate. The measures may be modified in later phases based on the results of environmental field monitoring of food, milk, and water. The following additional protective measures are not associated with either a numerical intervention level (GC or OILs) and may be directed in combination with any of the protective and precautionary measures listed in Annex L:

a. protective clothing
b. respiratory protection
c. self-decontamination

Population monitoring and medical management would begin during the Urgent Response Phase of a nuclear response (See Annex S).

N.2 EARLY RESPONSE PHASE

The Early Response Phase begins (and, by extension, the Urgent Response Phase ends) once the nuclear release is brought under control (though not necessarily contained), permitting environmental radiation surveys to continue, radiological sampling to be undertaken, and decision-making to be based on actual dose rate and radiological contamination measurements.

Evacuation and/or temporary relocation are the preferred protective measures during the Early Response Phase to prevent external exposure from deposited radioactive particles (e.g., ground contamination) as well as to prevent internal exposure from inhalation of radioactive particulates.

Ingestion control measures shall be directed as appropriate based on the results of environmental field monitoring of food, milk, and water.

A technical assessment of actual environmental radiation monitoring results applied against OILs (Annex Q.2) should inform the decision-making process to determine the need for both ingestion and exposure control measures.

The developing situation should be continuously re-assessed, and appropriate decisions made to apply protective measures and rescind those no longer necessary.

During the Early Response Phase, the CEM or designate (e.g. PEOC Commander) may recommend other, practical dose reduction measures to the public. Such measures may be implemented in combination with the measures described in this Annex or may simply be recommended to provide an additional level of protection against possible radioactive material present in the air or on the ground. Implementation of these measures are not based on GC or OILs and include:

a. Respiratory protection, such as covering of the nose and mouth with available material that can filter particulates when present in the air
b. Self-decontamination, including removing and bagging contaminated clothing, showering, and decontaminating surfaces of critical areas and objects
c. Staying indoors to the extent that it is practical, e.g., only conducting outdoor tasks when necessary (e.g., seeking medical attention, buying foodstuff and necessities).

Advice regarding these measures shall be issued through the emergency bulletin process (See Section 8.7.2.2) as well traditional public communications methods.

All technical assessments shall be evaluated in the context of operational factors and public policy (e.g., psychosocial and economic) considerations. Such considerations shall provide an overall assessment of the risks and costs associated with various protective measures to provide for decision-making that optimizes public health, safety, and welfare.

The Early Response Phase starts to overlap with the Transition Phase once there is no potential for further releases and the need for early protective actions diminishes. The Early Response Phase ends at the moment the Recovery Phase begins.

Population monitoring and medical management would continue during the Early Response Phase of the response (See Annex S).

N.3 TRANSITION PHASE

The Transition Phase can begin when the radiological environment has been characterized to the point where the risk to recovery workers is fully understood and manageable. The transition to recovery in these areas may proceed independent of ongoing Early Response Phase activities, including those in other areas.

The imposition and rescinding of protective measures, as well as the distinction between the first three phases, may not be uniform across the affected area as contamination levels may differ. Relevant partners and affected communities should be consulted and engaged early and often when developing and implementing Recovery Phase long-term protective action strategies.

The practical dose reduction measures detailed in the Early Response Phase are also applicable to the Transition Phase.

During both the Transition Phase and the Recovery Phase, recovery operations will be controlled and coordinated through a recovery management organization (RMO). Establishing the RMO is not a mandatory action when the PEOC is activated but should be considered at the earliest practical opportunity. As the nuclear emergency stabilizes, recovery operations will require planning for the transfer of operational control to the RMO.

N.4 RECOVERY PHASE

Recovery is the final stage in responding to a nuclear emergency. The goal during the Recovery Phase is to return Ontario to a state of normalcy as quickly and effectively as possible, to the extent practically achievable. Recovery operations may begin before to the Transition Phase is complete. At this time, oversight of operations transfers from the PEOC Commander to the RMO Commander.

Recovery operations will be managed by the RMO (Figure N – 1). Recovery Phase decision-making should focus on adjusting or rescinding of protective measures imposed during the previous phases with the goal of reducing environmental radiation to appropriate levels established in consultation with relevant partners and affected communities, in order to improve living conditions and resume social and economic activities.

Recovery Phase decision-making should be based on ongoing environmental and assurance monitoring results and continuous re-assessment of the radiological situation in order to optimize long-term protective actions and manage the long-term radiological exposure situation.

The Recovery Phase is detailed in the Provincial Nuclear Recovery Operations Plan.

This diagram is for visual reference only. See the text below this figure for the full description.

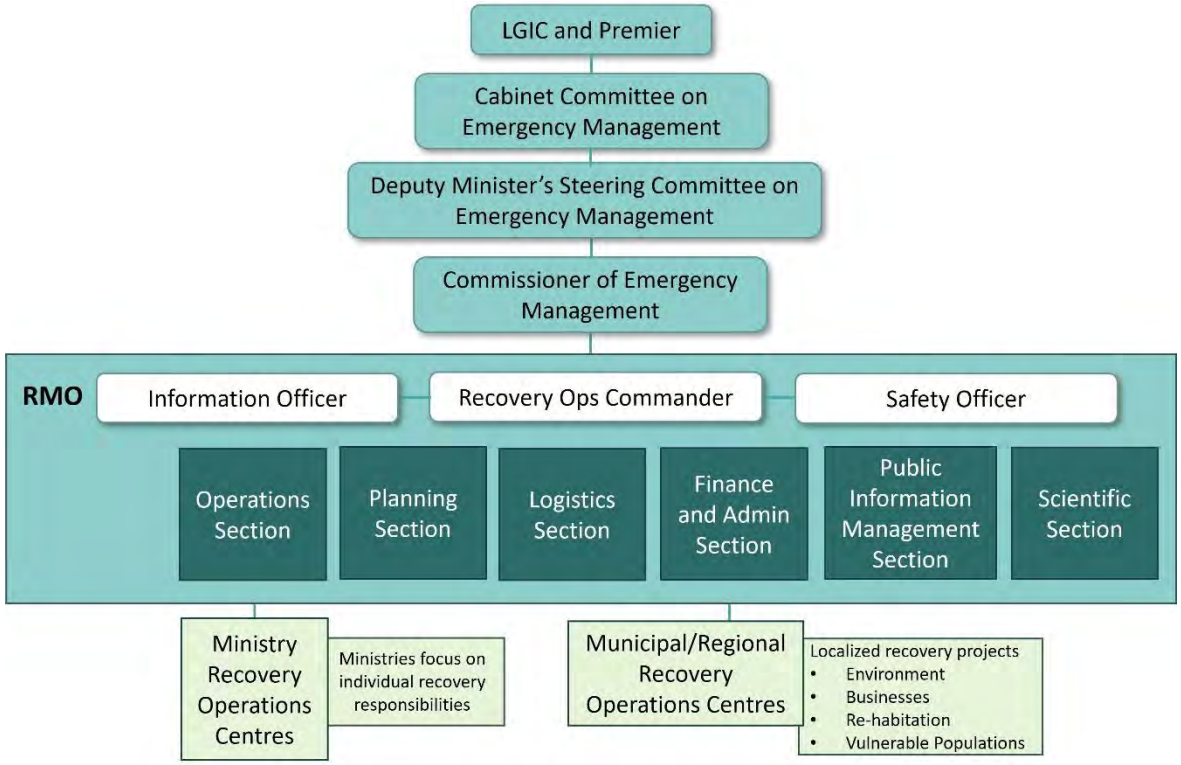

Figure N – 1: RMO Full Activation Organization

The diagram illustrates the Recovery Management Organization (RMO) structure during full activation. The Lieutenant Governor in Council and the Premier have the highest authority, followed by the Cabinet Committee on Emergency Management. Directly beneath them is the Deputy Minister's Steering Committee on Emergency Management, which connects to the Commissioner of Emergency Management. The Commissioner links to the RMO, which includes an Information Officer, Recovery Ops Commander, and a Safety Officer. Below this group are six functional sections: Operations, Planning, Logistics, Finance and Administration, Public Information Management, and Scientific. The Recovery Management Organization (RMO) links to Ministry Recovery Operations Centres and Municipal/ Regional Recovery Operations Centres. Ministry Recovery Operations Centres are ministries that focus on individual recovery responsibilities,

while Municipal and Regional Recovery Operations Centres oversee localized recovery projects.

ANNEX O. DESIGNATED/HOST MUNICIPALITIES

O.1 REACTOR FACILITIES AND THEIR DESIGNATED/HOST MUNICIPALITIES

Pursuant to EMCPA 3(4), the following municipalities (Table O – 1) are hereby designated as municipalities that must address nuclear emergencies in their municipal emergency plans either for the purposes of protecting their citizens from the hazard or in the capacity of designated host municipality.

Table O – 1: Reactor Facilities and their Designated/Host Municipalities

Reactor Facilities	Designated Municipalities	Designated Host Municipalities
Pickering Nuclear Generating Station	Regional Municipality of Durham City of Toronto	City of Peterborough
Bruce Nuclear Generating Station	Municipality of Kincardine	Town of Saugeen Shores
Darlington Nuclear Generating Station	Regional Municipality of Durham	City of Toronto City of Peterborough
Fermi 2 Power Plant (Michigan, U.S.)	Town of Amherstburg	City of Windsor Town of Essex

O.2 RECEPTION CENTRES

Municipal emergency plans for nuclear emergencies shall identify the location of facilities that will be used for the reception, care, and initial shelter of evacuees.

Evacuee monitoring and decontamination may be accomplished either in a Reception Centre that receives evacuees immediately upon leaving the emergency area or in a facility set up separately from the reception centre.

A Reception Centre is generally the first destination for evacuees. It is organized to perform many of the following functions:

a. registration and inquiry
b. allocation to Evacuation Centre
c. first-aid
d. monitoring and decontamination

Host Municipalities shall resource Reception Centres for the first three functions listed above.

In the event of a nuclear emergency, reactor facilities in Ontario shall provide the equipment and trained staff to perform monitoring and decontamination activities (pursuant to federal licensing requirements to provide offsite assistance).

EMO shall co-ordinate with appropriate response partners to establish arrangements for monitoring and decontamination for the Town of Amherstburg.

Municipal emergency plans for nuclear emergencies shall identify the roles and functions fulfilled by emergency workers at Reception Centres and include provisions for the selection, staffing, and resourcing of these facilities.

O.3 EVACUATION CENTRES

Evacuation Centres are facilities set up by designated host municipalities to provide shelter, food, and other services (e.g., family reunification and emergency social services) to people who have been evacuated as a result of a nuclear emergency.

Municipal emergency plans for nuclear emergencies shall identify the location of facilities that will be used for Evacuation Centres.

Municipal emergency plans for nuclear emergencies shall identify the roles and functions fulfilled by emergency workers at Evacuation Centres and include provisions for the selection, staffing and resourcing of these facilities.

O.4 EMERGENCY WORKER CENTRES

Emergency Worker Centres are facilities set up to monitor, decontaminate and control exposure of emergency workers to radiation.

Emergency workers are those who perform emergency services in support of an emergency response, but do not include those designated as nuclear energy workers (NEW). Emergency workers include:

a. Those required to remain in, or to enter, areas affected or likely to be affected by radiation from a nuclear emergency, and for whom special safety arrangements are required
b. Those who are required to provide response outside the affected areas
c. Those who are registered with an authorized responding organization
d. Police, firefighters, paramedic services, emergency social services workers, other essential services.

Municipal emergency plans for nuclear emergencies shall identify the location of facilities that will be used as Emergency Worker Centres. These locations should also be able to accommodate a command post for environmental monitoring, surveillance,

and sampling operations of the Environmental Radiation and Assurance Monitoring Group (ERAMG) Field Teams.

Municipal emergency plans for nuclear emergencies shall identify the roles and functions fulfilled by emergency workers at Emergency Worker Centres and include provisions for the selection, staffing, and resourcing of these facilities.

In the event of a nuclear emergency, reactor facilities (except Fermi 2) shall provide the equipment and trained staff to perform monitoring and decontamination activities (pursuant to federal licensing requirements to provide offsite assistance).

O.5 MONITORING AND DECONTAMINATION

A Monitoring and Decontamination Unit (MDU) is a deployable asset with equipment to monitor for contamination on an evacuees' person or vehicle. If contamination is identified, a decontamination process is used to reduce the amount of contamination to safe levels.

Reactor facilities in Ontario are responsible for providing resources, qualified staff and their training in order to meet their responsibility for the monitoring and decontamination of evacuees and emergency workers (per Annex G.4).

Similarly, once the emergency functions have ceased to be necessary, the Ontario reactor facility is responsible for restoring the monitoring and decontamination portion of any facility used to its pre-emergency state.

In the event of a nuclear emergency at the Fermi 2 NPP, EMO shall co-ordinate with appropriate response partners, including nuclear facilities in Ontario, to establish arrangements for monitoring and decontamination of both evacuees at the reception centre, and emergency workers at the EWC, as outlined in the Fermi 2 IP for the Town of Amherstburg and Town of Essex.

ANNEX P. ACCIDENT CHARACTERIZATIONS

Categorization of a nuclear event is the responsibility of the NGS operator. Potential categorizations are detailed below. Upon receiving a categorization, the CEM or designate (e.g. PEOC Commander) will determine an appropriate provincial offsite response level (see Table 1) as well as having additional specific actions to take.

P.1 REPORTABLE EVENT

A reportable event is an event affecting the reactor facility which would be of concern to the offsite authorities responsible for public safety. Triggers for reportable events include:

a. Any event or condition that reduces the reactor facility's capability to mitigate an emergency onsite, and which persists for longer than the allowable time limits
b. Any event or condition that reduces the reactor facility's capability to provide the agreed offsite emergency support, and which is expected to persist for over 8 hours, or actually does so
c. Natural, toxic, flammable, destructive or other phenomena which have the potential to lead to a minor break (to be defined in reactor facility procedures) in the physical integrity of the nuclear heat transport system boundary or the moderator system
d. Unexpected or unplanned activation of the emergency cooling injection system or the containment system (including box-up)
e. Declaration of a Station Emergency, with no potential for offsite effects
f. Any credible publicly announced threat to, or attempted or actual breach of, the facility's security that threatens its safe operation.

P.2 ABNORMAL INCIDENT

An abnormal incident is defined as an abnormal occurrence at the reactor facility which may have a significant cause, and/or may lead to more serious consequences. Triggers for Abnormal Incidents include:

a. A minor break in the physical integrity of the nuclear heat transport system boundary with no fuel failures (actual or likely)
b. Natural, toxic, flammable, destructive or other phenomena which have the potential to lead to the major (to be defined in reactor facility procedures) break specified in Item a. under Onsite Emergency
c. Activation of the emergency coolant injection system or the containment system (including box-up) due to a process system upset which is not reportable under any other category
d. Declaration of a Station Emergency due to an occurrence which has the potential to result in offsite effects.

P.3 ONSITE EMERGENCY

An onsite emergency is characterized as a serious malfunction that results or may result in an atmospheric release of radioactive material or is likely to result in a release at a later time. Triggers for onsite emergencies include:

a. A major break (to be defined in reactor facility procedures) in the physical integrity of the nuclear heat transport system boundary, the moderator system, or the irradiated (or spent) fuel handling and storage system, with fuel failures but with a fully intact and functioning containment system
b. An abnormal emission (to be defined in reactor facility procedures) of radioactive material to the atmosphere from any non-reactor source
c. An event or condition which has the potential to lead to a General Emergency criterion, concurrent with the loss of the ability to detect or control such a development
d. Hostile action in the protected area resulting in actual or potential loss of control over station safety or safety related systems but excluding reactor control systems.

A notification with this categorization must state whether an emission is ongoing or, if not, give a best estimate of when it is expected to commence.

P.4 GENERAL EMERGENCY

A general emergency is characterized as an ongoing or imminent atmospheric release of radioactive material as a result of a more severe accident. Triggers for general emergencies include:

a. Damage to reactor fuel leading to the release of radioactivity from the fuel coincident with the failure, impairment, or bypass of containment, resulting in an atmospheric emission or, a reasonable expectation of an emission within the next 12 hours
b. Hostile action in the protected area resulting in actual or imminent loss of the ability to achieve and maintain the reactor in a cold shutdown state.

A notification with this categorization must state whether an emission is ongoing or, if not, give a best estimate of when it is expected to commence.

ANNEX Q. BASIS FOR PROTECTIVE ACTIONS

Q.1 GENERIC CRITERIA

GC are used in the early stages of an emergency, prior to the availability of actual radiation monitoring data, when technical staff in the PEOC Scientific Section undertake dose projection modelling. GC values are calculated to reduce the risk of stochastic effects and are expressed in terms of equivalent dose (tissue or organ) and effective dose (whole body).

GC are compared against the dose projections modelled in order to determine the need for implementation of protective measures for the purposes of:

a. exposure control
b. ingestion control
c. population monitoring and medical management
d. trade control of foodstuff and other commodities
e. protection of emergency workers.

The GC used in this PNERP are implemented based on Health Canada's Generic Criteria and Operational Intervention Levels for Nuclear Emergency Planning and Response.

Table Q – 1: Generic Criteria for Exposure Control Measures

Projected Dose	Protective Measure
50 mSv (5 rem) in the first 7 days (H_T)	Iodine thyroid blocking
10 mSv (1 rem) of averted dose in the first 2 days (E)	Sheltering
100 mSv (10 rem) in the first 7 days (E or H_{fetus})	Evacuation
100 mSv (10 rem) in the first year* (E) Or 100 mSv in the full period of in utero development (H_{fetus})	Temporary relocation

Table Q – 2: Generic Criteria for Ingestion Control Measures

Projected Dose	Protective Measure
3 mSv/year (0.3 rem/year) (1 mSv/year (0.1 rem/year) for each of the following categories: drinking water, milk and other foods and beverages) (E)	Restriction of distribution and ingestion of potentially contaminated: • Drinking water • Milk • Other foodstuffs and beverages

* This is an initial value. As the response progresses, this generic criterion should be reduced.

Table Q – 3: Generic Criteria for Population Monitoring and Medical Management

Projected Dose	Protective Measure
• 100 mSv (10 rem) in one month (E) • 100 mSv (10 rem) for the full period of in utero development (H_{fetus})	• Population monitoring • Internal assessment • Medical follow-up (health screening)

Table Q – 4: Generic Criteria for Protecting Offsite Emergency Workers

Projected Dose	Protective Actions
50 mSv[†‡] over the duration of the response ($H_p(10)$ or E)[§]	Restriction of activities for individual workers

[†] This generic criterion is not appropriate for female workers who may be pregnant. These individuals should be excluded from emergency duties that would result in a dose greater than the relevant permissible dose for non-emergency situations.

[‡] While the employer is responsible for ensuring worker safety at all times, offsite emergency workers should be fully aware of the potential health risks associated with the dose they may receive, and they should accept these risks.

[§] $H_p(10)$ = personal dose equivalent at 10 mm

Q.2 OPERATIONAL INTERVENTION LEVELS

Operational Intervention Levels (OILs) are used once radiation monitoring data is available to determine the need to implement or adjust protective measures. OILs are classified as follows:

a. Plume exposure OILs (1γ and 2γ) are used to identify areas (beyond those for response actions that have been taken during the Urgent Response Phase) where the ground deposition of radioactive material warrants protection of the public frequenting or living in the area.
b. Skin monitoring OIL 4 γ is used to identify individuals with enough radioactive material on the skin to warrant response actions (such as decontamination). Only the public being evacuated or relocated is expected to possibly have sufficient radioactive material on the skin to warrant response actions.
c. Ingestion control OIL 3γ is used to identify in which areas food, milk, and drinking water may have become contaminated. Where OIL 3γ is exceeded, restrictions should be put in place until more detailed assessments can be completed.
d. The alpha/beta screening OILs 5α and 5β are used to confirm and adjust initial restrictions on food, milk and drinking water restrictions that were enacted based on OIL 3γ. If any of these OILs are exceeded, a radionuclide-specific assessment must be undertaken.
e. Radionuclide-specific OILs for ingestion control based on laboratory measurements are provided in Annex R. If the concentration of radionuclides in food, drinking water, or milk are exceeded, or if the summation condition is exceeded, then that item is not suitable for consumption and shall be disposed of.

   Summation condition: $\sum_i \left(\frac{C_i}{OIL_i}\right) \geq 1$

   Where,

   C_i is the concentration of radionuclide in the drinking water, milk or other foods or beverages (Bq/kg or Bq/L)
   OIL_i is the radionuclide specific value of OIL for radionuclide in drinking water, milk or other foods or beverages (Bq/kg or Bq/L).

f. The OILs have been derived for use in nuclear emergencies.
g. The OILs used in this PNERP are adopted from Health Canada's Generic Criteria and Operational Intervention Levels for Nuclear Emergency Planning and Response.

OILs are values that support decision-making post-release by quickly relating discrete measurements of contamination to GC, thereby identifying the need for, or confirming the adequacy of, protective actions.

The OILs presented in this section should ensure that doses do not exceed the GC in Annex Q.1. Measurements should be made in accordance with the details and

timeframes provided below and using properly calibrated equipment that is fit for purpose.

Table Q – 5: Operational Intervention Levels for External Exposure Control

OIL #	Protective Action	Measurement Details	Level	Timeframe for Action, Relative to Release
1_{γ}	Evacuation	Gamma dose rate (H*(10)), 1 m from the ground	1000 µSv/h (100 mrem/h)	Within 24h of the beginning of exposure
2_{γ}	Temporary relocation	Gamma dose rate (H*(10)), 1 m from the ground, measured within 10 days of reactor shutdown	100 µSv/h (10 mrem/h)	Initiate after evacuation
		Gamma dose rate (H*(10)), 1 m from the ground, measured more than 10 days after reactor shutdown	25 µSv/h (2.5 mrem/h)	

Table Q – 6: Operational Intervention Levels for Ingestion Control

OIL #	Protective Action	Measurement Details	Level	Timeframe for Action, Relative to Release
3_{γ}	Restriction of distribution and ingestion of potentially contaminated drinking water, milk and other food	Gamma dose rate (H*(10)) at 1 m from the ground	1 µSv/h (0.1 mrem/h)	Implement with Exposure Control and extend within days
5_{α}	Confirm ingestion controls (with lab measurements)	Gross alpha activity	See Annex Q	Initiate within a week to a month, depending on importance of local food and drinking water to the community
$5_{ß}$		Gross beta activity	See Annex Q	
6		Activity concentrations for specific radionuclides	See Table Q – 9	

Table Q – 7: Operational Intervention Levels for Population Monitoring and Medical Management

OIL #	Protective Action	Measurement Details	Level	Timeframe for Action, Relative to Release

4_{γ}	Personal decontamination and/or medical follow-up	Skin measurement at 10 cm from the hands and the face	1 µSv/h (0.1 mre m/h)	Implement concurrently with Exposure Control

Table Q – 8: Ingestion Control OILs for Alpha/Beta Screening

OIL	Action	Measurement details	Drinking water (mSv/y)	Milk (mSv/y)	Other foods and beverages (mSv/y)
5_{α}	Restrict distribution and ingestion pending radionuclide-specific analysis	Gross alpha (Bq/L)	1	1	3
$5_{ß}$	Restrict distribution and ingestion pending radionuclide-specific analysis	Gross beta (Bq/L)	10	30	30

Table Q – 9: Radionuclide-Specific OILS

Radionuclide	Symbol	Drinking Water (Bq/L)	Milk (Bq/L)	Other Foods and Beverages (Bq/kg)
Tritium (Hydrogen-3)	^{3}H	100000	30000	100000
Carbon-14	^{14}C	3000	3000	10000
Sulphur-35 (inorganic sulphur)	^{35}S	3000	3000	30000
Chromium-51	^{51}Cr	10000	30000	100000
Iron-55	^{55}Fe	1000	3000	10000
Iron-59	^{59}Fe	100	100	1000
Cobalt-58	^{58}Co	1000	1000	3000
Cobalt-60	^{60}Co	100	100	1000
Zinc-65	^{65}Zn	100	100	1000
Rubidium-86	^{86}Rb	100	300	1000
Strontium-89	^{89}Sr	300	300	1000
Strontium-90	^{90}Sr	30	30	100
Strontium-91	^{91}Sr	10000	10000	30000
Strontium-92	^{92}Sr	10000	10000	30000
Yttrium-90	^{90}Y	3000	1000	10000
Yttrium-91	^{91}Y	300	300	1000
Yttrium-92	^{92}Y	10000	10000	30000
Yttrium-93	^{93}Y	3000	3000	10000
Niobium-95	^{95}Nb	1000	1000	3000
Zirconium-95	^{95}Zr	1000	1000	3000
Zirconium-97	^{97}Zr	1000	1000	3000

Radionuclide	Symbol	Drinking Water (Bq/L)	Milk (Bq/L)	Other Foods and Beverages (Bq/kg)
Molybdenum-99	^{99}Mo	3000	10000	30000
Technetium-99	^{99}Tc	100	100	300
Technetium-99m	^{99m}Tc	1000	100000	300000
Ruthenium-103	^{103}Ru	1000	1000	3000
Ruthenium-105	^{105}Ru	1000	10000	100000
Ruthenium-106	^{106}Ru	100	100	300
Rhodium-105	^{105}Rh	3000	3000	30000
Silver-110m	^{110m}Ag	300	300	1000
Antimony-127	^{127}Sb	300	300	3000
Antimony-129	^{129}Sb	1000	10000	30000
Tellurium-127	^{127}Te	3000	30000	100000
Tellurium-127m	^{127m}Te	1000	1000	10000
Tellurium-129	^{129}Te	10000	100000	300000
Tellurium-129m	^{129m}Te	100	100	1000
Tellurium-131m	^{131m}Te	1000	3000	10000
Tellurium-132	^{132}Te	1000	1000	3000
Iodine-129	^{129}I	10	30	30
Iodine-131	^{131}I	100	100	300
Iodine-132	^{132}I	3000	10000	30000
Iodine-133	^{133}I	1000	1000	3000
Iodine-134	^{134}I	300	30000	100000
Iodine-135	^{135}I	1000	1000	3000
Cesium-134	^{134}Cs	100	300	1000
Cesium-136	^{136}Cs	300	300	3000
Cesium-137	^{137}Cs	100	300	1000
Barium-139	^{139}Ba	10000	30000	100000
Barium-140	^{140}Ba	300	300	3000
Lanthanum-140	^{140}La	300	300	3000
Lanthanum-141	^{141}La	1000	10000	30000
Lanthanum-142	^{142}La	1000	30000	100000
Cerium-141	^{141}Ce	1000	1000	3000
Cerium-143	^{143}Ce	100	3000	10000
Cerium-144	^{144}Ce	100	100	300
Praseodymium-143	^{143}Pr	300	300	3000
Neodymium-147	^{147}Nd	300	300	3000
Iridium-192	^{192}Ir	300	300	3000
Uranium-235	^{235}U	30	30	100
Neptunium-237	^{237}Np	3	3	10
Neptunium-239	^{239}Np	3000	3000	30000
Plutonium-238	^{238}Pu	1	1	3
Plutonium-239	^{239}Pu	1	1	3

Radionuclide	Symbol	Drinking Water (Bq/L)	Milk (Bq/L)	Other Foods and Beverages (Bq/kg)
Plutonium-240	^{240}Pu	1	1	3
Plutonium-241	^{241}Pu	100	100	300
Plutonium-242	^{242}Pu	1	1	3
Plutonium-244	^{244}Pu	1	1	3
Americium-241	^{241}Am	1	1	10
Curium-242	^{242}Cm	30	30	100
Curium-244	^{244}Cm	1	1	10

ANNEX R. RADIATION SAFETY AND EXPOSURE CONTROL STRATEGY

R.1 RADIATION PROTECTION CONSIDERATIONS

R.1.1 General

Dose control procedures for ERAMG field teams are detailed in the ERAMG Response Plan, and the procedures therein should be considered when developing dose control procedures for other emergency workers.

Equipment must be available to measure and/or determine the dose received by field staff including (as required):

a. Passive Dosimeter (e.g. TLD, OSLD) to all emergency workers
b. Direct Reading Dosimeters (DRD) (e.g., electronic personal dosimeter (EPD)) for all emergency workers or a representative individual in all groups at risk
c. radiation survey meters
d. air sampling equipment and contamination meters.

R.1.2 Exposure Control

Notwithstanding the permissible doses given (See Table R – 1), due diligence shall be exercised to maintain individual doses ALARA.

Table R – 1: Actions vs. Permissible Dose for Emergency Workers in the Urgent Response, Early Response, and Transition Phases

Action	Effective dose (mSv)	Equivalent dose to the skin (mSv)
Permissible dose for participation in control of emergency	50	500
Permissible dose for actions taken to minimize dose consequences, for members of the public, associated with the release of radioactive material.	100	1 000
Permissible dose for actions taken to prevent health effects of radiation that are fatal or life-threatening, or that result in permanent injury.	500	5 000
Permissible dose for actions taken to prevent the development of conditions that could significantly affect people and the environment.	500	5 000

All personnel participating as emergency workers shall be provided training to perform required responsibilities during an actual emergency and informed of:

a. The possible radiation hazards to which they might be exposed
b. The applicable rules, regulations, orders, directives, plans and procedures
c. The equipment, clothing, methods and measures required to reduce radiation doses
d. Their responsibilities, obligations and rights including the individual's responsibilities for personal and collective protection
e. Their expected radiation dose levels
f. The specified permissible dose and of the dose management control levels
g. The potential effects of radiation during pregnancy, the rights and obligations of a pregnant woman
h. That the designation shall only be effective during the nuclear emergency.

R.1.3 Turn-back Dose Rate

Turn-back dose rates are associated with Sector Safety Status, found in Table R – 2, and form the basis for worker safety decision making at the EWC. If the values associated with the dose rate are exceeded, the member must leave the area when safe to do so, or move away from the radiation source until the dose rate has dropped below the dose rate set by the EWC. In order to use turn-back dose rates, personnel must have:

a. A handheld dose rate meter may be carried by a representative individual in a group, which is to be closely monitored at all times when in an area of risk from radiation exposure; and/or
b. A DRD worn when at risk of radiation exposure. The DRD will be configured with both a dose and dose rate alarm and may be used to indicate excessive radiation levels independent of the handheld meter or in tandem with the handheld meter.

The DRD measures both the accumulated dose and the dose rate the individual is receiving from the surrounding environment. However, the DRD and the dose rate meter measure the incident rate of radiation; thus, the specified limit applies only to situations where the individual is wearing suitable clothing and respiratory protection.

Table R – 2: Guidelines for Assigning Sector Safety and Stay Times

Sector Status	Criteria
Red	> 1 rem/h (10 mSv/h) Recommended Stay Time – **1 hour**
Orange	25 mrem/h (250 µSv/h) to < 1 rem/h (10 mSv/h) Recommended Stay Time – **4 hours**
Yellow	0.1 mrem/h (1 µSv/h) to < 25 mrem/h (250 µSv/h) Recommended Stay Time – **4 hours**
Green	< 0.1 mrem/h (1 µSv/h) Recommended Stay Time – **No Limit**

Table R – 3 details some of the potential actions to be taken in the field after a release. Additional precautions are found in Table R – 4.

R.1.4 Personal Protective Equipment

PPE will be supplied to members who are tasked with conducting survey and sampling operation in any sector that may be contaminated following a nuclear release. Specifics on the type, quantities, and capabilities of specific PPE are available in the Provincial Nuclear Facilities, Equipment and Maintenance Manual. In general, PPE shall provide an outer layer to protect the skin and respiratory protection. The PPE shall be sealable at the wrists, ankles, waist, neck, and around the face. Tape is acceptable to create the seal. More detailed PPE requirements are found in Table R – 5.

R.2 SAFETY SECTOR STATUS TABLES

Table R – 3: Precautions for Each Sector Safety Status

Safety Status	Precautionary Measures for Emergency Workers
Green	No precautions necessary. No limit on stay period.
Yellow	a) Report to the Emergency Worker Centre (EWC) before entering the sector. b) Carry personal monitoring devices and observe all precautions prescribed by the EWC. c) If duties permit, remain under shelter or inside a vehicle. d) If working outside, wear appropriate personal protective equipment (PPE) as prescribed by the EWC. e) Stay in the sector shall be limited to 4 hours, or the time prescribed by the EWC. f) Report again to the EWC on leaving the sector.
Orange	a) Pregnant emergency workers shall not enter the sector. b) Report to the Emergency Worker Centre (EWC) before entering the sector. c) Carry personal monitoring devices and observe all precautions prescribed by the EWC. d) Dosimeters should be checked every hour. Exit from the sector if the reading reaches 40 mSv (4rem), or any lower personal limit prescribed by the EWC. e) If duties permit, remain under shelter or inside a vehicle. f) If working outside, wear appropriate personal protective equipment (PPE) as prescribed by the EWC. g) Stay in the sector shall be limited to 4 hours, or the time prescribed by the EWC. h) Report again to the EWC on leaving the sector.
Red	a) Pregnant emergency workers shall not enter the sector. b) Report to the Emergency Worker Centre (EWC) before entering the sector. c) Enter the sector accompanied by a qualified escort provided by the reactor facility and carry alarming personal dosimeters. Observe any precautions prescribed by the EWC. d) Dosimeters should be checked every 30 minutes. Exit from the sector if the reading reaches 40 mSv (4 rem), or any lower personal limit prescribed by the EWC. e) If duties permit, remain under shelter or inside a vehicle. If working outside, wear an outer garment such as a plastic raincoat or any other personal protective equipment (PPE) prescribed by the EWC. f) Stay in the sector shall be limited to 1 hour, or the time prescribed by the EWC. g) Report again to the EWC on leaving the sector.

Table R – 4: Recommended Reference Levels

Band (acute or per year)	Type of Situation
20-100 mSv	Emergency situations, where events with uncertain consequences require urgent protective actions such a sheltering and evacuation to minimize the impacts of possible radiation exposures.
1–20 mSv	Existing situations, where radioactivity is already present in the environment at the time actions are taken to reduce radiation exposures. If doses are optimized below this reference level it is safe to live in the contaminated area.

Note: From https://www.cnsc-ccsn.gc.ca/eng/resources/fact-sheets/reference-levels/

Table R – 5: Recommended PPE Based on Sector Safety Status

Sector safety status	Recommended PPE (to be determined in the moment)
RED	• Tyvek coveralls • Full-face piece air purifying respirator with a P-100 or high-efficiency particulate air (HEPA) filter • Rubber boot covers • Protective gloves • Electronic personal dosimeter (EPD) • Passive dosimeter • Hard hat, if required • Safety glasses, if required • Any other items as defined by the EWC supervisor
ORANGE	• Tyvek coveralls • N-100 mask **or** full-face piece air purifying respirator • Rubber boot covers • Protective gloves • Electronic personal dosimeter (EPD) • Passive dosimeter • Hard hat, if required • Safety glasses, if required
YELLOW	• N-100 mask • Disposable (paper) boot covers • Electronic personal dosimeter (EPD) • Passive dosimeter • Hard hat, if required • Safety glasses, if required • Protective gloves, if required
GREEN	• Passive dosimeter • Hard hat, if required • N95, if required • Safety glasses or face shield, if required • Protective gloves, if required

Table R – 6: Conversion Tables for Radiological Units

From Curie to Becquerel	From Becquerel to Curie
1 kilocurie (kCi) ≈ 37 terabecquerel (TBq)	1 terabecquerel (TBq) ≈ 27 curie (Ci)
1 curie (Ci) ≈ 37 gigabecquerel (GBq)	1 gigabecquerel (GBq) ≈ 27 millicurie (mCi)
1 millicurie (mCi) ≈ 37 megabecquerel (MBq)	1 megabecquerel (MBq) ≈ 27 microcurie (□Ci)
1 microcurie (μCi) ≈ 37 kilobecquerel (kBq)	1 kilobecquerel (kBq) ≈ 27 nanocurie (nCi)
1 nanocurie (nCi) ≈ 37 becquerel (Bq)	1 becquerel (Bq) ≈ 27 picocurie (pCi)
1 picocurie (pCi) ≈ 37 millibecquerel (mBq)	
From Rem to Sievert	**From Sievert to Rem**
1 kilorem (krem) = 10 sievert (Sv)	1 sievert (Sv) = 100 rem (rem)
1 rem (rem) = 10 millisievert (mSv)	1 millisievert (mSv) = 100 millirem (mrem)
1 millirem (mrem) = 10 microsievert (μSv)	1 microsievert (μSv) = 100 microrem (μrem)
1 microrem (μrem) = 10 nanosievert (nSv)	1 nanosievert (nSv) = 100 nanorem (nrem)

Table R – 7: Unit Prefixes

Large Prefixes	Small Prefixes
Tera (T) = x 10^{12}	Pico (p) = x 10^{-12}
Giga (G) = x 10^{9}	Nano (n) = x 10^{-9}
Mega (M) = x 10^{6}	Micro (μ) = x 10^{-6}
Kilo (k) = x 10^{3}	Milli (m) = x 10^{-3}

ANNEX S. POPULATION MONITORING

S.1 GENERAL

Population monitoring and medical management shall be required in the event of a nuclear release (such monitoring may be conducted in the absence of a release to provide assurances to the public).

The MOH outlines the methods by which population monitoring, decontamination, and medical management functions may be provided:

a. at a hospital when there are contaminated casualties
b. at personally determined evacuation destinations by undertaking self-decontamination
c. at a Monitoring and Decontamination Unit (MDU)

For nuclear emergencies, an Operational Intervention Level (OIL) based on the level of skin contamination for population monitoring and medical management is provided in Annex Q.2. Personal decontamination and medical follow-up shall be conducted for any person with skin contamination levels exceeding this value.

S.2 RESPONSIBILITY

The CEM or designate (e.g. PEOC Commander) shall, in consultation with the MOH and affected designated municipalities, consider the need for population monitoring, decontamination, and medical management to direct implementation of these measures as appropriate.

Population monitoring, decontamination, and medical management should be implemented according to the provisions of MOH's Radiation Health Response Plan, as appropriate, and by activating the monitoring and decontamination facilities administered by the designated municipalities and resourced by the reactor facilities.

The need for population monitoring, decontamination and medical management shall be communicated to the public via EBs and press releases issued by the CEM or designate (e.g. PEOC Commander) (Section 8.7).

Designated municipalities shall include provisions in their municipal plans for the establishment of Reception Centres which have Monitoring and Decontamination Units (MDUs) co-located. Ontario Reactor facilities shall resource and operate mobile MDUs and those co-located with Reception Centres according to applicable provisions in their emergency plans and procedures.

Medical personnel trained in the clinical management of radiation injuries may be designated in the preparedness phase, to the extent possible.

In accordance with the Radiation Health Response Plan (RHRP), the MOH shall coordinate the following, in order to assess long-term health impacts, if required:

a. A medical registry of affected people (public and workers).
b. Medical follow-up of the population affected.
c. Guidance for personal monitoring is provided in this Annex, as well as in the MOH RHRP.

S.3 CONTAMINATED DECEDENTS

There are situations in a nuclear or radiological emergency that could result in the need to manage contaminated decedents. While even a severe nuclear accident is not expected to cause offsite fatalities, an event simultaneous to a nuclear accident could have fatal consequences, as evidenced by the release of radioactive material from the Fukushima Daiichi Nuclear Power Plant. The nuclear accident itself resulted from the devastation caused by the 2011 earthquake and tsunami; the death of thousands of individuals was attributed to the natural disaster and not to the radioactive release. However, many bodies from the natural disaster still became contaminated after the fact due to the atmospheric releases.

As a result of the contamination, such decedents must be handled differently than non-radiologically contaminated decedents and consequently, special measures would be enacted.

ANNEX T. RECEPTION AND CARE OF EVACUEES

T.1 GENERAL

Reception Centres shall be set up to provide the following functions:

a. registration and inquiry
b. accommodation allocations at evacuation centres
c. first aid
d. monitoring and decontamination (co-location optional)

Efforts should be made not to separate households during evacuation (e.g., keep parents, children, elders together) to the extent practicable, as this maintains existing social support structures and reduces the burden placed on evacuation and reception centre staff.

Reception and Evacuation Centres may be operated separately or co-located.

T.2 RESPONSIBILITY

In a nuclear emergency, emergency social service needs (e.g., reception, registration and inquiry, shelter, food, clothing and personal services) for members of the public who have been evacuated from their homes, shall be assessed and provided by:

a. Designated/host municipalities identified under this plan;
b. Appropriate Ontario ministries (e.g., MMAH); or
c. Municipalities directed to provide support or assistance under a provincial emergency declaration (Section 8.6.3).

In a nuclear emergency, monitoring and decontamination facilities associated with Reception Centres shall be resourced and operated by reactor facilities (with the exception detailed in the Fermi 2 Implementing Plan).

In a radiological emergency, emergency social service needs (e.g., reception, registration and inquiry, shelter, food, clothing and personal services) for members of the public who have been evacuated from their homes:

a. May be assessed and provided under the provisions of existing municipal mutual assistance agreements; or
b. Shall be provided by municipalities ordered to provide support or assistance under a provincial emergency declaration (Section 8.6.3).

ANNEX U. PROTECTION AND CARE OF ANIMALS

U.1 GENERAL

Pursuant to EMCPA Section 7.0.2., provincial emergency orders can include evacuation of animals under a declared provincial emergency.

Any emergency that affects humans may affect their animals whether these are raised for foodstuff production, kept as companion or service animals, or for other purposes, such as in zoos.

U.2 RESPONSIBILITY

Provisions for the protection and care of animals should be included in public awareness and education materials in advance of a nuclear emergency.

Municipal emergency response plans for nuclear emergencies shall make provisions for the protection and care of all animals, including those left behind during an evacuation.

Designated municipalities should consult with the following for assistance in developing plans for the protection and care of animals:

a. OMAFA (provincial lead on farm animal disease)
b. the Ministry of Natural Resources (MNR) for issues pertaining to wildlife
c. Ontario Ministry of the Solicitor General (Mandate for Animal Welfare in Ontario)

During a nuclear or radiological emergency, the PEOC should coordinate assistance for the protection and care of animals with the following, as appropriate:

a. Ontario Ministry of the Solicitor General (mandate for Animal Welfare in Ontario)
b. Ontario Society for the Prevention of Cruelty to Animals (OSPCA)
c. OMAFA (provincial lead on farm animal disease)
d. MNR (for issues pertaining to wildlife).

ANNEX V. DISTRIBUTION LIST

V.1 NEMCC MEMBERSHIP

V.1.1 Federal Departments

a. Canadian Nuclear Safety Commission (CNSC)
b. Health Canada (HC)
c. Public Safety Canada (PS)
d. Transport Canada (TC)
e. Canadian Food Inspection Agency (CFIA)
f. Canadian Coast Guard (CCG)
g. Department of National Defence (DND) / Joint Task Force Central (JTFC)

V.1.2 Reactor Facilities

a. Ontario Power Generation (OPG)
b. Bruce Power Inc.
c. Chalk River - Canadian Nuclear Laboratories Limited (CNL)
d. Fermi 2 - DTE Electric

V.1.3 Designated / Host Municipalities

a. Regional Municipality of Durham
b. City of Toronto
c. City of Peterborough
d. Municipality of Kincardine
e. Town of Saugeen Shores
f. Town of Amherstburg
g. City of Windsor
h. Town of Essex

V.1.4 Provincial Ministries

a. Ministry of Agriculture, Food and Rural Agribusiness (OMAFA)
b. Ministry of the Attorney General
c. Ministry of Children, Community and Social Services (MCCSS)
d. Ministry of Energy and Mines (ENERGY)
e. Ministry of Environment, Conservation and Parks (MECP)
f. Ministry of Health (MOH)
g. Ministry of Labour, Immigration, Training and Skills Development (MLITSD)
    i. Radiation Protection Services, MLITSD
h. Ministry of Long-Term Care (MLTC)
i. Ministry of Municipal Affairs and Housing (MMAH)
j. Ministry of Natural Resources (MNR)

k. Ministry of Northern Economic Development and Growth (MND)
l. Ministry of the Solicitor General (SOLGEN)
m. Ministry of Transportation (MTO)
n. Ministry of Emergency Preparedness and Response (MEPR)

    i. Communications Branch
    ii. Legal Services Branch

V.2 OTHER INTERESTED PARTIES

V.2.1 Provincial

a. New Brunswick Emergency Measures Organization
b. Québec Ministry of Public Security (Ministère de la Sécurité publique du Québec)
c. Québec Ministry of Health and Social Services (Ministère de la Santé et des Services sociaux)

V.2.2 International

a. Michigan State Police, Emergency Management and Homeland Security Division
b. Michigan State Police – REP Program, Emergency Management and Homeland Security Division
c. New York State Department of Homeland Security and Emergency Services-REP Program
d. Ohio Emergency Management Agency

V.3 FIRST NATION AND INDIGENOUS PARTNERS

a. Algonquins of Ontario
b. Algonquins of Pikwakanagan
c. Alderville First Nation
d. Caldwell First Nation
e. Chippewas of Nawash First Nation
f. Curve Lake First Nation
g. Delaware Nation at Moraviantown
h. Hiawatha First Nation
i. Mississauga First Nation
j. Mississauga's of the Credit First Nation
k. Mississauga's of Scugog Island First Nation
l. Mohawks of the Bay of Quinte
m. Saugeen First Nation
n. Six Nations of the Grand River
o. Union of Indians, Anishinabek
p. Walpole Island First Nation

V.4 PNERP DARLINGTON NUCLEAR GENERATING STATION IMPLEMENTING PLAN

V.4.1 Ingestion Planning Zone Municipalities

a. Township of Brock
b. Town of Ajax
c. Municipality of Clarington
d. City of Kawartha Lakes
e. City of Oshawa
f. City of Pickering
g. Township of Scugog
h. Township of Uxbridge
i. Town of Whitby
j. Town of Cobourg
k. County of Northumberland
l. Township of Hamilton
m. Municipality of Port Hope
n. Township of Cavan Monaghan
o. Township of Ottonabee-South Monaghan
p. City of Markham
q. Regional Municipality of York
r. County of Peterborough
s. Town of Whitchurch Stouffville

V.5 PNERP PICKERING NUCLEAR GENERATING STATION IMPLEMENTING PLAN

V.5.1 Ingestion Planning Zone Municipalities

a. Township of Brock
b. Town of Ajax
c. City of Kawartha Lakes
d. Municipality of Clarington
e. City of Oshawa
f. City of Pickering
g. Township of Scugog
h. Township of Uxbridge
i. Town of Whitby
j. City of Brampton
k. Regional Municipality of Peel
l. City of Mississauga
m. Town of Bradford West Gwillimbury
n. Town of Aurora
o. Regional Municipality of York

p. County of Simcoe
q. Town of East Gwillimbury
r. Township of King
s. City of Markham
t. Town of Newmarket
u. Town of Richmond Hill
v. City of Vaughan
w. Town of Whitchurch-Stouffville

V.6 PNERP BRUCE POWER NUCLEAR GENERATING STATION IMPLEMENTING PLAN

V.6.1 Ingestion Planning Zone Municipalities

a. Municipality of Brockton
b. Township of Huron-Kinloss
c. Municipality of Arran-Elderslie
d. Municipality of South Bruce
e. Town of South Bruce Peninsula
f. Township of Chatsworth
g. Township of Georgian Bluffs
h. Town of Hanover
i. Municipality of West Grey
j. Township of Ashfield-Colborne-Wawanosh
k. Municipality of Morris-Turnberry
l. Township of North Huron
m. County of Huron
n. County of Grey
o. County of Bruce

V.7 PNERP FERMI 2 NUCLEAR POWER PLANT IMPLEMENTING PLAN

V.7.1 Ingestion Planning Zone Municipalities

a. Town of Essex
b. Township of Pelee
c. Municipality of Chatham Kent
d. Town of Kingsville
e. Town of Lakeshore
f. Town of Lasalle
g. Municipality of Leamington
h. Town of Tecumseh

V.8 TRANSBORDER IPZ MUNICIPALITEIS WITHIN A 80 KM RADIUS

V.8.1 Reactor Facility Ginna

a. Township of Alnwick/Haldimand
b. Prince Edward County
c. County of Northumberland
d. Municipality of Brighton
e. Town of Cobourg
f. Township of Cramahe
g. Township of Hamilton
h. Municipality of Port Hope

V.8.2 Reactor Facility Fitzpatrick and Nine Mile Point

a. Township of Frontenac Islands
b. Town of Greater Napanee
c. County of Prince Edward County
d. County of Frontenac
e. City of Kingston
f. County of Lennox and Addington
g. Township of Loyalist
h. Municipality of Brighton
i. County of Northumberland

V.8.3 Reactor Facility Perry

a. Municipality of Bayham
b. Municipality of Central Elgin
c. County of Elgin
d. Municipality of Chatham Kent
e. Norfolk County
f. Municipality of Dutton\Dunwich
g. Township of Malahide
h. Municipality of West Elgin

V.9 PNERP OTHER RADIOLOGICAL EMERGENCIES IMPLEMENTING PLAN

a. CNSC Licensed Facilities in Ontario

ANNEX W. GLOSSARY

(For other references see the Ontario Emergency Management Glossary of Terms)

Abnormal Incident: An abnormal occurrence that may have a significant cause and/or may lead to more serious consequences. (Source: CNSC Glossary)

Accident: Any unintended event, including operating errors, equipment failures or other mishaps, the consequences, or potential consequences, of which are significant from the point of view of protection or safety. With respect to nuclear criticality safety, the term accidents or accident sequences means events or event sequences, including external events that lead to violation of the sub-criticality margin (that is, to exceeding the upper subcritical limit). (Source: CNSC Glossary)

Activation: Decisions and actions taken to implement a plan, a procedure or to open an emergency operations centre. (Source: Ontario Emergency Management Glossary of Terms)

After-Action Report (AAR): A report that documents the performance of tasks related to an emergency, exercise or planned event and, where necessary, makes recommendations for improvements. (Source: Ontario Emergency Management Glossary of Terms)

Alert (U.S. emergency classification): Events are in progress or have occurred that involve an actual or potentially substantial degradation of the safety of the plant. Any releases of radioactive material are expected to be limited to a small fraction of the U.S. Environmental Protection Agency (EPA) Protective Action Guides (PAG) exposure levels.

Alerting: Informing the affected population, by means of an appropriate signal, that a nuclear emergency has occurred or is about to occur.

As Low As Reasonably Achievable (ALARA): A principle of radiation protection that holds that exposures to radiation are kept as low as reasonably achievable, social and economic factors taken into account. (Source: CNSC Glossary)

Automatic Action Zone (AAZ): A pre-designated area immediately surrounding a reactor facility where pre- planned protective actions would be implemented on the basis of reactor facility conditions with the aim of preventing or reducing the occurrence of severe deterministic effects. (Source: Canadian Standards Association (CSA N1600, General requirements for nuclear emergency management programs)

Becquerel (Bq): The International System of Units (SI) unit of radioactivity. One becquerel (Bq) is the activity of a quantity of radioactive material in which one nucleus decays per second. In Canada, the Bq is used instead of the non-SI unit curie (Ci). (Source: CNSC Glossary)

Beyond Design Basis Accident (BDBA): An accident less frequent and potentially more severe, than a design-basis accident. Note**:** For a reactor facility, a beyond-design-basis accident may or may not involve fuel degradation. (Source: CNSC Glossary)

CANDU Reactor: A Canadian-invented pressurized heavy-water reactor that uses heavy water (deuterium oxide) for moderator and coolant and natural uranium for fuel. "CANDU" is short for CANada Deuterium Uranium. Also called CANDU. (Source: CNSC Glossary).

Cloudshine: *Gamma radiation* from *radioactive materials* in an airborne *plume*.

Communications: Advisories, directives, information, and messages that are transmitted.
(Source: Ontario Emergency Management Glossary of Terms)

Community: A generic term that includes both municipalities and First Nations.
(Source: Ontario Emergency Management Glossary of Terms)

Containment (System): A series of physical barriers that exist between radioactive materials contained in a reactor facility and the environment. Containment usually refers only to the reactor and vacuum buildings, and integral systems such as dousing.

Contamination: Contamination refers to nuclear or hazardous substances on surfaces, or within solids, liquids or gases (including the human body), where their presence is unintended or undesirable, or to the process giving rise to their presence in such places. (Source: CNSC Glossary)

Contingency Planning Zone (CPZ): A pre-designated area surrounding a reactor facility, beyond the Detailed Planning Zone, where contingency planning and arrangements are made in advance, so that during a nuclear emergency, protective actions can be extended beyond the Detailed Planning Zone as required to reduce potential for exposure. (Source: CSA N1600, General requirements for nuclear emergency management programs)

Crop Control: See Produce and Crop Control.

Declaration of Emergency: A signed declaration made in writing by the Head of Council or the Premier of Ontario in accordance with the Emergency Management and Civil Protection Act. This declaration is usually based on a situation or an impending situation that threatens public safety, public health, the environment, critical infrastructure, property, and/or economic stability and exceeds the scope of routine community emergency response.

Notes:

- Municipal Declaration of Emergency: a declaration of emergency made by the Head of Council or a municipality, based on established criteria.

Provincial Declaration of Emergency: a declaration of emergency made by the Lieutenant Governor in Council or the Premier of Ontario, based on established criteria. (Source: Provincial Glossary)

Decontamination: Reduction or removal of radioactive contamination from materials or persons.

Design Basis Accident (DBA): Accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits. (Source: CSA N1600 – General requirements for nuclear emergency management programs)

Designated Host Municipality: The municipality assigned responsibility in the Provincial Nuclear Emergency Response Plan for the reception and care of people evacuated from their homes in a nuclear emergency (for list see Annex A).

Designated Municipality: A municipality in the vicinity of a reactor facility which has been designated under the Emergency Management and Civil Protection Act, as one that shall have a nuclear emergency plan (for list see Annex A).

Detailed Planning Zone: A pre-designated area surrounding a reactor facility, incorporating the AAZ, where pre-planned protective actions are implemented as needed on the basis of reactor facility conditions, dose modelling, and environmental monitoring, with the aim of preventing or reducing the occurrence of stochastic effects. (Source: Modified from CSA N1600 – General requirements for nuclear emergency management programs)

Deterministic Effects: Radiation-induced health effects including changes to cells and tissues that are certain to occur in an individual exposed to a radiation dose greater than some threshold dose, with a severity that increases with increasing dose. Now referred to as tissue reactions. (Source: Health Canada Glossary)

Direct Reading Dosimeter: An active dosimeter that displays cumulative dose and dose rate while in use, and may have an alarm that sounds at a preset dose and/or dose rate

Dose: A measure of the radiation received or “absorbed” by a target. The quantities termed absorbed dose, organ dose, equivalent dose, effective dose, committed equivalent dose or committed effective dose are used, depending on the context. The modifying terms are often omitted when they are not necessary for defining the quantity of interest.

Dose Management: Includes administrative controls to limit doses, monitor doses and

record doses received by emergency workers while fulfilling their duties related to nuclear emergency response.

Dose Projection: The calculation of projected dose (see Projected Dose).

Dose Rate: The amount of radiation dose which an individual would receive in a unit of time. In the context of this Plan, the measurement units are multiples or submultiples of the Sievert (or rem) per hour.

Dosimeter: An instrument for measuring and registering total accumulated exposure to ionizing radiation.

Duty Team (PEOC): The PEOC Duty Team's composition is flexible, based on the nature of the emergency response scenario. For a nuclear or radiological emergency, the Duty Team consists of Operations Section staff, Scientific Section staff, reactor facility operator representative(s), PIMS staff, and others as (and if) the emergency dictates.

Effective Dose (E): A quantity calculated by multiplying the equivalent dose received by irradiated tissues, by a tissue specific weighting factor that reflects the risk of radiation-induced cancer to that tissue. The effective doses can then be summed to obtain the effective dose absorbed by the body.

Emergency: A situation or an impending situation that constitutes a danger of major proportions that could result in serious harm to persons or substantial damage to property and that is caused by the forces of nature, a disease or other health risk, an accident or an act whether intentional or otherwise (EMCPA).

Emergency Bulletin: Directions to the public on appropriate protective and other measures to be taken during a nuclear or radiological emergency, which are issued by the province and broadcast through the media.

Emergency Exposure Situation: Emergency exposure situations arise as a result of an accident, a malicious act or other unexpected event, and require prompt action in order to avoid or to reduce adverse consequences. (Guidance on Planning for Recovery Following a Nuclear or Radiological Emergency)

Emergency Information Centre (EIC): A designated facility that is properly equipped to monitor and co-ordinate emergency information activities including the dissemination of information to the public. (Source: Ontario Emergency Management Glossary of Terms)

Emergency Planning Zone: The area in which implementation of operational and protective actions are or might be required during a nuclear emergency, in order to protect public health, safety, and the environment.

Emergency Public Information: Information about an emergency that can be disseminated in anticipation of an emergency or during an emergency. It may provide situational information or directive actions to be taken by the public.

Emergency Response Organization: A group (public, private or volunteer), trained in emergency response that may be called upon to respond to an emergency situation. (Source: Ontario Emergency Management Glossary of Terms)

Emergency Worker: A person performing emergency services to support emergency response.

Notes**:**

- Emergency workers can include the following: nuclear emergency workers required to remain in, or to enter, areas affected or likely to be affected by radiation from a nuclear emergency, and for whom special safety arrangements are required; emergency workers required to provide response outside the affected areas.
- This does not include nuclear energy workers.
- Emergency workers can include police, firefighters, ambulance and emergency social services workers, and other essential services.
- Those who are registered with an authorized responding organization.
- (Source: CSA N1600 — General requirements for nuclear emergency management programs)

Emergency Worker Centre: A facility set up to monitor and control radiation exposure to emergency workers.

Entry Control: The prevention of non-essential persons from entering a potentially dangerous area.

Environmental Decontamination: Reduction or removal of radioactive contamination from the environment.

Equivalent Dose: The absorbed dose multiplied by a weighting factor for the type of radiation giving the dose. Weighting factors for use in Canada are prescribed by the Canadian Nuclear Safety Commission. This term is also sometimes called weighted dose. Expressed in terms of sievert (or rem).

Evacuation: A directed protective action for the controlled displacement of the population from an area which has been or might become contaminated by radioactive substances to avoid exposure. (Source: CSA N1600 — General requirements for nuclear emergency management programs)

Evacuation Centre: A centre which provides affected people with basic human needs including accommodation, food, and water. (Source: Australian Emergency

Exclusion Zone: A parcel of land within or surrounding a reactor facility on which there is no permanent dwelling and over which a licensee has the legal authority to exercise control. (Source: CNSC Glossary)

Existing Exposure Situation: Existing exposure situations already exist when a decision on control must be taken, including prolonged exposure situations after emergencies. These situations include exposure to natural background, exposure due to residual radioactive material that derive from past practices that were never subject to regulatory control, and exposure due to the residual radioactive material deriving from a nuclear or radiological emergency. (Guidance on Planning for Recovery Following a Nuclear or Radiological Emergency)

Exposure: The act or condition of being subject to irradiation. Exposure can be either external exposure (irradiation by sources outside the body) or internal exposure (irradiation by sources inside the body).

Exposure Control: Emergency operations aimed at reducing or avoiding exposure to a plume or puff of radioactive material. Measures to deal with surface contamination and re-suspension might also be included.

Exposure Pathways: The routes by which radioactive material can reach or irradiate humans.

External Notification: The notification of organizations and agencies (not directly part of the emergency management organization) which may be affected by a nuclear or radiological emergency, or which may be required to assist in responding to it.

Food Control: Measures taken to prevent the consumption of contaminated foods that fall outside of any other ingestion control strategy (e.g., produce and crop control, livestock control). Measures may include controlling the supply of uncontaminated foodstuffs, placing foods in storage to permit radionuclide decay, and disposal of unusable stocks.

Fuel Failure: Any rupture of a fuel sheath such that fission products may be released. (Source: CNSC Glossary)

Gamma Radiation: Penetrating electromagnetic radiation emitted from an atom's nucleus. Also called gamma rays. (Source: CNSC Glossary)

General Emergency: Events at a nuclear power plant or onboard a nuclear-powered vessel resulting in an actual or substantial risk of a release of radioactivity or radiation exposure which warrants the implementation of protective actions offsite. (Source: Health Canada Glossary)

General Emergency (U.S. Emergency Classification): Events are in progress or have occurred that involve actual or imminent substantial core degradation or melting with potential for loss of containment integrity. Release of radioactive material can be reasonably expected to exceed PAG exposure levels offsite for more than the immediate site area.

Generic Criteria: Expressed as a projected dose, over a specified time period, above which protective actions are recommended to reduce the risk of stochastic effects.

Government Operations Centre: The Government Operations Centre (GOC) provides an all-hazards integrated federal emergency response to events (potential or actual, natural or human-induced, accidental or intentional) of national interest. It provides 24/7 monitoring and reporting, national-level situational awareness, warning products and integrated risk assessments, as well as national-level planning and whole-of-government response management. During periods of heightened response, the GOC is augmented by staff from other government departments/ agencies (OGD) and non-governmental organizations who physically work in the GOC and connect to it virtually.

Gray (Gy): The International System of Units (SI) unit of measurement used to express absorbed dose. One gray is defined as the absorption of 1 joule of ionizing radiation by 1 kilogram of matter. For gamma and beta radiation, the gray is numerically equal to the sievert. (Source: CNSC Glossary)

Groundshine: Gamma and/or beta radiation from radioactive material deposited on the ground.

Imminent Release: A radioactive emission that will occur in 12 hours or less.

Ingestion Control: Emergency response operations in which the main aim is to avoid or reduce the risk from ingestion of contaminated foodstuff and water.

Ingestion Planning Zone: A pre-designated area surrounding a reactor facility where plans or arrangements are made to:

- protect the food chain;
- protect drinking water supplies;
- restrict consumption and distribution of potentially contaminated produce, wild-harvested foods that are fished, foraged and hunted, milk from grazing animals, rainwater, animal feed; and
- restrict distribution of non-food commodities until further assessments.

(Source: CSA N1600 — General requirements for nuclear emergency management programs)

Initial Notification: The notification made by a reactor facility to provincial and/or municipal authorities upon the occurrence of an event or condition which has

implications for public safety or could be of concern to these authorities. The criteria and channels for making such notification are usually described in emergency plans.

Internal Notification: The notification by an organization to its personnel who are required to respond to an emergency.

Intervention Level: A radiation dose above which a specific protective action is generally justified. (Source: Health Canada Glossary)

Iodine Thyroid Blocking: The reduction or prevention of the absorption of radioiodine by the thyroid gland, which is accomplished by the intake of a stable iodine compound (such as potassium iodide) by people exposed or likely to be exposed to radioiodine.

Ionizing Radiation: For the purposes of *radiation* protection, *radiation* capable of producing ion pairs in biological material(s). *Ionizing radiation* is constantly present in the environment and includes the *radiation* that comes from both natural and artificial sources, such as cosmic rays, terrestrial sources (radioactive elements in the soil), ambient air (radon), and internal sources (food and drink). (Source: CNSC Glossary)

Joint Information Centre: A joint centre for the province, designated municipality, federal government, and the reactor facility or CNSC licensed facility, that is responsible for providing information on the emergency to the media and the public.

Key-Holing Strategy: A key-holing strategy refers to a targeted approach that involves selectively evacuating specific sectors or zones based on assessed risk levels and radiation exposure. This strategy prioritizes the safety of individuals in the most affected areas while minimizing disruption and resource allocation in less impacted zones.

By identifying and isolating critical sectors, emergency response teams can implement timely evacuation procedures, provide focused resources, and maintain effective communication with the public. The key-holing strategy aims to enhance the efficiency of evacuation operations, reduce potential contamination spread, and ensure the safety of both evacuees and emergency personnel during a nuclear incident.

Land Control: Control on the use of contaminated land for growing food products or animal feed.

Livestock Control: Quarantine of livestock in the affected area to prevent movement to other areas. Slaughter of such animals for food may be banned.

Megabecquerel: 10^6 becquerels. (Source: CNSC Glossary)

Microsievert (µSv): One-millionth of a sievert. (Source: CNSC Glossary)

Milk Control: Measures taken to prevent the consumption of contaminated milk in the area affected by a nuclear emergency, and its export outside the area until it has been

monitored. Collection and destruction of contaminated milk may also be involved.

Millisievert (mSv): One-thousandth of a sievert. (Source: CNSC Glossary)

Mitigate: Actions taken to reduce the adverse impacts of an *emergency* or *disaster*. Such actions may include diversion or containment measures to lessen the impacts of a flood or a spill. (Source: Ontario Emergency Management Glossary of Terms)

Municipality: "*Municipality*" means a geographic area whose inhabitants are incorporated (*Municipal Act*). (Source: Ontario Emergency Management Glossary of Terms)

Notification: Conveying to a person or an organization, by means of a message, warning of the occurrence or imminence of a nuclear or radiological emergency, usually includes some indication of the measures being taken or to be taken to respond to it.

Nuclear Emergency: An emergency that has led to or could lead to the release of radioactive material, or exposures to uncontrolled sources of radiation, which pose, or could pose, a threat to health and safety, property, and the environment. (Source: CSA N1600 — General requirements for nuclear emergency management programs)

Nuclear Establishment: A facility that uses, produces, processes, stores or disposes of a nuclear substance, but does not include a reactor facility. It includes, where applicable, any land, building, structures or equipment located at or forming part of the facility, and, depending on the context, the management and staff of the facility.

Nuclear Facility: A generic term covering both nuclear establishments and reactor facilities.

Nuclear Substance: Means:

- deuterium, thorium, uranium or an element with an atomic number greater than 92
- a derivative or compound of deuterium, thorium, uranium or of an element with an atomic number greater than 92
- a radioactive nuclide
- a substance that is prescribed as being capable of releasing nuclear energy or as being required for the production or use of nuclear energy
- a radioactive by-product of the development, production or use of nuclear energy
- a radioactive substance or radioactive thing that was used for the development or production, or in connection with the use, of nuclear energy

(Source Nuclear Safety and Control Act)

Offsite: Offsite refers to the area outside the boundary (fence) of a reactor facility.

Onsite: Onsite refers to the area inside the boundary (fence) of a reactor facility.

Onsite Emergency: A serious malfunction which results or may result in an atmospheric release of radioactive material or is likely to result in a release at a later time.

Operational Directive: Direction given by the Emergency Response Organization to implement operational measures.

Operational Intervention Level (OIL): A calculated value, measured by instruments or determined by laboratory analysis that corresponds to an intervention level.

Notes**:**

- OILs are typically expressed in terms of dose rates or of activity of radioactive material released, time integrated air concentrations, ground or surface concentrations, or activity concentrations of radionuclides in environmental, food, or water samples.
- An OIL is a type of action level that can be used immediately and directly (without further assessment) to determine the appropriate protective actions and other response actions on the basis of an environmental measurement.

(Source: Based on CSA N1600 — General requirements for nuclear emergency management programs)

Operational Measures: Measures undertaken by the Emergency Response Organization to deal with the emergency, including measures to enable or facilitate protective action for the public, e.g., public alerting, public direction, activation of plans, traffic control, emergency information, etc.

Operator: Holder of a subsisting licence issued pursuant to the Nuclear Safety and Control Act for the operation of a reactor facility.

Optimization: The process of determining a level of protection and safety that makes exposures and the probability and magnitude of potential exposures as low as reasonably achievable, with economic and social factors being taken into account.

Pasture Control: Measures taken to prevent animals from consuming contaminated feed and water. Measures may include removing milk- and meat-producing animals from pasture and from access to open water sources and supplying them with uncontaminated feed and water.

Personal Monitoring: The use of radiation monitoring devices to assess whether persons, and their belongings, including vehicles, are contaminated or not, and, if contaminated, the type and level of contamination.

Personal Protective Equipment: Clothing or other specialised equipment provided to an offsite emergency worker to prevent or reduce their exposure to radioactive material. (Source: Health Canada Glossary)

Planned Exposure Situation: Planned exposure situations involve the planned operation of a source (e.g., the operation of nuclear reactors) or planned activities that result in an exposure from a source (e.g., disposal of radioactive waste at an EWC). (Guidance on Planning for Recovery Following a Nuclear or Radiological Emergency)

Note: See definitions for AAZ, DPZ, CPZ, and IPZ. (Source: CSA N1600, General requirements for nuclear emergency management programs)

Plume: A cloud of airborne radioactive material that is transported in the direction of the prevailing wind from a reactor facility. A plume results from a continuing release of radioactive gases or particles. (This term may also be used for waterborne radioactive material resulting from a liquid emission. Where the context does not make it clear, this will be referred to as a waterborne plume).

Population Monitoring and Medical Management: The protective action strategy which includes population screening, decontamination, internal contamination assessment and medical follow-up. The purpose of this Protective Action Strategy is to reduce exposures to individuals. (Source: Health Canada Glossary)

Precautionary Measures: Measures directed to facilitate the application and effectiveness of protective measures.

Preparedness: Actions taken prior to an emergency or disaster to ensure an effective response. These actions include the formulation of emergency response plans, business continuity/continuity of operations plans, training, exercises, and public awareness and education. (Source: Ontario Emergency Management Glossary of Terms)

Prevention: Actions taken to stop an emergency or disaster from occurring. Such actions may include legislative controls, zoning restrictions, improved operating standards/procedures or critical infrastructure management. (Source: Ontario Emergency Management Glossary of Terms)

Probability: The likelihood of an event occurring that may result in an emergency, disaster or service disruption. (Source: Health Canada Glossary)

Produce and Crop Control: Measures taken to restrict the harvesting or processing of potentially or actually contaminated crops, vegetables and fruits. Measures include embargoing export outside the affected area; storage to allow radionuclide decay; destruction and disposal of contaminated produce

Projected Dose: The highest committed effective equivalent dose or committed

equivalent dose to a specified organ or tissue, likely to be received through all applicable exposure pathways by the most exposed member of the critical group in the area for which the projection is being made.

Protective Actions: Actions to mitigate the effects of a nuclear or radiological emergency. The actions include both precautionary measures and protective measures.

Protective Measures: Measures designed to protect against exposure to radiation during a nuclear or radiological emergency. The measures include sheltering, evacuation, potassium iodide (KI) pill ingestion, ingestion control, relocation, protective clothing, and respiratory protection.

Provincial Emergency Operations Centre (PEOC): A fully equipped facility maintained by Emergency Management Ontario that can be activated in response to, or in anticipation of, emergencies. The PEOC is staffed with appropriate representatives from ministries that have been delegated responsibilities for specified emergencies as well as EMO staff, and other partners in emergency management. It serves as a coordinating point-of-contact for the affected municipality, provincial, and federal interests. (Source: Ontario Emergency Management Glossary of Terms)

Provincial Nuclear Emergency Response Plan (**PNERP):** Short title for the Strategic Response Framework for Nuclear & Radiological Emergency Management. An approved emergency response plan for the Province of Ontario to respond to a nuclear or radiological emergency, and its consequences.

Public Alerting: See Alerting.

Public Awareness and Education Program: A program that provides focused information to a target audience to educate about protective actions to reduce the *risk* of life and property damage, in the event of an emergency. (Source: Ontario Emergency Management Glossary of Terms)

Radiation: The emission by a nuclear substance, the production using a nuclear substance, or the production at a reactor facility of, an atomic or subatomic particle or electromagnetic wave with sufficient energy for ionization. (Source: Health Canada Glossary)

Radioactive Material: For purposes of nuclear security, any material that emits one or more types of ionizing radiation, such as alpha or beta particles, neutrons or gamma rays. (Source: CNSC Glossary)

Radioiodine: A substance containing radioactive iodine in a chemical form that has a metabolic pathway similar to iodide, such as inorganic compounds and metabolic forms of organic iodine that are broken down in a living organism. Some examples are the radioisotopes iodine-125 and iodine-131. (Source: CNSC Glossary)

Radioisotope: A variation in the form of atoms, of the same chemical element, which are distinguished by the number of neutrons in the nucleus. The number of protons remains the same, but the number of neutrons differs. For example, uranium has 16 different isotopes. (Source: CNSC Glossary)

Radiological Emergency: Emergency caused by an actual or environmental hazard from ionizing radiation emitted by a source other than a reactor facility.

Radionuclide (or radioactive isotope or radioisotope): A naturally occurring or artificially created isotope of a chemical element having an unstable nucleus that decays, emitting alpha, beta and/or gamma rays until stability is reached.

Reactor Facility: A facility producing greater than 10 megawatts gross thermal energy from nuclear fuel and consisting of one or more reactor units.
Note: This includes nuclear power plants and research reactors greater than 10 megawatts gross thermal energy.

Reception Centre: Locations for the initial reception, monitoring, decontamination, and registration of evacuated members of the public, which provides or arranges for further emergency social services, humanitarian assessments and support.

Notes:

- A public Reception Centre is typically located in an existing facility, such as a community centre. Public Reception Centres should be beyond the Detailed Planning Zone boundary.
- Examples of emergency social services include emergency shelter, food, clothing, victim registration and inquiry and personal services.
- Examples of humanitarian support include, but are not limited, to housing and family reunification. (Source: Modified IAEA Safety Guide GS-G-2.1.)

Recovery: The short-term and long-term actions taken in order to restore, to an acceptable level, both the organizations involved in, and the communities affected by, the nuclear emergency and the associated response activities.
(Source: CSA N1600 — General requirements for nuclear emergency management programs)

Release: In the context of this plan, release refers to the emission of radioactive material to the environment from a reactor facility in the form of either an airborne or a liquid emission.

Representative Individual: An individual that due to his/her characteristics, habits and location of residence, is representative of the more highly exposed individuals in the population. May also be referred to as Representative Person. (Source Health Canada Glossary)

Reportable Event: An event affecting the reactor facility which would be of concern to the offsite authorities responsible for public safety.

Response: The actions taken during a nuclear or radiological emergency to reduce the magnitude of the hazard and manage its consequences, including the impact of the hazard on people, property, and the environment.
(Source: CSA N1600, General requirements for nuclear emergency management programs)

Response Sectors: The Detailed Planning Zone is subdivided into Response Sectors to facilitate the planning and implementation of protective measures.

Risk: The product of the probability of the occurrence of a hazard and its consequences. (Source: Ontario Emergency Management Glossary of Terms)

Self-decontamination: the removal of radioactive contamination from oneself through removal of clothing, washing of oneself and dressing in clean clothes. (Source: Centers for Disease Control and Prevention)

Severe Accident: A beyond design basis accident involving fuel degradation in the reactor core or wet storage bay.

Shall: Is used to express a requirement, i.e., a provision that the user is obliged to satisfy in order to conform to the PNERP.

Shelter-in-place: A directed protective action to take immediate refuge in an enclosed structure for protection from an airborne plume, deposited radioactive material, or both.

Notes**:**

- Shelter-in-place is a protective action which uses the shielding properties of buildings and their potential for ventilation control to reduce the radiation dose to people inside. Shelter-in-place has varying degrees of effectiveness depending on the type of building construction.
- Shelter-in-place should typically not extend beyond two days.
- Shelter-in-place is utilized as a protective action if there is insufficient time to safely evacuate an area; if the dose projected for an area is so low that evacuation is not required; or the risks of evacuation are higher than shelter-in-place (e.g., severe weather inhibits safe evacuation).

(Source: CSA N1600 — General requirements for nuclear emergency management programs)

Should: Is used to express a recommendation or that which is advised but not required in order to conform to the PNERP.

Shutdown State: A subcritical reactor state with a defined margin to prevent a return to criticality without external actions. (Source: CNSC Glossary)

SI: International System of Units; acronym from the French definition "Système International d'unités". (Source: modified from CNSC Glossary)

Sievert: The International System of Units (SI) unit of equivalent dose and effective dose, equal to 1 joule/ kilogram. (Source: CNSC Glossary)

Source Term: A generic term applied to the *radioactive material* released from a *reactor facility*. It includes the quantity and type of material released as well as the timing and rate of its release. It could apply to a release that was currently occurring, or one which had ended, or one which could take place in the future.

Site Area Emergency (U.S. Emergency Classification): Events are in progress or have occurred that involve actual or likely major failures of plant functions needed for protection of the public. Any releases of radioactive material are not expected to exceed PAG exposure levels, beyond the site boundary.

Stochastic Effects: *Radiation*-induced health effects, such as cancer and heritable diseases, which are associated with a statistical *risk* and where no threshold has been established. The *probability* of occurrence is proportional to the *dose* (the higher the *dose* the higher the *probability* of occurrence) but the severity of the effect is independent of *dose*. (Source: Health Canada Glossary)

Support Municipality: Pursuant to Section 7.0.2 (4) of the EMCPA, the LGIC may, by order, specify a municipality to act in a support capacity to provide assistance to designated municipalities.

Tissue Reactions: Radiation exposure can lead to various biological effects at the cellular and tissue levels, depending on the type, dose, and duration of exposure. Tissue reactions, also known as deterministic effects, are dose-dependent responses that manifest in specific organs or systems following exposure to ionizing radiation.

Transborder Emergency: A nuclear emergency involving a reactor facility or nuclear accident or event outside the borders of Ontario that might affect people and property in the province.

Unusual Events: (U.S. Emergency Classification)**:** Unusual events are in progress or have occurred that indicate a potential degradation of the level of safety of the plant. No releases of radioactive material requiring offsite response or monitoring are expected unless further degradation of safety systems occurs.

Urgent Protective Action: A protective action in the event of an emergency which must be taken promptly (normally within hours) in order to be effective, and the effectiveness of which will be markedly reduced if it is delayed. The most commonly

considered urgent protective actions in a nuclear or radiological emergency are evacuation, decontamination of individuals, sheltering, respiratory protection, iodine prophylaxis and restriction of the consumption of potentially contaminated foodstuffs. (Source: IAEA definition)

Venting: The release to the atmosphere of radioactive material from the containment of a reactor facility through systems designed for this purpose.

Vulnerable populations: Members of the public who have additional needs before, during, and after a nuclear or radiological emergency in one or more functional areas.

Notes**:**

- Functional areas can include, but are not limited to, the following:
  - maintaining independence;
  - communication;
  - transportation;
  - supervision; or
  - medical care.

- Individuals in need of additional assistance could include those who:
  - have disabilities;
  - are from diverse cultures;
  - have limited or no proficiency in the local official language; or
  - are transportation disadvantaged.

(Source: CSA N1600 — General requirements for nuclear emergency management programs)

Water Control: Measures taken to avoid the contamination of drinking water supplies and sources, and to prevent or reduce the consumption of contaminated water. Measures may include the isolation of drinking water systems to prevent system-wide contamination.

Weighted Dose: See Equivalent Dose. Expressed in terms of Sievert (or rem).

Wild-Harvested Food: A wild harvest represents food gathered from plants that grow naturally in an untouched environment and in their native habitat.

Wild-Harvested Food Control: Wild-Harvested Food Control refers to the systematic management and monitoring of food sources obtained from natural environments in the context of nuclear emergency response. This process includes the assessment of safety and contamination levels of wild-harvested foods—such as plants, mushrooms, and game—potentially impacted by nuclear incidents. It involves protocols for testing, guidelines for harvesting, and public advisories to ensure that consumption of these foods does not pose a health risk due to radioactive exposure or contamination. The

objective is to safeguard public health and mitigate the risks associated with consuming wild foods in areas affected by nuclear incidents.

Provincial Nuclear Emergency Response Plan (PNERP)

Implementing Plan for the Pickering Nuclear Generating Station (PNGS)

January 2019

Prepared by Office of the Fire Marshal and Emergency Management

Ministry of Community Safety and Correctional Services

Ontario

Executive Council of Ontario
Order in Council

On the recommendation of the undersigned, the Lieutenant Governor of Ontario, by and with the advice and concurrence of the Executive Council of Ontario, orders that:

Conseil exécutif de l'Ontario
Décret

Sur la recommandation de la personne soussignée, la lieutenante-gouverneure de l'Ontario, sur l'avis et avec le consentement du Conseil exécutif de l'Ontario, décrète ce qui suit:

WHEREAS section 8 of the *Emergency Management and Civil Protection Act*, as amended, requires the Lieutenant Governor in Council to formulate an emergency plan respecting emergencies arising in connection with nuclear facilities;

AND WHEREAS pursuant to O.C. 2317/2017, the *Provincial Nuclear Emergency Response Plan (PNERP) Master Plan 2017* was adopted by the Lieutenant Governor in Council as an emergency plan respecting emergencies arising in connection with nuclear facilities formulated under section 8 of the *Emergency Management and Civil Protection Act*, as amended;

AND WHEREAS pursuant to O.C. 619/2018 the Lieutenant Governor in Council adopted the "*Provincial Nuclear Emergency Response Plan (PNERP) Implementing Plan for the Pickering Nuclear Generating Station*" dated March 2018 (the "Pickering Implementing Plan, March 2018"), as an emergency plan under section 8 of the *Emergency Management and Civil Protection Act*, as amended;

AND WHEREAS the Pickering Implementing Plan, March 2018 has been amended;

THEREFORE the document entitled "*Provincial Nuclear Emergency Response Plan (PNERP) Implementing Plan for the Pickering Nuclear Generating Station (PNGS)" dated January 2019*, (the "Pickering Implementing Plan, January 2019") is hereby adopted as an emergency plan under section

8 of the *Emergency Management and Civil Protection Act*, as amended, and replaces the Pickering Implementing Plan, March 2018.

AND THAT Order in Council O.C. 619/2018 dated March 27, 2018 be revoked.

ATTENDU QUE l'article 8 de la *Loi sur la protection civile et la gestion des situations d'urgence*, dans sa version modifiée, exige que le lieutenant-gouverneur en conseil établisse un plan de mesures d'urgence relatif aux situations d'urgence liées aux installations nucléaires;

ATTENDU QU'en vertu du décret numéro 2317/2017, la lieutenante-gouverneure en conseil a adopté le *Plan provincial d'intervention en cas d'urgence nucléaire (PPIUN) – Plan directeur 2017* à titre de plan de mesures d'urgence relatif aux situations d'urgence liées aux installations nucléaires établi en vertu de l'article 8 de la *Loi sur la protection civile et la gestion des situations d'urgence*, dans sa version modifiée;

ATTENDU QU'en vertu du décret numéro 619/2018, la lieutenante-gouverneure en conseil a adopté le « *Plan provincial d'intervention en cas d'urgence nucléaire (PPIUN) – Plan de mise en œuvre pour la centrale nucléaire de Pickering* », daté de mars 2018 (le « Plan de mise en œuvre pour Pickering de mars 2018 »), comme plan de mesures d'urgence en vertu de l'article 8 de la *Loi sur la protection civile et la gestion des situations d'urgence*, dans sa version modifiée;

ATTENDU QUE le Plan de mise en œuvre pour Pickering de mars 2018 a été modifié;

EN CONSÉQUENCE, le document intitulé « *Plan provincial d'intervention en cas d'urgence nucléaire (PPIUN) – Plan de mise en œuvre pour la centrale nucléaire de Pickering (PCNP)* », daté de janvier 2019 (le « Plan de mise en œuvre pour Pickering de janvier 2019 »), est par les présentes adopté comme plan de mesures d'urgence en vertu de l'article 8 de la *Loi sur la protection civile et la*

gestion des situations d'urgence, dans sa version modifiée, et remplace le Plan de mise en œuvre pour Pickering de mars 2018.

EN OUTRE, le décret numéro 619/2018, daté du 27 mars 2018, est révoqué.

Recommended: Minister of Community Safety and Correctional Services
Recommandé par: Ministre de la Sécurité communautaire et des Services correctionnels

Concurred: Chair of Cabinet
Appuyé par: Le président/la présidente du Conseil des ministres,

Approved and Ordered:
Approuvé et décrété le: JAN 3 1 2019

Lieutenant Governor
La lieutenante-gouverneure

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Foreword

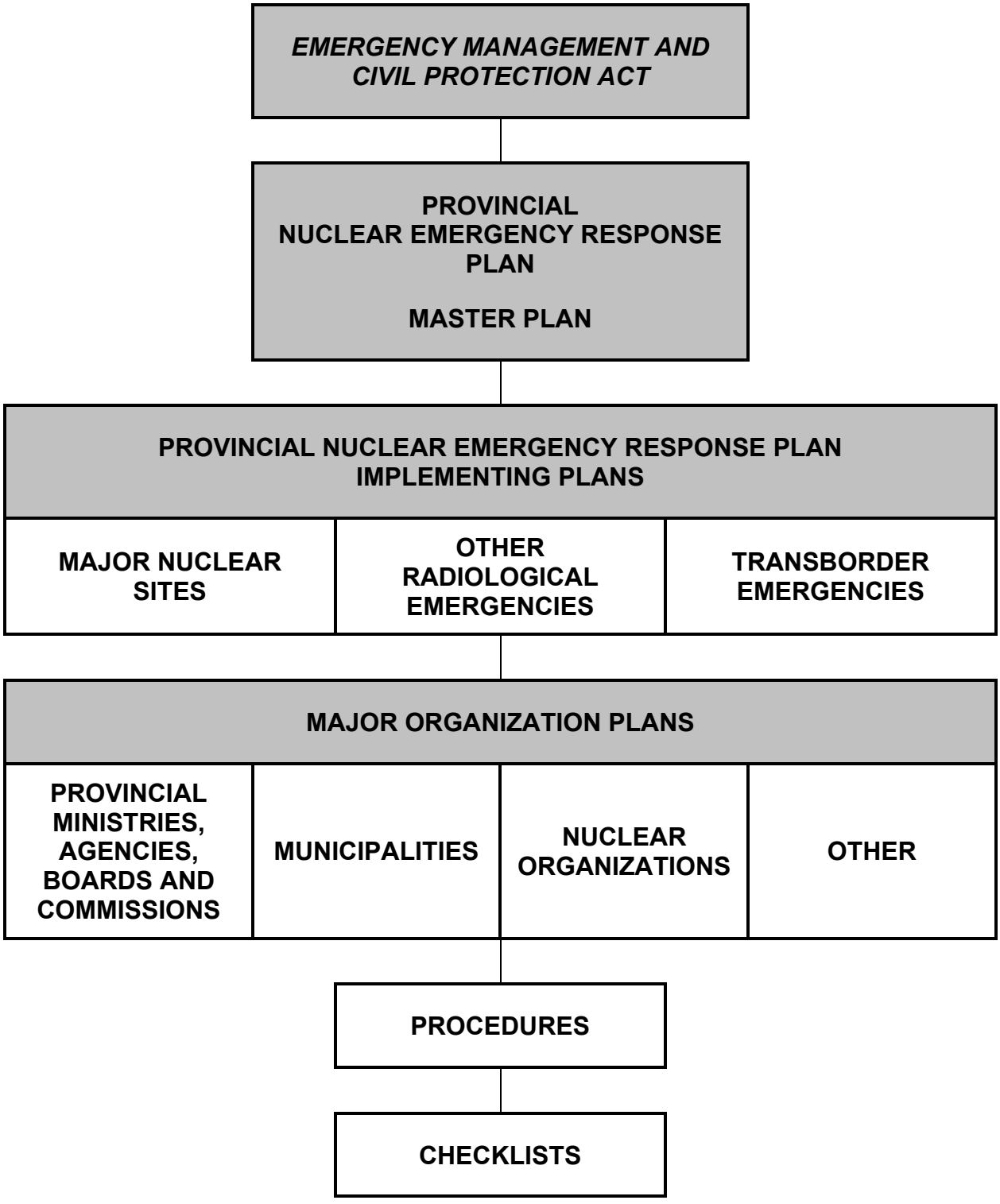
The province of Ontario's Nuclear Emergency Response Plan has been developed pursuant to **Section 8** of the *Emergency Management and Civil Protection Act*, R.S.O. 1990, c. E. 9 (hereafter referred to as the *Emergency Management and Civil Protection Act* or *EMCPA*). The current edition of this plan supersedes and replaces all older versions which should be destroyed.

Holders of the *Provincial Nuclear Emergency Response Plan Implementing Plan for the Pickering Nuclear Generating Station* are responsible for keeping it updated by incorporating amendments, which may be issued from time to time.

This plan is administered by the **Minister of Community Safety and Correctional Services of Ontario**. All comments and suggestions relating to it should be directed to:

Program Manager, Planning and Exercises
Office of the Fire Marshal and Emergency Management
Ministry of Community Safety and Correctional Services
25 Morton Shulman Avenue
Toronto ON
M3M 0B1
Phone: 647-329-1100
E-mail: askEMO@ontario.ca
Website: http://www.ontario.ca/emo (English version)
http://www.ontario.ca/gdu (French version)

Figure I: Nuclear and Radiological Emergency Response Planning Structure



Nuclear and Radiological
Emergency Response Planning Structure

The structure for nuclear and radiological emergency response planning in Ontario, illustrated on the previous page, consists of the following components:

a) The **Provincial Nuclear Emergency Response Plan (PNERP)**; developed pursuant to Section 8 of the EMCPA and subject to Cabinet approval:

- **The Master Plan**: sets out the overall principles, policies, basic concepts, organizational structures and responsibilities.

- **The Implementing Plans**: the elements of the **Master Plan** are applied to each major nuclear site, *transborder emergencies* and other types of *radiological emergencies*, and detailed provincial implementing plans developed. The major organization plans (as per **Figure I**) should be consistent with the requirements of these implementing plans.

b) **Major organization plans**: Each major organization involved (e.g., provincial ministries, agencies, boards and commissions, municipalities, and nuclear organizations, etc.) develops its own plan to carry out the relevant roles, responsibilities and tasks agreed to by them and consistent with their mandate. These plans are based on, and *should* be consistent with the **PNERP** and its Implementing Plans.

c) **Procedures**: Based on all of the above plans, procedures are developed for the various *emergency* centres to be set up and for the various operational functions required.

d) **Checklists**: The culmination of the planning process is the development of checklists based on the requirements of the procedures, e.g., individual position or function-specific checklists.

All emergency organizations involved in the preparation and implementation of the *Provincial Nuclear Emergency Response Plan should* employ common terminology. The terminology contained in the Glossary, **Annex D**, *should* be used for this purpose by all concerned. Words or phrases defined in the Glossary are italicized within the text of this document. Further reference information can be found in the Incident Management System (IMS) doctrine at www.ontario.ca/ims.

Acronyms and Abbreviations

AAZ	Automatic Action Zone
ALARA	As Low As Reasonably Achievable
ARGOS	Accident Reporting and Guidance Operational System
BDBA	Beyond Design Basis Accident
Bq	Becquerel
BWR	Boiling Water Reactor
CANDU	Canada Deuterium Uranium (reactor)
CCEM	Cabinet Committee on Emergency Management
CEOC	Community Emergency Operations Centre
CEOF	Corporate Emergency Operations Facility
CFIA	Canadian Food Inspection Agency
CMOH	Chief Medical Officer of Health
CNSC	Canadian Nuclear Safety Commission
CPZ	Contingency Planning Zone
CRL	Chalk River Laboratories
CSA	Canadian Standards Association
DBA	Design Basis Accidents
DNGS	Darlington Nuclear Generating Station
DPZ	Detailed Planning Zone
ECCC	Environment and Climate Change Canada
ECI	Emergency Coolant Injection
EIC	Emergency Information Centre

EMCPA	Emergency Management and Civil Protection Act
EME	Emergency Mitigating Equipment
EMST	Environmental Monitoring and Surveillance Team
ENDM	Ministry of Energy, Northern Development and Mines
EOC	Emergency Operations Centre
EPZ	Emergency Planning Zone
ERAMG	Environmental Radiation and Assurance Monitoring Group
FADS	Filtered Air Discharge System
FNEP	Federal Nuclear Emergency Plan
FNEP TAG	FNEP Technical Assessment Group
GC	Generic Criteria
GOC	Government Operations Centre
Gy	Gray
HC	Health Canada
HIRA	Hazard Identification Risk Assessment
IAEA	International Atomic Energy Agency
IMS	Incident Management System
INES	International Nuclear Event Scale
IPZ	Ingestion Planning Zone
ITB	Iodine Thyroid Blocking
KI	Potassium Iodide
km	Kilometre
LGIC	Lieutenant Governor in Council

LOCA	Loss-of-Coolant Accident
MCCSS	Ministry of Children, Community and Social Services
MCSCS	Ministry of Community Safety and Correctional Services
MDU	Monitoring and Decontamination Unit
MEOC	Ministry Emergency Operations Centre
Met	Meteorology, meteorological
MLDP	Modèle Lagrangien de Dispersion de Particules
MMAH	Ministry of Municipal Affairs and Housing
MNRF	Ministry of Natural Resources and Forestry
MECP	Ministry of Environment, Conservation and Parks
MOHLTC	Ministry of Health and Long-Term Care
MOL	Ministry of Labour
MOU	Memorandum of Understanding
MTO	Ministry of Transportation, Ontario
MW	Megawatts
NAADS	National Alert Aggregation and Dissemination System
NGS	Nuclear Generating Station
NIG	Nuclear Incident Group
NRCan	Natural Resources Canada
OIL	Operational Intervention Level
OMAFRA	Ontario Ministry of Agriculture, Food and Rural Affairs
OPP	Ontario Provincial Police
PCEIO	Provincial Chief Emergency Information Officer

PEOC	Provincial Emergency Operations Centre
PLERP	Provincial Liquid Emission Response Plan
PNERP	Provincial Nuclear Emergency Response Plan
PNGS	Pickering Nuclear Generating Station
PPE	Personal Protective Equipment
RD	Radiological Device
RDD	Radiological Dispersal Device
RED	Radiological Exposure Device
RHRP	Radiation Health Response Plan
RIMPUFF	Risø Mesoscale PUFF
SAMG	Severe Accident Management Guidelines
URI	Unified RASCAL Interface
USA	United States of America
UTCC	Unified Transportation Coordination Centre
UTMP	Unified Transportation Management Plan

Chapter 1
SCOPE AND AUTHORITY

1.1 Aim

The aim of the **Provincial Nuclear Emergency Response Plan (PNERP) Implementing Plan for the Pickering Nuclear Generating Station (PNGS)** is to describe the measures that *should* be undertaken to *mitigate* the off-site effects of a *nuclear emergency* at PNGS.

1.2 Scope

1.2.1 This implementing plan *shall* be read and applied in the context of the **PNERP Master Plan**.

1.2.2 In case of any apparent differences between the provisions of the **PNERP Master Plan** and this Implementing Plan, the latter being more detailed and specific is applicable.

1.2.3 Together, these two plans focus on provincial level actions and should therefore be supplemented by the appropriate municipal plans and other supporting plans and procedures (see **Sections 1.3** and **1.4** below).

1.2.4 OFMEM *shall* establish and communicate guidance to verify stakeholder conformance with the PNERP Master Plan and this Implementing Plan.

1.3 Designated and Support Municipalities

1.3.1 Designated Municipalities

a) The City of Toronto and the Regional Municipality of Durham are the *Designated Municipalities* in the *Detailed Planning Zone* with respect to PNGS (**PNERP Master Plan, Annex A**).

b) The City of Peterborough is a *Designated Host Municipality* with respect to PNGS (**PNERP Master Plan, Annex A**).

c) In this document the terms “municipal” and “*Municipality*” *shall* include, unless the context indicates otherwise, the *Designated Municipality*, as well as the local police services and local boards whose area of operation includes the area covered by the municipal plans.

d) Pursuant to **Section 3(4)** of the *EMCPA*, as *Designated Municipalities,* the Regional Municipality of Durham, the City of Toronto, and the City of Peterborough *shall* formulate plans to *mitigate* the *off-site* consequences of nuclear emergencies at PNGS.

e) These plans *shall* also contain, where applicable, arrangements for the provision of services and assistance by municipal departments, local police services, fire services, paramedic services, hospitals and local boards.

f) The plans prepared by the *Designated Municipalities* and by these other organizations, are collectively referred to as “municipal plans” in this document.

g) As required by **Section 5** of the *EMCPA*, plans of *Lower-tier Municipalities* within the Regional Municipality of Durham *shall* conform to the plans of their *Upper-tier Municipality*.

h) As required by **Section 8** of the *EMCPA*, municipal *nuclear emergency response* plans *shall* conform to the **PNERP** and be subject to the approval of the Solicitor General (this function is fulfilled by the Minister of Community Safety and Correctional Services). The Solicitor General may make such alterations as considered necessary for the purpose of coordinating the municipal plan with the province’s plan.

1.3.2 Support Municipalities

a) In the event of a declared *emergency*, the Lieutenant Governor in Council (LGIC) or the Premier may order a *Municipality* to provide support or assistance to *Designated Municipalities* or to affected *municipalities*. Such orders, if made, would be authorized by **Sections 7.0.2(4)** or **7.0.3** of the *EMCPA*.

b) Support and assistance may include, but *shall* not be limited to, personnel, equipment, services and material.

1.4 Supporting Plans and Procedures

1.4.1 Other jurisdictions and organizations that have, or are assigned, some responsibility for responding to a PNGS *emergency should* develop appropriate plans or procedures for carrying out their roles and tasks. They include:

a) Provincial ministries, including:

i. MOHLTC and the Radiation Health Response Plan

ii. MTO and the Unified Transportation Management Plan

iii. MCSCS and the Environmental Radiation and Assurance Monitoring Group (ERAMG) Plan

iv. MCSCS and the Provincial Liquid Emission Response Plan (PLERP)

b) Municipal departments, local police services, local boards and other agencies assigned roles and responsibilities in the municipal plans.

c) The PNGS *operator's nuclear emergency* plan and *emergency* procedures.

1.4.2 Radiation Health Response Plan (RHRP)

a) The MOHLTC issues the **RHRP** as an organizational plan under the **PNERP**.

b) The RHRP establishes the roles and responsibilities, operational concepts and *response* principles for coordinating the provincial *response* of health organizations during a *nuclear emergency*.

1.4.3 Unified Transportation Management Plan (UTMP)

a) The PNGS **UTMP**, an organizational plan under the **PNERP**, *shall* be issued by the Ministry of Transportation (MTO), for the management of evacuating traffic in the *Detailed Planning Zone* as well as the traffic impact beyond it.

b) Representatives of the OPP, Metrolinx, Regional Municipality of Durham, City of Toronto, local police services, road authorities and *emergency* services *shall* cooperate with MTO in the development and maintenance of the **UTMP** and in its implementation during a *nuclear emergency* response through the Unified Transportation Coordination Centre (UTCC).

c) The **UTMP** *shall* be designed to meet the requirements of the provincial and municipal *nuclear emergency* plans. For specific guidance see the following:

i. **Section 3.1.3** Unified Transportation Coordination Centre (UTCC)

ii. **Section 3.4** Telecommunications

iii. **Section 4.3** Internal Notifications

iv. **Section 4.6** Early Phase Response

v. **Section 5.3.1** Evacuation

vi. **Section 6.6** Entry Control

vii. **Section 6.7** Transportation Management

1.4.4 Environmental Radiation and Assurance Monitoring Group (ERAMG) Plan

The Environmental Radiation and Assurance Monitoring Group (ERAMG) Plan issued by MCSCS (OFMEM) *shall* describe the means by which the environment, water, milk and foodstuffs are sampled and analyzed during a nuclear or radiological emergency, to determine their safety.

1.4.5 Provincial Liquid Emission Response Plan (PLERP)

The **PLERP** for PNGS is an organizational plan issued by the MCSCS (OFMEM) to *mitigate* the effects of a waterborne *release* from a *reactor facility* resulting in discharges with above normal levels of radioactivity. Additional guidance on the application of the PLERP coincident with a *nuclear emergency* is provided in **Section 6.10**.

Chapter 2
THE PLANNING BASIS

2.1 General

2.1.1 This implementing plan details the response to an *emergency* at the Pickering Nuclear Generating Station (PNGS).

2.1.2 PNGS is located at latitude 43° 48' North and longitude 79° 04' West, approximately 35 kilometres east of Toronto (City Hall) on the northern shore of Lake Ontario.

2.1.3 PNGS consists of eight Canada Deuterium Uranium (CANDU) nuclear reactors, grouped as PNGS-A and PNGS-B, each with four reactors. Each of the reactor units was built with an electric power generation capacity of 540 megawatts. **Figure 2.1** shows a schematic diagram of a CANDU reactor.

2.1.4 Two of the four reactors at PNGS-A are de-fuelled and in a safe shutdown state since 1997.

2.2 The Hazard

2.2.1 If an *accident* were to occur at PNGS, the most probable result would be that its effects would be confined within the station boundary because of the facility's designed safety systems, structures and components.

2.2.2 *Nuclear emergency preparedness* requires a planning basis which considers both *design basis accidents,* and significantly less probable *beyond design basis accidents* (BDBAs), including *severe accidents* and multi-unit scenarios where applicable. For a detailed explanation regarding the basis for these reference *accidents*, refer to **PNERP Master Plan, Annex L - PNERP Planning Basis Background**.

2.2.3 Design Basis Accidents (DBA)

a) The DBA *release* provides the main platform for detailed planning and is generally characterized by one or more of the following:

i. Station *containment* systems function normally allowing *radiation* to start decaying prior to a controlled *release*.

ii. Sufficient time would be available to alert the public and implement *protective measures* prior to a *release*.

iii. The main radiological hazard to people would be external *exposure* to, and inhalation of, *radionuclides*.

iv. Filter systems function to remove almost all of the radioactive particulate and *radioiodine*. As a result, the *plume* would be mostly comprised of inert noble gasses which would dissipate and do not pose a *contamination* hazard.

v. *Radiation doses* to the public would likely be below the *generic criteria* (GC) as defined in **PNERP Master Plan, Annex E**.

vi. Environmental *contamination* would be limited to very low levels.

vii. Low-level radioactive *releases* to the environment could occur on and off for some time (e.g., days or weeks).

b) An example of a *design basis accident* scenario is a *Loss-of-Coolant Accident* (LOCA), with the following typical progression:

i. The reactor building would “box-up” preventing any immediate *releases*. A “box-up” is a condition whereby all possible release pathways to the environment, such as ventilation stacks, are closed.

ii. Duct connections from the reactor building to the vacuum building would open, thereby reducing the former’s internal pressure to below atmospheric pressure, drawing *radioactive material released* from the damaged reactor fuel into the vacuum building. During this holdup time (e.g., 48 hours) the contained radioactivity would decay.

iii. If at any stage the pressure in the *containment* system nears atmospheric pressure, the contained radioactivity may be vented through filters to the environment. Such *venting* could be intermittent or continuous but may last for weeks (see **Section 4.6.6**). The level of radioactivity being *released* would progressively decline with time.

iv. Suitable meteorological conditions may make it possible to vent some of this contained radioactivity through filters in a direction away from populated areas. It may be possible to do this several times.

2.2.4 Beyond Design Basis Accidents (BDBA)

a) One or more of the following may define a BDBA:

i. Station *containment* systems may be impaired leading to significantly reduced hold up time and decay of *radioactive materials*.

ii. An early *release* of radioactivity from a BDBA with limited warning time.

iii. An uncontrolled *release* of radioactivity from a BDBA with limited warning time.

iv. The *plume* could include *radioiodine* and particulates along with noble gases.

v. *Radiation doses* could potentially be high.

vi. Environmental *contamination* could be quantitatively significant in both extent and duration.

vii. The area affected could extend beyond the *Detailed Planning Zone*.

viii. A multi-unit *accident* (i.e., an *accident* involving more than one reactor).

b) BDBAs which go unmitigated may evolve into *severe accidents* involving fuel degradation in the reactor core.

c) The response to BDBAs, including *severe accidents*, is facilitated by dedicated emergency mitigating equipment in addition to the measures already in place to respond to DBAs (see **Section 2.2.3** above) and the ability to expand their function.

d) The following additional planning and *preparedness* actions *shall* be conducted to *mitigate* the much less probable, but possibly more severe, *off-site* effects of BDBAs:

i. pre-distribution of Potassium Iodide (KI) pills (see **Section 5.3.3)**

ii. automatic, default actions to initiate public *alerting* (see **Section 6.2**) and to direct the implementation of protective actions, including *sheltering-in-place* (see **Section 5.3.4**) and *evacuation* (see **Section 5.3.1**)

iii. timely dispatch of aerial and ground monitoring teams to determine areas of *contamination* (see **Section 4.7.3**)

iv. priority *evacuations* for those closest to the hazard (see **Section 5.3.1**)

v. extension of protection actions to the Contingency Planning Zone (CPZ), if required, to reduce potential for exposure

vi. *radiation* monitoring and, if necessary, *decontamination* of persons (see **Section 6.9**)

vii. medical assessment, treatment and counselling as required (see **Section 6.9**)

2.3 Protective Actions

2.3.1 The protective actions available for minimizing the *radiation* hazard in a *nuclear emergency* include:

a) precautionary measures

b) exposure control measures

c) ingestion control measures

2.3.2 These measures are listed in **Table 2.1** below and defined in the glossary (**Annex D**).

2.3.3 The operational use of these measures is described in appropriate sections of this plan.

Table 2.1 Protective Actions for a Nuclear Emergency Response

Precautionary Measures	Exposure Control Measures	Ingestion Control Measures
• Closing of beaches, recreation areas, etc. • Closing of workplaces and schools • Suspension of non-critical patient admissions in hospitals	• Sheltering-in-place • Iodine Thyroid Blocking • Evacuation	• Milk control • Water control • Pasture control • Produce and crop control • Livestock control

2.4 Planning Zones

2.4.1 Automatic Action Zone (AAZ)

a) The AAZ is a pre-designated area immediately surrounding a *reactor facility* where pre-planned protective actions would be implemented by default on the basis of *reactor facility* conditions with the aim of preventing or reducing the occurrence of severe *deterministic effects*.

b) The PNGS AAZ is the area immediately surrounding the *reactor facility* extending out to an approximate radius of 3 kilometres.

c) The *Automatic Action Zone* comprises *Detailed Planning Zone* (DPZ) *response sectors* P1 and P2 (see **Figure 2.2**) and includes the residential areas on the east and west side of Frenchman's Bay and the Brock Road industrial area respectively.

2.4.2 Detailed Planning Zone (DPZ)

a) The DPZ is a pre-designated area surrounding a *reactor facility*, incorporating the *Automatic Action Zone*, where pre-planned protective actions are implemented as needed on the basis of *reactor facility* conditions, *dose* modelling, and environmental monitoring, with the aim of preventing or reducing the occurrence of *stochastic effects*.

b) The PNGS DPZ is the area immediately surrounding the *reactor facility* extending out to an approximate radius of 10 kilometres.

c) The *Detailed Planning Zone* for PNGS is shown in **Figure 2.2**. It includes an area of the Regional Municipality of Durham bounded generally by Lynde Creek to the east, the 5th Concession to the north, and the Durham-Toronto boundary to the west, but excluding Pickering Nuclear Generating Station. It also includes an area of Toronto from the Durham-Toronto boundary west to Morningside Avenue and extending to Finch and Steeles Avenues in the north. The zone extends out into Lake Ontario to a radius of approximately 10 km from PNGS. The exact boundaries of the zone can be determined from **Annex A**.

2.4.3 Contingency Planning Zone (CPZ)

a) The CPZ is a pre-designated area surrounding a *reactor facility*, beyond the *Detailed Planning Zone* (see **Paragraph 2.4.2 c)** above), where contingency planning and arrangements are made in advance, so that during a *nuclear emergency*, protective actions can be extended beyond the *Detailed Planning Zone* as required to reduce potential for *exposure*.

b) The PNGS CPZ is shown in **Figure 2.3** and includes the area between 10 and 20 kilometres surrounding the *reactor facility*.

c) Additional CPZ guidance is provided in **Annex C**.

2.4.4 Ingestion Planning Zone (IPZ)

a) The IPZ (see **Figure 2.3)** is a pre-designated area surrounding a *reactor facility* where plans or arrangements are made to:

i. protect the food chain

ii. protect drinking water supplies

iii. restrict consumption and distribution of potentially contaminated produce, wild-grown products, milk from grazing animals, rainwater, animal feed

Note: Wild-grown products can include mushrooms and game.

iv. restrict distribution of non-food commodities until further assessments are performed

b) The PNGS IPZ encompasses all areas of the Regional Municipality of Durham, York, Peel, the City of Toronto, and City of Kawartha Lakes that lie within a 50 km radius of the Pickering Nuclear Generating Station. The *Ingestion Planning Zone* includes the *Automatic Action Zone*, *Detailed Planning Zone and Contingency Planning Zone*. **Figure 2.3** also shows the sub-zones of the *Ingestion planning zone*.

2.5 Response Sectors

2.5.1 The *Detailed Planning Zone* for PNGS is divided into 25 *response sectors* which fall into the following sector rings around the station:

Automatic Action Zone	Sectors P1 and P2 and lake sector P23
Inner Ring	Sectors P3 through P14 and lake sector P24
Outer Ring	Sectors P15 through P22, and lake sector P25

2.5.2 The boundaries of the DPZ *response sectors* are shown in **Figure 2.2**, and are detailed in **Annex A**.

2.6 Planning Data, Interface and Support

2.6.1 Planning Times for Radioactive Emissions

a) The time interval between the occurrence of an *accident* at PNGS and the commencement of an emission depends on the condition and functioning of the station *containment* system and on the effectiveness and timing of the actions taken by station *operators* to reduce the rate of vacuum structure repressurization thus prolonging the holdup and decay of *radioactive material* within *containment*.

b) For a normally functioning *containment* system, a minimum interval of 2 days (between the occurrence of the *accident* and the commencement of an emission) can be used for planning purposes.

c) When controlled *venting* is required (see **Section 4.6.6**), intermittent filtered releases of varying durations could continue for many weeks.

d) In the exceptional situation where the *containment* system was impaired, an emission could commence much earlier; in some cases, very soon after the *accident* and, the emission may be continuous.

2.6.2 Municipal Planning Data

Designated Municipality nuclear emergency plans *shall* detail the planning data necessary to undertake an effective *nuclear emergency* response. This data *should* be organized according to *planning zones*, sub-zone and response sector, and include:

a) population estimates (see **Annex B**)

b) institutional data

c) critical infrastructure

2.6.3 Evacuation Time Estimates

a) *Evacuation* time estimate studies *shall* be prepared and regularly updated to facilitate transportation planning and the management of transportation during a response.

b) *Evacuation* time estimate studies *shall* be developed in accordance with NUREG/CR-7002, **Criteria for Development of Evacuation Time Estimate Studies**, or similar standard.

c) *Evacuation* time estimates *shall* be based on current census data and future population growth projections until end of life of the *reactor facility* and take into consideration shadow *evacuations*[1].

d) The Province, *Designated Municipalities* and reactors facilities *shall* agree on their respective role in the development and maintenance of public evacuation time estimates and where they are documented.

e) *Nuclear emergency* response plans of provincial ministries, Designated Municipalities and reactor facilities shall identify:

   i. Their respective role in the development and maintenance of public *evacuation* time estimates.

   ii. Where public *evacuation* time estimates are documented.

2.6.4 Interface and Support

a) The PNGS operator *shall* provide an interface and support for OFMEM in accordance with the PNERP Master Plan and regulatory requirements[2].

b) For PNGS, this ensures that in the unlikely event of containment *venting* (see **Section 4.6.6**):

   i. There is a designated person onsite at all times with the authority for *venting*.

   ii. The Province, *Designated Municipalities* and Canadian Nuclear Safety Commission (CNSC) are consulted before undertaking any *venting* activity, unless *venting* must be performed in an urgent manner to protect the structural integrity of containment. In such a case, every effort *shall* be made to inform these stakeholders as early as possible.

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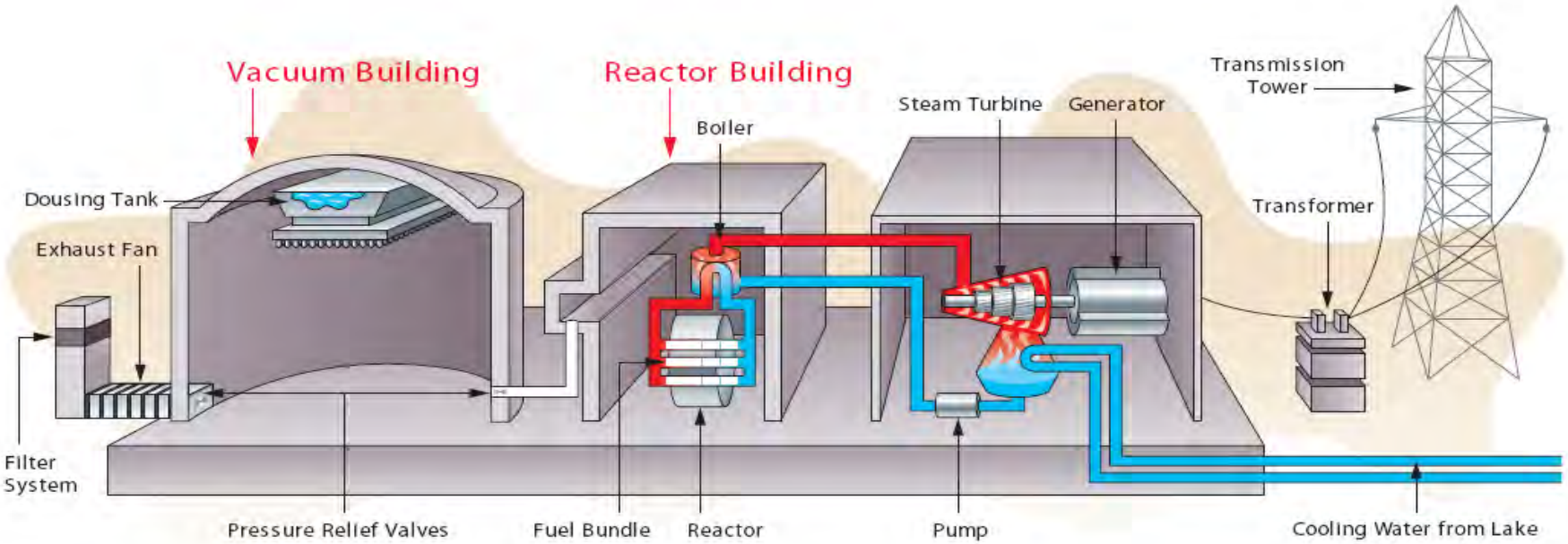
[1] "Shadow evacuation" is the term used to describe when people beyond the officially declared evacuation zone who are not directly affected by a nuclear emergency choose to voluntarily leave the area.

[2] Clause 2.2.4 of REGDOC 2.10.1 Nuclear Emergency Preparedness and Response.

Figure 2.1: Nuclear Generating Unit Schematic of a CANDU Pressurized Heavy Water Reactor[3]



[3] Nuclear Generating Unit Schematic – CANDU Pressurized Heavy Water Reactor is a generic diagram.

Figure 2.2: Detailed and Contingency Planning Zones

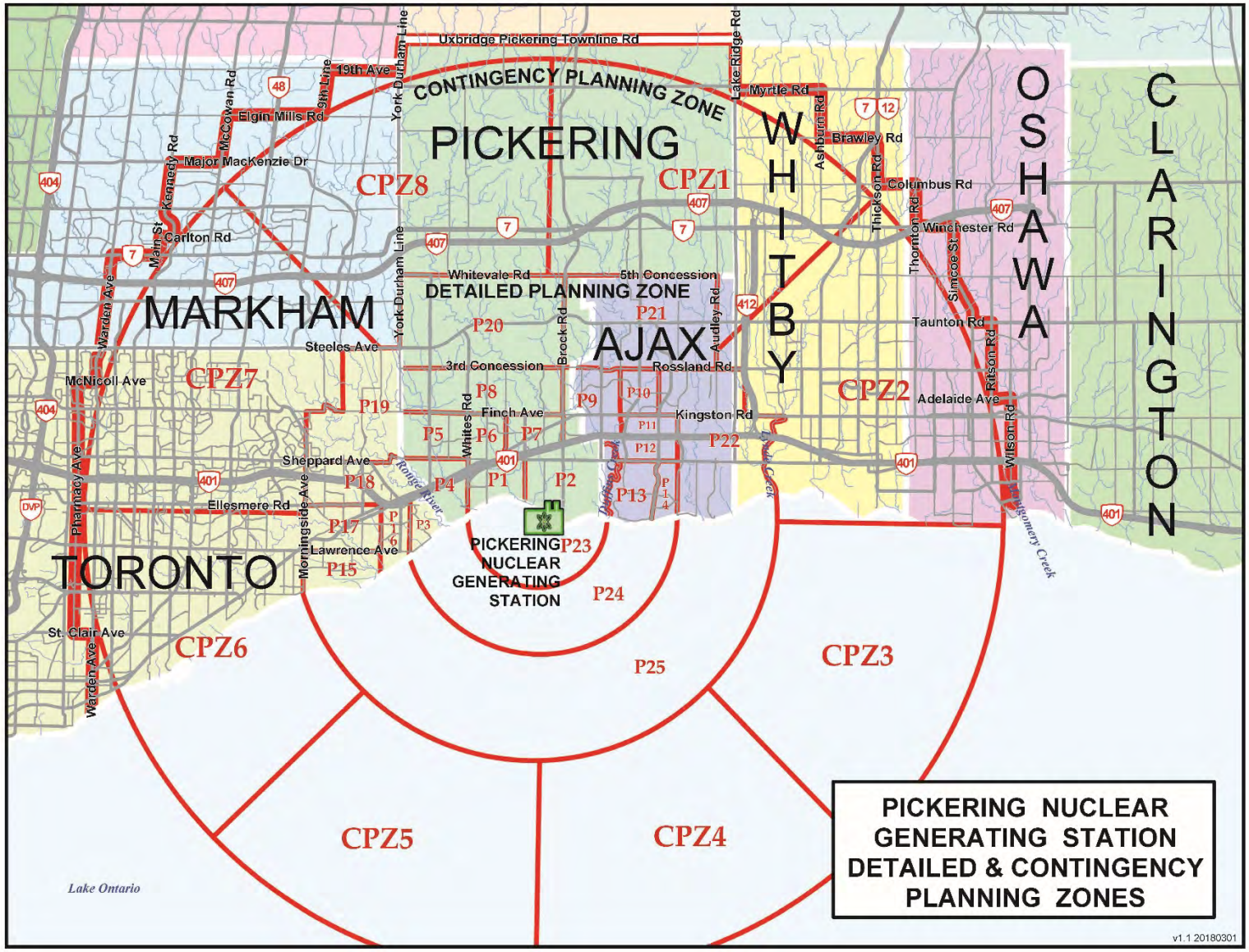
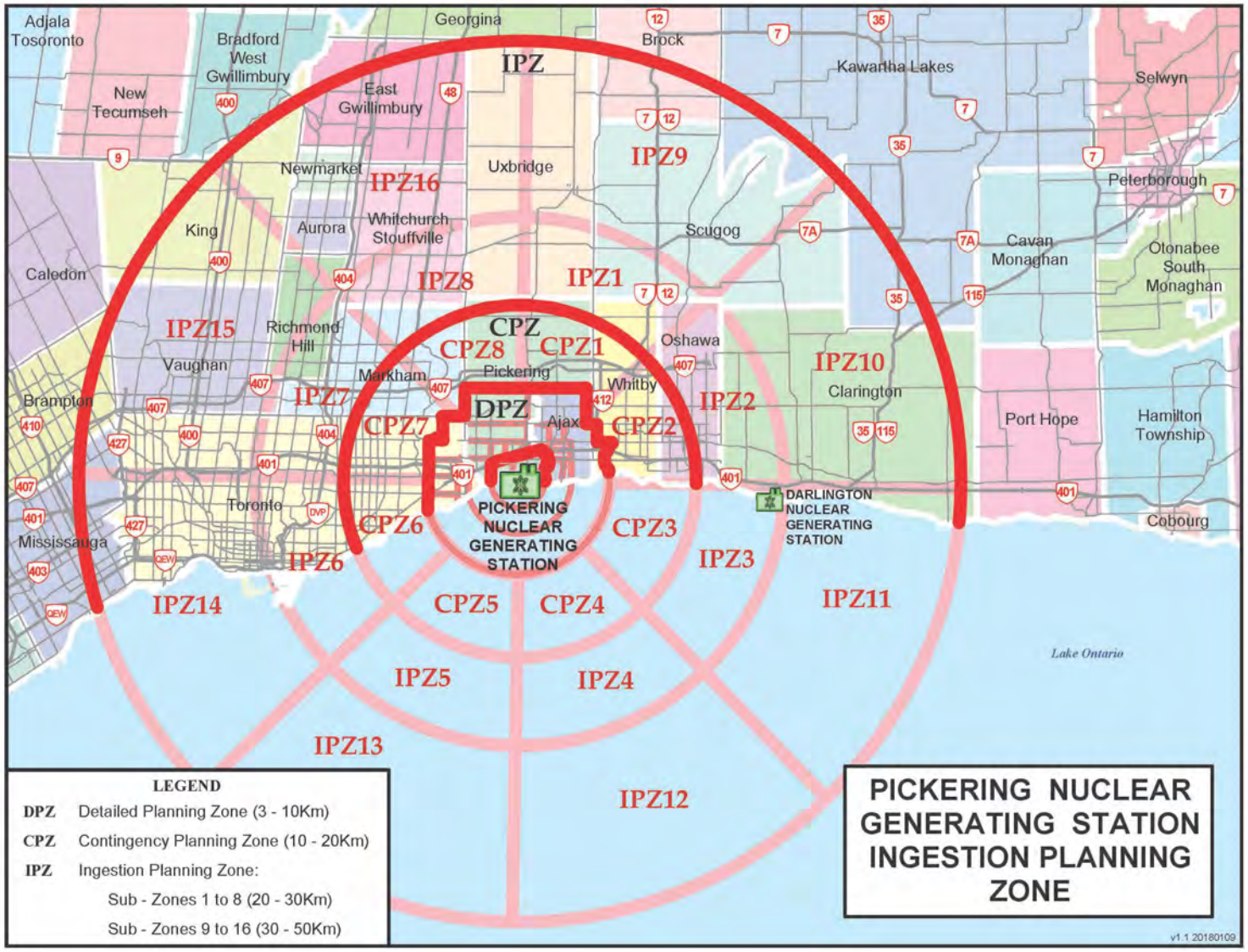


Figure 2.3: Ingestion Planning Zone



Chapter 3
EMERGENCY RESPONSE ORGANIZATION AND FACILITIES

3.1 Emergency Response Organization

3.1.1 The provincial *Emergency Response Organization* for managing a *nuclear emergency* at Pickering Nuclear Generating Station is shown in **Figure 3.1** and detailed in the **PNERP Master Plan, Chapter 4**.

3.1.2 Liaison Arrangements

a) To ensure liaison and coordination between different elements of the *Emergency Response Organization*, the following arrangements and agreements *shall* be made:

i. Each federal department and provincial ministry with a role in the *emergency* response to provide a representative to join the PEOC.

ii. The PNGS *operator* to provide:

- a liaison officer to the Municipal EOC (see **Paragraph 3.1.2 b)** below for functions)
- a corporate liaison representative to join the PEOC Operations Group
- a technical support staff to support the PEOC Nuclear Incident Group (NIG) if requested and resources are available

iii. Provincial staff to be deployed to join the Municipal Emergency Operations Centres (EOCs).

b) The role of the PNGS *operator* liaison officer at the Municipal EOC is to act as a link to the station in respect of the following types of issues:

i. PNGS operator support to the Municipalities, e.g., Monitoring and Decontamination Units and Emergency Worker Centres.

ii. Requests for mutual assistance (e.g., for additional *personal monitoring* resources, potassium iodine tablets, fire or paramedic services at Pickering Nuclear Generating Station etc.).

iii. Coordination of the *evacuation* of non-essential station staff, and of the movement of essential staff to and from the site.

iv. Facilitating the work of the *off-site* field monitoring teams.

v. Providing situational updates related to the *emergency* at the *reactor facility*.

vi. Providing technical briefings to the Municipal EOC staff in order to clarify the context within which the operational situation may be understood.

3.1.3 Unified Transportation Coordination Centre (UTCC)

A UTCC *shall* be set up and staffed for a PNGS *emergency* to implement the **UTMP** upon *notification* of either a partial or full *activation* response by the province.

3.1.4 Provincial Ministry Offices

The following regional, district and area offices of provincial ministries *shall* be prepared to respond to the *emergency* and provide the necessary assistance to the *Designated Municipalities*, as required by the **PNERP Master Plan**, **Annex I** and detailed in municipal plans or, as directed by their respective ministries:

a) Agriculture, Food and Rural Affairs - Brighton Office and Lindsay Office

b) Children, Community and Social Services - Toronto Region Office and Central East Region Office

c) Community Safety and Correctional Services, OPP - Highway Safety Division and Central Region

d) Environment, Conservation and Parks - Central Region Toronto District Office, Peterborough District Office and York Durham District Office

e) Labour - Central Region Office, Mississauga and Radiation Protection Service, Toronto

f) Municipal Affairs and Housing - Central Region Municipal Services Office

g) Natural Resources and Forestry - Aurora District

h) Transportation - Central Region, East Region and the Emergency Management and Planning Office

3.1.5 Designated Municipality Organization

Emergency plans for the *Designated Municipalities* (City of Toronto, Regional Municipality of Durham and City of Peterborough) *shall* describe their municipal *emergency response organizations* and its activation.

3.2 Contingency Provisions

3.2.1 The PEOC Commander issues *operational directives* to the *emergency* management and *response* organization through the centres in the tier below (see **Figure 3.1**). However, if for any reason, any of these centres is not functioning or is not responsive, the PEOC Commander may issue *operational directives* directly to any other element of the *emergency* management and *response* organization.

3.2.2 Similarly, *response* organizations are responsible for taking appropriate actions according to their respective plans, procedures and the requirements of the situation.

3.3 Municipal Emergency Facilities

a) Municipal *nuclear emergency* plans *shall* identify the location of the following *emergency* facilities and *shall* include provisions for their selection, staffing and resourcing:

- Reception Centres
- Evacuation Centres
- Monitoring and decontamination for evacuees may be accomplished either in a Reception Centre or may be set up separately
- Emergency Worker Centres (EWC). EWC locations should also be able to accommodate a command post for environmental monitoring operations of the Environmental Radiation and Assurance Monitoring Group (ERAMG)
- Emergency Information Centre

b) Municipal *nuclear emergency* plans *shall* also identify the location of alternate Municipal emergency facilities outside the CPZ.

3.4 Telecommunications

3.4.1 All stakeholder emergency plans *shall* describe how their *emergency* centres and facilities *are* linked via primary and backup communication systems which enable email and transfer of *emergency* information.

3.4.2 The PNGS *operator shall* establish primary and backup *communications* between its Corporate Emergency Operations Facility (CEOF) and the following centres:

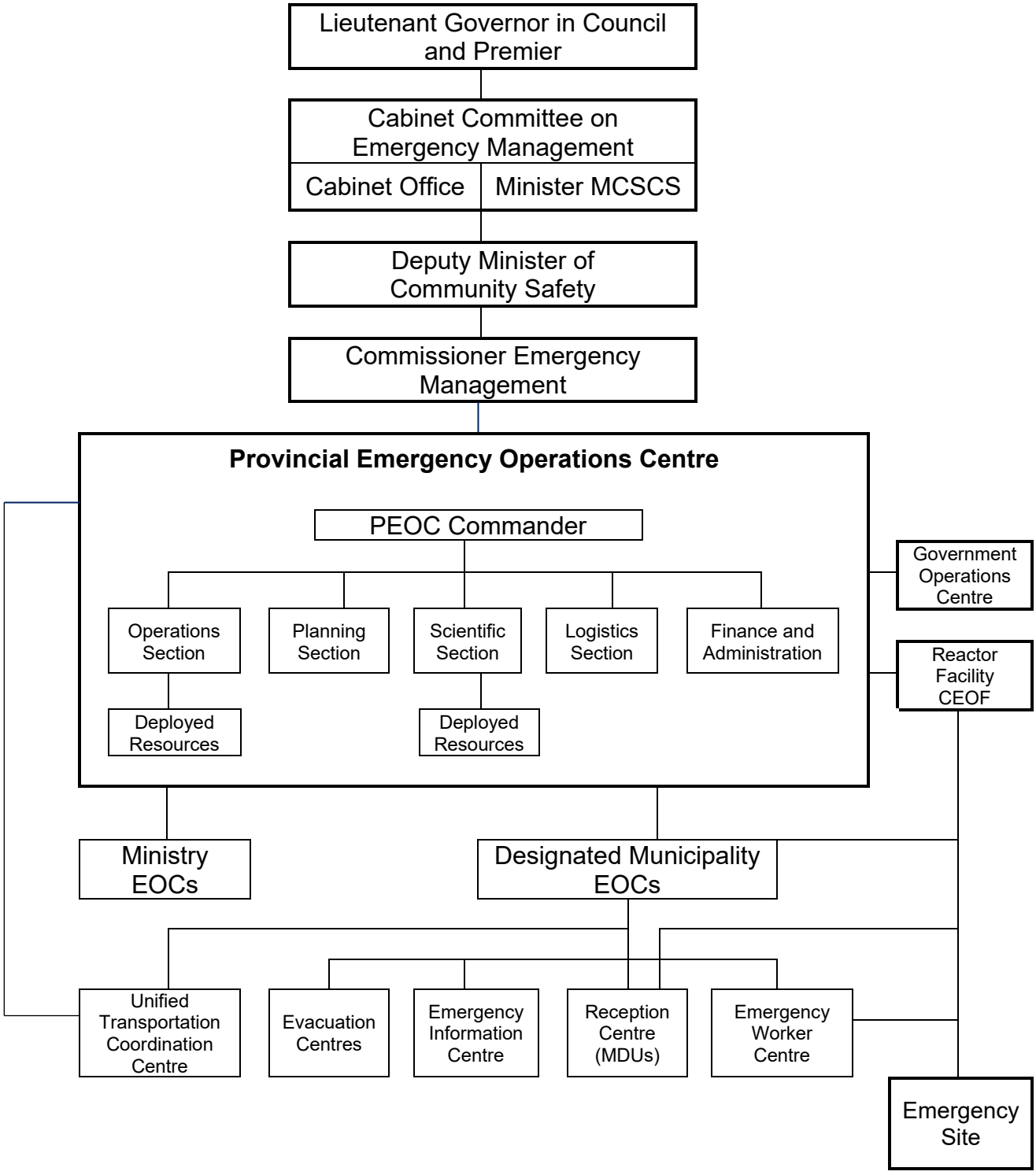
a) *Provincial Emergency Operations Centre* (PEOC)

b) Pickering Nuclear Generating Station (PNGS)

c) Municipal Emergency Operations Centres (EOCs)

3.4.3 All organizations and agencies involved in responding to a PNGS *nuclear emergency shall* ensure the availability of backup telecommunications systems.

Figure 3.1: Provincial Nuclear and Radiological Emergency Response Organizational Structure



Chapter 4
NOTIFICATIONS AND RESPONSE

4.1 Initial Notification

4.1.1 According to responsibilities outlined in federal legislation, (see **PNERP**, **Master Plan Paragraph 5.2.1**) regulations and agreement with the provincial government, the Pickering Nuclear Generating Station (PNGS) *operator shall* notify the pre-designated contact points (see **Paragraph 4.1.12** below) in provincial and municipal *emergency* organizations as soon as conditions arise at the facility which require such *initial notification* under the criteria described in **Table 4.1** and, as incorporated in facility procedures.

4.1.2 The form and content of the *initial notification shall* be determined by the Commissioner of Emergency Management.

4.1.3 The PNGS *operator shall* make a *notification* to the designated provincial and municipal contact points within 15 minutes of categorizing the event.

4.1.4 The initial *notification* message from the PNGS *operator shall* include:

a) the *notification* category

b) recommended default *protective measures*

4.1.5 Where more than one criteria are applicable, the highest category triggered *shall* be reported in the *notification*. The *notification* message *shall* not be delayed to permit an accurate assessment of the applicable category.

4.1.6 In the case of a *General Emergency* or *On-Site Emergency notification*, the message must state whether an emission is ongoing or if not, give a best estimate of when it is expected to commence and the wind direction at the time of the *notification*.

4.1.7 During the initial stage (see **Paragraph 4.1.8** below) of an *emergency*, the *General Emergency* notification category (received by the PEOC Duty Officer) initiates the implementation of *off-site* default *protective measures*.

4.1.8 The initial stage of an *emergency* is defined as the earlier of:

a) The first 4 hours after the *initial notification*, or

b) Once ongoing reporting by the PNGS CEOF to the PEOC Scientific Section is established.

4.1.9 If during the initial stages of an *emergency*, the assessment of the *on-site* situation changes to warrant a different category (and to recommended *protective measures)* than those initially notified, then the PNGS *operator shall* immediately issue such changes to the designated provincial and municipal contact points.

4.1.10 Once ongoing reporting is established between the PNGS CEOF and the PEOC Scientific Section, any changes to the *notification* category or recommended *protective measures shall* be included in the regular data transmittals sent by the PNGS CEOF to the PEOC Scientific Section and *shall* no longer be transmitted to the provincial and municipal contact points.

4.1.11 The PNGS *operator* cannot terminate or cancel a *notification* once it has been made. Such a *notification shall* automatically lapse when the provincial response to it is formally terminated (see **Paragraph 4.2.3** below).

4.1.12 Contact Points

a) Contact points and phone numbers shall be pre-determined and routinely validated to ensure availability.

b) The provincial contact point *shall* be the PEOC Duty Officer.

c) The following municipal contact points *shall* be set out in the municipal plans:

i. A contact point to receive an *initial notification* anytime, 24 hours per day, 7 days per week.

ii. A municipal *emergency* response staff person who can be contacted anytime 24 hours per day, 7 days per week for passage of information and monitoring of the situation following the initiation of a *notification*.

4.2 Initial Provincial and Municipal Response

4.2.1 The initial provincial response to a *notification* from PNGS *shall* depend on the category (and other relevant information) contained in the *notification* message (see **Table 4.1**).

4.2.2 Within 15 minutes of the receipt of an *initial notification*, the PEOC Commander *shall* decide on the initial response level to be adopted and inform the municipal contact point(s). This level *should* normally be the one linked to the category of the *notification* received (see **Table 4.2**) unless another level is judged to be more appropriate.

4.2.3 The PEOC Commander or PEOC Operations Chief may adopt another provincial response level as appropriate including termination of the provincial response. All *stakeholders shall* be notified of any such change.

4.2.4 The initial (and any subsequent) response level to be adopted by the *Municipalities* and other organizations *shall* be communicated by the PEOC Commander (see **Paragraph 4.2.2** above) to all *stakeholders*. The general municipal response for each level is outlined in **Table 4.2**; the specific response *shall* be described in the municipal plans.

4.3 Internal Notifications

4.3.1 Each organization or agency required to respond to a *nuclear emergency shall* have an internal *notification* system to inform all concerned staff of the imminence or occurrence of an *emergency* under this plan, and of the appropriate response to the *notification*.

4.3.2 Each jurisdiction and organization receiving a *notification* of an *activation response* (either partial or full) *shall* issue an appropriate *internal notification* to its units and individuals who are required to respond. The *notification shall* indicate the level of *activation* to be adopted.

4.3.3 The PEOC and each jurisdiction and organization required to respond and issue an internal or *external notification* (see **Section 4.4** below) *shall* prepare a *notification* procedure and list of recipients.

4.3.4 PEOC Notifications

a) If the PEOC is to be activated (whether fully or partially), then the PEOC Commander *shall* issue an appropriate *notification* (including an indication of the level of *activation*) to at least one pre-designated contact point in each of the following jurisdictions and organizations:

   i. City of Toronto

   ii. Regional Municipality of Durham

   iii. each provincial-level organization required to respond to the *emergency*

   iv. *Host Municipalities*

   v. Pickering Nuclear Generating Station

   vi. PNGS *Operator* (CEOF)

b) Additionally, the PEOC Commander *shall* notify:

   i. PEOC staff

   ii. Emergency Information Section staff

4.4 External Notifications

4.4.1 Additional organizations or agencies which might be affected by a *nuclear emergency* under this plan, or which may be required to assist in responding to it, *should* be notified at an appropriate stage by their links in the *Emergency Response Organization*. As such, upon adoption of an *activation response* (partial or full), *external notifications shall* be carried out as detailed below. The *notification* must indicate the level of *activation* being adopted.

4.4.2 The PEOC Commander *shall* ensure the following are notified:

a) the federal *Government Operations Centre* (GOC) and the Federal Nuclear Emergency Plan (FNEP) Duty Officer who *shall* then complete the *notifications* listed in **Paragraph 4.4.3** below

b) the Province of Quebec (Sûreté du Quebec)

c) the State of New York Emergency Management Agency

d) the State of Ohio Emergency Management Agency

e) the State of Michigan Emergency Management Agency

f) Canada News Wire/National Alert Aggregation and Dissemination System (NAADS)

g) GO Transit

h) Bell Canada

i) wireless phone providers

4.4.3 As directed by the FNEP Duty Officer, the federal Government Operations Centre (GOC) shall notify:

a) Natural Resources Canada (NRCan)

b) Royal Canadian Mounted Police (RCMP)

c) Privy Council Office (PCO)

d) Transport Canada

e) Department of National Defence (DND)

f) Canadian Coast Guard (which *shall* notify the US Coast Guard under agreed protocols)

g) CNSC Duty Officer

h) Global Affairs Canada

i) Canadian Food Inspection Agency (CFIA)

j) Canada Border Services Agency (CBSA)

k) Indigenous Services Canada (ISC)

l) Air Traffic Control

m) CN Rail

n) CP Rail

o) VIA Rail

p) U.S. Department of Homeland Security

q) international organizations under existing agreements, conventions and departmental *emergency* plans

4.4.4 Other agencies and organizations *shall* be notified by the following:

a) Ministry of Agriculture, Food and Rural Affairs *shall* notify the Dairy Farmers of Ontario.

b) Ministry of Children, Community and Social Services *shall* notify the Red Cross, Ontario Zone.

c) Ministry of Natural Resources and Forestry *shall* notify:

i. Central Lake Ontario Conservation Authority

ii. Toronto and Region Conservation Authority

d) Municipal plans *shall* include provisions for the following *external notifications*:

i. school boards

ii. paramedic services

iii. local utilities (e.g., hydro, gas, water)

iv. local branches of voluntary organizations

4.5 Provincial Response Levels

4.5.1 The provincial response level adopted depends on the *notification* category received from PNGS (see **Table 4.2**).

4.5.2 Reportable Event and Abnormal Incident

a) In the event of a *Reportable Event notification* from PNGS the provincial response level adopted *should* be Routine Monitoring, unless the PEOC Commander decides otherwise.

b) In the event of an *Abnormal Incident notification* from PNGS, the provincial response level adopted *should* be Enhanced Monitoring, unless the PEOC Commander decides otherwise.

c) In these cases, the *notifications* and level of staffing *shall* proceed according to **Table 4.2**, unless the PEOC Commander decides otherwise.

d) The PEOC Commander *shall* ensure the applicable *stakeholders* are notified when a response to a Reportable Event or *Abnormal Incident* has been terminated.

4.5.3 On-site Emergency

An On-site Emergency can result in either a partial or full *activation* provincial response (see **Figure 4.1**), depending on the source of the *accident* and the prospects for the resulting emission:

a) An On-site Emergency with no emission occurring *shall* normally result in a partial *activation* response.

b) An On-site Emergency with a (non-reactor) emission occurring or expected within 12 hours *shall* normally result in a full *activation* response.

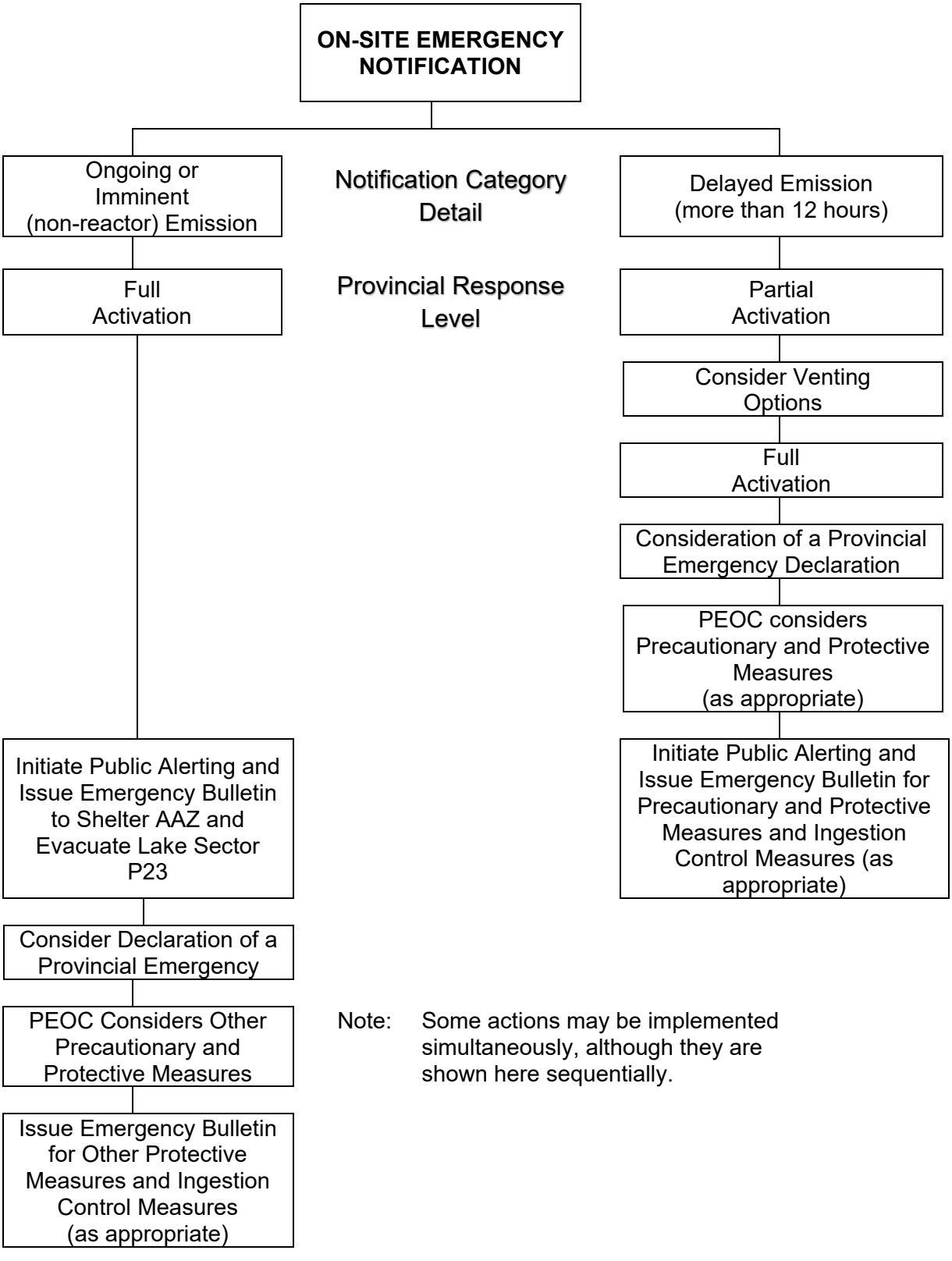
4.5.4 General Emergency

A *General Emergency notification* from PNGS *shall* result in a full *activation* response (see **Figure 4.2**) as it denotes an emission from a reactor can result due to fuel and *containment* failures.

4.5.5 The remainder of this chapter therefore deals with the operational response to an *accident* at PNGS which results in, or has the potential to result in, an emission of *radioactive material* to the atmosphere, and therefore requires a partial or full *activation response*.

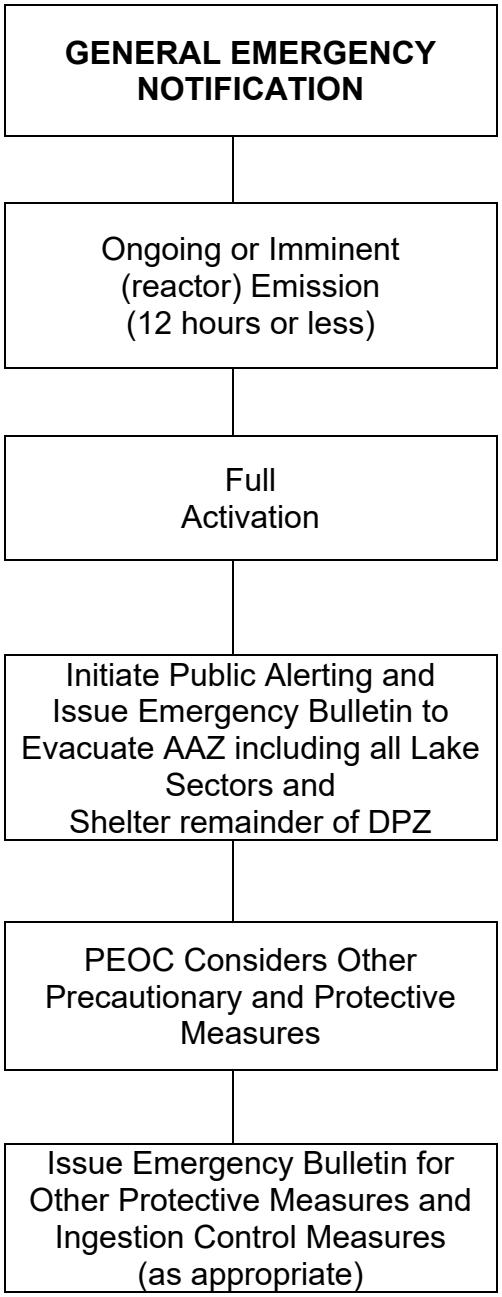
4.5.6 The partial and full *activation* response to a *nuclear emergency* is described below in relation to the three successive phases defined by the **PNERP Master Plan, Section 5.9**.

Figure 4.1: Initial Provincial Response to an On-Site Emergency Notification



Note: Some actions may be implemented simultaneously, although they are shown here sequentially.

Figure 4.2: Initial Provincial Response to a General Emergency Notification



Provincial Response Level

Note: Some actions may be implemented simultaneously, although they are shown here sequentially.

4.6 Early Phase Response

4.6.1 The early phase:

a) Begins with an *initial notification* of an *emergency* prior to, or during a radioactive *release.*

b) Lasts anywhere from hours to days and in the event of, an ongoing or imminent emission, *should* involve the implementation of automatic default *protective measures* (see **Table 4.3**).

c) Ends when the radioactive *release* is brought under control and reliable environmental *radiation* monitoring is available to be used for protective action decision-making.

d) *Shall* be terminated and the intermediate phase *shall* begin based on the criteria in **Section 4.7** below.

4.6.2 The operational response in this phase differs depending on whether the initial provincial response level is partial or full *activation*.

4.6.3 Partial Activation

a) A PEOC partial *activation* response (see **Figure 4.1**) is adopted when it is expected that a radioactive *release* will occur at some point in the future and therefore protective or *operational measures* (other than monitoring and assessment of the situation) are not likely to be required within 36 hours.

b) When the PEOC is partially activated, initial actions include:

i. *Notification of the emergency* management organization and set up and full staffing of the PEOC and the Municipal EOCs to monitor and assess the situation on a continuous basis.

ii. *Activation* of the Ministry EOCs and Unified Transportation Coordination Centre and staffing as appropriate to the situation.

iii. *Activation* of the *Emergency Information Centres* (EICs) with staffing at an appropriate level. Provincial staff to be dispatched, as appropriate.

iv. All *emergency response* personnel not immediately required *should* be placed on standby. This provision *should* ensure that personnel can be quickly contacted when needed to report to their duty stations.

v. Other *emergency* centres *should* be readied to a level where they can become fully operational without undue delay, when required. Specific levels of readiness *shall* be described in the municipal plans.

vi. Consideration *shall* be given to issuing an *emergency* bulletin(s) and news release(s).

c) Technical assessments of the *accident* situation and projected *radiation doses shall* be carried out on a regular basis by the PEOC Scientific Section, as described in **Section 4.6.5** below.

d) The technical assessments carried out by the PEOC Scientific Section, as well as inputs from the other PEOC sections *shall* be compiled by the PEOC Planning Section Chief into recommendations for protective action decision-making, to the PEOC Commander.

e) The PEOC Commander, in consultation with the PEOC Command Section and select stakeholder organizations (including MOHLTC, *Designated Municipalities* and others deemed appropriate), *shall* consider and decide on the need for *operational measures* as well as future protective actions and ensure that all stakeholders are so informed.

f) If the *emergency* situation is resolved and the potential for *off-site* consequences is eliminated, the PEOC Commander *shall* downgrade the provincial response level, as appropriate.

g) Alternatively, the PEOC Commander *shall* upgrade the response level to full *activation* when a radioactive emission seems likely to occur in 36 hours or less or, as deemed appropriate.

4.6.4 Full Activation

a) A PEOC full *activation* response *should* be adopted as a result of:

i. An *initial notification* from PNGS stating that an emission is ongoing or imminent (On-site or *General Emergency*), or

ii. An escalation of an existing *emergency* situation, where an emission is now expected in 36 hours or less

b) The following actions *shall* be initiated upon adoption of a full *activation response*:

i. All emergency operations centres, Emergency Information Centres, Reception Centres, Evacuation Centres, Emergency Worker Centres and Monitoring and Decontamination Units are fully staffed and operational.

ii. All emergency response personnel from **Paragraph i.** above immediately report to their places of duty.

iii. Public alerting is initiated and emergency bulletins issued concurrently (see **Sections 6.2** and **6.4**).

iv. Operational directives (or emergency orders) issued for protective actions per **c)** or **d)** below, as appropriate.

v. PEOC Commander advises government of the need for a provincial emergency declaration (see **PNERP Master Plan Section 1.5.1**).

vi. Provincial Chief Emergency Information Officer (PCEIO) *shall* consider establishing a *Joint Information Centre* as necessary (**Paragraph 6.5.2 e)**).

c) Where the full *activation* response level is adopted as a result of an *initial notification* from PNGS that an emission is ongoing or imminent (see **Paragraph (a) (i)** above), the default actions noted in **Table 4.3** *shall* be implemented, unless there are good reasons for modifying the response. This default response is undertaken due to the potential lack of detailed information or plant data together with a lack of available time for analysis.

d) Where an escalating event results in the upgrade to a full *activation* response (see **Paragraph (a) (ii)** above), data gathering and analysis are already being undertaken by the *emergency response organization*, with a protective action decision-making process in place. Therefore, in this escalating type of scenario, the technical and operational assessments and the recommendation process (see **Section 4.6.5** below) replaces the need for default actions.

4.6.5 PEOC Scientific Section Technical Assessments

In the early phase, prior to or during a radioactive emission, the PEOC Scientific Section *shall* undertake technical assessments in accordance with PEOC Scientific Section procedures which *shall* serve as input into the PEOC Planning Section recommendations for Command decision-making. Examples of technical assessments include:

a) Accident Assessments

The Nuclear Incident Group (NIG) of the PEOC Scientific Section *shall* receive hourly plant status and data from the PNGS EOC, through agreed to transmission systems (and backup), and *shall* on an ongoing basis:

i. Evaluate the status of relevant station systems and make ongoing assessments of possible *accident* progressions, considering both positive and negative outcomes (see **paragraph ii** below).

ii. Monitor the progress of station vacuum structure repressurization and continually forecast the date and time when its pressure could reach, firstly, the minimum level required for the Filtered Air Discharge System (FADS) operation and, secondly, the level at which FADS operation becomes necessary.

iii. Analyze *venting* data and make projections to inform *venting* decision-making by the PEOC Commander and stakeholders (see **Section 4.6.6** below).

iv. Liaise with CEOF staff and undertake a technical projection of the maximum distance from the *reactor facility* at which the *generic criteria* (see **PNERP Master Plan, Annex E, Appendix 1**) for *protective measures* against the *plume* are likely to be reached during the anticipated duration of the emission (allowance *should* be made for the effects of early *venting*, if applicable):

- *evacuation*
- *sheltering-in-place*
- *iodine thyroid blocking*

b) Condition of Station Systems

i. **Table 4.3** describes four main categories for the condition of station systems along with some examples of each. It can provide a baseline for making appropriate judgements or, if time and adequate information are not available, it may be used to determine default measures.

ii. In an actual event, the estimate of station conditions may not conform exactly to the various sets of conditions given in **Table 4.3**. In such a case, the default *protective measures* listed in the table, may be appropriately modified.

iii. The PEOC Scientific Section Chief *shall* determine and make recommendations to the PEOC Commander for approval on:

- the set of *protective measures* that best match the current conditions
- whether current meteorological conditions warrant any change to the distance out to which *protective measures* are advised
- the DPZ and CPZ response sectors likely to be affected by the emission

c) Exposure Levels

The PEOC Scientific Section *shall* make an assessment as to whether the *dose* in any sector(s) is likely to require the *activation* of the **Radiation Health Response Plan** (see **Paragraph 6.9.7**).

d) Subsequent Technical Assessments

As more data and projections become available, the PEOC Scientific Section shall continuously update the assessments made in order to establish whether any additional *protective measures* are required.

4.6.6 Venting of Containment

a) During a *Design Basis Accident* (DBA), the holdup period of any radioactive material within the station *containment* structure (e.g., vacuum building) permits the *venting* of said contained radioactivity in a controlled manner and in a safe direction, i.e., over the lake (refer to **PNERP Master Plan, Annex G**).

b) During a *Beyond Design Basis Accident* (BDBA), the holdup period of any radioactive material may be significantly reduced and radioactive material may be *released* in an uncontrolled manner. BDBAs are categorized as *General Emergencies* and as such the PEOC assumes full *activation* as described in **Section 4.6.4** above.

c) For all *accidents*, the PEOC Commander may decide at any time to upgrade to full *activation* as conditions warrant however the Commander *should* upgrade no later than 36 hours prior to *venting*.

d) For all *accidents*, the PNGS *operator shall* include, in each hourly report to the PEOC, an estimate of the time at which the vacuum building pressure would reach the minimum level at which a venting system can be operated.

e) The time interval between the occurrence of the *accident* and *containment* pressure reaching this minimum level may depend on the condition and behaviour of the *containment* system. With no impairment to *containment*, this time interval is expected to be about 48 hours (an impaired *containment* could significantly reduce that time).

f) For all accidents, the PNGS operator *shall* consult with the Province, *Designated Municipalities* and CNSC before undertaking any *venting* activity, unless *venting* must be performed in an urgent manner to protect the structural integrity of containment (see **Paragraph 2.6.4 b)**). In such a case, every effort shall be made to inform these stakeholders as early as possible.

g) The PEOC Commander *should* consider, in consultation with the PNGS *operator*, the CNSC, City of Toronto and Regional Municipality of Durham, whether *venting* over Lake Ontario (according to the considerations in the **PNERP Master Plan, Annex G**), would be feasible and advisable. Detailed procedures for such decision-making *should* be developed in consultation with the above agencies and incorporated in the PEOC procedures for the PEOC Scientific, Operations and Command Sections, as appropriate.

h) Environmental Radiation Monitoring

If *venting* over Lake Ontario, ground monitoring teams from PNGS *shall* complete radiological surveys following the shoreline, out to 20 km on either side of the plant to detect any “blow back” of radioactivity towards land during *venting*. Any radioactivity detected *shall* be immediately reported by the PNGS *operator* to the PEOC.

4.6.7 Early Phase Protective Action Decision-Making

a) The PEOC Scientific Section’s technical assessment of the situation *should* produce a projection of the maximum distance from PNGS at which the *generic criteria* for *evacuation*, sheltering-in-place and KI ingestion are likely to be reached during the anticipated duration of the emission.

b) The PEOC Planning Section *shall* undertake an evaluation of this technical assessment, taking into account operational and public policy considerations, and shall prepare a preliminary assessment regarding the need to implement these measures, proposed timings, and the area within which these measures should be taken.

c) These assessments *shall* be continually updated and, as soon as a reasonably certain picture of the *evacuation* (and other *protective measures*) distance is achieved, the PEOC, through the Command Section, *shall* consult with

applicable stakeholders (*Designated Municipalities*, host and support *Municipalities*, federal departments, the *reactor facility*).

d) Command decisions on protective action *shall* be communicated to the *emergency response organization* (see PEOC Command Section procedures) and the applicable *emergency* bulletin(s) *shall* be issued (per PEOC Operations Section procedures).

4.7 Intermediate Phase Response

4.7.1 The intermediate phase begins once the radioactive *release* has been brought under control and reliable environmental *radiation* monitoring is available for use in protective action decision-making.

4.7.2 Following the radioactive emission, the PEOC Scientific Section's input into the protective action decision-making process *shall* no longer be based on modelling the *projected doses* or on default measures but rather on the tangible results of environmental *radiation* monitoring.

4.7.3 The PEOC Scientific Section *shall* undertake, and continuously update, the following assessments:

a) *Off-site* environmental *radiation* monitoring undertaken by the ERAMG *shall* produce a picture of the *contamination* situation.

b) The PEOC Scientific Section Chief *shall* make technical recommendations for protective action (*exposure* and *ingestion control* measures) based on the results of the actual *contamination* levels as compared against the *Operational Intervention Levels* (OILs) (per the **PNERP Master Plan, Annex E, Appendix 2**).

c) The PEOC Scientific Section Chief *shall* make recommendations regarding sector safety status on behalf of *emergency workers* operating in the area.

d) The intermediate phase operations of the PEOC Scientific Section *shall* be detailed in the Scientific Section procedures.

4.7.4 Intermediate Phase Protective Action Decision-Making

a) The PEOC Planning Section *shall* undertake an assessment of these Scientific Section technical recommendations, in light of operational and public policy considerations, and *shall* prepare recommendations for the PEOC Commander regarding the *protective measures*, areas where they *should* be implemented, and implementation timings.

b) These assessments *shall* be continually updated and, as soon as a reasonably certain picture of the *evacuation* (and other *protective measures*) distance is achieved, the PEOC Commander shall advise all stakeholders of the protective action strategy to be undertaken. If time is available, the PEOC Commander shall undertake prior consultation with applicable stakeholders on the protective action strategy recommendations.

c) Planning for the management of radioactive waste (see **Section 6.11**) generated by the *emergency should* preferably begin during the intermediate phase.

4.8 Transition to the Recovery Phase

4.8.1 During the *recovery* phase actions will commence to restore the affected area to pre-*emergency* conditions and to scale back the *emergency response organization*.

4.8.2 As there may not be a clear distinction between phases, with *emergency* response operations occurring in all three, planning for *recovery* should begin as soon as practical.

4.8.3 Stakeholder *recovery* plans *should* include measures to address the following as applicable to their organization:

a) *recovery* organization structure

b) care for persons exposed or contaminated

c) long-term relocation

d) resettlement and return of evacuees

e) long-term support for those living in contaminated areas

f) *decontamination* and reconstruction of property damaged during the *emergency*

g) economic impact issues and improvement plans

4.8.4 Stakeholder *recovery* plans *should* be prepared in advance and conform to the provincial *recovery* plan.

Table 4.1: Initial Notification Categories and Criteria

CATEGORY	CRITERIA (Alternative)	EXAMPLES
REPORTABLE EVENT	1. Any event or condition that reduces the *reactor facility's* capability to *mitigate* an *emergency on-site*, and which persists for longer than the allowable time limits.	1A. Level 1 or 2 impairment of a special safety system, which persists for more than 4 hours. 1B. Entry into emergency operating procedures.
	2. Any event or condition that reduces the *reactor facility*'s capability to provide the agreed *off-site emergency* support, and which is expected to persist for over 8 hours, or actually does so.	2. Reduced ability to: A. Carry out *off-site* field monitoring. B. Provide *source term* data. C. Provide required *off-site ERO* staff.
	3. Natural, toxic, flammable, destructive or other phenomena which have the potential to lead to a minor* break in the physical integrity of the nuclear heat transport system boundary or the moderator system.	3A. Equipment failure. 3B. Extreme environmental conditions. 3C. Earthquake. 3D. Fire or explosion.
	4. Unexpected or unplanned *activation* of the emergency cooling injection system or the *containment* system (including box-up).	4A. Unexpected or unplanned *activation* of the ECI system component that does not result in injection. 4B. Unexpected or unplanned *activation* of the *containment* system. 4C. Excludes events initiated during testing.
	5. Declaration of a Station Emergency, *with no potential for off-site effects*.	5. As per nuclear emergency procedures.
	6. Any credible publicly announced threat to, or attempted or actual breach of, the facility's security that threatens its safe operation.	6A. A publicized bomb threat. 6B. A breach or attempted breach of the protected area.
ABNORMAL INCIDENT	1. A minor* break in the physical integrity of the nuclear heat transport system boundary with no *fuel failures* (actual or likely).	1. LOCA on one or more units, without fuel failures and with or without *containment* impairment.
	2. Natural, toxic, flammable, destructive or other phenomena which have the potential to lead to the major* break specified in Item 1 under *On-Site Emergency*.	2. Similar causes as for # 3 under REPORTABLE EVENT.
	3. *Activation* of the emergency cooling injection system or the *containment* system (including box-up) due to a process system upset which is not reportable under any other category.	3A. *Activation* of the ECI system component that results in injection. 3B. *Activation* of the *containment* system on high activity or pressure.
	4. Declaration of a Station Emergency due to an occurrence which has the potential to result in *off-site* effects.	

* To be defined in *reactor facility* procedures.

Table 4.1: Initial Notification Categories and Criteria (continued)

CATEGORY	CRITERIA (Alternative)	EXAMPLES
ON-SITE EMERGENCY (Note: A *notification* with this category **must** state whether an emission is ongoing or, if not, give a best estimate of when it is expected to commence).	1. A major* break in the physical integrity of the nuclear heat transport system boundary, the moderator system, or the irradiated (or spent) fuel handling and storage system, with *fuel failures** but **with a fully intact and functioning containment system.** 2. An abnormal emission* of *radioactive material* to the atmosphere from any non-reactor source. 3. An event or condition which has the potential to lead to a ***General Emergency*** criterion, concurrent with the loss of the ability to detect or control such a development. 4. *Hostile action* in the protected area resulting in actual or potential loss of control over station safety or safety related systems **but excluding reactor control systems**.	1A. LOCA with *fuel failures* on one or more units. 1B. Fuelling machine *accident*. 2. Spent fuel bay *accident*. 3A. Loss of all AC power. 3B. Extreme environmental conditions. 3C. Earthquake damage. 3D. Fire or explosion. 3E. Entry requirements met for SAMG or EME to maintain fuel cooling.
GENERAL EMERGENCY (Note: A *notification* with this category **must** state whether an emission is ongoing or, if not, give a best estimate of when it is expected to commence).	1. Damage to reactor fuel leading to the *release* of radioactivity from the fuel **coincident with** the failure, impairment, or bypass of *containment*, resulting in an atmospheric emission or, a reasonable expectation of an emission within the next 12 hours. 2. *Hostile action* in the protected area resulting in actual or imminent loss of the ability to achieve and maintain the reactor in a cold *shutdown state*.	1. LOCA with actual or imminent *fuel failures* on one or more units and impaired *containment*. 2. Inability to control or maintain reactivity control or fuel cooling.

* To be defined in *reactor facility* procedures.

Table 4.2: Initial Provincial and Municipal Response

INITIAL NOTIFICATION	INITIAL PROVINCIAL RESPONSE	INITIAL MUNICIPAL RESPONSE
REPORTABLE EVENT	ROUTINE MONITORING 1. *Provincial Emergency Operations Centre* (PEOC) *shall* notify the municipal contact point(s), *reactor facility operator*, and others as appropriate, and *shall* monitor the situation. 2. PEOC Scientific staff is consulted, if appropriate. 3. If and when appropriate, Emergency Information Section (EIS) staff issues news release(s).	*Emergency* response staff remain in contact with the PEOC, and monitor event.
ABNORMAL INCIDENT	ENHANCED MONITORING 1. PEOC *should* adopt Enhanced Monitoring and *shall* inform the municipal contact point(s), *reactor facility operator*, and any other organizations affected. 2. *External notifications* to Michigan, New York, Ohio and Quebec are made. 3. PEOC to set up a duty team consisting of operations staff, scientific staff, *reactor facility operator* representative(s), EIS staff, and others as required. 4. If and when appropriate, EIS staff *shall* issue news release(s). 5. Provincial staff are notified to remain available to report in for duty.	*Emergency* response staff monitor event, preferably from Municipal Emergency Operations Centres (EOCs).
ON-SITE EMERGENCY (*No emission occurring*)	PARTIAL ACTIVATION 1. PEOC *should* adopt partial *activation* response (for details, see **Section 4.6.3**), and *shall* initiate the appropriate internal and *external notifications* (**Section 4.3** and **Section 4.4** respectively), including the municipal contact points and the host communities. 2. If a reactor emission is expected to occur in 36 hours or less, PEOC *should* consider adopting full *activation* response and consider the need to implement the immediate measures per *General Emergency* below. 3. PEOC *shall* be fully staffed. Consideration *shall* be given to issuing an *emergency bulletin* (**Section 6.4)**, news release or both. 4. Ministry EOCs and Unified Transportation Coordination Centre (UTCC) to be established and appropriately staffed.	1. Issue *notification* placing municipal *Emergency Response Organization* on standby. 2. Municipal EOCs fully staffed. 3. *Emergency Information Centres* (EICs) to be established. 4. Other emergency centres readied to become operational without undue delay.

Table 4.2: Initial Provincial and Municipal Response (Continued)

INITIAL NOTIFICATION	INITIAL PROVINCIAL RESPONSE	INITIAL MUNICIPAL RESPONSE
ON-SITE EMERGENCY *(Emission Ongoing or expected within 12 hours)*	FULL ACTIVATION 1. PEOC *should* notify and require the municipal contacts to activate the *public alerting* system (**Section 6.2**). 2. PEOC *should* adopt full *activation* (**Section 4.6.4**), and *shall* initiate the appropriate internal and *external notifications* (**Section 4.3** and **Section 4.4** respectively), including the host *community*. 3. PEOC *shall* issue the appropriate *emergency bulletin* (**Section 6.4**). 4. PEOC *shall* issue *operational directives* implementing the following *operational measures*, unless there are good reasons for modifying this response, for: a) Sheltering (**Section 5.3.4**) in the *Automatic Action Zone.* b) Suspension of road and rail traffic through the *Automatic Action Zone*. c) Clearance of all boaters in Lake Sector P23. 5. PEOC *shall* assess the situation for further action. 6. PEOC *shall* issue further *emergency bulletins*, as appropriate (**Section 6.4**). 7. EIS staff *shall* issue news releases, as appropriate. 8. UTCC and Ministry EOCs *shall* be established.	1. Initiate public alerting. 2. Issue notification activating municipal Emergency Response Organization. 3. Municipal EOCs, EICs and other centres to be activated and operational. 4. Implement operational directives, as issued by the PEOC.

Table 4.2: Initial Provincial and Municipal Response (Continued)

INITIAL NOTIFICATION	INITIAL PROVINCIAL RESPONSE	INITIAL MUNICIPAL RESPONSE
GENERAL EMERGENCY (*Emission ongoing or expected within 12 hours)*	FULL ACTIVATION 1. PEOC *shall* notify and ensure that the municipal contacts have activated the public *alerting* system (**Section 6.2**). 2. PEOC *shall* issue the appropriate *emergency bulletin* (**Section 6.4**). 3. PEOC *shall* issue *operational directives* implementing the following *operational measures* for: a) Suspension of road, rail and air traffic throughout the *Automatic Action Zone.* b) *Evacuation* of the *Automatic Action Zone* and Lake Sectors P23 through P25 unless there are good reasons for modifying this *response*. c) *Precautionary measures* in the DPZ. 4. If emission is ongoing or, if *evacuations* will not be completed prior to emission, issue *operational directives* implementing the *operational measures* for: a) Evacuees to report for *radiation* monitoring or, if not possible, to evacuate to a destination beyond the DPZ and to undertake self-*decontamination*. b) Ingestion of KI pills (**Section 5.3.3**) in the *Automatic Action Zone.* c) Sheltering (**Section 5.3.4**) in the rest of the *Detailed Planning Zone*. Otherwise, take this action 4 hours (or, at a time deemed appropriate) before the expected time of commencement of the emission. 5. PEOC *shall* adopt full *activation* (**Section 4.6.4**), and *shall* initiate the appropriate internal and *external notifications* (**Section 4.3** and **Section 4.4** respectively), including the host *community*. 6. PEOC *shall* assess the situation for further action. 7. PEOC *shall* issue further *emergency bulletins*, as appropriate (**Section 6.4**). 8. EIS staff *shall* issue news releases, as appropriate. 9. Ministry EOCs and UTCC to be established.	1. Initiate public *alerting*. 2. Issue *notification* activating municipal *Emergency Response Organization*. 3. Municipal EOCs, EICs and other centres activated and fully staffed. 4. Implement *operational directives*, as issued by the PEOC.

Table 4.3: Default Protective Measures

CONDITION OF STATION SYSTEMS	EXAMPLES	DEFAULT PROTECTIVE MEASURES Issue Immediate Operational Directives
A. Intermediate to severe core damage with an accompanying loss of the *containment* function.	*Either*: 1. Failure of reactor shutdown, or 2. LOCA and failure of ECI, or 3. LOCA causing early flow stagnation in a core pass. *Combined with either:* a) Large hole in the containment envelope (e.g., airlock doors open, multiple airlock seal failures), or b) An emission pathway bypassing containment.	1. **Evacuation** of the *Automatic Action Zone*, all other *Detailed Planning Zone* sectors likely to be affected by the emission, and the area beyond the *Detailed Planning Zone* likely to be affected by the emission up to a distance of 20 km from the reactor. 2. **Iodine Thyroid Blocking**: All evacuees from the *Detailed Planning Zone* to ingest a KI dose. 3. **Personal Monitoring**: All evacuees from the *Detailed Planning Zone* to proceed to a facility for *personal monitoring* or to self-decontaminate at destination. 4. **Sheltering**: All sectors likely to be affected by the emission, which are not immediately evacuating, to shelter. Also, all sectors and areas adjacent (in the same ring) to sectors and areas being evacuated *should shelter-in-place*.
B. Intermediate level of core damage and a loss of the filtered pathway.	*Either*: 1. LOCA and failure of ECI, or 2. LOCA and failure of emergency coolant recovery. *Combined with either:* a) Containment envelope impairment resulting in loss of pressure control, or b) Impairment of the FADS, including a reduction in filter efficiency.	1. **Evacuation** of the *Automatic Action Zone* and all other *Detailed Planning Zone* sectors likely to be affected by the emission. 2. **Iodine Thyroid Blocking**: All evacuees to ingest a KI dose. 3. **Personal Monitoring**: All evacuees to proceed to a facility for *personal monitoring* (ongoing emission only) or to self-decontaminate at destination. 4. **Sheltering**: All sectors likely to be affected by the emission, which are not immediately evacuating, to shelter. Also, all sectors adjacent (in the same ring) to those being evacuated *should shelter-in-place*.

Table 4.3: Default Protective Measures (continued)

CONDITION OF STATION SYSTEMS	EXAMPLES	DEFAULT PROTECTIVE MEASURES Issue Immediate Operational Directives
C. Intermediate to severe fuel damage with *containment* envelope impairment leading to early *venting*.	*Either*: 1. LOCA causing flow stagnation in a core pass, or 2. LOCA and failure of ECI, or 3. LOCA and failure of emergency coolant recovery, or 4. End-fitting or other failure and fuel ejection from a channel, or 5. LOCA in fuelling machine. *Combined with:* A loss of *containment* pressure control requiring early *venting*.	1. **Evacuation** of the *Automatic Action Zone* and all sectors in the Inner Ring likely to be affected by the emission. 2. **Personal Monitoring**: All evacuees to proceed to a facility for *personal monitoring* (ongoing emission only) or to self-decontaminate at destination. 3. **Sheltering**: All sectors in the Inner Ring adjacent to those being evacuated *should shelter-in-place*. Sectors beyond this zone likely to be affected by the emission to also *shelter-in-place*.
D. All other events or conditions likely to lead to an emission.	Spent fuel bay *accident*.	1. **Sheltering** the *Automatic Action Zone* and evacuate Lake Sector P23.

Chapter 5
PROTECTIVE ACTION RESPONSE STRATEGY

5.1 Protective Action Response Strategy

During the response to a *nuclear emergency*, the PEOC *shall* implement a protective action response strategy to protect the public and responding *emergency workers* from the effects of a radioactive emission. Protective actions include:

a) precautionary measures

b) exposure control protective measures

c) ingestion control protective measures

d) additional measures to protect the public

5.2 Precautionary Measures

The PEOC Commander *shall* direct as appropriate, any or all of the following *precautionary measures* in the *Detailed Planning Zone* (or part thereof) and adjacent areas (e.g., CPZ). Consideration *shall* also be given to the most suitable timing for the measures (in the case of a delayed emission it may be appropriate to delay the application of some of them) and issue the necessary bulletin(s) and directions for their implementation. These measures are:

a) closing of beaches, recreation areas, etc.

b) closing of workplaces and schools

c) suspension of admissions of non-critical patients in hospitals

d) *entry control* (see **Section 6.6**)

e) clearing the milk storage of dairy farms

f) banning consumption of any item of food or water that may have been exposed outdoors

g) banning consumption and export of locally produced milk, meat, produce, milk- and meat-producing animals

h) removing milk- and meat-producing animals from outside pasture and exposed water sources

5.3 Exposure Control Protective Measures

5.3.1 Evacuation

a) *Evacuation* time estimates (see **Section 2.6.3**) *should* be used to inform decision-making regarding the implementation of *evacuation* strategies.

b) All available routes will be utilized to evacuate the public.

c) Shadow *evacuations* may occur spontaneously in areas contiguous to the *Detailed Planning Zone* and subsequently contribute to the *Detailed Planning Zone evacuation* time.

d) Contamination

i. In the event of a delayed emission, evacuees are not expected to be contaminated nor require monitoring and *decontamination*.

ii. In the event of an ongoing or imminent emission, evacuees exposed to the radioactive emission can be expected to have varying levels of *contamination*.

iii. *Contamination*, where found, would be in the form of loose particulate on people, their belongings and vehicles.

iv. Internal *contamination* may be present in individuals exposed to a radioactive emission.

v. Given the population density, self-*decontamination* may be the primary means of *decontamination*, if required.

vi. Monitoring and *decontamination* facilities are required for those evacuees who are not able to self-decontaminate as well as for those who desire assurance monitoring.

e) Transportation

i. During a *nuclear emergency*, traffic density and volume on major arterial roads and highways are significantly increased and therefore, travel times in all directions are significantly longer than normal.

ii. Integrated and multi-modal transportation management is required to ensure that *evacuations* can proceed as smoothly as possible.

f) Family Reunification Prior to Evacuation

i. Families will want to reunite and evacuate together, as far as practical.

ii. The feasibility of family reunification depends on the time of day at the onset of the *emergency* and on the urgency for *evacuations* to proceed (i.e., timing of the emission).

iii. Factors affecting family reunification include workplace location, school children, residents of hospitals, long-term care homes or other institutions, etc.

g) Mass Care

i. The majority of evacuees will make their own arrangements for care and lodging. Mass care arrangements are required for those evacuees without such resources.

ii. Evacuees requiring public or privately provided accommodation, may need assurances that these accommodations are not contaminated.

h) Protection and Care of Animals

i. Pursuant to **Section 7.0.2. (4)**, of the *EMCPA*, Municipal *emergency* response plans *should* make provisions for the protection and care of all animals, including those left behind during an *evacuation*.

ii. *Designated Municipalities should* request assistance as necessary from the following to develop plans for the protection and care of animals:

- Ontario Society for the Prevention of Cruelty to Animals (OSPCA) (whose mandate is to protect all animals in Ontario)
- OMAFRA (provincial lead on farm animal disease (OIC 1157/2009))
- the Ministry of Natural Resources and Forestry (MNRF) for issues pertaining to wildlife

iii. The PEOC *should* provide assistance to the *stakeholders* above as required for the protection and care of animals.

i) Directing Evacuations

i. *Evacuations should* be directed by response sector or *planning zone* ring of sectors and include information detailing the boundaries of the *evacuation* area by readily identifiable roads or landmarks.

ii. Evacuees who may have been exposed to an emission *shall* be directed either to proceed to a Monitoring and *Decontamination* Unit (MDU) or to self-decontaminate upon reaching their destination. Information on locations for monitoring *shall* be provided at the time of the *emergency*.

iii. Evacuees who are not at *risk* of being contaminated *shall* be instructed to leave the *Detailed Planning Zone* and not be directed to an MDU or to self-decontaminate.

iv. Evacuees *shall* be permitted to evacuate the affected area in the direction and to the destination of their choosing, subject to restrictions (due to weather, traffic conditions etc.) announced by the PEOC Commander through the *emergency bulletins*.

v. The responsibility for the expeditious movement of evacuees via the provincial transportation network is identified in the Unified Transportation Management Plan.

vi. The Unified Transportation Coordination Centre *shall* monitor the provincial transportation network utilized by evacuees and inform the PEOC Commander of any issues impacting the *evacuation*.

j) Evacuation Arrangements

i. Municipal plans for the City of Toronto and the Regional Municipality of Durham *shall* include arrangements for mass *evacuation* transportation and medical transfers.

ii. The *evacuation* of the affected public *should* be facilitated by the planning and *preparedness* undertaken in advance, including:

- transportation management (e.g., Ministry of Transportation)
- *reception* and *evacuation centres* (e.g., *Designated Municipalities*)
- long-term housing (e.g., multi-ministry and multi-jurisdictional planning group)

- health issues (led by the Local Public Health Units and Medical Officers of Health in conjunction with the MOHLTC, Local Health Integration Networks (LHINs) and Paramedic Services)

iii. Medical assistance required during an *evacuation* is the responsibility of the *emergency* medical services and hospitals under municipal arrangements and *should* be detailed in the municipal plans.

iv. *Designated Municipalities* and *Host Municipalities shall* include provisions for the reception and care of evacuees in their *emergency* plans.

v. The PNGS *operator shall* include provisions for the monitoring and *decontamination* of evacuees in its *emergency* plan and associated procedures (see **Section 6.9**).

vi. *Emergency* plans of the schools in the *Detailed Planning Zone should* provide for the movement of staff and students to pre-arranged host schools and, if necessary, to Monitoring and *Decontamination* Units for prior monitoring and *decontamination*. Evacuated students are the responsibility of their school staff until collected from the host school by their guardians, or parents.

vii. *Emergency* plans of hospitals, long-term care homes, and other institutions in the *Detailed Planning Zone should* include provisions for the transfer of staff/residents/patients to an appropriate facility outside the *Detailed Planning Zone*, with which prior arrangements have been made. Provisions *should* also be made to take staff/residents/patients to Monitoring and *Decontamination* Units, if necessary.

viii. As it may not be possible or desirable to evacuate some of these persons, special arrangements *shall* be made for the care of staff/residents/patients remaining behind, as identified in the municipal plans.

k) Pickering Nuclear Generating Station (PNGS) - Evacuation

PNGS prepares its own *evacuation* plans for non-essential *on-site* personnel. During an *emergency*, the actual *evacuation* of *on-site* personnel *should* be carried out in consultation with the PEOC Commander. Where time permits without compromising the safety of station staff (i.e., delayed or imminent emission), the timing and sequence of *on-site evacuations should* be agreed to in advance with the PEOC Commander.

5.3.2 Temporary Relocation

a) Temporary relocation:

i. Is the displacement of people from their homes for a period beyond one week and up to one year to avoid chronic *exposure* to *radiation*, usually from ground *contamination*. Beyond one-year, the PEOC *should* consider permanent resettlement.

ii. Can be directed post-*release*, during the intermediate *response* phase, based on actual measured *contamination* levels.

iii. Can be directed as a subsequent measure to *evacuation*, or *sheltering-in-place*, or as a separate measure.

iv. Is determined following analysis of environmental *radiation* monitoring results and assessment against *Operational Intervention Levels* (OILs; **PNERP Master Plan Annex E, Appendix 2**).

b) The *evacuation* arrangements described in **Section 5.3.1 j)** above *shall* be considered and applied as appropriate for the implementation of temporary relocation.

c) The PEOC *should* consider socioeconomic factors before recommending temporary relocation as the potential impacts of this action may not be justified in areas where the OIL for relocation is minimally exceeded.

5.3.3 Iodine Thyroid Blocking

a) It is the responsibility of the PNGS *operator* to procure adequate quantities of KI pills for the *Detailed Planning Zone* population (**PNERP Master Plan, Section 6.5.2**).

b) *Designated Municipalities shall* detail in their plans the means by which they facilitate:

i. Availability of KI pills for *Detailed Planning Zone* institutions and for *emergency* centres (Emergency Worker, Reception and *Evacuation Centres* and MDUs).

ii. Availability of KI pills for any members of the *Detailed Planning Zone* population who may wish to possess a supply.

c) Other operational responsibilities regarding *iodine thyroid blocking* (stocking, distribution and administration) are described in the **Radiation Health Response Plan**, as prepared by MOHLTC.

d) The Chief Medical Officer of Health *shall* decide when to administer KI in consultation with the PEOC Commander.

5.3.4 Sheltering-in-Place

The need for future sheltering-in-place as a protective measure *should* be broadcast through the *emergency* bulletin as soon as that need is identified. The timing to actually issue an operational directive for sheltering-in-place (or, in the event of a declared *emergency*, advise that *emergency* orders have been made) *shall* be ultimately made by the PEOC Commander (as a general guidance, however, the *emergency* bulletin to direct this protective measure *should* be issued at least 4 hours prior to the expected emission time) following escalation to a full *activation* response.

5.4 Ingestion Control Measures

5.4.1 Before an emission commences, appropriate *ingestion control* measures *should* be directed as a precaution within and, if necessary, areas adjacent to the *Detailed Planning Zone* (e.g., the CPZ).

5.4.2 After an emission commences, precautionary *ingestion control* measures *should* be reviewed by the PEOC Scientific Section and adjusted as necessary once environmental monitoring results become available.

5.4.3 If environmental monitoring indicates the need, appropriate *ingestion control* measures *should* be considered in areas known or suspected to be contaminated.

5.4.4 Based on the data produced by ground monitoring, additional *ingestion control* measures *should* be considered, where necessary, while the original *precautionary measures* may be lifted where appropriate.

5.5 Additional Measures to Protect the Public

5.5.1 The PEOC Commander may recommend other, practical *dose* reduction measures to the public. Such measures may be implemented in combination with the measures described above or, may simply be recommended to provide an additional level of protection against possible *radionuclides* present in the air or on the ground but which do not meet the *generic criteria* or OILs. Such measures include:

a) Respiratory protection, such as covering of the nose and mouth with available material that can filter particulates when present in the air.

b) Self-*decontamination*, including removing and bagging contaminated clothing, showering, and decontaminating surfaces of critical areas and objects.

c) Staying indoors to the extent that it is practical, e.g., only conducting outdoor tasks when necessary (e.g., seeking medical attention, buying *foodstuff* and necessities).

5.5.2 Detailed advice regarding these measures *shall* be provided for in public awareness and education materials as well as in *emergency bulletins*.

Chapter 6
OPERATIONAL RESPONSE

6.1 General

6.1.1 Operational response strategies are employed during a *nuclear emergency* response in order to facilitate the implementation of *protective measures* against the effects of a radioactive emission.

6.1.2 Responsibility and strategies for operational response implemented during a PNGS *emergency* are described below.

6.2 Public Alerting

6.2.1 Public Alerting System Activation and Responsibilities

a) Whenever the public *alerting* system is to be activated, the PEOC Commander *shall* concurrently issue an *emergency bulletin* (see **Section 6.4**) to the broadcast media. The bulletin *shall* contain specific instructions on what actions the public *should* take and where to get more information. *Emergency bulletins should* be continuously repeated.

b) In case of a *General Emergency notification* from PNGS stating that an emission is ongoing or imminent, the municipal contact points of the *Designated Municipalities should immediately* activate the public *alerting* system. Reference to any other authority is not required.

c) In all other cases, the PEOC Commander *shall* decide when to activate the public *alerting* system and issue the necessary instructions to the *Designated Municipalities*.

6.2.2 Public *alerting* systems used to implement this **PNERP** *shall* conform to the following principles:

a) The Regional Municipality of Durham and City of Toronto, as the *Designated Municipalities* in the PNGS *Detailed Planning Zone* (see **PNERP Master Plan, Annex A**) *shall* make provisions, in their *nuclear emergency* plans, for a *public alerting* system which *shall* ensure that their *Automatic Action Zone* populations that may be required to undertake the default or immediate *protective measures* of (e.g., *sheltering-in-place*, *evacuation*, and ingestion of KI) can be alerted within 15 minutes of initiation.

b) Municipal plans *shall* detail how this requirement is met and, pursuant to **Section 5** of the *EMCPA*, plans of *Lower-tier Municipalities* whose populations reside within the required *alerting* area, *shall* conform to the Municipal Plan.

c) The Regional Municipality of Durham, City of Toronto, and the province *shall* include provisions in their *nuclear emergency response* plans to coordinate the timing of *public alerting*, public direction and *emergency* information. This *should* ensure that the population receives timely and accurate information on what *protective measures* to take once they have been alerted of an *emergency*.

d) The Regional Municipality of Durham and City of Toronto *shall* ensure an initial evaluation of any new *public alerting* system is completed to verify that the requirements under this **PNERP** are met. Further, regular integrated testing of existing *public alerting* systems *shall* be included as a component of municipal *exercise* programs.

e) Where the public *alerting* area includes more than one *Municipality*, the selected system(s) *shall* be compatible or integrated in order to ensure consistency in timing, type of signal and other key implementation specifications.

f) Such a public *alerting* system, coupled together with *emergency bulletins*, *should* ensure that the population within the *Detailed Planning Zone* is notified in an effective and timely manner.

6.2.3 The public *alerting* system for a PNGS *emergency shall*, in addition to adhering to the principles in **Section 6.2.2** above, meet the following requirements:

a) The Regional Municipality of Durham's *nuclear emergency response* plan *shall* describe how the public *alerting* system has the capability to issue a public alert to practically 100%[4] of the population, located both indoors and outdoors, within the *Automatic Action Zone*, within 15 minutes[5].

b) The Regional Municipality of Durham and the City of Toronto's *nuclear emergency* response plans *shall* describe how the public *alerting* system has

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[4] The term "practically 100%" means that the public alert can be heard or received by everyone in the *alerting* area unless exceptional circumstances provide an impediment.

[5] The focus of the public *alerting* system capability is on issuing the public alert. For example, if an auto dialer is used it needs to demonstrate that it has the capability to make all of the calls required within the specified timeframe.

the capability to issue a public alert to the population, located both indoors and outdoors, in the *Detailed Planning Zone* within 15 minutes.

c) The provincial Alert Ready program *shall* also be used to alert all populations within and beyond the DPZ[6].

6.2.4 The *operators* of the Pickering NGS, pursuant to the *Nuclear Safety and Control Act*, *shall* provide resources and assistance to the Regional Municipality of Durham and the City of Toronto to establish and maintain a public *alerting* system in their *Detailed Planning Zone*.

6.3 Technical Assessments - PEOC Scientific Section

6.3.1 The PEOC Scientific Section is responsible for assessing the radiological consequences of the *nuclear emergency* through the operations of its Nuclear Incident Group (NIG) and the Environmental Radiation Assurance and Monitoring Group (ERAMG).

6.3.2 In the early phase of the *emergency* the majority of the assessments undertaken shall be by the NIG (**Section 4.6.5**). The ERAMG shall be focussed on ensuring the availability of baseline *radiation* information.

6.3.3 Once the emission has ceased, the ERAMG shall begin operations to determine the level and extent of radioactive *contamination* (**Section 4.7.3**).

6.4 Public Direction - Emergency Bulletins

6.4.1 The responsibility for issuing *emergency bulletins* rests with the PEOC Commander and may be delegated to the PEOC Operations Chief.

6.4.2 The aim of public direction is to communicate, directly to the affected public through *emergency* bulletins, the direction and guidance regarding protective actions they *should* take in order to ensure their safety and welfare. Public direction principles are described in the **PNERP Master Plan, Section 7.3**.

6.4.3 *Emergency bulletins* issued during a partial *activation* response level, before an emission is expected to occur, *should* be informative and permissive, whereas *emergency* bulletins issued once a full *activation* response has been adopted *should* be increasingly directive.

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[6] The provincial Alert Ready program ensures that emergency bulletins are broadcast in a timely manner via radio, television and mobile devices.

6.4.4 While the need for future sheltering *should* be broadcast through *emergency* bulletin as soon as that need is identified, the actual sheltering directive *should* be made, via *emergency* bulletin, at least 4 hours prior to the expected emission time.

6.4.5 At a partial *activation* response, the *emergency* bulletin *shall* include the following information, as applicable:

a) date and time of expected emission

b) sectors (by geographical description) which may be affected

c) applicable precautionary and *protective measures* for the affected sectors or area and applicable timings (in the case of a delayed emission it may be appropriate to delay the application of some of them)

d) public inquiry phone number(s) and websites

6.4.6 As successive *emergency bulletins* are issued, as much additional information as possible *should* be provided which may encourage those who can leave early to evacuate.

6.4.7 Once a full *activation* response level has been adopted and an emission is expected in 36 hours or less, *emergency* bulletins *should* ensure that they include directions regarding:

a) date and time of expected emission

b) *precautionary measures* directed in the applicable zone(s)

c) *protective measures* and the affected sectors or zones

d) *Reception Centres* which can receive evacuees without accommodation

e) KI pill ingestion details and availability information, as applicable

f) public inquiry phone number(s) and websites

6.4.8 Marine Notification and Public Direction

a) The PEOC Commander *shall* ensure the Canadian Coast Guard is notified whenever the PEOC receives a *notification* (partial or full *activation*) under this plan (see **Paragraph 4.4.3 f)**) and they, in turn, *shall* notify the U.S. Coast Guard.

b) In the case of a full *activation response*, the Canadian Coast Guard *shall* broadcast an *emergency* message through their radio stations to marine craft

on the marine radio channel. The message *should* notify all marine craft in the vicinity of the *emergency* and direct them to remain clear of the *Detailed Planning Zone*.

c) Municipal plans *shall* detail how the Regional Municipality of Durham and Toronto Police Marine Units assist in notifying and evacuating marine craft that do not have radios on board.

6.4.9 The public awareness and education program for nuclear emergencies *shall* include information regarding the means by which public direction will be communicated.

6.5 Emergency Public Information

6.5.1 Lower Level Response

When the *off-site response* adopted is Routine Monitoring or Enhanced Monitoring, (see **Table 4.2**), all news releases pertaining to the event and prepared on behalf of the province, *shall* be issued by the Director Communications Branch, MCSCS who acts as the Provincial Chief Emergency Information Officer (PCEIO).

6.5.2 Higher Level Response

a) When the *off-site response* adopted is partial *activation* or full *activation*, (see **Table 4.2**), the Director of Communications Branch, MCSCS, assumes their role as PCEIO, establishing the provincial Emergency Information Section (EIS), on behalf of the province.

b) The *Designated Municipalities*, the *reactor facility operator* and the federal government each have their own *emergency* information operations.

c) In order to ensure the coordination and consistency of all *emergency* information issued to the public, all *stakeholders should* inform the provincial EIS if they plan to issue news releases or other *emergency* information materials.

d) *Stakeholders should* coordinate the release and content of *emergency information* for public release with the provincial Emergency Information Section.

e) *Stakeholder emergency* plans *should* include provisions for supporting the operation of a *Joint Information Centre* (JIC) if established by the PCEIO (e.g., public affairs/spokesperson).

6.5.3 The Provincial Emergency Information Section (EIS)

a) The provincial EIS, located in Toronto, *shall* ensure that the province's *emergency* information is coordinated with the *emergency* information produced and disseminated by the *Designated Municipalities*, nuclear *operator*, federal partners and other *stakeholders* to ensure consistent messaging.

b) *Stakeholders should* share *emergency* information prior to release wherever possible and practicable.

c) The EIS functions include:

   i. Coordinating all of the provincial *communications* related to the *nuclear emergency*.

   ii. Issuing provincial *emergency* information.

   iii. Sharing and coordinating *emergency* information with Municipal EICs to ensure continuity and uniformity of messaging.

   iv. Sharing copies of all news releases, fact sheets, and other public information materials with EICs prior to release to the public, if or when possible.

   v. Sending a liaison officer(s) to the Municipal EICs, if so requested.

6.5.4 Municipal Emergency Information Centre (EIC)

a) Emergency plans for the *Designated Municipalities shall* each describe how an EIC is established at a partial or full *activation response*.

b) The EIC is responsible for the collection, dissemination and monitoring of local *emergency* information.

c) The *Designated Municipalities* may invite the PNGS *operator*, neighbouring *Municipalities*, federal, and provincial liaison officers to participate in the EIC operation.

d) The functions of the EIC include:

   i. Issuing news releases and other public information documents to the local media and residents describing the *emergency* and *response* measures.

ii. Keeping the provincial EIS and JIC informed regarding the development and distribution of news releases and other public information documents to local residents and media.

iii. Keeping the EIS apprised of local public perceptions, rumours, and reactions.

iv. Assisting media covering the *emergency*.

v. Monitoring local media to ensure that local news is being correctly transmitted to the public by the media and confirming this with the EIS.

vi. Arranging media briefings, as required, to communicate “key messages” to the public.

6.5.5 Public Inquiry

a) Provincial public inquiries *shall* be coordinated by the EIS and include use of the Service Ontario’s hotline.

b) The *Designated Municipalities shall* be responsible for establishing their own public inquiry operation.

6.6 Entry Control

6.6.1 Management of the main transportation routes shall be coordinated by the PEOC as follows:

a) In the case of marine, air and rail, through the relevant coordinating agency in the PEOC (federal liaison, MTO, OPP).

b) In the case of road traffic, by the Unified Transportation Coordination Centre (UTCC) via the MCSCS representative in the PEOC.

6.6.2 In the event of an ongoing emission or one that is imminent, the PEOC Commander *should* consider the following *entry control* measures and notify the proper authorities for implementation as appropriate:

a) Suspension of through traffic on main road and rail routes going through the *Detailed Planning Zone* (Hwy. 401, Hwy. 2, the CN and CP lines).

b) Suspension of marine traffic in the *Detailed Planning Zone* area (P23 - P25) of Lake Ontario.

c) Aircraft *should* be kept clear of the *Detailed Planning Zone*.

6.7 Transportation Management

6.7.1 A Unified Transportation Management Plan (UTMP) *shall* be developed for the *Detailed Planning Zone* as well as the arterial roads that provide access to this zone. During an *emergency*, the Unified Transportation Coordination Centre (UTCC) (see **Paragraph 3.1.3**) *shall* be responsible for implementing the UTMP.

6.7.2 The Unified Transportation Coordination Centre (UTCC) *shall* operate in coordination with the Municipal Emergency Operations Centres (EOCs), and the *Provincial Emergency Operations Centre* (PEOC).

6.7.3 The UTMP *shall* be designed to allow implementation in incremental stages consistent with the agreed upon evacuation time estimate data and the provincial emergency response levels. For example, a staged approach may include:

a) Stage 1: The aim in this stage could be to keep traffic flowing smoothly on the main *evacuation* routes and ensure that these routes remain open.

b) Stage 2: UTMPs could prevent traffic from entering the *Detailed Planning Zone* and divert traffic around it. However, access *should* be allowed for *emergency workers* who have tasks to perform in the *Detailed Planning Zone*. Stage 1 measures *should* continue.

c) Stage 3: Could be initiated when it appears that particular sectors are likely to be evacuated. Additional resources *should* be deployed to ensure that *evacuation* proceeds smoothly beyond the *Detailed Planning Zone* boundary. Stages 1 and 2 measures *should* continue.

6.7.4 The timing and order of sector *evacuations shall* be determined by the PEOC Commander, in coordination with the UTCC.

6.7.5 The UTMP *shall* provide, where applicable, for the priority *evacuation* of any *response sectors* if and when ordered.

6.7.6 *Operational directives* implementing *evacuations* (or *emergency* orders issued in the event of a declared *emergency*) *shall* be accompanied by *emergency bulletins* issued by the PEOC Commander.

6.8 Emergency Worker Safety

6.8.1 At the commencement of an *emergency* resulting in the *activation* of this plan, the *response sectors* in the *Detailed Planning Zone shall* be assumed to carry the following default safety status (**PNERP Master Plan**, **Annex H**), based on the category of the *notification* initiated by PNGS:

a) On-Site Emergency Notification with an ongoing emission:

i. Sectors P1, P2, P23 - ORANGE

ii. All other sectors - GREEN

b) General Emergency Notification with an ongoing emission:

i. Sectors P1, P2, P23 - RED

ii. Sectors P3 through P14, and P24 - ORANGE

iii. All other sectors - GREEN

c) All other cases:

If there is no ongoing emission, then the sector safety status for all sectors *should* be GREEN and *should* remain GREEN until an emission commences.

6.8.2 The PEOC Scientific Section Chief *shall* make recommendations on sector safety status to the PEOC Commander for approval and update recommendations as data becomes available.

6.8.3 The PEOC Commander *shall* reassign safety status to all sectors and update them periodically as soon as relevant data is available.

6.8.4 During the course of an emission over land, safety sector status updates *shall* be done on an hourly basis and promptly communicated by the PEOC Commander to all *stakeholders*.

6.8.5 It is the responsibility of each organization with *emergency workers* operating or required to operate in the *Detailed Planning Zone* to ensure that they are kept apprised of the current safety status of *response sectors*.

6.8.6 The municipal plans *shall* provide for the setting up of *Emergency Worker Centres* (EWCs), as appropriate (**PNERP Master Plan, Paragraph 7.10.3**).

6.8.7 The PNGS *operator* is responsible for the monitoring and *decontamination* aspect of EWCs, the relevant details of which are provided in their plans and procedures.

6.8.8 *Emergency workers* who need to enter a sector *shall* first report to an EWC, where they will be provided with *personal monitoring* devices and briefed on the health risks and precautions they *should* observe and any maximum time limit on their stay in the sector (see **Paragraph 6.8.1** above).

6.8.9 If an emission is ongoing, *emergency* services (police, fire and paramedic services) who are required to operate in the *Automatic Action Zone* (before an *Emergency Worker Centre* is functioning) *should* carry and use the following equipment:

a) personal protective equipment (e.g., respiratory protection, gloves, etc.)

b) dosimetry

c) stable iodine tablets (one tablet to be ingested prior to entering a RED sector)

d) a card listing the default safety status of sectors (see **Paragraph 6.8.1** above) and the precautions to be taken for each safety status (**PNERP Master Plan, Annex H**)

6.8.10 Municipal plans *shall* detail how these *emergency* services obtain these items, appropriately store them, and maintain such equipment so that it is readily available when needed. The PNGS *operator shall* provide assistance in obtaining and maintaining items **Paragraph 6.8.9 a)** and **b)** above.

6.9 Population Monitoring and Medical Management

6.9.1 The PNGS *operator shall* resource five Monitoring and *Decontamination* Units (MDUs) as follows:

a) Three MDUs *should* be located at fixed sites (e.g., at *Reception Centres*).

b) Two MDUs *should* be mobile facilities, stored at a PNGS *operator* facility and transportable, when required, to locations which have been pre-designated.

6.9.2 *Designated Municipalities* and the PNGS *operator shall* collaborate to identify in their respective emergency plans, multiple sites within the DPZ and CPZ which could host mobile MDUs, to ensure the availability of infrastructure and amenities to support their operation.

6.9.3 Fixed and pre-designated sites for mobile units *shall* be selected so as to provide, as far as practical, monitoring and *decontamination* options for all directions surrounding the *Detailed Planning Zone*.

6.9.4 Fixed and mobile MDUs *shall* provide both assurance monitoring, for those who have undertaken self-*decontamination*, as well as monitoring and *decontamination* for those evacuees who either require or desire it upon evacuating the *Detailed Planning Zone*.

6.9.5 Mobile MDUs can also be deployed to support fixed MDUs if additional capacity is required at those locations.

6.9.6 The MOHLTC is responsible for leading and coordinating the health *response* and maintaining health services during *nuclear and radiological emergencies*. As such, the MOHLTC *shall* develop arrangements, in coordination with the PNGS *operator*, hospitals, *Designated Municipalities* and their public health units, to track evacuees for the purposes of *contamination* assessments (internal and external) and to provide follow up with those affected.

6.9.7 The **Radiation Health Response Plan (RHRP)** *shall* be fully activated through the MOHLTC EOC when it seems likely that the incident may result in high *radiation exposures* to some persons necessitating medical management.

6.10 Provincial Liquid Emission Response Plan (PLERP)

6.10.1 The main *radiation* (e.g., tritium) *exposure* pathway for a liquid emission from PNGS is through *contamination* of a water supply source, with the resulting hazard being the subsequent ingestion of contaminated water.

6.10.2 If a liquid emission has occurred at PNGS in conjunction with an event that meets the *notification* category system as detailed in this implementing plan, then it *shall* be managed within this implementing plan.

6.10.3 If a liquid emission has occurred at PNGS and has not occurred in conjunction with an event that meets the *notification* category system as detailed in this implementing plan, then it *shall* be managed in accordance with the PNGS *operator's* liquid emission response plan.

6.10.4 A liquid emission *response* undertaken pursuant to the **PLERP**, may shift to a **PNERP** *response* if events escalate to a magnitude where it is deemed appropriate by the PEOC Commander.

6.11 Management of Radioactive Waste

6.11.1 For large waste volumes and high radioactive levels, existing disposal facilities may be insufficient or unsuitable, thereby necessitating alternative means of disposal, including the potential for construction of new facilities.

6.11.2 The following factors *should* be considered in the determination of waste disposal siting (existing or new):

a) proximity to the incident area

b) proximity to residential areas or commercial districts

c) proximity to transportation corridors

d) for newly designated sites, the existing level of *contamination* and potential for remediation

e) consistency with national and international standards and practices for the management and control of radioactive waste; and

f) solutions to protect the health and safety of persons and the environment

6.11.3 When appropriate, the PEOC Commander *shall* appoint a working group to implement a waste management plan, including representatives from:

a) provincial ministries (e.g., MEPC, ENDM, MOL and MTO)

b) federal departments (e.g., CNSC and Health Canada)

c) municipal public works departments

d) *reactor facility* specialists

e) private sector organizations, as applicable

ANNEXES

ANNEX A: RESPONSE SECTOR BOUNDARIES

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ANNEX C: CONTINGENCY PLANNING ZONE GUIDANCE

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ANNEX A
RESPONSE SECTOR BOUNDARIES

(Reference: **Section 2.4.2**)

SECTOR	SECTOR BOUNDARY (north; east; south; west)
P1	Area bounded on the north by Route 401, on the east by Liverpool road, on the south by Lake Ontario, and on the west by Whites Road. East and West side of Frenchman's Bay and the Brock Road industrial area.
P2	Area bounded on the north by Route 401, on the east of Duffins Creek, on the south by Lake Ontario, and on the west by Liverpool Road. East and West side of Frenchman's Bay and the Brock Road industrial area.
P3	Area bounded on the north by Route 401, on the east by the Rouge River, on the south by Lake Ontario, and on the west by East Avenue.
P4	Area bounded on the north by Sheppard Avenue, on the east by Whites Road, on the south by Lake Ontario, and on the west by Rouge River.
P5	Area bounded on the north by Finch Avenue, on the east by Whites Road, on the south by Sheppard Avenue, and on the west by Scarborough Pickering Townline Road.
P6	Area bounded on the north by Finch Avenue, on the east by Dixie Road, on the South by Route 401, and on the west by Whites Road.
P7	Area bounded on the north by Finch Avenue, on the east by Brock Road, on the south by Route 401, and on the west by Dixie Road.
P8	Area bounded on the north by 3rd Concession Road, on the east by Brock Road, on the south by Finch Avenue, and on the west by Scarborough Pickering Townline Road.
P9	Area bounded on the north by Rossland Road, on the east by Ravenscroft Road and Rotherglen Road, on the south by Route 401, and on the west by Brock Road.
P10	Area bounded on the north by Rossland Road, on the east by Old Harwood Avenue, on the south by Kingston Road, and on the west by Ravenscroft Road and Rotherglen Road.
P11	Area bounded on the north by Kingston Road, on the east by Pickering Beach Road, on the south by Route 401, and on the west by Rotherglen Road.

SECTOR	SECTOR BOUNDARY (north; east; south; west)
P12	Area bounded on the north by Route 401, on the east by Pickering Beach Road, on the south by Bayly Street, and on the west by Duffins Creek.
P13	Area bounded on the north by Bayly Street, on the east by Harwood Avenue, on the south by Lake Ontario, and on the west by Duffins Creek.
P14	Area bounded on the north by Bayly Street, on the east by Pickering Beach Road, on the south by Lake Ontario, and on the west by Harwood Avenue.
P15	Area bounded on the north by Lawrence Avenue, on the east by Centennial Road, on the south by Lake Ontario, and on the west by Morningside Avenue.
P16	Area bounded on the north by Route 401, on the east by East Avenue, on the south by Lake Ontario, and on the west by Centennial Road.
P17	Area bounded on the north by Ellesmere Avenue, on the east by Centennial Road, on the south by Lawrence Avenue, and on the west by Morningside Avenue.
P18	Are bounded on the north by Sheppard Avenue, on the east by the Little Rouge River, on the south by Ellesmere Avenue, and on the west by Morningside Avenue.
P19	Area bounded on the north by Old Finch Avenue and Steeles Avenue, on the east by Scarborough Pickering Townline Road, on the South by Sheppard Avenue, and on the west by Morningside Avenue.
P20	Area bounded on the north by Whitevale Road, on the east by Brock Road, on the south by 3rd Concession Road/Rossland Road, and on the west by Markham Pickering Townline Road/York Durham line.
P21	Area bounded on the north by Whitevale Road, on the east by Audley Road, on the south by Rossland Road, and on the west by Brock Road.
P22	Area bounded on the north by Rossland Road, on the east by Halls Road and Lynde Creek, on the south by Lake Ontario, and on the west by Harwood Avenue and Pickering Beach Road.

ANNEX B
POPULATION DATA

(Reference: **Paragraph 2.6.2**)

SECTOR	POPULATION[7]
P1	8,900
P2	4,740
P3	5,675
P4	8,330
P5	17,190
P6	11,140
P7	15,900
P8	15,620
P9	17,810
P10	16,190
P11	9,150
P12	3,670
P13	8,810
P14	9,420

[7] Population estimates based on the 2016 Canada Census.

SECTOR	POPULATION
P15	10,898
P16	7,224
P17	14,604
P18	19,407
P19	9,541
P20	1,830
P21	36,450
P22	25,630
PNGS	4,500
TOTAL (with PNGS)	**277,374**
TOTAL (without PNGS)	**272,874**

ANNEX C
CONTINGENCY PLANNING ZONE GUIDANCE

(Reference: **Section 2.4.3**)

General

1. The *Contingency Planning Zone* (CPZ) is a pre-designated area surrounding a *reactor facility*, beyond the *Detailed Planning Zone*, where contingency planning and arrangements are made in advance, so that during a *nuclear emergency*, *protective measures* can be extended beyond the *Detailed Planning Zone* as required to reduce potential for *exposure*.

2. The CPZ included within the PNERP Master Plan and Implementing Plans is aligned with new standards and guidance documents, including the Canadian Standards Association's (CSA) **N1600 General Requirements for Nuclear Emergency Management Programs**, International Atomic Energy Agency's (IAEA) **General Safety Requirements (GSR) Part 7**.

3. The CPZ is intended to be used as necessary in the event of very low probability, *severe accident* situations where the area affected could extend beyond the *Detailed Planning Zone*.

4. The CPZ does not require the same level or type of detailed arrangements as the *Automatic Action Zone or Detailed Planning Zone*, in so far as there are no default or pre-planned *protective measures* associated with the CPZ.

5. *Response* activities within the CPZ may occur in the event of a limited and localized radiological release and based on the results received from environmental *radiation* monitoring activities.

6. The distribution of *Iodine Thyroid Blocking* pills should be undertaken in a manner consistent with the processes established for the *Ingestion Planning Zone*.

7. Public Education requirements are consistent with the processes stipulated for the *Ingestion Planning Zone*.

8. The designation of additional primary emergency facilities beyond those designated in the *Detailed Planning Zone* is not required (e.g., Emergency Operations Centres (EOCs), *Emergency Information Centres* (EICs), Reception Centres, Evacuation Centres, MDU facilities, etc.). However, *municipalities* are required to identify and document in their municipal plans, those sites that could be used as a back-up or

alternative location in the event that the primary emergency facility becomes unavailable.

Operational Response Activities Within the CPZ

9. Operational *response* activities within the CPZ *should* unfold using the planning, communication, assessment and command and control mechanisms set out in this document and the PNERP Master Plan. For example, *public alerting* and emergency communications would be conducted using existing processes and systems as established for the *Ingestion Planning Zone* and beyond.

10. Operational response activities within the CPZ would be focused on monitoring dose rates from deposition (e.g., *groundshine*) in order to determine which specific locations or areas beyond the *Detailed Planning Zone* may require the imposition of *exposure control* measures (e.g., *evacuation, sheltering-in-place*, *Iodine Thyroid Blocking* (ITB), etc.).

11. In the event of a radiological release, the PEOC would undertake the following functions:

    a) The PEOC would determine and advise *stakeholders* on the direction of the radioactive plume and likely *radioactive material* deposition locations.

    b) The PEOC would direct field sampling teams to measure for *radioactive material* deposition in suspected locations.

    c) The PEOC Scientific Section would employ its existing mechanisms, processes and procedures to assess environmental radiation monitoring results and analyze the data received from the field sampling teams to identify the size and boundaries for the response activities within the eight CPZ sub-zones **(**see **Figure 2.2)** and to make protective action recommendations to the PEOC Commander, consistent with the results received and in line with this plan's guiding principles **(PNERP Master Plan, Section 1.2)**.

    d) The PEOC Commander would promulgate protective actions using existing communication methods specified in this Implementing Plan.

12. *Municipalities* would be required to identify any emergency facilities that may be at risk of exposure to a radioactive plume during the emergency. *Municipalities* and the PEOC would then collaborate to determine which previously identified alternate facilities would be used to support the *response*. In the event of an emergency where all previously identified alternative facilities are unavailable, the PEOC will identify and source appropriate alternate facilities and communicate the location of these facilities to the *Emergency Response Organization*.

13. The emergency information function would be engaged to advise the public and *stakeholders* which areas of the CPZ have been impacted and what protective actions are required.

ANNEX D
NUCLEAR / RADIOLOGICAL GLOSSARY

(Reference: **Paragraph 2.3.2**)

(For other references see Provincial Glossary)

Abnormal Incident: An abnormal occurrence that may have a significant cause and/or may lead to more serious consequences. (Source: CNSC Glossary)

Accident: Any unintended event, including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are significant from the point of view of protection or safety. With respect to nuclear criticality safety, the term *accidents* or *accident* sequences means events or event sequences, including external events that lead to violation of the sub-criticality margin (that is, to exceeding the upper subcritical limit). (Source: CNSC Glossary)

Activation: Decisions and actions taken to implement a plan, a procedure or to open an emergency operations centre. (Source: Provincial Glossary)

Alerting: Informing the population, by means of an appropriate signal, that a *nuclear emergency* has occurred or is about to occur.

As Low As Reasonably Achievable (ALARA): A principle of *radiation* protection that holds that *exposures* to *radiation* are kept as low as reasonably achievable, social and economic factors taken into account. (Source: CNSC Glossary)

Automatic Action Zone (AAZ): A pre-designated area immediately surrounding a *reactor facility* where pre- planned protective actions would be implemented by default on the basis of *reactor facility* conditions with the aim of preventing or reducing the occurrence of severe *deterministic effects*. (Source: Canadian Standards Association (CSA) *N1600, General requirements for nuclear emergency management programs*)

Becquerel (Bq): The International System of Units (SI) unit of radioactivity. One *becquerel* (Bq) is the activity of a quantity of *radioactive material* in which one nucleus decays per second. In Canada, the Bq is used instead of the non-SI unit curie (Ci). (Source: CNSC Glossary)

Beyond Design Basis Accident (BDBA): An *accident* less frequent and potentially more severe than a *design-basis accident*. Note**:** For a *reactor facility*, a *beyond-design-basis accident* may or may not involve fuel degradation. (Source: CNSC Glossary)

Boiling Water Reactor (BWR): A common type of light-water reactor, where water is allowed to boil in the core, generating steam directly in the reactor vessel to generate electrical power. (Source: CNSC Glossary)

Buffer Zone: An area beyond the *Restricted Zone*, where limited areas of radioactivity are detected. The *buffer zone* is initially delineated based on results of preliminary environmental *radiation* monitoring. *Ingestion control* measures may be applied within this zone, based on guidance provided by the *Operational Intervention Levels* (OILs) and, in accordance with direction provided by the Environmental Radiation and Assurance Monitoring Group (ERAMG).

CANDU Reactor: A Canadian-invented pressurized heavy-water reactor that uses heavy water (deuterium oxide) for moderator and coolant and natural uranium for fuel. “CANDU” is short for CANada Deuterium Uranium. Also called CANDU. (Source: CNSC Glossary)

Cloudshine: *Gamma radiation* from *radioactive materials* in an airborne *plume*.

Communications: Advisories, directives, information and messages that are transmitted. (Source: Provincial Glossary)

Community: A generic term that includes both *Municipalities* and First Nations. (Source: Provincial Glossary)

Containment (System): A series of physical barriers that exist between *radioactive materials* contained in a *reactor facility* and the environment. *Containment* usually refers only to the reactor and vacuum buildings, and integral systems such as dousing.

Contamination: Contamination refers to nuclear or hazardous substances on surfaces, or within solids, liquids or gases (including the human body), where their presence is unintended or undesirable, or to the process giving rise to their presence in such places. (Source: CNSC Glossary)

Contingency Planning Zone (CPZ): A pre-designated area surrounding a *reactor facility*, beyond the *Detailed Planning Zone*, where contingency planning and arrangements are made in advance, so that during a *nuclear emergency,* protective actions can be extended beyond the *Detailed Planning Zone* as required to reduce potential for *exposure*. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)
Note: The actual CPZ for each *reactor facility* is specified in the relevant implementing plans of the *Provincial Nuclear Emergency Response Plan*.

Crop Control: *See* **Produce and Crop Control**.

Declaration of Emergency: A signed declaration made in writing by the Head of Council or the Premier of Ontario in accordance with the *Emergency Management and Civil Protection Act.* This declaration is usually based on a situation or an impending situation that threatens public safety, public health, the environment, critical infrastructure, property, and/or economic stability and exceeds the scope of routine *community emergency response*.
Notes:
1 Municipal Declaration of *Emergency*: a declaration of *emergency* made by the Head of Council or a *Municipality*, based on established criteria.
2 Provincial Declaration of *Emergency*: a declaration of *emergency* made by the Lieutenant Governor of Council or the Premier of Ontario, based on established criteria.
(Source: Provincial Glossary)

Decontamination: Reduction or removal of radioactive *contamination* in or on materials, persons or the environment.

Design Basis Accident (DBA): *Accident* conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the *release* of *radioactive material* are kept within authorized limits. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Designated Host Municipality: The *Municipality* assigned responsibility in the *Provincial Nuclear Emergency Response Plan* for the reception and care of people evacuated from their homes in a *nuclear emergency*.

Designated Municipality: A *Municipality* in the vicinity of a *reactor facility* which has been designated under the *Emergency Management and Civil Protection Act*, as one that *shall* have a *nuclear emergency* plan (*for list see* **PNERP Master Plan, Annex A**).

Detailed Planning Zone: A pre-designated area surrounding a *reactor facility*, incorporating the *Automatic Action Zone*, where pre-planned protective actions are implemented as needed on the basis of *reactor facility* conditions, *dose* modelling, and environmental monitoring, with the aim of preventing or reducing the occurrence of *stochastic effects*. (Source: Modified from *CSA N1600, General requirements for nuclear emergency management programs*)

Deterministic Effects: *Radiation*-induced health effects including changes to cells and tissues that are certain to occur in an individual exposed to a *radiation dose* greater than some threshold *dose*, with a severity that increases with increasing *dose*. Now referred to as tissue reactions. (Source: Health Canada Glossary)

Disaster: A serious disruption to an affected area, involving widespread human, property, environmental and / or economic impacts, that exceed the ability of one or more affected communities to cope using their own resources. (Source: Provincial Glossary)

Dose: A measure of the *radiation* received or “absorbed” by a target. The quantities termed *absorbed dose*, organ *dose*, equivalent *dose*, effective *dose*, committed equivalent *dose* or

committed *effective dose* are used, depending on the context. The modifying terms are often omitted when they are not necessary for defining the quantity of interest.

Dose Management: Includes administrative controls to limit *doses*, monitor *doses* and record *doses* received by *emergency workers* while fulfilling their duties related to *nuclear emergency response*.

Dose Projection: The calculation of *projected dose* (*see Projected Dose*).

Dose Rate: The amount of *radiation dose* which an individual would receive in a unit of time. In the context of this plan, the measurement units are multiples or submultiples of the Sievert (or rem) per hour.

Dosimeter: An instrument for measuring and registering total accumulated *exposure* to *ionizing radiation*.

Drill: supervised instruction intended to test, develop, maintain, and practice the skills required in a particular *emergency response* or *recovery* activity.
Note: A *drill* can be a component of an *exercise*. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Effective Dose (E): A quantity calculated by multiplying the equivalent *dose* received by irradiated tissues, by a tissue specific weighting factor that reflects the *risk* of *radiation*-induced cancer to that tissue. The effective *doses* can then be summed to obtain the effective *dose* absorbed by the body.

Emergency: A situation or an impending situation that constitutes a danger of major proportions that could result in serious harm to persons or substantial damage to property and that is caused by the forces of nature, a disease or other health *risk*, an *accident* or an act whether intentional or otherwise (*EMCPA*). (Source: Provincial Glossary)

Emergency Action Level: Pre-determined criteria related to *on-site* conditions (e.g., plant parameters) which trigger the implementation of protective actions, particularly in the *Automatic Action Zone*. (Source: Health Canada Glossary)

Emergency Bulletin: Directions to the public on appropriate protective and other measures to be taken during a *nuclear or radiological emergency*, which are issued by the province and broadcast through the media.

Emergency Information (EI): Information about an *emergency* that can be disseminated in anticipation of an *emergency* or during an *emergency*. It may provide situational information or directive actions to be taken by the public. (Source: Provincial Glossary)

Emergency Information Centre (EIC): A designated facility that is properly equipped to monitor and co-ordinate *emergency* information activities including the dissemination of information to the public. (Source: Provincial Glossary)

Emergency Response Organization: A group (public, private or volunteer), trained in *emergency response* that may be called upon to respond to an *emergency* situation. (Source: Provincial Glossary)

Emergency Worker: A person performing *emergency* services to support *emergency response*.
Notes**:**
1 *Emergency workers* can include the following: *nuclear emergency workers* required to remain in, or to enter, areas affected or likely to be affected by *radiation* from a *nuclear emergency*, and for whom special safety arrangements are required; *emergency workers* required to provide *response* outside the affected areas.
2 This does not include nuclear energy workers.
3 *Emergency workers* can include police, firefighters, paramedic services and *emergency* social services workers, and other essential services.
(Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Emergency Worker Centre: A facility set up to monitor and control *radiation exposure* to *emergency workers*.

Entry Control: The *prevention* of non-essential persons from entering a potentially dangerous area.

Environmental Decontamination: See *Decontamination*.

Equivalent Dose: The *absorbed dose* multiplied by a weighting factor for the type of *radiation* giving the *dose*. Weighting factors for use in Canada are prescribed by the Canadian Nuclear Safety Commission. This term is also sometimes called *weighted dose*. Expressed in terms of Sievert (or rem).

Evacuation: A directed protective action for the controlled displacement of the population from an area which has been or might become contaminated by radioactive substances to avoid *exposure*. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Evacuation Centre: A centre which provides affected people with basic human needs including accommodation, food and water. (Source: Australian Emergency Management Glossary)

Exclusion Zone: A parcel of land within or surrounding a *reactor facility* on which there is no permanent dwelling and over which a licensee has the legal authority to exercise control. (Source: CNSC Glossary)

Exercise: A simulated *emergency* in which players carry out actions, functions, and responsibilities that would be expected of them in a real *emergency*. *Exercises* can be used to validate plans and procedures, and to practice *prevention*, mitigation, *preparedness*, *response*, and *recovery* capabilities.

Exposure: The act or condition of being subject to irradiation. *Exposure* can be either external *exposure* (irradiation by sources outside the body) or internal *exposure* (irradiation by sources inside the body).

Exposure Control: *Emergency* operations aimed at reducing or avoiding *exposure* to a *plume* or *puff* of *radioactive material*. Measures to deal with surface *contamination* and re-suspension might also be included.

Exposure Pathways: The routes by which *radioactive material* can reach or irradiate humans.

External Notification: The *notification* of organizations and agencies (not directly part of the *emergency* management organization) which may be affected by a *nuclear emergency*, or which may be required to assist in responding to it.

Far Incident: A transborder nuclear *accident* or event anywhere in the world which could affect Ontario, other than a Near Incident (*see Near Incident*).

Food Control: Measures taken to prevent the consumption of contaminated *foodstuffs* and control of including the supply of uncontaminated *foodstuffs*. Where appropriate, such control may include *foodstuff* storage to permit *radionuclide* decay, diversion of *foodstuff* to non-human, non-*foodstuff* chain use or disposal of unusable stocks.

Foodstuff: Food or drink intended for human consumption, including (a) an ingredient of food or drink intended for human consumption or (b) any animal or plant, or any of its parts, from which food or drink, or an ingredient of food or drink, intended for human consumption may be derived.

Fuel Failure: Any rupture of a fuel sheath such that fission products may be *released*. (Source: CNSC Glossary)

Gamma Radiation: Penetrating electromagnetic *radiation* emitted from an atom's nucleus. Also called gamma rays. (Source: CNSC Glossary)

General Emergency: Events at a nuclear power plant or onboard a nuclear-powered vessel resulting in an actual or substantial *risk* of a *release* of radioactivity or *radiation exposure* which warrants the implementation of protective actions off site. (Source: Health Canada Glossary)

Generic Criteria: Expressed as a *projected dose*, over a specified time period, above which protective actions are recommended to reduce the *risk* of *stochastic effects*.

Government Operations Centre: The federal government organization located in the National Capital Region which directs the mobilization and delivery of national support to the affected province in the case of an event in or near Canada, or which co-ordinates federal actions in the case of an international event.

Gray (Gy): The International System of Units (SI) unit of measurement used to express *absorbed dose*. One gray is defined as the absorption of 1 joule of *ionizing radiation* by 1 kilogram of matter. For *gamma* and beta *radiation*s, the gray is numerically equal to the Sievert. (Source: CNSC Glossary)

Groundshine: *Gamma* and/or beta *radiation* from *radioactive material* deposited on the ground.

Guaranteed Shutdown State: A reactor is considered to be in this state when there is sufficient negative reactivity to ensure sub-criticality in the event of any process failure, and approved administrative safeguards are in place to prevent net removal of negative reactivity.

Helper: Member of the public who willingly and voluntarily helps in the *response* to a *nuclear or radiological emergency*. (Source: *IAEA General Safety Requirements (GSR) Part 7*)

Hostile Action: Any deliberate action, or threat of action, which could cause a *nuclear emergency*.

Imminent Release: A radioactive emission that will occur in 12 hours or less.

Ingestion Control: *Emergency response* operations in which the main aim is to avoid or reduce the *risk* from ingestion of contaminated *foodstuff* and water.

Ingestion Planning Zone: A pre-designated area surrounding a *reactor facility* where plans or arrangements are made to:

a) protect the food chain;
b) protect drinking water supplies;
c) restrict consumption and distribution of potentially contaminated produce, wild-grown products, milk from grazing animals, rainwater, animal feed; and
   Note: Wild-grown products can include mushrooms and game.
d) restrict distribution of non-food commodities until further assessments

(Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Initial Notification: The *notification* made by a *reactor facility* to Provincial and/or municipal authorities upon the occurrence of an event or condition which has implications for public safety, or could be of concern to these authorities. The criteria and channels for making such *notification* are usually described in *emergency* plans.

Internal Notification: The *notification* by an organization to its personnel who are required to respond to an *emergency*.

Intervention Level: A *radiation dose* above which a specific protective action is generally justified. (Source: Health Canada Glossary)

Iodine Thyroid Blocking: The reduction or *prevention* of the absorption of *radioiodine* by the thyroid gland, which is accomplished by the intake of a stable iodine compound (such as potassium iodide) by people exposed or likely to be exposed to *radioiodine*.

Ionizing Radiation: For the purposes of *radiation* protection, *radiation* capable of producing ion pairs in biological material(s). *Ionizing radiation* is constantly present in the environment and includes the *radiation* that comes from both natural and artificial sources, such as cosmic rays, terrestrial sources (radioactive elements in the soil), ambient air (radon), and internal sources (food and drink). (Source: CNSC Glossary)

Joint Information Centre: A joint centre for the province, *Designated Municipality*, federal government and the *reactor facility* or *nuclear establishment* that is responsible for providing information on the *emergency* to the media and the public.

Land Control: Control on the use of contaminated land for growing food products or animal feed.

Livestock Control: Quarantine of livestock in the affected area to prevent movement to other areas. Slaughter of such animals for food may be banned.

Loss-of-Coolant Accident (LOCA): A type of reactor *accident* that results from a loss of coolant due to a break in the primary heat transport system. (Source: CNSC Glossary)

Lower-tier Municipality: A *Lower-tier Municipality* is the most basic unit of local government and includes townships, towns, and cities within a county or region, but excludes Single-tier *Municipalities*. (Source: Provincial Glossary)

Malevolent Act: An illegal action or an action that is committed with the intent of causing wrongful harm. (Source: CNSC Glossary)

Megabecquerel: 10^6 *becquerel*s. (Source: CNSC Glossary)

Microsievert (µSv): One-millionth of a sievert. (Source: CNSC Glossary)

Milk Control: Preventing the consumption of locally produced milk in the area affected by a *nuclear emergency*, and its export outside the area until it has been monitored. Collection of contaminated milk, its diversion to other uses, or its destruction, may also be involved.

Millisievert (mSv): One-thousandth of a sievert. (Source: CNSC Glossary)

Ministry Action Group (MAG): The *Ministry Action Group* (MAG) is composed of the deputy minister or designate of the ministry, the senior ministry official appointed to the ministry's emergency management program committee, the ministry's emergency management program coordinator; and such other ministry employees as may be appointed by the minister. The group *shall* direct the ministry's *response* in an *emergency*, including the implementation of the ministry's *emergency* plan. (Source: Provincial Glossary)

Mitigate: Actions taken to reduce the adverse impacts of an *emergency* or *disaster*. Such actions may include diversion or *containment* measures to lessen the impacts of a flood or a spill. (Source: Provincial Glossary)

Municipality: "*Municipality*" means a geographic area whose inhabitants are incorporated (*Municipal Act*). (Source: Provincial Glossary)

Near Incident: A transborder nuclear *accident* or event at a site within 80 km of Ontario.

Notification: Conveying to a person or an organization, by means of a message, warning of the occurrence or imminence of a *nuclear emergency*, usually includes some indication of the measures being taken or to be taken to respond to it.

Nuclear Emergency: An *emergency* that has led to or could lead to the *release* of *radioactive material*, or *exposures* to uncontrolled sources of *radiation*, which pose, or could pose, a threat to health and safety, property, and the environment. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Nuclear Establishment: A facility that uses, produces, processes, stores or disposes of a *nuclear substance*, but does not include a *reactor facility*. It includes, where applicable, any land, building, structures or equipment located at or forming part of the facility, and, depending on the context, the management and staff of the facility.

Nuclear Facility: A generic term covering both *nuclear establishments* and *reactor facilities*.

Nuclear Substance: As defined in the (Federal) *Nuclear Safety and Control Act*.

Off-site: *Off-site* refers to the area outside the boundary (fence) of a *reactor facility*.

On-site: *On-site* refers to the area inside the boundary (fence) of a *reactor facility*.

Operational Directive: Direction given by the *Emergency Response Organization* to implement *operational measures*.

Operational Intervention Level (OIL): A calculated value, measured by instruments or determined by laboratory analysis that corresponds to an *intervention level*.

Notes**:**
1 *OILs* are typically expressed in terms of *dose rates* or of activity of *radioactive material released*, time integrated air concentrations, ground or surface concentrations, or activity concentrations of *radionuclides* in environmental, food, or water samples.
2 An *OIL* is a type of action level that can be used immediately by default and directly (without further assessment) to determine the appropriate protective actions and other *response* actions on the basis of an environmental measurement.

(Source: Based on *CSA N1600, General requirements for nuclear emergency management programs*)

Operational Measures: Measures undertaken by the *Emergency Response Organization* to deal with the *emergency*, including measures to enable or facilitate protective action for the public, e.g., *public alerting*, public direction, *activation* of plans, traffic control, *emergency* information, etc.

Operator: Holder of a subsisting licence issued pursuant to the *Nuclear Safety and Control Act* for the operation of a *reactor facility*.

Optimization: The process of determining a level of protection and safety that makes *exposures* and the *probability* and magnitude of potential *exposures* as low as reasonably achievable, with economic and social factors being taken into account.

Pasture Control: Removing milk- and meat-producing animals from pasture and from access to open water sources, and supplying them with uncontaminated feed and water.

Personal Monitoring: The use of *radiation* monitoring devices to assess whether persons, and their belongings, including vehicles, are contaminated or not, and, if contaminated, the type and level of *contamination*.

Personal Protective Equipment: Clothing or other specialized equipment provided to an *off-site emergency worker* to prevent or reduce their *exposure* to *radioactive material*. (Source: Health Canada Glossary)

Planning Zone: The area in which implementation of operational and protective actions are or might be required during a *nuclear emergency*, in order to protect public health, safety, and the environment.
Note**:** See definitions for *Automatic Action Zone*, *Detailed Planning Zone*, *Contingency Planning Zone*, and *Ingestion Planning Zone*.
(Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Plume: A cloud of airborne *radioactive material* that is transported in the direction of the prevailing wind from a *reactor facility*. A *plume* results from a continuing *release* of radioactive gases or particles. (This term may also be used for waterborne *radioactive material* resulting from a liquid emission. Where the context does not make it clear, this will be referred to as a waterborne *plume*).

Population Monitoring and Medical Management: The protective action strategy which includes population screening, *decontamination*, internal *contamination* assessment and medical follow-up. The purpose of this Protective Action Strategy is to reduce *exposures* to individuals. (Source: Health Canada Glossary)

Precautionary Measures: Measures which will facilitate the application and effectiveness of *protective measures*.

Preparedness: Actions taken prior to an *emergency* or *disaster* to ensure an effective *response*. These actions include the formulation of *emergency response* plans, business continuity/continuity of operations plans, training, *exercises*, and public awareness and education. (Source: Provincial Glossary)

Prevention: Actions taken to stop an *emergency* or *disaster* from occurring. Such actions may include legislative controls, zoning restrictions, improved operating standards/procedures or critical infrastructure management. (Source: Provincial Glossary)

Probability: The likelihood of an event occurring that may result in an *emergency*, *disaster* or service disruption. (Source: Health Canada Glossary)

Produce and Crop Control: Restrictions on the harvesting or processing of potentially or actually contaminated crops, vegetables and fruits. Measures include: embargoing export outside the affected area; storage to allow *radionuclide* decay; diversion to non-food chain use; destruction and disposal of contaminated produce.

Projected Dose: The highest committed effective equivalent *dose*, or committed equivalent *dose* to a specified organ or tissue, likely to be received through all applicable *exposure pathways* by the most exposed member of the *critical group* in the area for which the projection is being made.

Protective Measures: Measures designed to protect against *exposure* to *radiation* during a *nuclear emergency*.

Provincial Emergency Operations Centre (PEOC): A fully equipped facility maintained by the Office of the Fire Marshal and Emergency Management (OFMEM) that can be activated in *response* to, or in anticipation of, emergencies. The PEOC is staffed with appropriate representatives from ministries that have been delegated responsibilities for specified emergencies as well as OFMEM staff, and other *stakeholders*/partners in *emergency* management. It serves as a coordinating point-of-contact for the affected *Municipality*, provincial, and federal interests. (Source: Provincial Glossary)

Provincial Nuclear Emergency Response Plan (**PNERP):** A Cabinet approved *emergency response* plan for *reactor facility* emergencies mandated under the *Emergency Management and Civil Protection Act* and maintained by the province of Ontario. (Source: Provincial Glossary)

Public Alerting: See *Alerting*.

Public Awareness and Education Program: A program that provides focused information to a target audience to educate about protective actions to reduce the *risk* of life and property damage, in the event of an *emergency*. (Source: Provincial Glossary)

Puff: A *plume* of short duration. The distinction between a *puff* and a *plume* is a matter of time. The upper limit on the duration of a *puff* is half an hour (*See also* **Plume**).

Radiation: The emission by a *nuclear substance*, the production using a *nuclear substance*, or the production at a *reactor facility* of, an atomic or subatomic particle or electromagnetic wave with sufficient energy for ionization. (Source: Health Canada Glossary)

Radioactive Material: For purposes of nuclear security, any material that emits one or more types of *ionizing radiation*, such as alpha or beta particles, neutrons or *gamma* rays. (Source: CNSC Glossary)

Radioiodine: A substance containing radioactive iodine in a chemical form that has a metabolic pathway similar to iodide, such as inorganic compounds and metabolic forms of organic iodine that are broken down in a living organism. Some examples are the *radioisotopes* iodine-125 and iodine-131. (Source: CNSC Glossary)

Radioisotope: A variation in the form of atoms, of the same chemical element, which are distinguished by the number of neutrons in the nucleus. The number of protons remains the same, but the number of neutrons differs. For example, uranium has 16 different isotopes. (Source: CNSC Glossary)

Radiological Emergency: *Emergency* caused by an actual or environmental hazard from *ionizing radiation* emitted by a source other than a *reactor facility*.

Radiological Device (RDs): Could be lost or stolen radioactive sources which may be in locations resulting in *radiation exposure* and/or *contamination* of the public, *contamination* of a site and/or *contamination* of *foodstuff* and water supplies.

Radiological Dispersal Device (RDDs): A device that causes the dissemination of *radioactive material*.

Radionuclide (or radioactive isotope or *radioisotope*): A naturally occurring or artificially created isotope of a chemical element having an unstable nucleus that decays, emitting alpha, beta and/or *gamma* rays until stability is reached.

Reactor Facility: A facility producing greater than 10 megawatts gross thermal energy from nuclear fuel and consisting of one or more reactor units.
Note: This includes nuclear power plants and research reactors greater than 10 megawatts gross thermal energy.

Reception Centre: Locations for the initial reception, monitoring, *decontamination*, and registration of evacuated members of the public, which provides or arranges for further *emergency* social services, humanitarian assessments and support.
Notes:
1) A public *Reception Centre* is typically located in an existing facility, such as a *community* centre. Public *Reception Centres should* be beyond the *Detailed Planning Zone* boundary.
2) Examples of *emergency* social services include *emergency* shelter, food, clothing, victim registration and inquiry and personal services.
3) Examples of humanitarian support include, but are not limited to housing and family reunification. (Source: Modified IAEA Safety Guide GS-G-2.1.)

Recovery: The short-term and long-term actions taken in order to restore, to an acceptable level, both the organizations involved in, and the communities affected by, the *nuclear*

emergency and the associated *response* activities. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Release: In the context of this plan, *release* refers to the emission of *radioactive material* to the environment from a *reactor facility* in the form of either an airborne or a liquid emission.

Representative Individual: An individual that due to his/her characteristics, habits and location of residence, is representative of the more highly exposed individuals in the population. May also be referred to as Representative Person. (Source Health Canada Glossary)

Response: The actions taken during a *nuclear emergency* to reduce the magnitude of the hazard and manage its consequences, including the impact of the hazard on people, property, and the environment. (Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Response Sectors: The *Detailed Planning Zone* is subdivided into *Response Sectors* to facilitate the planning and implementation of *protective measures*.

Restoration: Operations to restore conditions to normal after a *nuclear emergency*.

Restricted Zone: The area, within which *exposure control* measures are likely to be needed, based on the results of field monitoring. (Source: Provincial Glossary)

Risk: The product of the *probability* of the occurrence of a hazard and its consequences. (Source: Provincial Glossary)

Severe Accident: A *beyond design basis accident* involving fuel degradation in the reactor core or wet storage bay.

Shall: Is used to express a requirement, i.e., a provision that the user is obliged to satisfy in order to conform to the **PNERP**.

Shelter-in-place: A directed protective action to take immediate refuge in an enclosed structure for protection from an airborne *plume*, deposited *radionuclides*, or both.
Notes**:**

1) *Shelter-in-place* is a protective action which uses the shielding properties of buildings and their potential for ventilation control to reduce the *radiation dose* to people inside. *Shelter-in-place* has varying degrees of effectiveness depending on the type of building construction.
2) *Shelter-in-place should* typically not extend beyond two days.
3) *Shelter-in-place* is utilized as a protective action if there is insufficient time to safely evacuate an area; if the *dose* projected for an area is so low that *evacuation* is not required; or the *risks* of *evacuation* are higher than *shelter-in-place* (e.g., severe weather inhibits safe *evacuation*).

(Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Should: Is used to express a recommendation or that which is advised but not required in order to conform to the **PNERP**.

Shutdown State: A subcritical reactor state with a defined margin to prevent a return to criticality without external actions. (Source: CNSC Glossary)

SI: International System of Units. (Source: CNSC Glossary)

Sievert: The International System of Units (SI) unit of equivalent *dose* and effective *dose*, equal to 1 joule/ kilogram. (Source: CNSC Glossary)

Source Term: A generic term applied to the *radioactive material released* from a *reactor facility*. It includes the quantity and type of material *released* as well as the timing and rate of its *release*. It could apply to a *release* that was currently occurring, or one which had ended, or one which could take place in the future.

Special Group: A group for which special constraints arise in the application of a protective measure, such as intensive care patients in hospitals and institutions, bedridden patients in long-term care homes, people with disabilities and/or special needs and inmates.

Stakeholder: A person, group, *community*, or organization that has a role in the management of a *nuclear emergency.* (Source: Based on *CSA N1600, General requirements for nuclear emergency management programs*)

Stochastic Effects: *Radiation*-induced health effects, such as cancer and heritable diseases, which are associated with a statistical *risk* and where no threshold has been established. The *probability* of occurrence is proportional to the *dose* (the higher the *dose* the higher the *probability* of occurrence) but the severity of the effect is independent of *dose*. (Source: Health Canada Glossary)

Support Municipality: Pursuant to **Section 7.0.2 (4)** of the EMPCA, the LGIC may, by order, specify a *Municipality* to act in a support capacity to provide assistance to *Designated Municipalities*.

Transborder Emergency: A *nuclear emergency* involving a *reactor facility* or nuclear *accident* or event outside the borders of Ontario that might affect people and property in the province.

Upper-tier Municipality: An upper-tier *Municipality* is a county or region. Upper-tier *Municipality* means a *Municipality* of which two or more *Lower-tier Municipalities* form part for municipal purposes (*Municipal Act*). (Source: Provincial Glossary)

Venting: The *release* to the atmosphere of *radioactive material* from the *containment* of a *reactor facility* through systems designed for this purpose.

Vulnerable populations: Members of the public who have additional needs before, during, and after a *nuclear emergency* in one or more functional areas.
Notes**:**

1) Functional areas can include, but are not limited to, the following:
    a) maintaining independence;
    b) communication;
    c) transportation;
    d) supervision; or
    e) medical care
2) Individuals in need of additional assistance could include those who:
    a) have disabilities;
    b) are from diverse cultures;
    c) have limited to no proficiency in the local official language; or
    d) are transportation disadvantaged

(Source: *CSA N1600, General requirements for nuclear emergency management programs*)

Water Control: Measures taken to avoid the *contamination* of drinking water supplies and sources, and to prevent or reduce the consumption of contaminated water.

Weighted Dose: See *Equivalent Dose*. Expressed in terms of Sievert (or rem).

IAEA
International Atomic Energy Agency

IAEA SAFETY STANDARDS

No. SSG-4 (Rev. 1)

for protecting people and the environment

Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants

SPECIFIC SAFETY GUIDE

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

The publications by means of which the IAEA establishes standards are issued in the **IAEA Safety Standards Series**. This series covers nuclear safety, radiation safety, transport safety and waste safety. The publication categories in the series are **Safety Fundamentals**, **Safety Requirements** and **Safety Guides**.

Information on the IAEA's safety standards programme is available on the IAEA web site:

http://www-ns.iaea.org/standards/

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at: Vienna International Centre, PO Box 100, 1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by email to Official.Mail@iaea.org.

RELATED PUBLICATIONS

The IAEA provides for the application of the standards and, under the terms of Articles III and VIII.C of its Statute, makes available and fosters the exchange of information relating to peaceful nuclear activities and serves as an intermediary among its Member States for this purpose.

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DEVELOPMENT AND APPLICATION OF LEVEL 2 PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR POWER PLANTS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world".

IAEA SAFETY STANDARD SERIES No. SSG-4 (Rev. 1)

DEVELOPMENT AND APPLICATION OF LEVEL 2 PROBABILISTIC SAFETY ASSESSMENT FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2025

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International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
tel.: +43 1 2600 22529 or 22530
email: sales.publications@iaea.org
www.iaea.org/publications

© IAEA, 2025

Printed by the IAEA in Austria
June 2025
STI/PUB/2105
https://doi.org/10.61092/iaea.buld-d6ia

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Development and application of level 2 probabilistic safety assessment for nuclear power plants / International Atomic Energy Agency.
Description: Vienna : International Atomic Energy Agency, 2025. | Series: safety standards series, ISSN 1020–525X ; no. No. SSG-4 (Rev. 1) | Includes bibliographical references.
Identifiers: IAEAL 25-01739 | ISBN 978-92-0-137424-0 (paperback : alk. paper) | ISBN 978-92-0-137524-7 (pdf) | ISBN 978-92-0-137624-4 (epub)
Subjects: LCSH: Nuclear power plants — Safety measures. | Nuclear power plants. | Nuclear reactors — Safety measures. | Nuclear power plants — Risk assessment.
Classification: UDC 621.039.58 | STI/PUB/2105

FOREWORD

by Rafael Mariano Grossi
Director General

The IAEA's Statute authorizes it to "establish…standards of safety for protection of health and minimization of danger to life and property". These are standards that the IAEA must apply to its own operations, and that States can apply through their national regulations.

The IAEA started its safety standards programme in 1958 and there have been many developments since. As Director General, I am committed to ensuring that the IAEA maintains and improves upon this integrated, comprehensive and consistent set of up to date, user friendly and fit for purpose safety standards of high quality. Their proper application in the use of nuclear science and technology should offer a high level of protection for people and the environment across the world and provide the confidence necessary to allow for the ongoing use of nuclear technology for the benefit of all.

Safety is a national responsibility underpinned by a number of international conventions. The IAEA safety standards form a basis for these legal instruments and serve as a global reference to help parties meet their obligations. While safety standards are not legally binding on Member States, they are widely applied. They have become an indispensable reference point and a common denominator for the vast majority of Member States that have adopted these standards for use in national regulations to enhance safety in nuclear power generation, research reactors and fuel cycle facilities as well as in nuclear applications in medicine, industry, agriculture and research.

The IAEA safety standards are based on the practical experience of its Member States and produced through international consensus. The involvement of the members of the Safety Standards Committees, the Nuclear Security Guidance Committee and the Commission on Safety Standards is particularly important, and I am grateful to all those who contribute their knowledge and expertise to this endeavour.

The IAEA also uses these safety standards when it assists Member States through its review missions and advisory services. This helps Member States in the application of the standards and enables valuable experience and insight to be shared. Feedback from these missions and services, and lessons identified from events and experience in the use and application of the safety standards, are taken into account during their periodic revision.

I believe the IAEA safety standards and their application make an invaluable contribution to ensuring a high level of safety in the use of nuclear technology. I encourage all Member States to promote and apply these standards, and to work with the IAEA to uphold their quality now and in the future.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered 'overarching' requirements, are expressed as 'shall' statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

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[1] See also publications issued in the IAEA Nuclear Security Series.

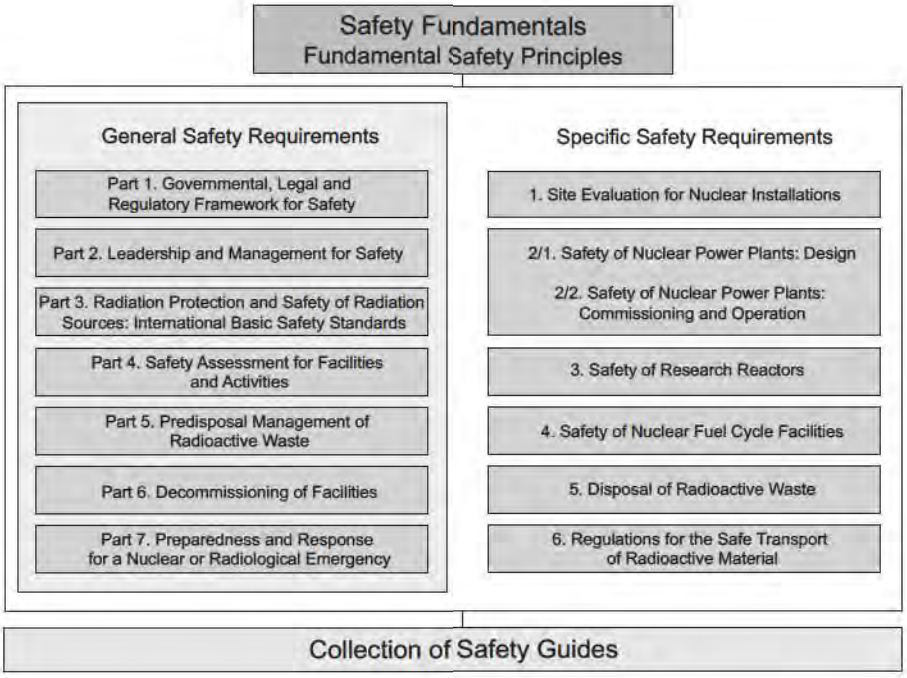

FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as 'should' statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be

used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five Safety Standards Committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the Safety Standards Committees and may provide comments on draft standards. The membership of the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards.

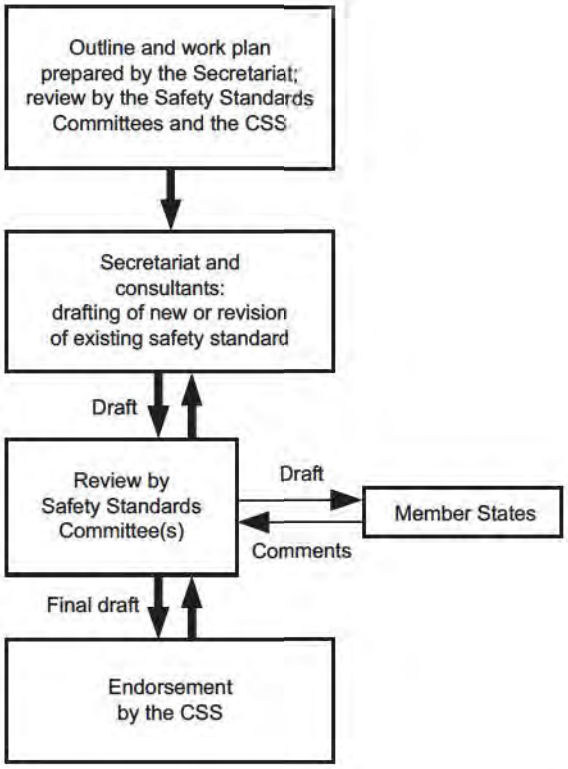

FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as they appear in the IAEA Nuclear Safety and Security Glossary (see https://www.iaea.org/resources/publications/iaea-nuclear-safety-and-security-glossary). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

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1. INTRODUCTION

BACKGROUND

1.1. IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [1], establishes principles to ensure the protection of workers, the public and the environment, now and in the future, from harmful effects of ionizing radiation. These principles emphasize the need to assess and control the inherent risk posed by nuclear installations. In particular, para. 3.22 of SF-1 [1] states:

> "To determine whether radiation risks are as low as reasonably achievable, all such risks, whether arising from normal operations or from abnormal or accident conditions, must be assessed (using a graded approach) a priori and periodically reassessed throughout the lifetime of facilities and activities."

1.2. Several IAEA Safety Requirements publications establish general and specific requirements on risk assessment for nuclear power plants. Paragraph 4.13 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2], states that "The safety assessment shall include a safety analysis, which consists of a set of different quantitative analyses for evaluating and assessing challenges to safety by means of deterministic and also probabilistic methods."

1.3. Requirement 15 of GSR Part 4 (Rev. 1) [2] states that "**Both deterministic and probabilistic approaches shall be included in the safety analysis.**" Paragraph 4.55 of GSR Part 4 (Rev. 1) [2] further states:

> "The objectives of a probabilistic safety analysis are to determine all significant contributing factors to the radiation risks arising from a facility or activity, and to evaluate the extent to which the overall design is well balanced and meets probabilistic safety criteria where these have been defined."

1.4. Requirement 42 of IAEA Safety Standards No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [3], states:

> "**A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to enable the challenges to safety in the various categories of plant states to be evaluated and assessed.**"

1.5. Paragraph 5.76 of SSR-2/1 (Rev. 1) [3] further states (footnote omitted):

> "The design shall take due account of the probabilistic safety analysis of the plant for all modes of operation and for all plant states, including shutdown, with particular reference to:
>
> (a) Establishing that a balanced design has been achieved such that no particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the overall risks, and that, to the extent practicable, the levels of defence in depth are independent;
> (b) Providing assurance that situations in which small deviations in plant parameters could give rise to large variations in plant conditions (cliff edge effects) will be prevented;
> (c) Comparing the results of the analysis with the acceptance criteria for risk where these have been specified."

Thus, a probabilistic safety assessment (PSA) contributes to assessing and verifying that a balanced design of the nuclear power plant has been achieved in relation to the overall risk from potential internal initiating events and internal and external hazards, and to preventing cliff edge effects.

1.6. PSA has been shown to provide important safety insights in addition to those provided by deterministic analysis. PSA provides a methodological approach for identifying accident sequences that can follow from a broad range of initiating events and it includes a systematic and realistic determination of accident frequencies and consequences. In international practice, three sequential levels of PSA are generally recognized:

(1) In Level 1 PSA, the design and operation of the plant are analysed in order to identify the sequences of events that can lead to core and/or fuel damage, and the corresponding core and/or fuel damage frequencies are estimated. Level 1 PSA provides insights into the strengths and weaknesses of structures, systems and components (SSCs) important to safety and of the procedures in place or envisaged to prevent core and/or fuel damage. Further recommendations are provided in IAEA Safety Standards Series No. SSG-3 (Rev. 1), Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [4].
(2) In Level 2 PSA, the chronological progression of core and/or fuel damage sequences identified in Level 1 PSA is evaluated, including a quantitative assessment of phenomena arising from severe damage to reactor fuel and/

or to spent fuel. Level 2 PSA identifies ways in which associated releases of radioactive material from fuel can result in releases to the environment. It also estimates the frequency and other relevant characteristics of releases of radioactive material to the environment. This analysis provides additional insights into the relative importance of accident prevention and mitigation measures and the physical barriers (e.g. containment) to the release of radionuclides to the environment.

(3) In Level 3 PSA, public health and other societal consequences are estimated, such as the contamination of land or food from the accident sequences that lead to a release of radioactive material to the environment.

1.7. Each PSA level could also have a different scope depending on the range of initiating events (internal and/or external to the plant) and modes of operation of the plant that are to be considered.

1.8. Level 2 PSA is a structured process. Although there may be differences in the approaches to performing a Level 2 PSA (see para. 2.6), the general main steps, in the context of the overall PSA (see para. 1.6), are shown in the two central boxes of Fig. 1 and can be described as follows:

(a) Level 1 PSA provides information on the accident sequences that lead to core and/or fuel damage and hence provides the starting point for Level 2 PSA. The accident sequences identified by Level 1 PSA might not include information on the status of the SSCs dedicated to ensuring the confinement function (e.g. the containment associated systems in pressurized water reactors) that mitigate the effects of severe accidents.

(b) The interface between Level 1 PSA and Level 2 PSA is where the accident sequences leading to core and/or fuel damage are grouped into plant damage states (PDSs) based on similarities in the plant conditions that determine the further accident progression. Some extended event trees can complete the information provided by Level 1 PSA.

(c) An accident progression event tree[1] is used to model accident progression in order to identify accident sequences that challenge the SSCs dedicated to ensuring the confinement function and that lead to releases of radioactive material to the environment.

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[1] Accident progression event trees are also termed 'containment event trees'. The term 'accident progression event tree' has been chosen throughout this Safety Guide, as was done in the ASAMPSA2 project [5], because it is more generally applicable.

(d) Source term[2] analysis is used to determine the quantities and timings of radioactive material released to the environment from each release category[3].

1.9. For practical purposes, a number of grouping tasks sometimes need to be performed for Level 2 PSA. Therefore, the process for carrying out Level 2 PSA depends on the approach selected for the grouping indicated in the two central boxes of Fig. 1, as described below:

(a) The grouping (binning) of the core and/or fuel damage sequences (extended to include the status of SSCs dedicated to ensuring the confinement function) into the PDSs that form the starting point for the Level 2 PSA. Some methodologies use a multi-step process of grouping and regrouping similar PDSs into a condensed set of PDSs to be taken forward into the accident progression event tree analysis.
(b) The grouping (binning) of the severe accident sequences identified in the accident progression event tree analysis into release categories.

1.10. Further grouping or regrouping of the release categories into a condensed set[4] to be used for Level 3 PSA may be needed. The interface between Level 2 PSA and Level 3 PSA is not addressed in detail in this publication, although it is briefly mentioned in Section 15. Level 1 PSA and Level 2 PSA of varying scopes and levels of detail have been performed for almost all nuclear power

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[2] The term 'source term' is to be understood as defined in the IAEA Nuclear Safety and Security Glossary [6] as "The amount and isotopic composition of radioactive material released (or postulated to be released) from a facility." This term is used in modelling releases of radionuclides to the environment, in particular in the context of accidents at nuclear installations or releases from radioactive waste in repositories. A more detailed definition provided in Ref. [7] is as follows:

> "The characteristics of a radionuclide release at a particular location including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, location relative to local obstacles that would affect transport away from the release point, and the temporal variations in these parameters (e.g. time of release, duration, etc.)".

[3] A release category is a group of accident progression sequences that would generate a similar source term for release to the environment. The categories are defined by attributes in relation to the release (see Section 10).

[4] Some methodologies use the term 'source term categories' to denote the final condensed set of release categories used for source term calculations. It should be noted, however, that the term 'source term category' is generally used synonymously with 'release category' and is understood by the definition provided in footnote 3. See also footnote 15.

plants in operation or under construction worldwide, whereas Level 3 PSA has been performed only for some nuclear power plants in some States.

1.11. This Safety Guide was prepared on the basis of a systematic review of relevant IAEA publications, including Refs [1–4, 8, 9] and an International Nuclear Safety Advisory Group report [10].

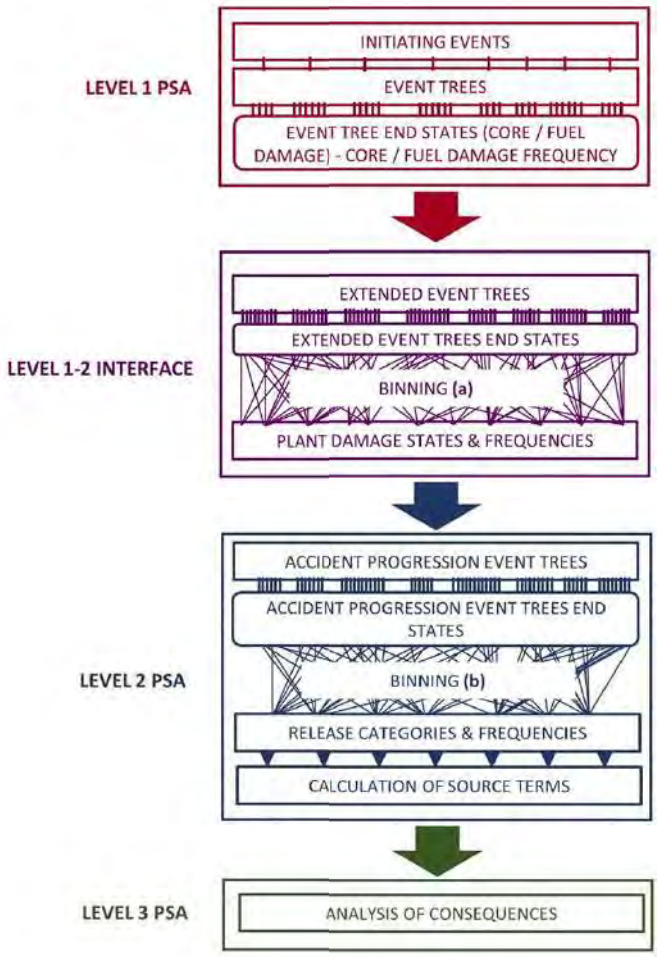

FIG. 1. General overview of the development of a typical Level 2 PSA in the context of the overall PSA.

1.12. This Safety Guide supersedes IAEA Safety Standards Series No. SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants[5].

OBJECTIVE

1.13. The main objective of this Safety Guide is to provide recommendations for meeting the requirements of GSR Part 4 (Rev. 1) [2] on performing or managing a Level 2 PSA project and the development of the Level 2 PSA for a nuclear power plant; this Safety Guide therefore complements SSG-3 (Rev. 1) [4]. It aims to promote a standard framework, standard terms and a standard set of documents for PSAs to facilitate regulatory and external peer review, in particular for Level 2 PSA results. Another objective is to provide a consistent, reliable means of ensuring the effective fulfilment of obligations under Article 14 of the Convention on Nuclear Safety [11].

1.14. The recommendations presented in this Safety Guide are based on internationally recognized good practices for current water cooled reactors. However, they are not intended to pre-empt the use of equivalent new or alternative methods. On the contrary, the use of any method that achieves the objectives of Level 2 PSA is acceptable. Although the recommendations provided in this Safety Guide are intended to reflect a technology-inclusive methodology, the details of the analysis methods might change as understanding of severe accident phenomena improves or in order to adapt to a particular reactor technology. Most of the phenomenology described as examples in this Safety Guide is directly applicable to current water cooled reactors, such as pressurized water reactors or boiling water reactors, but the phenomenology for other particular reactor technologies needs to be investigated and identified. For example, for molten salt reactors with liquid fuel, the concept of core melt is not meaningful.

SCOPE

1.15. This Safety Guide addresses the necessary methodological technical features of Level 2 PSA for nuclear power plants (both existing and new plants) in relation to its application, with an emphasis on the procedural steps and essential

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[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-4, IAEA, Vienna (2010).

elements of the PSA rather than on details of the modelling methods. This Safety Guide includes all of the steps in the Level 2 PSA process, up to and including the determination of the detailed source terms needed as input into a Level 3 PSA.

1.16. The scope of a Level 2 PSA addressed in this Safety Guide includes all modes of normal operation of the plant (i.e. startup, power operation, shutting down, shutdown, maintenance, testing and refuelling) and considers the Level 1 PSA results obtained for all potential initiating events and potential hazards (i.e. a full scope Level 1 PSA as described in SSG-3 (Rev. 1) [4]), namely: (a) internal initiating events caused by random component failures and human error, (b) internal hazards and (c) external hazards, both natural and human induced, as well as combinations of hazards, such as consequential (subsequent) events, correlated events and unrelated (independent) events. If the objectives and scope of the Level 2 PSA are limited, only the relevant recommendations provided in this Safety Guide apply.

1.17. If the aim of the PSA is to determine all of the contributions to risk to public health and society, in the calculation of the source term the PSA needs to take into account the potential for release from other sources of radioactivity at the plant, such as irradiated fuel and stored radioactive waste. Such an aim is not covered by this Safety Guide, which focuses on releases of radioactive material resulting from severe accidents in the reactor and the spent fuel pool (SFP). This Safety Guide also covers the development of a multi-unit Level 2 PSA for sites where several units are located and is suitable for use where regulatory requirements compel such an assessment as part of the quantification of the source term at the site level.

1.18. Different plant designs have different provisions to prevent or limit the release of radioactive material following a severe accident. Most designs include a containment structure or building (hereinafter referred to as 'containment') as one of the passive features dedicated to ensuring the confinement function. The phenomena associated with severe accidents are also very much influenced by the reactor technology, design and composition of the reactor core. The recommendations provided in this Safety Guide are intended to be technology-inclusive to the extent possible. However, the number and content of the various steps of the analysis assume the existence of some type of containment with related passive features and the phenomena associated with the nuclear reactor technology used.

1.19. Recommendations relating to the performance, project management, documentation and peer review of a PSA and the implementation of a management

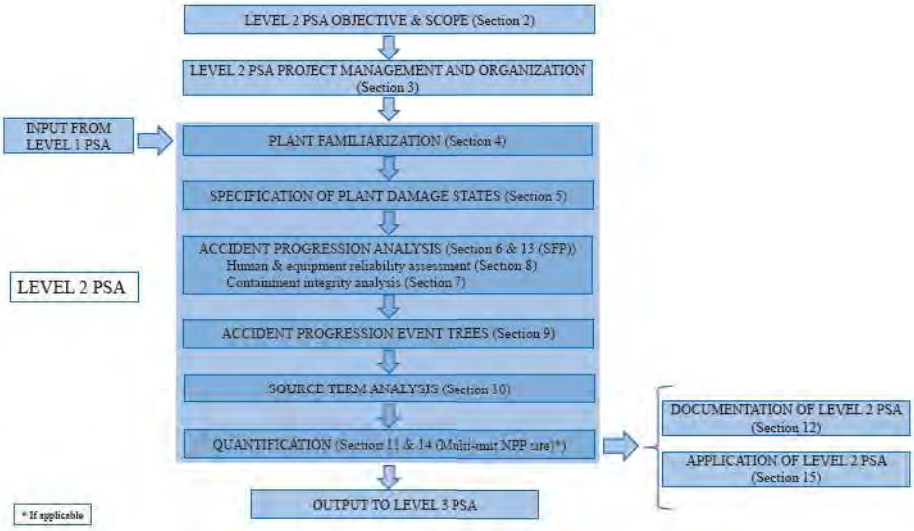

FIG. 2. Main steps in the performance of Level 2 PSA.

system in accordance with IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [8] are provided in SSG-3 (Rev. 1) [4] and are therefore not addressed in detail here. This Safety Guide addresses only the aspects of PSA that are specific to Level 2 PSA.

STRUCTURE

1.20. Sections 2–12 of this Safety Guide provide recommendations on the performance of Level 2 PSA, with each section corresponding to a major procedural step in Level 2 PSA as shown in Fig. 2. Section 13 provides recommendations on the performance of Level 2 PSA for the SFP. Section 14 provides recommendations on the performance of Level 2 PSA for a site with multiple nuclear power plants. Section 15 provides recommendations on the use and applications of a Level 2 PSA. The Appendix gives an overview of human reliability analysis in Level 2 PSA. Annex I describes various types of computer code available for simulation of severe accidents and PSA studies. Annex II presents a sample plan of activities and an outline of documentation for a Level 2 PSA. Annex III provides information on the common risk metrics used in Level 2 PSA with examples from several Member States.

2. GENERAL CONSIDERATIONS RELATING TO THE PERFORMANCE AND USE OF LEVEL 2 PSA

OBJECTIVES OF LEVEL 2 PSA

2.1. Requirement 4 of GSR Part 4 (Rev. 1) [2] states:

> "**The primary purposes of the safety assessment shall be to determine whether an adequate level of safety has been achieved for a facility or activity and whether the basic safety objectives and safety criteria established by the designer, the operating organization and the regulatory body…have been fulfilled.**"

2.2. The main objective of Level 2 PSA is to determine whether there are sufficient safety provisions[6] that have been designed in a balanced way to manage a severe accident and to mitigate the effects of such an accident to ensure that sufficient protection of people and the environment has been achieved. Level 2 PSA, in combination with Level 1 PSA, contributes to demonstrating the practical elimination[7] of plant event sequences that could lead to an early radioactive release or a large radioactive release. Recommendations on the implementation of selected requirements in SSR-2/1 (Rev. 1) [3] related to defence in depth and practical elimination of plant event sequences leading to early radioactive releases or large radioactive releases are provided in SSG-88 [12]. The sufficiency of these provisions is normally demonstrated by compliance with numerical safety goals (see paras 2.16 and 2.17), whereas the balanced design is demonstrated by analysis of the individual contributions of the safety provisions to the overall risk

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[6] For the purposes of this Safety Guide, 'safety provisions' are considered to be design solutions applied to structures, systems and components and related operational strategies.

[7] The concept of practical elimination applies to plant event sequences that could lead to unacceptable consequences (i.e. an early radioactive release or a large radioactive release) that cannot be mitigated by reasonably practicable means. Practical elimination implies that those plant event sequences have to be demonstrated to be either physically impossible or, with a high level of confidence, extremely unlikely to arise by the implementation of safety provisions in the form of design and operational features (see IAEA Safety Standards Series No. SSG-88, Design Extension Conditions and the Concept of Practical Elimination Concept in the Design of Nuclear Power Plants [12]).

profile. Safety provisions to manage severe accidents and mitigate their effects could include the following:

(a) Systems provided specifically to mitigate the effects of a severe accident, such as in-vessel or ex-vessel molten core retention features, hydrogen mixing devices or hydrogen recombiners, or filtered containment venting systems;
(b) The inherent strength of the containment or the capability for confinement and retention of radioactive material within dedicated SSCs, and the use of non-permanent equipment (e.g. diesel power generators, pumps) for accident management;
(c) Guidance to plant operators on severe accident management.

2.3. The objectives of Level 2 PSA should be defined. These might include the following:

(a) To gain insights into the progression of severe accidents and the performance of the confinement function, ensured by dedicated SSCs (e.g. the containment), to minimize the release of radioactive material.
(b) To identify plant specific challenges and vulnerabilities of the dedicated SSCs ensuring the confinement function with regard to severe accidents.
(c) To provide an input into the resolution of specific regulatory concerns and into the decision making process for a given application.
(d) To provide an input into determining compliance with probabilistic safety goals, or with probabilistic safety criteria, if these have been set. The most common probabilistic safety goals or criteria relate to large release frequencies and/or large early release frequencies, as further explained in para. 2.17.
(e) To identify the major failure modes of dedicated SSCs ensuring the confinement function (e.g. containment failure modes) and their frequencies, and to estimate the associated frequencies and magnitudes of radioactive releases.
(f) To provide an input into the development of off-site emergency preparedness and response arrangements.
(g) To provide an input into the development of plant specific accident management guidance and strategies.
(h) To provide an input into determining plant specific options with regard to design and accident management guidelines and strategies aiming to risk reduction.
(i) To provide an input into the prioritization of research activities for the minimization of risk significant uncertainties.

(j) To provide an input into Level 3 PSA consistent with the PSA objectives.
(k) To provide an input into the environmental impact assessment for the plant.
(l) To contribute to demonstrating the practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release.
(m) To gain insights into possible cliff edge effects leading to radioactive releases.
(n) To inform the choice of representative severe accidents for deterministic analysis.

2.4. The objectives reflecting the intended uses and applications of the Level 2 PSA should therefore be clearly specified at the beginning of the Level 2 PSA project. In particular for the design stage, the detail of Level 2 PSA should be sufficient to achieve the above mentioned objectives due to the difficulty or impossibility of implementing design safety features to manage severe accidents at a later stage.

SCOPE AND APPROACHES FOR LEVEL 2 PSA

2.5. Requirement 1 of GSR Part 4 (Rev. 1) [2] states:

> "**A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out at a particular stage for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity.**"

Furthermore, Requirement 14 of GSR Part 4 (Rev. 1) [2] states that "**The performance of a facility or activity in all operational states and, as necessary, in the post-operational phase shall be assessed in the safety analysis.**"

2.6. In undertaking a Level 2 PSA, there are two types of approaches likely to be encountered, depending on the overall objective of the PSA project and the software capabilities for developing the probabilistic models. The first is an integrated approach in which the Level 1 and Level 2 PSA models are developed, linked and quantified in a single software tool. The second is a separated approach, in which the Level 1 and Level 2 PSA models are not developed, linked or quantified in a single software tool such that additional steps to transfer data, information and results from Level 1 to Level 2 are needed. Reference [5] provides information on the advantages and disadvantages of each approach. In either approach, when linking the Level 1 and Level 2 PSA models, typically via the specification and

quantification of PDSs, it should be ensured that the Level 2 PSA fully takes into account the initial and boundary conditions from the Level 1 PSA model and the dependencies between the Level 1 PSA and the Level 2 PSA.

2.7. The scope of a Level 2 PSA should be determined by its defined objectives (see paras 2.2 and 2.3) and its specific intended uses and applications (see para. 15.2). Appropriate consideration should be given to the significance of key uncertainties associated with phenomena and modelling (see Section 11). Care should be taken to avoid distorting the conclusions of the Level 2 PSA through models and assumptions that are systematically biased towards particular outcomes (often for the sake of conservatism).

2.8. Commonly, the Level 2 PSA is developed as a base model for internal events. Such a base model should be used as a basis for extension to internal and external hazards.

2.9. If the starting point of a Level 2 PSA is an existing Level 1 PSA, then its output might not explicitly cover all of the features that need to be taken into account. For example, if the objective of the Level 1 PSA was the quantification of core damage frequency, then the status of the dedicated SSCs ensuring the confinement function (e.g. the containment and its associated systems) might not have been directly addressed and therefore will have to be determined as part of the Level 2 PSA or as part of the modelling of the interface between Level 1 and Level 2 PSA (e.g. specification and quantification of the PDSs).

2.10. If the scope of the PSA includes internal and/or external hazards (e.g. fire, earthquake, human induced hazards (see SSG-3 (Rev. 1) [4] and Ref. [13])), their potential impact on the confinement function and the dependent failures they could cause should be taken into account as part of the Level 2 PSA if they have not been previously taken into account in the Level 1 output. Examples of such dependent failures include failures in the containment isolation system due to cable fire, and damage of containment structures due to seismic events or transportation accidents.

2.11. If the SFP is located inside the same containment as the reactor (e.g. in some pressurized water reactor and boiling water reactor designs), the Level 2 PSA should consider the combined consequences of severe accident phenomena induced by the reactor core and the SFP for the containment and source term calculations. Recommendations on Level 2 PSA for the SFP are provided in Section 13.

2.12. If the scope of the PSA includes sources of radioactivity other than the reactor and the SFP (e.g. refuelling pool, transport casks, liquid radioactive waste, dry long term storage facility for spent fuel) that are located outside of the containment (e.g. reactor containment), then the potential risk of release from those sources should be considered. As stated in para. 1.17, releases from other sources of radioactivity at the plant, such as irradiated fuel and stored radioactive waste, are not considered in this Safety Guide.

2.13. Any analysis and assumptions associated with a Level 2 PSA should be as realistic as possible, commensurate with the intended uses and applications of the Level 2 PSA, and should include an assessment of uncertainties, consistent with the intent and scope of the study being undertaken.

2.14. If the scope of the PSA study considers Level 3 PSA, the scope of the Level 2 PSA should consider the inputs needed to conduct the Level 3 PSA.

2.15. If several reactor units (e.g. power and/or research reactors) are located at the site, the scope of the Level 2 PSA might include the impact of severe accidents for the accident management of more than one unit on the site and the corresponding aggregation of risk for these units on the site. Recommendations on conducting a Level 2 PSA for multi-unit nuclear power plants are provided in Section 14.

REFERENCE VALUES, PROBABILISTIC SAFETY GOALS OR CRITERIA AND RISK METRICS FOR LEVEL 2 PSA

2.16. The general recommendations related to reference values, probabilistic safety goals or criteria and risk metrics used in PSA provided in paras 2.10–2.15 of SSG-3 (Rev. 1) [4] are applicable to Level 2 PSA and are not repeated here. Paragraphs 2.17–2.19 of this Safety Guide provide recommendations on meeting Requirement 4 of GSR Part 4 (Rev. 1) [2] in relation to reference values and risk metrics for Level 2 PSA.

2.17. Level 2 PSA risk metrics should provide information that is meaningful and sufficient to facilitate the use and interpretation of the PSA with regard to the risk profile of the nuclear power plant. This could be represented by a sufficiently low frequency of occurrence of releases from the containment above a certain magnitude of fission products. Often, a temporal element is also included such that effective public safety measures for sheltering or evacuation can be undertaken. Large release frequency and large early release frequency are the most common risk metrics used in Level 2 PSA, but there is variation among States (see

Annex III). A large release means a release of radioactive material from the plant with significant off-site impacts necessitating off-site emergency arrangements. The release can be specified in a number of ways, including the following:

(a) As absolute quantities (in becquerels) of the most significant radionuclides released;
(b) As a fraction of the inventory of the core;
(c) As a specified dose to the most exposed person off the site;
(d) As a release resulting in unacceptable consequences.

2.18. In defining Level 2 PSA risk metrics, adequate information should be provided to understand the meaning of qualitative concepts or terms to be derived as quantitative values. In particular, the terms 'large', 'early', 'release', 'exceedance' and 'frequency' should be taken into account. When defining Level 2 PSA risk metrics, the following should be considered:

(a) The defined limits related to unacceptable consequences in terms of dose to the general public and the impact on the environment regarding limits in space and time;
(b) The evaluation of capabilities associated with severe accident management programmes and with the emergency plan to effectively arrest and control severe accident progression and implement off-site emergency response actions;
(c) The design capabilities of SSCs to effectively retain and reduce the energy, the quantity and the physical form of the chemical elements and the radionuclides contained in the fuel, the reactor core and the reactor coolant system;
(d) The radionuclides (and their radiotoxicity) that could be liberated as a result of the accident sequences;
(e) The expected probability of occurrence of severe accidents within a specific time frame;
(f) The uncertainties associated with the assumptions and the results of the Level 2 PSA study.

2.19. The following should also be taken into consideration in defining Level 2 PSA risk metrics:

(a) Probabilistic safety goals or criteria currently in use in other States;
(b) Operating experience feedback;
(c) The relationship between defined safety goals and different PSA levels (e.g. between core damage frequency and large early release frequency);

(d) Implications of the exceedance of probabilistic safety goals or criteria;
(e) Strategies to cope with the exceedance of probabilistic safety goals or criteria.

LIVING PSA

2.20. Requirement 24 of GSR Part 4 (Rev. 1) [2] states that "**The safety assessment shall be periodically reviewed and updated**", while Requirement 12 of SSR-2/2 (Rev. 1) [14] states:

> "**Systematic safety assessments of the plant, in accordance with the regulatory requirements, shall be performed by the operating organization throughout the plant's operating lifetime, with due account taken of operating experience and significant new safety related information from all relevant sources.**"

2.21. In the operating lifetime of a nuclear power plant, modifications are often made to the design of SSCs or to the way the plant is operated. Such modifications could have an impact on the level of risk associated with the plant which, in the context of this Safety Guide, is represented by the risk metrics associated with Level 2 PSA. Additional statistical data on the frequencies of initiating events, the probabilities of component failure and severe accident management guidelines will become available during plant operation. Likewise, new information and state of the art methods, tools and research results related to severe accidents may become available, which may change some of the assumptions made in the analysis and hence the risk estimates given by the Level 2 PSA. Consequently, the PSA should be kept up to date throughout the lifetime of the plant to ensure that it remains relevant to the decision making process. A PSA that undergoes periodic updating is termed a 'living PSA'. In updating a PSA, account should be taken of changes in the design and operation of the plant (e.g. operating procedures and practices, emergency operating procedures, maintenance policies, operator training and accident management practices), changes to external facilities or sources of external hazards, new technical information, more sophisticated PSA methods and tools that become available, changes in industry operating experience and new plant specific data derived from the operation of the plant (e.g. data to be used for the assessment of initiating event frequencies or component failure probabilities). The updating of a PSA should be initiated by a specified process, and the status of the PSA should be reviewed periodically to ensure that it is maintained as a representative model of the plant and is fit for purpose.

2.22. Data should be collected throughout the lifetime of the plant to check and, if necessary, update the PSA. These should include data on operating experience, in particular data on initiating events, data on component failures and unavailability during periods of testing, maintenance and repair, and data on human performance. The results from the PSA should be periodically reassessed in the light of new data.

2.23. The development of a living PSA should be encouraged as an aid to the decision making process in the normal operation of the plant. Many issues, such as evaluation of the change in risk associated with a change to the plant or a temporary change in the allowed outage time of a component, can be supported by arguments derived from a PSA. Experience has shown that this type of living PSA can be of substantial benefit to the operating organization, and its use is generally welcomed by regulatory bodies [15].

USE OF PSA IN THE DECISION MAKING PROCESS

2.24. Although the scope of the Safety Guide is limited to consideration of Level 2 PSA, this section describes some issues relevant to PSA in general, in order to provide a complete picture of the capabilities of the PSA methodology and its results. Some statements in this section do not represent explicit recommendations; rather, they provide supporting information to facilitate understanding of the context of other statements and recommendations provided in other sections of the Safety Guide.

2.25. Paragraphs 2.26–2.27 provide recommendations on meeting Requirement 1 of GSR Part 4 (Rev. 1) [2], which relates to a graded approach to the scope and level of detail of safety assessment, and Requirement 14 of GSR Part 4 (Rev. 1) [2], which relates to the scope of the safety analysis for a PSA.

2.26. Quantitative results of PSA are often used to verify compliance with probabilistic safety goals or criteria, which, depending on the scope of the PSA, are usually formulated in terms of quantitative estimates of core damage frequency or fuel damage frequency, frequencies of different types of radioactive release and societal risks, necessitating the performance of a Level 1, Level 2 or Level 3 PSA, respectively. Probabilistic safety goals or criteria do not usually specify which hazards and plant operational states are to be addressed (i.e. their scope and level of detail). Therefore, in order to use the PSA results for verification of compliance with existing probabilistic safety goals or criteria, a full scope PSA involving a comprehensive list of initiating events and hazards and all plant operational states should be performed unless the probabilistic safety goals or criteria are formulated to specify a PSA of

limited scope, or alternative approaches are used to demonstrate that the risk from those initiating events and hazards and operational states that are not in the model does not threaten compliance with the probabilistic safety goals or criteria.

2.27. A major advantage of PSA is that it provides an explicit framework for the analysis of uncertainties in risk estimates. The identification of sources of uncertainty and an understanding of their implications on the PSA model and its results should be considered an inherent part of any PSA so that, when the results of the PSA are to be used to support a decision, the impact of the uncertainties can be taken into account.

2.28. Requirement 23 of GSR Part 4 (Rev. 1) [2] states:

> "**The results of the safety assessment shall be used to specify the programme for maintenance, surveillance and inspection; to specify the procedures to be put in place for all operational activities significant to safety, and for responding to anticipated operational occurrences and accidents; to specify the necessary competences for the staff involved in the facility or activity; and to make decisions in an integrated, risk informed approach.**"

2.29. PSA should be used during the lifetime of the plant to provide an input into decision making in combination with the results and insights of deterministic safety analyses, assessment of engineering safety features and considerations of defence in depth.

2.30. PSA can provide useful insights and inputs for various interested parties, such as operating organizations (e.g. management, as well as engineering, operations and maintenance personnel), regulatory bodies, technical support organizations, designers and vendors, for making decisions on the following:

(a) Design modifications and plant modifications;
(b) Optimization of plant operation and maintenance;
(c) Safety analysis and research programmes;
(d) Regulatory issues;
(e) Scenarios to be focused on during emergency preparedness drills;
(f) Development of severe accident management guidelines;
(g) Optimization of training.

2.31. Where the results of the PSA are to be used in support of the decision making process, a formal framework for doing so should be established. The details of the decision making process will depend on the purpose of the particular

PSA application, the nature of the decision to be made and the PSA results to be used [16]. If numerical results from the PSA are to be used, reference values against which these results can be compared should be established.

2.32. The PSA should address the actual design or, in the case of a plant under construction or modification, the intended design or operation of the plant as part of the periodic safety reviews. These reviews should be clearly identified as the basis for the analysis. The status of the plant can be fixed as it was on a specific date or as it will be when the agreed modifications are completed. This needs to be done to provide a clear reference point for completion of the PSA. Later changes can be addressed in the framework of the periodic safety reviews, as part of a living PSA programme, as described in paras 2.20–2.23.

2.33. For a plant in the design stage, the results of PSA should be used as part of the design process to assess the level of safety. In this case, the insights gained from PSA should be considered in combination with the insights gained from the assessment of engineering safety features and deterministic safety analysis to make decisions about the safety of the plant. Such decisions should be the result of an iterative process aimed at ensuring that national requirements and probabilistic safety goals or criteria are met, that the design is balanced, and that the risk is as low as reasonably achievable.

2.34. For a plant in the design stage or at a periodic safety review stage, the results of the PSA (including uncertainties, importance analysis and sensitivity studies) should be compared with the probabilistic safety goals or criteria if these have been specified in national regulations or guidelines. Such a comparison should be performed for all probabilistic safety goals or criteria defined for the plant, including those that address system reliability, core damage frequency, fuel damage frequency, radioactive release frequency, health effects for the public and other off-site consequences such as land contamination and restrictions on foodstuffs. If numerical probabilistic safety criteria or goals are not specified, risk reduction possibilities can nevertheless be examined based on the Level 2 PSA results.

2.35. The PSA should aim to identify all accident sequences that contribute to risk in a non-negligible way. If the PSA does not address all significant contributions to risk, or if its scope is reduced (see para. 2.25) (e.g. if it omits external hazards or shutdown states), then the insights from the PSA about the level of risk from the plant, the balance of the risk contributors and the need for changes to be made to the design or operation to reduce risk might be limited. Such limitations should be acknowledged when using PSA to support decision making and should be addressed by alternative analyses as necessary.

2.36. The results of the PSA should be used to identify weaknesses in the design or operation of the plant and in actions considered in severe accident management strategies (see also IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [17]). These weaknesses can be identified by considering the contributions to the risk from groups of initiating events and the importance measures[8] for SSCs and for human errors. If the results of the PSA indicate that changes could be made to the design or operation of the plant to reduce risk, the changes should be incorporated where reasonably achievable (e.g. taking the relative costs and benefits of any modifications into account).

3. PROJECT MANAGEMENT AND ORGANIZATION FOR PSA

3.1. Requirement 22 of GSR Part 4 (Rev. 1) [2] states that "**The processes by which the safety assessment is produced shall be planned, organized, applied, audited and reviewed.**" The recommendations on project management and organization for PSA provided in section 3 of SSG-3 (Rev. 1) [4] are also applicable to Level 2 PSA and are therefore not repeated here. Only those aspects that are particularly important for Level 2 PSA are presented in this section.

DEFINITION OF OBJECTIVES OF THE LEVEL 2 PSA PROJECT

3.2. Paragraphs 3.3–3.5 provide recommendations on meeting Requirement 4 of GSR Part 4 (Rev. 1) [2] with regard to defining the objectives of the Level 2 PSA project.

3.3. Differing end uses place differing emphases and needs on the various inputs into, and components of, a Level 2 PSA. The objectives of the Level 2 PSA project should be set out fully at the beginning of the project and should be in agreement with the main objective of the Level 2 PSA and intended purposes, as described in Section 1.

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[8] Typical importance measures used in probabilistic safety assessment are Fussell–Vesely importance, risk reduction worth, risk achievement worth and Birnbaum importance (see para. 5.171 of SSG-3 (Rev. 1) [4]). These importance measures provide a perspective on how an individual basic event, groups of basic events, credited systems and groups of initiating events contribute to the overall risk profile.

3.4. The limitations of both Level 1 PSA and Level 2 PSA should be identified, taken into account and documented in the Level 2 PSA project, with the objectives, intended uses and applications of the Level 2 PSA all being taken into account.

3.5. The objectives of the Level 2 PSA project should be understandable and achievable by the users of the Level 2 PSA. In this context, previous experiences from the management of other Level 2 PSA projects are very valuable and should be gathered.

DETERMINATION OF THE SCOPE OF THE LEVEL 2 PSA PROJECT

3.6. Paragraphs 3.7–3.8 provide recommendations on meeting Requirements 1 and 14 of GSR Part 4 (Rev. 1) [2] in relation to the scope of the Level 2 PSA project.

3.7. The scope of the Level 2 PSA project should be determined by the overall scope of Level 2 PSA, as described in paras 2.5–2.15. The scope of the Level 2 PSA project should follow a graded approach to defining the scope and the methods used for modelling the severe accident phenomena and for the contribution of the SSCs to the risk of a radioactive release depending on their source (see para. 1.17). A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs that are part of an installation containing other potential sources of radioactive releases (e.g. fault tree and event tree development, assumptions related to human reliability analysis or equipment reliability data, fragility curves (if applicable) and reliability of digital instrumentation and control systems, including computer based systems used to control the process in the installation).

3.8. In compliance with para. 2.14, consideration should be given to the input requirements for a Level 3 PSA, as applicable, when determining the scope of the Level 2 PSA project. The ultimate product of a Level 2 PSA is a description of a number of challenges to the containment, a description of the possible responses of that containment and an assessment of the consequent releases considering the source term calculations described by the release categories definitions, frequency and characterization of their magnitude. The description includes the inventory of material released, its physical and chemical characteristics, and information on the time, energy, duration and location of the releases. Subsidiary products of the Level 2 PSA are a description of a number of challenges to the dedicated SSCs ensuring the confinement function (e.g. the containment), and a description of the possible responses of those SSCs.

PARTICULAR FEATURES OF PROJECT MANAGEMENT FOR LEVEL 2 PSA

3.9. Requirement 5 of GSR Part 4 (Rev. 1) [2] states that "**The first stage of carrying out the safety assessment shall be to ensure that the necessary resources, information, data, analytical tools as well as safety criteria are identified and are available**." Recommendations on the decisions that PSA project managers should take and on the supervision, coordination and implementation of various tasks are provided in paras 3.3–3.9 of SSG-3 (Rev. 1) [4]. Those recommendations are also applicable to Level 2 PSA and are not repeated here. One aim of project management for Level 2 PSA is to ensure that the PSA being produced does indeed represent the plant in its 'as is' condition and reflects realistic operating practices to the extent possible and that it does take account of recent developments in methods, models and data.

3.10. If the starting point is an existing Level 1 PSA, then coordination with the Level 1 PSA management team should be established. If the starting point is to develop jointly a Level 1 PSA and Level 2 PSA, a single management team could be established.

3.11. Although the basic framework and methods of Level 2 PSA are well established, the analysis in Level 2 PSA demands high levels of expertise and technical resources. Even when many resources are employed, analyses of both the behaviour of the containment during the severe accident and the radiological source terms are subject to large uncertainties associated with phenomena. The expertise and technical resources needed should be considered, in accordance with the scope of the Level 2 PSA project, in the selection of computer codes (see paras 3.17 and 3.18), and in the selection, training and qualification of personnel (see paras 3.19–3.22).

3.12. To meet the requirements established in GSR Part 2 [8], a management system for the project should be implemented with due consideration given to the safety implications of the results of the Level 2 PSA and its intended uses. In particular, the application of expert judgment should be justified and managed through a controlled and documented process. Provisions should be made by the Level 2 PSA project management team for establishing independent review processes or performing comparative studies, as appropriate (see paras 3.24–3.29). Recommendations on the technical review of relevant aspects of the analysis, project documentation and configuration control are provided in Section 12.

3.13. The Level 2 PSA project management team should aim to ensure that the insights gained from performing the analysis relating to plant vulnerabilities and severe accident management are properly understood by the plant management and operating staff, so that the operating organization gains ownership of the Level 2 PSA, and by the regulatory body and other relevant interested parties.

3.14. Recommendations related to the establishment of a quality assurance programme for the development of Level 2 PSA studies as part of the duties of the Level 2 PSA project management team are defined in paras 3.15 and 3.16 of SSG-3 (Rev. 1) [4] and are not repeated here. This quality assurance process should include activities related to the independent review performed for the Level 2 PSA (see paras 3.24–3.29).

3.15. Taking into account the stage in the plant lifetime (e.g. design, operation) and the objectives to be reached, the Level 2 PSA project management team should specify what prior information is key for the successful development of the Level 2 PSA. Such information might be provided, for example, on the following:

(a) Selection of staff and responsibilities (see paras 3.19–3.22);
(b) Scope and level of detail to be achieved, including the predefinition of a sufficient number of PDSs and/or of release categories (see paras 3.6–3.8);
(c) Planning and scheduling of project activities, including identifying the need for research related activities, software development, verification and validation, and training;
(d) Availability and collection of plant data in relation to SSCs, severe accident phenomena, human factors, emergency operating procedures and/or severe accident management guidelines, and internal and external hazards [13];
(e) Modelling assumptions related to the PSA (e.g. integral, mechanistic or dedicated computer codes);
(f) Procedures for using expert judgement;
(g) Definition of the format and amount of information to be presented as the Level 2 PSA results, including the uncertainty and importance analysis and sensitivity studies;
(h) Scope and structure of the documentation for the Level 2 PSA (see Section 12).

SELECTION OF SOFTWARE, APPROACHES AND METHODS

3.16. Requirement 18 of GSR Part 4 (Rev. 1) [2] states that "**Any calculational methods and computer codes used in the safety analysis shall undergo**

verification and validation." The selection of computer codes to be used for the Level 2 PSA should follow the recommendations for PSA in general provided in paras 2.5 and 2.6 of SSG-3 (Rev. 1) [4]. In addition, specific codes for the probabilistic modelling on Level 2 PSA should be considered advantageous to have the possibility to deal with multiple point branches as well as with correct quantification of success branches.

3.17. The computer codes selected for use in Level 2 PSA should go through a process of verification and validation covering all severe accident phenomena encountered during the accident progression. Models and correlations introduced in the computer codes used for severe accident progression analysis for Level 2 PSA should be verified and validated by experiments and/or benchmarking to ensure a sufficient level of confidence in the results obtained and to minimize the uncertainties introduced by the simplifications and assumptions related to the physical phenomenon considered. However, it should be recognized that the level to which verification and validation can be performed for severe accident progression analysis codes is much lower than for other codes used to support PSA, such as the thermohydraulic codes used to support the success criteria for the credited systems[9] in the Level 1 PSA. This is because there is, in general, a limited applicability of experimental results to real reactor conditions, as it is not always possible to conduct experiments that reflect the extreme conditions that occur in a severe accident and the scale of the geometry of the reactor coolant system and the reactor containment.

3.18. The selection of an integrated or separated approach (see para. 2.6) and the methods for accident progression event tree construction (see Section 9) should be consistent with the scope and objectives of the Level 2 PSA project.

TEAM SELECTION FOR A LEVEL 2 PSA PROJECT

3.19. In the selection of the Level 2 PSA team, it should be ensured that there is an adequate level of expertise in the following areas:

(a) Knowledge of the design and operation of the plant;
(b) Knowledge of severe accident phenomena and challenges to the containment;

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[9] 'Credited systems' are systems credited in the PSA, which include operating and standby safety systems and non-safety systems whose operation during an accident can help prevent an undesired end state (e.g. core damage, fuel damage). Also 'credited SSCs' is a term used in this publication to specify particular structures or components credited in the PSA.

(c) Knowledge of physics regarding radioactive material;
(d) Knowledge of PSA in general, and of Level 2 PSA techniques in particular.

The depth of the team's expertise can be different depending on the stage in the lifetime of the plant at which the Level 2 PSA is performed, the scope of the Level 2 PSA and the intended applications of the Level 2 PSA. To the extent possible, there should be extensive participation of the plant engineers and utility personnel or designers (e.g. if performed at the design stage) as well as probabilistic safety analysts specialized in severe accident phenomena and other Level 2 PSA disciplines.

3.20. The Level 2 PSA project management team should provide working arrangements that ensure good interaction and communication between all of the members of the team, including project managers and analysts. Effective communication throughout the project is essential. In addition, the Level 2 PSA project management team should aim to ensure that, as the analysis progresses and insights are developed, the approaches to the different technical areas are modified as necessary to ensure that the analysis is progressing in a coherent way and that there is a balance of efforts across all topics.

3.21. The following recommendations apply to Level 2 PSA team members using computer codes:

(a) They should have adequate training on the computer code(s) used.
(b) They should have a sound knowledge of the models and methods implemented in the computer code(s).
(c) They should have sufficient understanding of the limitations of the computer code(s) in relation to the phenomena to be modelled.
(d) They should be familiar with the guidance on the computer code(s) and the procedures for implementing the models in the computer codes.
(e) They should have adequate capacity to evaluate the results of the computer code(s).

3.22. For a nuclear power plant in operation, the following members should be considered for inclusion in the Level 2 PSA team:

(a) A technical project leader responsible for coordination among all of the project experts;
(b) Experts in, and staff working on, the design and operation of the plant (particularly the containment associated systems), the emergency operating procedures and the severe accident management guidelines;

(c) Experts in severe accident phenomena, performance of the containment, uncertainties associated with severe accidents, chemical and physical processes governing accident progression, loads generated over the containment, releases of radionuclides and computer codes for the analysis of severe accidents;
(d) Experts in the structural design, the load bearing capacity and the failure modes of the containment;
(e) Experts in developing event tree analysis, fault tree analysis, human reliability analysis, uncertainty analysis and statistical methods, in particular for Level 2 PSA;
(f) Experts in processes for expert elicitation and judgement for Level 2 PSA computer codes and Level 1 PSA.

INDEPENDENT VERIFICATION

3.23. Requirement 21 of GSR Part 4 (Rev. 1) states that "**The operating organization shall carry out an independent verification of the safety assessment before it is used by the operating organization or submitted to the regulatory body.**"

3.24. The main objective of the independent verification of the Level 2 PSA is to confirm that the methods and approaches selected, the probabilistic and deterministic models used, and assumptions and data considered have been applied in an adequate manner to meet the applicable safety objectives and requirements regarding the PSA. It is considered good practice to conduct an internal independent verification as a quality assurance process integrated into project management when developing the Level 2 PSA. This internal independent verification, if conducted, should look at each aspect of the Level 2 PSA (e.g. data and computers codes used, interface with Level 1 PSA, supporting studies, assumptions in the event trees, event tree quantifications, results interpretation) to ensure that it is of a sufficient quality for its purposes. This internal independent verification process may help to identify some sources of uncertainty (see, e.g., paras 5.14, 6.24–6.27, 7.23–7.29, 8.19–8.23. and 11.21–11.24).

3.25. Since the development of the Level 2 PSA and the design of the nuclear power plant may be conducted in parallel as part of the iterative design process of the nuclear power plant, the licensee should carry out an independent verification (e.g. peer review) to ensure that the Level 2 PSA results relate only to the design and operation of the nuclear power plant as submitted to the regulatory body for approval (i.e. in accordance with the scope of the document to be submitted to

the regulatory body) and comply with relevant regulatory requirements related to reference values and risk metrics for Level 2 PSA.

3.26. The independent verification of the Level 2 PSA performed by or on behalf of the licensee should be conducted by a group of experts or an institution different from that which developed the Level 2 PSA (e.g. a group or institution external to the licensee, sometimes from a different State) to ensure that the Level 2 PSA conforms to current, internationally recognized good practices. The licensee should ensure independence between the organization conducting the Level 2 PSA and the group of experts or the institution in charge of the independent verification.

3.27. Level 2 PSA uses a number of analytical methods and approaches to model complex phenomena with their associated uncertainties, based on computational tools that might have limited resources for validation and use of expert judgement. Therefore, the licensee should ensure that the group of experts or institution in charge of the independent verification has an adequate level of expertise to satisfactorily evaluate the data, assumptions and models (i.e. deterministic and probabilistic) considered in the Level 2 PSA (see para. 3.29).

3.28. The results of the independent verification of the Level 2 PSA should be reported in a separate document, which is to be presented to the regulatory body upon request.

3.29. The report compiling the results of the independent verification of the Level 2 PSA should consider the assessment of the technical adequacy of the following:

(a) The development, grouping and quantification of PDSs;
(b) The analysis of accident progression and the associated systems;
(c) The models of phenomena that could occur in relation to the behaviour of the containment of the nuclear power plant following core damage;
(d) The accident progression event tree models and supporting models as well as the methods for solution of the logic models;
(e) The probability development (e.g. phenomena probabilities based on data or expert judgement);
(f) The development, grouping, quantification and source term characterization of the release categories;
(g) Supporting calculations, correct and appropriate application of codes;
(h) Structural analysis and/or fragility curves;
(i) The models for considering the human reliability analysis;
(j) The consideration of equipment reliability, taking into account the equipment qualification or survivability (in particular for severe accidents scenarios);

(k) The uncertainties and the sensitivity analysis performed (e.g. the bases for the selection of probability distributions of uncertain parameters and assignment of their distribution parameters).

4. FAMILIARIZATION WITH THE PLANT DESIGN AND SEVERE ACCIDENT MANAGEMENT

IDENTIFICATION OF DESIGN ASPECTS IMPORTANT TO SEVERE ACCIDENTS

4.1. This section provides recommendations on meeting Requirements 6–13 of GSR Part 4 (Rev. 1) [2] for Level 2 PSA. Before starting the analysis, the entire Level 2 PSA team should become familiar with the design and operation of the plant. The aim should be to identify and highlight plant SSCs and operating procedures that can influence the progression of severe accidents, the response of the containment and the transport of radioactive material inside that containment. Design features that can influence the progression of a severe accident and Level 2 PSA depend on the particular reactor design and technology, and might include coolers employing fans or water, the containment heat removal system, containment sprays and/or filtered containment venting systems, the containment exhaust system and suppression pools, a dedicated set of steam relief valves and hydrogen control systems (i.e. ignitors, recombiners). This familiarization process should cover the reactor building, auxiliary building, secondary containment and other relevant structures and buildings, which differ depending on the reactor technology and design. For existing plants, familiarization with the plant should include a plant walkdown with the participation of operating personnel and plant technical staff. Interviews with operating personnel and plant technical staff fulfilling relevant roles from a Level 2 PSA perspective should also be conducted.

4.2. The specific reactor technology and plant features that might influence the progression of a severe accident should be identified and characterized. Examples of the features that should be identified for light water reactors are as follows:

(a) The area under the reactor pressure vessel. This is important with regard to the behaviour of molten core material after it exits the bottom of the reactor pressure vessel, since the area influences the extent to which the molten core material spreads and its coolability.

(b) The flow paths from the area under the reactor pressure vessel to the main containment volume. Restrictions to the flow or other geometric aspects of the flow path reduce the extent to which core debris is dispersed following a lower head failure. This is particularly important for high pressure melt ejection in a light water reactor.
(c) A highly compartmentalized containment configuration. This limits the extent to which combustible gases mix and become distributed in the containment atmosphere.
(d) Features that could lead to containment bypass sequences.

These and other plant specific design features should be identified for further investigation.

4.3. Examples of key design features of the plant that are significant in respect of the progression and mitigation of severe accidents are listed in Table 1. In addition to plant features, relevant operating procedures and severe accident management guidelines should also be considered.

TABLE 1. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES

Key plant and/or containment design feature	Comment
Reactor	
Reactor type	Boiling water reactor, pressurized water reactor, advanced gas cooled reactor or other
Power level	Total thermal power at steady state
Type of fuel mix/type of cladding	Oxide, mixed oxide/zircaloy, stainless steel, ceramic or other
Core	
Mass of fuel and mass of cladding	Actual operational values
Fuel assembly geometry	Actual operational values
Type and mass of control rods	Actual operational values

TABLE 1. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES (cont.)

Key plant and/or containment design feature	Comment
Spatial distribution of reactor power	Typically axial and radial peaking factors
Decay heat	Total decay heat level as a function of time
Radioactive material inventory	Full inventory of radionuclides in the core
Reactor coolant system	
Reactor coolant and moderator types	Water, heavy water, CO_2, helium and others
Reactor coolant system coolant/ moderator volume	As designed and fabricated
Accumulator volume and pressure set point and number	Actual operational values for each type of accumulators
Reactor coolant system depressurization devices and procedures	Set point and procedures
Pressure relief capacity	Actual operational values
Isolation of containment penetrations connected to the reactor coolant system	Potential for containment bypass
Systems actuation mechanism	Passive or active
Safety systems injection volume and pressure set point	Actual operational values
Containment [a]	
Containment geometry	Shape and separation of internal volumes
Containment free volume	As-built, taking into account displacement by structures

TABLE 1. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES (cont.)

Key plant and/or containment design feature	Comment
Containment design pressure and temperature	A realistic assessment of maximum capacity is needed for the PSA
Containment design leakage and conditions of leakage	Actual operational values
Containment material construction	Steel, concrete, metallic liner and other
Operating pressure and temperature	Actual operational values
Hydrogen control mechanisms	Provision of inertness, ignitors, passive recombiners, other
Suppression pool volume	Water available for containment pressure control or fission product retention
Containment cooler capacity and set points	Actual operational values
Containment associated systems	Actual operational values, logic
Concrete aggregate of each containment structure	Specify chemical content
Design of cavity, keyway or pedestal	Dispersive versus non-dispersive
Flooding potential of cavity or pedestal	Flooded or dry
Sump(s), volume filters and location(s)	Geometric details, identification of materials (e.g. for painting or pipe insulation) potentially affecting sump filter clogging
Heat removal paths from reactor and containment	Layout, location and operation
Configuration of heat sink	Operational procedure

TABLE 1. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES (cont.)

Key plant and/or containment design feature	Comment
In-containment refuelling water storage tank or refuelling water storage tank or other in-containment water storage tank	Location, volume and operation
Proximity of containment boundaries	Distance from reactor pressure vessel and cavity or pedestal
Containment venting procedure and location	Location of vent line and actuation procedure
Response to external hazards	Structural damage due to seismic events, flooding events or transportation events
Potential for containment isolation failure	Penetration arrangements and reliability of seal materials for containment isolation
Potential for cooling of molten core	The design of some plants (recent or backfitted) includes features for cooling of the spread molten core
	Spent fuel pool
Geometry	Shape, separation into sections, specific layout
Capacity and arrangement	Number of maximum and actual stored spent fuel assemblies, rack design, loading pattern (if any)
Decay heat	Total decay heat in normal storage conditions and for emergency unloaded core
Radioactive material inventory	Full inventory of radionuclides in the fuel stored in the SFP
Design parameters	Nominal coolant temperature and level, maximum allowed coolant temperature, minimum allowed coolant level

TABLE 1. EXAMPLES OF KEY PLANT AND/OR CONTAINMENT DESIGN FEATURES (cont.)

Key plant and/or containment design feature	Comment
Safety features	Nominal and minimum flow rate, coolant inventory, soluble absorber concentration, nominal and maximum temperature of the coolant
Materials and composition	Steel, concrete, other

[a] The specific information listed here might change in some areas for plants without a pressure retaining containment (e.g. nominal leak rate needs to be included for plants with structures that provide a confinement function) or with a different type of containment.

CONSIDERATIONS REGARDING MULTIPLE UNITS OR MULTIPLE INSTALLATIONS WITH RADIOACTIVE SOURCES ON A SITE

4.4. Paragraphs 4.5–4.9 are intended to provide an overview of key aspects to be identified from the plant familiarization perspective when performing a Level 2 PSA for a site where multiple units or multiple installations with radioactive sources are located. Recommendations on Level 2 PSA for multi-unit sites are provided in Section 14.

4.5. Site organizational aspects should be identified and recognized as an important aspect affecting the modelling of Level 2 PSA for multi-unit sites.

4.6. Performance shaping factors considered for the human reliability analysis should take into account conditions related to field operations and environmental conditions when several units are at different stages of the severe accident progression or induced by the impact of internal or external hazards and their combinations (see also Ref. [13]).

4.7. Considerations related to equipment (either installed or non-permanent) and systems that are shared among or common to all units on the site, should be identified.

4.8. Potential common cause failures among similar equipment at different units should be identified for the purpose of the development of the Level 2 PSA model for multi-unit sites.

4.9. The availability, capability and accessibility of the ultimate heat sink and of electrical power supply sources for multiple units on a site should be considered.

REVIEW OF STRATEGIES TO COPE WITH SEVERE ACCIDENT ASSOCIATED PHENOMENA

4.10. Paragraphs 4.11–4.14 provide an overview of the key aspects to be considered from the plant familiarization perspective in relation to strategies to cope with severe accident associated phenomena when performing a Level 2 PSA.

4.11. As part of plant familiarization, the analyst should collect available documentation on the strategies implemented at the plant to cope with severe accident associated phenomena and should become familiar with the priorities and actions contained within these strategies. Strategies developed to cope with severe accident progression generally include those aimed at (a) protecting the confinement function, including preventing the containment bypass, and (b) if applicable, protecting the reactor building where the SFP is located. Depending on the reactor design, strategies should also address protection of the proper functioning of filtered venting systems in the auxiliary building, and management of leakage of liquid effluent from the reactor containment in case of recirculation of contaminated water outside the containment. During the progression of a severe accident involving the degradation of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors), two important strategies are considered: first, in-vessel cooling and retention of damaged fuel (e.g. in-vessel melt retention for some reactor technologies such as water cooled, metal cooled and molten salt); and second, ex-vessel cooling and retention of damaged fuel (e.g. ex-vessel corium[10] cooling for some water cooled reactor designs) (see also paras 4.13 and 4.14).

4.12. The analyst should also be familiar with other strategies related to severe accident management. These strategies may include the SFP and the long term phase of the severe accident, such as the control of combustible gases in the atmosphere of the containment, the control of the pressure inside the containment

[10] Corium is the material formed during the meltdown of a nuclear reactor. It is composed of nuclear fuel (uranium or plutonium) and material that melts on contact with the fuel.

and the control of radioactive releases from the containment. SSG-54 [17] provides recommendations on the long term phase of the severe accident related to the use of non-permanent equipment, the maintenance and inspection of non-permanent equipment, waste management due to long term actions such as water treatment, limits on dose rates to ensure the operator actions, and the availability of electricity, compressed air or water sources.

In-vessel melt retention

4.13. For water cooled reactors, the in-vessel melt retention strategy is intended to ensure a passive and/or active flooding of the reactor pressure vessel cavity up to a level high enough to ensure and maintain, with sufficient confidence, the integrity of the reactor pressure vessel by cooling it from outside and, if necessary, by additional cooling of the corium inside by in-vessel water injection. For other reactor technologies, in-vessel melt retention might be defined slightly differently, depending on the specifics of that reactor technology (e.g. non-pressurized reactor coolant system). Information related to the success of this strategy should be collected during the familiarization task, for example on the following:

(a) Design of safety provisions for passive and/or active reflooding of the reactor cavity;
(b) Design of safety provisions for in-vessel water injection, where applicable (e.g. to reduce the risk of the focusing effect[11]);
(c) Reactor pressure vessel insulation (e.g. for water cooled reactors, this allows consideration of sufficient water circulation between the reactor pressure vessel wall and the insulation, and the evacuation path of the produced steam to the upper volume of the containment);
(d) Design solutions related to reactor pressure vessel internals (e.g. a large mass of steel in the corium relocated in the lower plenum may reduce the risk of the focusing effect at the reactor pressure vessel wall);
(e) Reactor pressure vessel design (e.g. geometry, thickness and low bottom reactor pressure vessel penetrations);
(f) Water inventory available (i.e. affecting the time delay of corium arrival in the lower plenum and therefore reducing the residual heat removal).

[11] The 'focusing effect' phenomenon involves a thin metal layer on top of a large oxidic pool. If the radiative heat transfer on top is inadequate to discharge the thermal power received from below, the temperature of the metal layer rises, and increasing amounts of energy flow are directed to the vessel wall. This focusing increases as the metal layer thickness decreases [18]. This effect can induce reactor vessel rupture.

Ex-vessel corium cooling

4.14. For water cooled reactors, ex-vessel corium cooling can consist of a passively controlled and gradual spreading of the corium outside of the reactor pressure vessel on a surface, allowing effective corium cooling by passively and actively injecting water up and down the corium layer. Information related to the success of this strategy should be collected during the familiarization process, for example on the following:

(a) Analysed reactor vessel failure modes that support the strategy;
(b) Operator actions that support the strategy;
(c) The configuration of the corium spreading surface (relevant to reducing downward heat flux to the concrete and hence reducing its ablation);
(d) Potential sequences for a wet reactor cavity (relevant to the risk of steam explosion);
(e) Functional criteria (e.g. timing, volume, automatic or manual actuation) for safety provisions needed for residual heat removal from the corium spreading surface (e.g. passive, active or a combination of both);
(f) The chemical composition of materials in the corium spreading surface (e.g. provisions to reduce the risk of recriticality or to facilitate the corium cooling).

COLLECTION OF INFORMATION IMPORTANT TO SEVERE ACCIDENT PROGRESSION ANALYSIS

4.15. Requirement 19 of GSR Part 4 (Rev. 1) [2] states that "**Data on operational safety performance shall be collected and assessed.**" When the PSA team has developed a general understanding of the plant design, phenomena[12] and features that might influence severe accidents and releases of radioactive material, the quantitative data that are necessary to perform the plant specific analysis should be collected and organized. The data necessary for the Level 2 PSA depend in part on the scope of the analyses and the nature of the computational tools. For example, the amount and type of input data collected may depend on the plant specific computer model used to calculate accident progression. Detailed architectural and construction data for the containment structure should be collected to develop plant specific model calculations of the containment performance if such calculations are necessary for the scope of the containment performance analysis.

[12] Phenomena could be obtained from a Phenomena Identification and Ranking Table analysis for severe accidents, if available.

4.16. Data should be obtained from sources such as the following:

(a) Design documents and/or plant licensing documents, such as the safety analysis report, technical specifications and system descriptions;
(b) As-built drawings;
(c) Plant specific normal operating, maintenance or test procedures;
(d) Information on automatic actuations in the plant;
(e) Emergency operating procedures and severe accident management guidelines;
(f) Engineering calculations or analysis reports;
(g) Observations during plant walkdowns and/or walkdown reports;
(h) Construction standards;
(i) Regulatory requirements;
(j) Vendor manuals;
(k) Other relevant plant documents.

References to the source(s) of data should be recorded as part of the PSA documentation.

4.17. If the intent is to use data from a reference plant in the development of the Level 2 PSA, the plant specific data should be compared with reference plant values. Such a comparison is of great value in determining whether the two plants are in fact similar and would therefore likely have similar vulnerabilities. Table 2 lists examples of design features of the plant and containment of water cooled reactors for comparison with those of other plants and how they can be used. However, considering that there are always differences from the reference plant, great care should be applied when drawing conclusions from such a comparison.

TABLE 2. SAMPLE COMPARISON OF PLANT AND CONTAINMENT DESIGN CHARACTERISTICS OF WATER COOLED REACTORS

Parameter and design feature	Significance or comparability
Ratio of reactor power to reactor coolant system volume	Accident progression times, time for recovery actions
Ratio of reactor power to containment volume	Scaling of containment loads
Ratio of zirconium mass to containment free volume	Potential for combustion and scaling of containment loads

TABLE 2. SAMPLE COMPARISON OF PLANT AND CONTAINMENT DESIGN CHARACTERISTICS OF WATER COOLED REACTORS (cont.)

Parameter and design feature	Significance or comparability
Core catcher	Spreading area, corium cooling devices
Vessel cavity to containment pathways	Potential for melt ejection and dispersal in containment at high pressure
Concrete aggregate (composition)	Non-condensable gas generation and radioactive release during molten core–concrete interaction; efficiency of corium cooling by water submersion.

5. INTERFACE WITH LEVEL 1 PSA: GROUPING OF SEQUENCES

5.1. This section provides recommendations on the interface between Level 1 PSA and Level 2 PSA. It addresses the analysis of results and information from the Level 1 PSA that needs to be performed to provide the necessary input for the Level 2 PSA. The Level 1 PSA and Level 2 PSA interface depends on the methodology chosen for the Level 2 PSA, the modelling software used and the reactor technology.

5.2. According to SSG-3 (Rev. 1) [4], Level 1 PSA identifies a large number of accident sequences that lead to core damage. It is neither practical nor necessary to individually treat each accident sequence when assessing accident progression, containment response and radionuclide release within Level 2 PSA. Accident sequences should be grouped together into PDSs in such a manner that all accidents within a given PDS can be treated in the same way for the purposes of the Level 2 PSA. If necessary, the accident sequence models in the Level 1 PSA should be adjusted to take account of the specific needs of the Level 2 PSA. The PDSs should represent groups of accident sequences that have similar accident timelines, containment status and containment system (un)availability status and that generate similar loads on the containment, thereby resulting in a similar event progression and similar radiological source terms. Attributes of accident progression that will influence the chronology of the accident, the progression of the core damage, the containment response or a radioactive release to the

environment should be identified. The attributes of the PDSs provide boundary conditions for the performance of severe accident progression analysis.

5.3. The grouping of Level 1 PSA sequences into PDSs may involve some assumptions and simplifications that introduce additional uncertainties. The analyst should keep track of these assumptions and simplifications (see para. 11.22).

5.4. Care should be taken to ensure that assumptions that could minimize the source term are not introduced when sequences from the Level 1 PSA are mapped and transferred to the Level 2 PSA and that no sequences are lost or duplicated. The latter can be achieved by quantifying the frequencies of all of the PDSs and validating their sum against the core and/or fuel damage frequency determined from Level 1 PSA. Justifications for any numerical deviations should be given.

5.5. The success criteria for a system credited in Level 2 PSA should specify the system mission time to reach a controlled state[13] or to fulfil the modelled system function. The mission time should be defined adequately for capturing the severe accident progression, the time needed for design features to effectively cope with severe accidents, including possible cliff edge effects, and to ensure that the residual risk accrued after the mission time is negligible.

5.6. Paragraphs 5.7–5.25 provide examples of the attributes that may need to be taken into account in defining PDSs. Examples of such attributes for water cooled reactors are given in Table 3.

PLANT DAMAGE STATES FOR PSA FOR INTERNAL INITIATING EVENTS DURING FULL POWER CONDITIONS

5.7. If the Level 2 PSA is developed following a separated approach (see para. 2.6), the Level 1 PSA does not account for specific aspects relevant to the specification of PDSs. For example, the Level 1 PSA might not have addressed the status of containment associated systems or other systems that do not directly affect the determination of core damage (i.e. they do not contribute to the success criteria for preventing core damage). In such cases, the Level 1 PSA should be expanded to take into account the missing aspects in the specification of PDSs

[13] As defined in SSR-2/1 (Rev. 1) [3], a 'controlled state' is a plant state, following an anticipated operational occurrence or accident conditions, in which the fundamental safety functions can be ensured and which can be maintained for a time sufficient to effect provisions to reach a safe state.

TABLE 3. EXAMPLES OF ATTRIBUTES OF PLANT DAMAGE STATES FOR WATER COOLED REACTORS

Initiating event	Large loss of coolant accident Small loss of coolant accident Safety or relief valve stuck open Transient, such as: — Reactor trip — Loss of off-site power — Loss of electrical bus — Loss of feedwater — Loss of service water — Steam line break — Feedwater line break — Anticipated transient without scram Bypass event (loss of coolant accident in interfacing system or steam generator tube rupture) Reactivity accidents (e.g. homogenous or heterogenous dilution, accidental withdrawal or ejection of control rods)
Reactor coolant system status	Reactor coolant system pressure at core damage: — High (relief valves are challenged) — Medium (above low pressure coolant injection head) — Low (including method of depressurization) Status of safety relief valves Reactor coolant system integrity (shutdown states): — Vessel head removed — Nozzle dams installed — Safety valves removed — Vent open Reactor coolant system coolant inventory (shutdown states): — Full power inventory — Flooded refuelling cavity — Mid-loop operations in a pressurized water reactor State of fuel in the reactor for decay heat: — Operating power level — Pre or post refuelling — Time since reactor shutdown
Status of emergency cooling system and other cooling systems (timing of core damage and ability to prevent further core damage progression)	Injection fails to start (no injection, early damage) Coolant injection initially successful, but recirculation cooling fails (later core damage) Emergency core cooling functionality after core damage or breach of reactor pressure vessel Steam generator cooling availability

TABLE 3. EXAMPLES OF ATTRIBUTES OF PLANT DAMAGE STATES FOR WATER COOLED REACTORS (cont.)

Status of the containment's engineered safety features	Sprays (if any): — Operate at all times — Fail on demand — Initially operate, but fail on switchover to recirculation cooling Suppression pool (if any): — Effective at all times — Ineffective (pool drained or bypassed early) — Bypassed late Fan or water coolers (if any): — Operate at all times — Fail on demand — Fail late Venting systems: — Operate at all times — Fail on demand — Fail late Status of containment inerting systems (if any) Status of means for management of combustible gases, i.e. monitoring and control by passive means (e.g. autocatalytic recombiners) and active means (e.g. igniters) — Fully available as designed — Degraded — Fully failed Containment passive heat removal system (if any): — Available — Unavailable — In operation — Failed
Status of support systems	Electrical alternating current and direct current power Component cooling Instrument air Heating, ventilation and air conditioning Availability and accessibility of mitigating systems
Containment status	Intact and isolated at the onset of core damage Intact, but not isolated at the onset of core damage Bypassed Structural failure or enhanced leakage (with indication of size and location of leakage)[a]

TABLE 3. EXAMPLES OF ATTRIBUTES OF PLANT DAMAGE STATES FOR WATER COOLED REACTORS (cont.)

Status of secondary containment (reactor building or enclosure building)	Intact and isolated at the onset of core damage Intact, but not isolated at the onset of core damage Structural failure or enhanced leakage[a]

[a] This includes any external hazards that might damage containment structures.

(see Table 3). One method for incorporating such missing systems into the Level 1 PSA is to develop bridge trees (referred to as 'extended event trees' in Fig. 1) that link to Level 2 PSA system models, thereby capturing important dependencies (e.g. support systems, operator performance).

5.8. If the Level 2 PSA is developed as part of an integrated Level 1–Level 2 PSA (see para. 2.6), many of the PDS characteristics listed in paras 5.11–5.13 will be implicitly available for the Level 2 PSA model. Such an approach may allow for a reduction in the number of PDSs needed. In any case, even though the structure of the PDSs could be simpler in an integrated Level 1 PSA and Level 2 PSA model, the analyst should verify that simplifications or assumptions in the Level 1 PSA model will not screen out possible PDSs contributing to radioactive releases.

5.9. Generally, PDSs can be divided into two main classes: those in which radioactive material is released from the reactor coolant system to the containment and those in which the containment is either bypassed or is ineffective at the time of core damage. Thus, the containment status (e.g. intact and isolated, intact and not isolated, failed or bypassed) should be specified for the PDS and, for PDSs where the containment is bypassed, the type and size of the bypass (e.g. loss of coolant accident in interfacing systems, steam generator tube rupture) should be specified. If the reactor building or secondary containment is likely to have a major influence on the source term, then its status at the time of core damage should be specified for the PDS. For PDSs with intact containment, an accident progression event tree analysis should be performed. For PDSs where the containment is failed, not isolated or bypassed, only a source term analysis may be necessary, though a simplified event tree may need to be provided in the model. However, for some accidents, an accident progression event tree may be needed to address possible plant features that can reduce the source term (e.g. scrubbing of fission products by means of a water pool or water spray, actions for reducing

or isolating the containment bypass) or to quantify possible additional damage to the containment that increases the releases.

5.10. The characteristics specified for the PDSs are generally left to the discretion of the analyst. Examples of characteristics are given in paras 5.11–5.13. It should be noted that the level of detail of characteristics needed to define the PDSs depends on the approach used for the development of Level 1 PSA and Level 2 PSA (see para. 2.6). In either approach, the definition and selection of characteristics specified for the PDSs should be justified.

Plant damage states without containment bypass

5.11. In specifying PDSs without containment bypass, account should be taken of the equipment and system failures or scenario features, including operator actions identified within Level 1 PSA, that could affect either the challenge to the containment or the release of radioactive material. Depending on the reactor technology, the following might need to be taken into account (see Table 3 for further details for water cooled reactors):

(a) Type of initiating event, which can, for example, affect the rate of discharge of reactor coolant in the containment, the progression of the core damage and of hydrogen generation, and the timing of the release of radioactive material.
(b) Failures of the credited systems (e.g. reactor protection system, residual heat removal system, emergency core cooling system) that have occurred, leading to core damage.
(c) Extent of fuel damage.
(d) The timing at which core damage occurs (e.g. early or late relative to the time of reactor trip).
(e) The reactor coolant system pressure at the onset of core damage and the status of safety valves or relief valves and other components that could change the pressure in the reactor pressure vessel before failure of the lower head of the reactor pressure vessel[14].
(f) The pressure in the reactor pressure vessel after the onset of core damage, which affects the possibility of temperature and pressure induced failures of the reactor coolant system (e.g. creep rupture of piping and steam generator

[14] In some cases, it is useful to group separately the sequences with high pressure at the onset of core damage but where the pressure is naturally reducing (e.g. loss of coolant accidents, small consequential leakages) and can result in low reactor coolant system pressure before the lower head failure.

tubes, thermal seizure of a safety or relief valve in the open position). The pressure is affected by the initiating event and the functionality of any depressurization system.

(g) The pressure in the reactor pressure vessel at the time of lower head failure, which is related to (but not necessarily the same as) the pressure at the onset of core damage and may affect the mode of discharge of debris to the containment. This, in turn, could present a challenge to containment integrity if, for instance, high pressure melt ejection and direct containment heating ensue.

(h) The integrity of the containment (e.g. intact, failed, isolation failure, bypassed due to a steam generator tube rupture (in a pressurized water reactor) or a loss of coolant accident at interfacing systems).

(i) Loss of coolant accident with or without pressure suppression capability (e.g. for a boiling water reactor).

(j) The state of the suppression pool (e.g. subcooled, saturated) when core damage occurs (for a boiling water reactor).

(k) The availability of the containment associated systems (e.g. containment sprays, heat removal systems and hydrogen mixing devices or recombiners).

(l) Initial and boundary conditions including the availability of alternating current and direct current power and associated recovery times.

(m) Actions by operating personnel that have been attempted and failed.

5.12. The status of the engineered safety features[15] of the containment is of high importance in determining the response of the containment, and such safety features should be taken into account in the grouping of accident sequences into PDSs, as they might influence processes such as cooling, the removal of radioactive material, or the mixing of combustible gases in the containment. For some reactor designs, the status of electrical power supply is important in grouping of accident sequences into PDSs because it impacts the availability of accident management systems. The details on how these characteristics are taken into account may depend on the methodology used for linking Level 1 and Level 2 PSA, although these issues should be addressed irrespective of the methodology applied.

Plant damage states with containment bypass

5.13. For PDSs with containment bypass, the main consideration should be the identification of attributes that are associated with the attenuation of concentrations of radioactive material along the release pathway or that affect the

[15] The attributes listed in Table 3 can be adjusted, as appropriate, for plants with structures that provide a confinement function rather than pressure retaining containment.

timing of release. This should include the type of initiating event, the status of the emergency core cooling system (including failure time), whether the leak pathway is isolable after a given time period, and whether scrubbing of fission products can be justified (e.g. scrubbing in the secondary side during a steam generator tube rupture accident or scrubbing through a flooded auxiliary building during an interfacing system loss of coolant accident).[16] For leaks into the auxiliary building or an equivalent building, the status of emergency exhaust filtration systems, heating, ventilation and air conditioning, or whether or not the leak is submerged, could be significant and should be taken into account.

Final selection of plant damage states

5.14. If the consideration of all factors and parameters that affect the Level 2 PSA results in an excessive number of potential PDSs, then they should be reduced to a manageable number. Two approaches could be taken. The first is to combine similar PDSs and perform a bounding analysis to select a representative sequence that characterizes the PDSs for the purpose of the Level 2 PSA. For instance, if the Level 2 PSA relies on time consuming physical calculations, it could be possible to run a manageable number of these calculations and attribute the outcomes of one calculation to several PDSs that are similar with regard to accident progression. The second approach is to use a frequency cut-off as a means of screening out less important PDSs.[17] The analyst should justify that such screening does not introduce a significant underestimation of the risk calculated by the Level 2 PSA; careful evaluation is therefore necessary before introducing a frequency screening criterion at the PDS level. This is especially true when dealing with PDSs that could involve an early radioactive release or a large radioactive release to the environment (see also para. 5.24). In any case, during the selection process, account should be taken of the degree of variability and uncertainty introduced in the Level 2 PSA by the grouping of accident sequences into PDSs. Consideration should be given to how this affects the specific objectives of the PSA, bearing in mind that for some PSA applications (e.g. risk monitors), the screening out approach could be inappropriate.

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[16] Isolation of containment bypass might not prevent severe accident progression scenarios.

[17] In some States, a cut-off value in terms of percentage of the total risk metric (large release frequency or large early release frequency) is established to distinguish significant PDSs from less important PDSs.

PLANT DAMAGE STATES FOR LOW POWER AND SHUTDOWN MODES OF OPERATION

5.15. Differences in the Level 2 PSA with respect to the mode of operation and power level when the initiating event occurs result primarily from differences in the reactor coolant inventory and in the status of both the reactor coolant system and the reactor containment. The PDSs specified for full power conditions should be reviewed and adapted for low power and shutdown modes; direct use of the PDSs specified for Level 2 PSA for full power conditions might not be possible.

5.16. Additional PDSs, different from those for a PSA for full power operation, should be specified in a Level 1–Level 2 PSA interface for a low power and shutdown PSA to capture the unique conditions that could have a major impact on plant behaviour in severe accidents. For example, additional PDSs may be necessary for conditions unique to certain shutdown states such as those with the reactor vessel head removed or with the containment hatches opened. The following additional accident sequence characteristics should be considered in specifying the PDSs for low power and shutdown (see also para. 9.34 of SSG-3 (Rev. 1) [4]):

(a) Decay heat level (based on time since shutdown from power operation);
(b) Conditions that determine the time taken to restore containment isolation and the potentially reduced effectiveness (i.e. leaktightness) of the containment during this time;
(c) Integrity of the reactor coolant system pressure boundary with the reactor vessel head removed, nozzle dams installed, safety valves removed and reactor coolant system vent open;
(d) Coolant inventory in the reactor coolant system;
(e) Closure status of the containment and associated manual actions to close it prior to core damage.

CONSIDERATIONS FOR INTERNAL AND EXTERNAL HAZARDS IN LEVEL 2 PSA

5.17. In order to extend the scope of Level 2 PSA to include internal and external hazards such as fire, seismic hazards and external flooding, the potential impact of the hazards on systems necessary for mitigation of severe accidents, including systems that support operator actions, as well as the impact on the integrity of the containment, should be taken into account if those aspects have not yet been taken into account in the Level 1 PSA. For example, in a Level 2 PSA for fire, the

cables associated with the systems for ensuring the confinement function should be tracked to assess the impact of fire scenarios; or operator actions in the main control room may need to be failed if the main control room is assumed abandoned. In a Level 2 PSA for seismic events, the impact of the seismic event on the core cooling system, water storage tanks, systems ensuring the confinement function, and other mitigating systems and equipment should be taken into account. This could lead in some cases to the specification of a new set of distinct PDSs. The analyst should consider the need to introduce new PDSs and possibilities for assimilating new PDSs into existing ones; for instance, some failures of the systems for ensuring the confinement function due to internal or external hazards could be assimilated into already defined isolation failures of systems for ensuring the confinement function.

5.18. The potential impact of hazards on the systems for ensuring the confinement function should be taken into account as part of the Level 2 PSA, if those aspects have not yet been taken into account in the Level 1 PSA.[18]

5.19. The analyst may decide to credit human actions that occur before or soon after core damage in the Level 1 PSA and capture these human actions as part of an attribute of the PDSs for the Level 2 PSA (see para. 8.4). When extending the Level 2 PSA for other internal and external hazards, the environmental or physical conditions introduced by the hazards may interfere with these human actions and therefore should be taken into consideration when specifying the PDSs for other internal and external hazards.

5.20. The analysis process to be conducted in order to consider hazards and their combinations for Level 1 PSA is described in section 6 of SSG-3 (Rev. 1) [4]. This process is applicable to Level 2 PSA and it is not repeated here. For Level 2 PSA, single as well as combined hazards have the potential to result in accident sequences induced by common cause initiators that might impact the confinement function, such as the following:

(a) A beyond design basis earthquake resulting in a station blackout and a containment failure, possibly with consequential internal fire or flooding;
(b) A combination of external flooding and high winds that might lead to the loss of the heat sink together with the loss of off-site power;

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[18] Typical examples of impacts of hazards are failures of the isolation function of systems ensuring the confinement function due to internal fire, explosion or flooding at the plant, damage of the containment due to seismic events, aircraft crashes or external explosions (blasts).

(c) An aircraft crash causing a common loss of off-site power and emergency diesel generator failure that results in not only a plant transient but also an accident sequence with containment bypass and releases of radionuclides (via air or water).

5.21. In order to be widely applicable, the Level 2 PSA for hazards should be based on the Level 1 PSA covering these hazards, as described in section 6 of SSG-3 (Rev. 1).

5.22. The development of a Level 2 PSA for hazards depends on the scope set for the Level 2 PSA but can also be influenced by the results of the Level 1 PSA for hazards. In particular, if there is a low level of knowledge associated with the Level 1 PSA results, the relevance of extending this PSA to Level 2 should be analysed with regard to safety issues, feasibility and ease of analysing insights from it.

5.23. If the Level 2 PSA is based on a Level 1 PSA with a more limited scope or details, these limitations should be taken into account in the application of the Level 2 PSA.

5.24. Any hazards, single or combined, that were screened out from further (bounding or detailed) analysis within the Level 1 PSA should also be reassessed, consistent with paras 6.17–6.19 of SSG-3 (Rev. 1) [4], taking into account that para. 6.19 of SSG-3 (Rev. 1) [4] states that "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." To determine if such hazards should be taken into account in Level 2 PSA, their ability to affect the confinement function should be considered. In this context, there should be a distinction between the following:

(a) Hazards for which the site and plant specific screening has demonstrated that a detailed analysis is not needed but that a bounding assessment of the Level 1 PSA PDSs (i.e. core and/or fuel damage) is sufficient, and for which detailed accident sequences do not have to be modelled, but a bounding assessment of the radioactive release frequencies (i.e. large release frequency or large early release frequency) is sufficient;
(b) Hazards for which detailed accident sequences need to be modelled and quantified within Level 2 PSA.

5.25. Based on the identification and screening of hazards performed within Level 1 PSA, those combined hazards, for which either simple or detailed

probabilistic analyses have been performed in the framework of Level 1 PSA, should be analysed within Level 2 PSA. In line with the guidance provided in section 6 of SSG-3 (Rev. 1) [4] for Level 1 PSA, relevant combined hazards should be treated in the same manner as single hazards when developing accident sequences to be analysed.

6. SEVERE ACCIDENT PROGRESSION ANALYSIS

6.1. Severe accident progression analysis typically consists of different groups of analyses performed in different phases of a Level 2 PSA project. Early in the project, severe accident progression analysis is used to understand the general core damage accident progression for key initiating events, providing a starting point for the Level 2 PSA analysis [19]. Subsequently, analyses are performed to support the specification of PDSs, which assists the event tree analysts in establishing which systems and accident progression features are most important for determining the plant response and, hence, need to be included in the PDS specifications or in the accident progression event tree model, as appropriate for the methods being applied in the PSA. For example, investigations may provide insights into the variation of accident progression when different numbers of injection trains are operating or into the influence of primary and secondary pressure status (in a pressurized water reactor), or they may indicate the effect of changes in the volume of water injected to containment on ex-vessel molten core behaviour. A third area where severe accident progression analyses provide support is in the assessment of specific phenomena, where accident analysis results may be used as an input to phenomenological probability calculations (see Section 9) and help to specify the timing for human action events included in the logic models. Finally, severe accident progression analyses support the grouping of accident progression event tree sequences into release categories (see Section 11), which is a process similar to the definition of PDSs, and the quantitative characterization of the radiological release associated with each release category.

6.2. This section presents recommendations for the Level 2 PSA studies that provide information on the possible severe accident progression paths, such as accident timing, main phenomena, conditions for manual actions and automatic actions, effects of systems actuation and loads (e.g. pressure, temperature, radiation) induced on the containment boundaries (e.g. building structure, penetrations, seals). The results of these studies are used in containment integrity analysis (see Section 7), in human and equipment reliability analysis (see Section 8), in the

development of accident progression event trees (see Section 9) and in support of source term analysis (see Section 10).

6.3. Plant specific deterministic analysis of the progression of accidents inside the reactor should be considered the preferred method for evaluating severe accident progression and effects in the reactor vessel, in the reactor containment and in auxiliary buildings.

6.4. Taking into account the modelling of the plant, the automatic actuations and human actions with appropriate simulation tools, severe accident progression analysis should be a significant part of the resources planned to develop a first version of a Level 2 PSA.

6.5. Severe accident progression analysis should be performed by teams with experience in the application of severe accident codes; if necessary, appropriate training should be included in the project (see paras 3.19–3.22 on team selection for the Level 2 PSA project).

6.6. In addition, generic studies of severe accident phenomena and containment response reported in the literature for similar plants could also be used to complement the scope of plant specific calculations to include a broader set of conditions.

6.7. The analysis of the progression of the severe accidents identified by the Level 1 PSA and grouped in specific PDSs should provide key information such as fuel uncovering kinetics, hydrogen production, vessel failure, risk of explosion, risk of basemat penetration by corium, and the amplitude and kinetics of radioactive release.

ANALYSIS OF SEVERE ACCIDENTS INVOLVING REACTOR CORE DAMAGE

6.8. Each identified PDS representing a significant contributor to core damage (see para. 5.14 and footnote 17) should be mapped to a specific representative calculation; some calculations can represent more than one PDS, if justified. Calculations could also be performed for those PDSs that have a low occurrence frequency but the potential to result in large and/or early releases of radionuclides to the environment. Such PDSs typically involve either direct containment bypass or early failure of the primary and/or secondary containment. If detailed calculations are performed for PDSs with high occurrence frequencies and for

those with severe consequences, a sufficiently wide range of information is usually generated to estimate the response of the plant for other PDSs that are not addressed in detail.

6.9. If relevant, Level 2 PSA should also consider assessing reactivity accident scenarios resulting in prompt criticality accidents leading to reactor core damage and potential damage to the containment integrity.

6.10. Severe accident progression analysis for the reactor should be performed using one or more computer codes for severe accident simulation (see Annex I for examples of computer codes for water cooled reactors).

6.11. To meet Requirement 18 of GSR Part 4 (Rev. 1) [2], decisions on the computer code(s) used to perform detailed analyses and the number of calculations to be made should be based on the objective of the Level 2 PSA. In making these decisions, the following should also be taken into account:

(a) The code(s) should be capable of modelling most of the initiating events that were considered in Level 1 PSA and phenomena that might occur during the progression of the accident according to the state of the art.
(b) Interactions between various physicochemical processes should be correctly addressed in the computer code(s).
(c) The code(s) should be verified and validated against the severe accident phenomena analysed by them, and a validation matrix should be available.
(d) Validation and benchmarking, and the associated documentation, should be sufficient to support the necessary severe accident progression analyses (see also Ref. [20]).

6.12. The analysts should be adequately trained in the use of the code(s) to be applied and should be aware of the technical limitations and weaknesses of the selected code(s) (see para. 3.21).

6.13. The analyses of severe accidents should first cover key sequences for each PDS that leads either to a successful controlled state of the plant, where sufficient safety systems or safety features have operated correctly so that all of the required safety functions necessary to cope with the sequences have been fulfilled, or to filtered containment venting (if provided) or to a degraded state[19] with one or

[19] A 'degraded state' is considered to be a state following a severe accident, in which the safety functions performed by the containment and its associated systems are affected.

several containment failures. Second, the remaining sequences in the accident progression event tree should be quantified for confirmation purposes.

6.14. To support the Level 2 PSA, deterministic analyses of both the integral behaviour of the plant during a severe accident as well as individual phenomena analyses of the severe accident sequences under consideration should be performed. Integral analyses start with the initiating event and end in accordance with appropriate criteria, depending on the purpose of the analysis. Examples of criteria for when to terminate analyses are (a) when the cumulative release of radionuclides into the environment has stabilized, (b) after corium stabilization (in-vessel or ex-vessel), or (c) after a predetermined mission time has elapsed. The analysis of individual phenomena should be supported by severe accident progression analyses, as necessary (see also Section 10). Some examples of individual phenomena for water cooled reactors are as follows:

(a) Structural–mechanical behaviour of components of the reactor coolant system in the event of high-pressure severe accident scenarios;
(b) Interaction of the core, core structures and corium with coolant inside and outside the reactor pressure vessel (e.g. quenching, steam production, steam explosion, hydrogen generation and induced effects on the plant);
(c) Ex-vessel cooling of the reactor pressure vessel for in-vessel melt retention;
(d) High pressure melt ejection and direct containment heating;
(e) Hydrogen and carbon monoxide generation by molten core concrete interactions, flow and distribution in the reactor containment and mitigation means to cope with combustion behaviour;
(f) Ex-vessel corium cooling;
(g) Criticality accident effects;
(h) Containment pressurization.

6.15. In general, the analyses should be performed in a best estimate manner with regard to applied codes, models, model parameters and boundary conditions. Conservative assumptions, which are in common use for design basis accident analysis, might not be useful or appropriate in severe accident progression analyses for Level 2 PSA because, for example, conservative assumptions might distort the results and the risk insights and consequently might lead to deviation from optimal severe accident management strategies.

6.16. Accident management measures for both prevention of core damage and mitigation should be considered in the severe accident progression analyses with realistic timing for human actions.

6.17. All severe accident progression analyses (including descriptions of input decks, boundary conditions, assumptions and results) should be part of the Level 2 PSA documentation in order to provide the justification for the probabilistic accident progression (i.e. accident progression event tree) models. Key variables are typically catalogued at important points in time and recorded as time dependent plots for detailed study.

ANALYSIS OF INTERACTIONS BETWEEN THE REACTOR AND THE SPENT FUEL POOL

6.18. Analysis of the severe accidents in SFPs that are identified by the Level 1 PSA and grouped in specific PDSs can be performed on the same basis as the analysis of reactor accidents. Section 13 provides recommendations on severe accident progression analysis in the SFP.

6.19. Depending on the SFP location (i.e. inside the reactor containment, outside the reactor containment but inside the reactor building, or outside the reactor building), severe accident progression analysis should provide information on the interactions between the reactor and the SFP: there may be mechanisms through which a reactor accident can induce a SFP accident and vice versa. The outcome of this analysis may be additional accident scenarios (involving both the reactor and the SFP) being added to the Level 2 PSA.

6.20. If the SFP is inside the reactor containment and the joint accident progression event tree sequences involving the reactor and the SFP are not demonstrated to be negligible, severe accident progression analysis should be performed to show the combined impact of reactor and SFP accidents on conditions in the containment (e.g. pressure, temperature, corium spreading, inflammable gas, steam production in the containment, acceleration of evaporation of SFP water) and on radioactive releases.

6.21. If the SFP is located inside the reactor building but outside the reactor containment, then the severe accident progression analyses should consider the impact of severe accident progression inside the reactor building and its effects on the SFP and on radioactive releases.

SEVERE ACCIDENT PROGRESSION ANALYSIS FOR LOW POWER AND SHUTDOWN MODES

6.22. Specific analysis should be performed for low power and shutdown modes of reactor operation, depending on the reactor technology, in order to capture specificities that have implications for the accident progression and for source term calculations. Reference should be made to the plant operational states from Level 1 PSA. The following might apply in low power and shutdown modes:

(a) The core decay heat level might be lower.
(b) It might be possible to open the reactor coolant system with a lower coolant inventory (in this case, high pressure core melt is impossible).
(c) It might be possible to open the containment.
(d) Interconnection between the reactor and the SFP might be possible, thus allowing the use of common systems and common severe accident management strategies for ensuring decay heat loads in each of these locations and for possible fuel assembly handling.
(e) In water cooled reactors, when the reactor pressure vessel head is closed (and provided that the decay heat level and operating configuration are similar to those for full power), core melt accident phenomena might be considered very similar to the sequences that occur in full power mode. When the reactor pressure vessel head is open, the analyst should consider that some of the Level 2 PSA issues become irrelevant compared to full power mode, while others come into existence. Certain phenomena might not occur (e.g. direct containment heating, induced steam generator ruptures, alpha mode failure). In some reactor designs, during the majority of shutdown states when the reactor pressure vessel is open, there is a water pool connecting both the reactor and the spent fuel pool.

6.23. Specific analysis should be considered for accidents occurring when the reactor and SFP are connected (see Section 13).

IDENTIFICATION OF SOURCES OF UNCERTAINTY

6.24. The analyst should be aware of the technical limitations and weaknesses of the computer code(s) selected for modelling severe accident progression. The analyst should identify any possible lack of information on plant design or procedures and any lack of information from systems and components qualification.

6.25. Known areas of uncertainty in the modelling of severe accidents inside water cooled reactors are shown in Table 4, together with potential implications on the modelling.

6.26. A plant specific list of uncertain parameters to be varied in the frame of the uncertainty and sensitivity analyses should be derived. Care should be taken if including parameters such as correlation coefficients and parameters for modelling the phenomenology of severe accidents in the corresponding computer codes, which are established as part of the computer code validation procedure. The list of parameters should be established on the basis of a systematic process and good practices should be followed in performing severe accident uncertainty analysis.

6.27. The uncertainties related to calculated key variables (e.g. peak pressures and temperatures, total mass of corium, mass of combustible hydrogen, timing of major events) should be documented for use in the models for quantification of accident progression (e.g. accident progression event tree).

TABLE 4. EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO SEVERE ACCIDENT PROGRESSION FOR WATER COOLED REACTORS

Type of severe accident event	Related phenomena
In-vessel core degradation	Formation of flow blockages in core 'Ballooning' of cladding and rod failure Relocation and solidification of molten fuel Oxidation and hydrogen generation Relocation of corium into lower head of reactor vessel Corium stratification inside lower plenum of reactor vessel (metallic/oxidized layers, focusing effect for thermal flux) Ex-vessel cooling (for in-vessel retention)
In-vessel forced or natural circulation	Circulation flows in reactor coolant system loops influenced by the presence of water in cold leg (direct or counter-current steam flows) Competing mechanisms of degradation and failure of reactor coolant pump seal

TABLE 4. EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO SEVERE ACCIDENT PROGRESSION FOR WATER COOLED REACTORS (cont.)

Type of severe accident event	Related phenomena
In-vessel corium–water interactions (energetic and non-energetic)	Effect of in-vessel water injection (quenching) after recovery of systems or components: pressure peak, hydrogen generation, fuel cooling depending on core degradation progression and water flow rate Potential for terminating in-vessel fuel degradation Recriticality Steam explosion, high pressure failure of the reactor pressure vessel Release and dispersion of radioactive material
Failure of primary circuit	Break size Break location
Failure mechanisms of reactor pressure vessel and loop boundaries	Melt penetration and cooling within vessel head penetrations Local or global failure of lower head of reactor pressure vessel: mechanical (creep) or melting failure Impact of ex-vessel cooling Heat-up and creep rupture of reactor coolant system pressure boundary (hot leg nozzle, pressurizer surge line and steam generator tubes) Impact of possible steam generator tubes flaws
High pressure melt ejection and/or direct containment heating	Trapping of melt debris on containment structures Heat release by zirconium oxidation and additional hydrogen production Debris transport from the vessel/cavity to the containment atmosphere Hydrogen combustion simultaneously with heat transfer to containment atmosphere (pressurization) Releases of radioactive material
Ex-vessel corium–water interactions (energetic and non-energetic)	Debris fragmentation and quench (cooling) Quasi-static increase in containment pressure (steam spike) Local dynamic loads to containment from steam explosion (reactor cavity) and possible damage to structures Releases of radioactive material

TABLE 4. EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO SEVERE ACCIDENT PROGRESSION FOR WATER COOLED REACTORS (cont.)

Type of severe accident event	Related phenomena
Core–concrete interactions	Erosion of containment structure (basemat) by debris Generation of non-condensable and/or combustible gas (e.g. CO, CO_2 and H_2) Lateral/axial spreading of debris and potential for contact with containment boundary Corium spreading Corium coolability Effects of presence of metal within the melt or within the concrete, melt stratification (metallic/oxidized layers) Releases of radioactive material
Hydrogen and carbon monoxide combustion	Mixing and/or stratification of flammable gas in containment atmosphere Local increase of concentrations (e.g. due to strengthened steam condensation in cold containment regions) Steam or nitrogen inerting Hydrogen and carbon monoxide recombination, ignition time and combustion, flame acceleration and transition from deflagration to detonation Heat losses to structures Containment structure response to combustion pressure wave (e.g. open doors or blow-out panels, displacement of water pools) Transport and distribution of combustible gas in secondary buildings and containment venting systems

7. CONTAINMENT INTEGRITY ANALYSIS

7.1. The content of this section is based on experience with water cooled reactor technologies with containment. It presumes the existence of some type of passive structure with the capability to withstand some of the conditions resulting after severe damage to the reactor core and thus retaining a large portion of the radioactive material. The most common version of such a passive structure in many plant designs is a containment building or a steel containment vessel, which includes systems associated with containment. As such, the applicability

of this section depends on the reactor technology and design. In the case of non-water-cooled reactors with such a containment structure, containment integrity analysis may be conducted depending on the design features and loading conditions on the containment.

7.2. The containment integrity analysis includes both deterministic and probabilistic analysis methods and should address the following:

(a) The capability of the containment to maintain its leaktightness and structural integrity under internal loads (see paras 7.4–7.11);
(b) The potential for loss of containment leaktightness due to failure mechanisms induced by severe accident phenomena, such as erosion of concrete structures by direct interaction with molten core debris (i.e. not for in-vessel melt retention) (see paras 7.12–7.16 and 9.3–9.6);
(c) The potential for containment isolation failure or containment bypass resulting in a direct leakage pathway to the environment (see paras 7.17–7.20 and 9.3–9.6).

7.3. Containment integrity might also be impacted by hazards. For instance, high level seismic events might directly cause a loss of containment leaktightness. Therefore, a review should be conducted of the impacts caused by hazards within the scope of PSA in order to identify any impacts on the containment structure that need to be captured. Usually this is done as part of the Level 1 and Level 2 PSA (see paras 5.17–5.25).

ANALYSIS OF REACTOR CONTAINMENT PERFORMANCE

Analysis of containment performance under internal loads

7.4. In general, the goal of the containment structural analysis with regard to internal loads is to assess the probability of containment failure as a function of pressure and/or temperature under severe accident conditions[20], known as

[20] Examples of phenomena leading to overpressurization loads could be combustion of combustible gases such as hydrogen combustion (i.e. deflagration and detonation) and carbon monoxide combustion.

a fragility curve[21] or a fragility (hyper)surface[22] (see Ref. [21] for examples). Typically, material properties vary with containment structure temperature; consequently, pressure driven failure modes will be affected by temperature conditions. Usually, an enveloping temperature is chosen (on the basis of severe accident progression analyses) and the overpressure analysis is then performed assuming those temperature conditions. The development of containment fragility curves should include consideration of pressure driven and temperature driven failure modes that are applicable to the containment design under consideration in the Level 2 PSA. This analysis should include an identification of potential failure modes and their respective locations.

7.5. A realistic characterization of the leakage area associated with each failure mode should be developed. Failure criteria are also needed for each failure mode. Typically, in terms of leakage area, a classification is made into small area failures (usually designated as 'leaks') and large area failures (usually designated as 'ruptures' and/or 'catastrophic ruptures'). Criteria specified for containment leak rates (e.g. threshold values expressed as volume percentage per day) or containment failure under severe accident conditions (e.g. a realistic allowed structural deformation) should be defined. Design criteria for the containment are generally not adequate best estimate measures of capacity of the containment because of the safety factor built into such values. Furthermore, containment design limits might not take into account the harsh environmental conditions that can develop inside the containment during a severe accident, for which additional failure modes often need to be considered.

7.6. To generate a realistic assessment of containment performance limits, detailed information on the structural design of the containment and containment penetrations should be collected. The following are examples of important features of the structural design of the containment and containment penetrations that should be taken into account for water cooled reactors:

(a) Containment material:
    (i) Steel;
    (ii) Concrete:
        — Prestressed;
        — Post-tensioned;

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[21] A fragility curve represents the probability of containment failure as a function of one variable, such as pressure or temperature.

[22] A fragility surface represents the probability of containment failure as a function of two or more variables together, such as pressure and temperature.

— Reinforced;
— With or without steel liner;
— With or without resin (e.g. epoxy) liner.

(b) Containment penetrations:
  (i) Equipment hatch(es);
  (ii) Personnel hatch(es);
  (iii) Piping penetrations;
  (iv) Electrical penetrations;
  (v) Atmosphere purge line(s);
  (vi) Vent line(s).

(c) Other aspects:
  (i) Geometrical shape of containment (sphere, cylinder, rectilinear);
  (ii) Geometric details of the containment structure, penetrations and hatches;
  (iii) Geometrical discontinuities (e.g. transition from cylindrical shell to top head and basemat);
  (iv) Steel or resin (e.g. epoxy) liner anchorages;
  (v) Details of any reinforcements around penetrations;
  (vi) Materials and their properties;
  (vii) Interactions with other surrounding structures.

7.7. This step of the Level 2 PSA is aimed at developing a best estimate assessment of the ultimate strength of the containment. This can be achieved by making plant specific structural calculations that account for the containment design features listed in para. 7.6. However, depending on the scope of the Level 2 PSA, use can be made of existing calculations for plants with similar containment designs. In this case, the PSA documentation should provide a thorough justification for the use of existing calculations. Items to address in this justification include similarities and differences of the designs, the applicability of the existing structural response analyses to the plant under consideration and the basis for any adjustments or extrapolations made.

7.8. When applying the fragility curves in the accident progression event tree models, calculations should be made of the probability of containment failure for each of the different leakage area categories defined in accordance with para. 7.5, to the extent that these separate calculations are necessary to realistically address the objectives of the Level 2 PSA. In calculating the probability of a rupture failure mode, credit may be taken for two basic approaches used in PSA studies to characterize the loss of containment integrity, namely the use of the 'threshold model' and the 'leak before break model'. The threshold model defines a threshold pressure, with associated uncertainties, at which the containment

is expected to fail. This failure is represented as a large rupture and with the potential for a significant and rapid blowdown from the containment atmosphere to the environment. In the leak before break model, containment leakage failure modes occur at pressures below the pressure at which a larger failure mode occurs. The precise treatment depends on the rate of addition of mass and energy to the containment. If this rate of addition is lower than or equal to the leakage rate associated with a leak failure, containment pressurization will be stopped if such a failure occurs, thereby preventing the larger rupture failure mode from occurring.

7.9. If plant specific calculations are necessary, containment performance analyses should be based on validated structural models supported by data and reasonable failure criteria. In particular, the failure criteria used in plant specific calculations should be justified. The use of containment failure experiments may be useful for this purpose (see para. 7.24). In the analysis, fragility curves should be developed for static pressure loads. Best practice is to include pressure ramp rates, localized heat loads and localized or global dynamic pressure loads. Dynamic pressure loads may be addressed in a simplified way, for example by a single degree of freedom calculation. Ageing of structures should also be taken into account. The supporting analyses provide an engineering basis for containment failure mode, location, size and ultimate pressure and/or temperature capabilities of the containment structure.

7.10. Large penetrations (e.g. equipment hatches, personnel hatches) and singularity zones can be relative weak points of the reactor building in severe accident conditions. The impact of these penetrations on containment capacity should be considered in the reactor building structural analysis. This may include performing global analysis followed by local analysis for these zones, with adequate detail to capture the local mechanical behaviours. Usually, such local analyses take boundary conditions from the global containment structural model. As an example, the behaviour of the equipment hatch closure system is to be studied under severe accident conditions [22, 23].

7.11. While internal pressure loading is the principal determinant of potential containment failure, consideration should also be given in the Level 2 PSA to the possible effects of temperature and radiation on the containment performance. The temperature of the containment could affect the strength characteristics of the structural materials [23] and cause degradation of penetration seal materials. The impact of radiation on penetration seal or gasket materials could also affect the tightness properties of the seals in a severe accident environment, in particular if they are directly exposed to radiation.

Analysis of containment leaktightness due to other failure mechanisms induced by severe accident phenomena

7.12. Containment leaktightness might also be affected by failure mechanisms induced by severe accident phenomena. Examples of phenomena to consider are induced fires (e.g. graphite fires), steam explosions (e.g. instantaneous vaporization of water induced by its contact with molten corium), chemical attacks (e.g. chemical reactions affecting containment structure integrity) and direct contact between molten core debris and the containment structure. Recommendations related to the consequences of molten core debris and containment structures for the containment integrity analysis are provided in paras 7.13–7.16.

7.13. The effects of extensive erosion of concrete structures due to long term exposure and to attack by molten core debris (molten core–concrete interactions) should be examined if calculations of severe accident progression suggest that extensive erosion is possible as a result of such interactions (see Section 6). This may be of particular concern for the response of the containment basemat but also, depending on the plant design, the response of the containment wall or reactor pressure vessel support structure (e.g. concrete pedestal).

7.14. Extensive erosion of concrete structures may lead to containment leakage or failure if the structures interacting with the molten core debris are significantly weakened due to this interaction. Failure criteria for these mechanisms, which are generally expressed in terms of reduction of concrete thickness, should be defined and justified.

7.15. The consequences of an extensive erosion of concrete structures should be examined. For example, the response of a reactor pressure vessel support structure (e.g. concrete pedestal), containment wall or floor to a complete or partial penetration by core debris should be examined if calculations of severe accident progression suggest such levels of erosion are possible.

7.16. Potential locations for melt-through of the containment (e.g. penetrations, sump suction lines) should be identified and analysed.

Analysis of initial and induced containment isolation failure and containment bypass

7.17. The potential for containment isolation failure should be assessed as part of the analysis of reactor containment performance. All of the containment penetrations should be modelled, or a careful justification should be provided to justify that

some penetrations can be screened out. Screening criteria may be applied in order to focus on the penetrations that are most likely to result in significant releases. For instance, containment isolation might not be modelled for normally closed lines if the isolation valves would not be opened during the accident (e.g. due to the initiating event or type A human failure event (see para. 8.1)) or for closed loop systems inside the containment, if closed loop integrity would not be threatened during the accident. Any plant operating feedback regarding containment isolation valve leakages should be taken into account.

7.18. Possible failure modes of valves should be taken into account, such as failure to close and spurious opening. Leaks through closed valves may also be considered, depending on the goals of the Level 2 PSA (as low levels of leakage are not likely to cause large releases). The dependencies involved in containment isolation should also be taken into account (e.g. power supply to motor valves, automatic isolation signals). In addition, manual recovery operator actions can be credited, if justified.

7.19. The potential for containment bypass (i.e. release from the core to the environment without being able to credit containment) should be assessed through interface with the Level 1 PSA (see Section 5) or through accident progression event tree logic (see Section 9). The bypass paths should be identified by a rigorous search of systems that are located outside the containment and linked to reactor coolant loops or containment atmosphere. In general, containment bypass might result either from an initiating event (e.g. steam generator tube rupture, loss of coolant accident in interfacing system), from an induced event due to accident conditions (e.g. severe accident induced steam generator tube rupture for a pressurized water reactor) or from a failure (e.g. leak, incorrect circuit configuration) in an emergency cooling line outside of the containment.

7.20. Potential containment failure modes due to isolation failure or bypass that have not already be taken into account in the Level 1 PSA should be addressed in the Level 2 PSA.

CONTAINMENT INTEGRITY ANALYSIS FOR LOW POWER AND SHUTDOWN MODES

7.21. For low power and shutdown modes of operation, the primary considerations relevant to containment integrity analysis are associated with the potential for large containment penetrations to be present, such as an open equipment hatch. If such penetrations are present, the assessment of containment internal loading

and the effects on concrete structures due to molten core debris interactions are not applicable to the assessment of containment integrity, since the containment is in a bypassed state. However, the degradation of structures due to interactions with core debris may still be relevant to other parts of the Level 2 PSA. If the containment remains sealed during low power and shutdown modes of operation, then the previous recommendations within this section are applicable.

Analysis of containment isolation failure during shutdown

7.22. With regard to containment isolation failure, particular attention should be paid to the shutdown mode of operation, during which the following might apply:

(a) Personnel airlocks or equipment hatches might be open at the time of the initiating event. The closure of such very large penetrations before the onset of core damage should be precisely evaluated. In particular for an equipment hatch, the analyst should take into account aspects such as whether the closure is requested by emergency operating procedures, the time taken to perform the closure and the dependencies (e.g. power supply) involved.
(b) Isolation lines that are normally closed at power might be open and might not be subject to automatic isolation.

CHARACTERIZATION OF UNCERTAINTIES

Characterization of uncertainties related to containment performance under internal loads

7.23. In determining the structural performance of the containment, the uncertainties associated with estimation of the structural capacities necessary for withstanding extremes of pressure and/or temperature should be assessed. Uncertainties arise in the evaluation of the ultimate strength capacity as the result of several factors, including the following:

(a) Material variability, which includes uncertainties about intrinsic properties of the materials, such as the behaviour laws, yield and tensile strengths, and the influence of the temperature on the mechanical characteristics;
(b) Modelling uncertainty, which includes uncertainties about the geometry of the model (e.g. position and section of the materials), the material failure models considered, and the reliability of calculations performed.

7.24. Material variability and modelling uncertainty can be determined as part of the structural capacity assessment by techniques for uncertainty quantification and propagation in order to establish a failure pressure and/or temperature distribution function. Statistical feedback based on test samples from construction sites may be useful for assessing material variabilities. Benchmarks from mock-ups or feedback experience from pressure tests (if available) may be useful for assessing modelling uncertainty. One example for assessing modelling uncertainty is provided in Ref. [21]. Experiments are supported by analytical predictions of containment failure pressure; these analytical predictions are useful for transferring knowledge at the reactor scale (see, e.g., Ref. [22]). Alternatively, expert judgement supported by simple analysis could be used to establish this failure pressure and/or temperature distribution for various credible failure modes (leaks and ruptures).

7.25. Containment failure is typically described by a composite fragility curve calculated by combining the individual containment failure mode fragility curve. Individual fragility curves are also needed to calculate leak and rupture failures, as described in para. 7.8. These individual fragility curves may also be applied in the accident progression event tree task when competing leak and rupture failure modes are evaluated. Each fragility curve should be characterized by a best estimate (median) failure pressure, a parameter representing the material variability and a parameter representing the modelling uncertainty (see para. 7.23).

Characterization of uncertainties related to concrete structure erosion by molten core debris

7.26. Uncertainties associated with the resistance of concrete structures to extensive erosion by molten core debris should be assessed. In a similar way, as described in para. 7.23, uncertainties are the result of several factors, including material variability and modelling uncertainty.

7.27. Uncertainties affecting the development of the molten core–concrete interaction phenomenon include the availability of water (presence before vessel failure or injection after vessel failure), containment geometry, corium temperature, amount and composition of core debris, decay power and type of concrete used for the basemat construction.

7.28. The molten core–concrete interactions phenomenology is rather complex, and various situations might occur as a result of accident progression. Assessment of the probability of extensive erosion of structures should take into account the uncertainties affecting the molten core–concrete interactions calculations.

These uncertainties can be assessed, for example by using Ref. [24], which comprehensively summarizes various topics related to these interactions, such as available experiments, plant application, simulation tools and models, and uncertainties.

Characterization of uncertainties related to containment isolation failure and containment bypass

7.29. The uncertainties associated with estimation of containment isolation failure and containment bypass should be assessed. This assessment may include uncertainties associated with the following:

(a) Data used (e.g. component reliability data, unavailability due to maintenance, opening duration of large penetrations such as equipment hatches or personnel airlocks);
(b) Human reliability analysis (see Section 8);
(c) Severe accident phenomena modelling (see Section 6).

8. HUMAN AND EQUIPMENT RELIABILITY ANALYSIS

HUMAN RELIABILITY ANALYSIS

8.1. Human failure events can be classified in the same way for Level 2 PSA as for Level 1 PSA (see paras 5.97–5.122 of SSG-3 (Rev. 1) [4]):

(a) Type A human failure events are those that occur before the initiating event and that have the potential to lead to the failure or unavailability of credited systems. Level 2 PSA may include Type A human failure events associated with the systems not considered in Level 1 PSA.
(b) Type B human failure events are those that could lead to an initiating event. These events are included in the Level 1 PSA but are not relevant to Level 2 PSA.
(c) Type C human failure events are those that might occur during critical actions taken by operating personnel after the occurrence of an initiating event. Identifying and analysing Type C human failure events is the main task performed in human reliability analysis for Level 2 PSA. Paragraphs 8.2–8.12 provide recommendations on Type C human failure events in Level 2 PSA. Examples of Type C human failure events include actions

to arrest and mitigate a core damage condition, protect containment from failure, and terminate or mitigate fission product releases.

8.2. Human actions that impact the severe accident progression or the release of radionuclides should be modelled in Level 2 PSA. In general, these actions are identified in the following:

(a) Emergency operating procedures, specifically the actions taken under these procedures that are not modelled in Level 1 PSA but are relevant in Level 2 PSA (e.g. manual containment isolation, containment isolation in shutdown mode, reactor coolant system depressurization, application of criteria for entering severe accident management guidelines).
(b) Severe accident management guidelines, including:
    (i) Operator actions that can be implemented without the support and/or approval of response organizations;
    (ii) Operator actions that can be implemented only with the support and/or approval of response organizations (or technical support centre).
(c) Strategies and guidelines for the deployment of non-permanent equipment or additional strategies not considered in emergency operating procedures or severe accident management guidelines, if such strategies and guidelines are established.

8.3. These human actions are most often taken by operators (in the control room) or by field workers (outside the control room). However, some actions can also be taken by external teams (i.e. external to the plant organization) specially trained to handle severe accidents (e.g. an external response organization that transports non-permanent equipment from another (often distant) location to the site to mitigate the event).

8.4. Some operator actions considered in the Level 1 PSA human reliability analysis may be considered for applicability in the Level 2 PSA. Some actions may be considered failed in the context of Level 1 PSA but could become feasible in the Level 2 PSA if an extended time window is available. This is because the criteria for core damage considered in Level 1 PSA are more restrictive than the criteria applied in Level 2 PSA for arresting the accident progression.

8.5. Human actions (e.g. manual opening of pressure operated relief valves) that are needed before or soon after the onset of core damage can be represented in the extended accident sequence event trees in the Level 1 PSA model if it can be justified that the performance of the action is feasible. In such cases, the status

of such human actions (success or failure) should be reflected either explicitly by an attribute of a PDS or implicitly via their impact on the status of the attributes already defined for the PDS. Other relevant severe accident management actions that are not represented in Level 1 PSA should be incorporated into the Level 2 PSA. Typically, such actions are those expected to occur later in the severe accident sequence, such as refilling steam generators to reduce the releases to the environment via damaged steam generator tubes, restarting the low pressure injection after a high temperature induced break in primary circuit boundaries (primary cooling system) or opening the containment venting line to relieve containment pressure.

8.6. Human reliability analysis for Level 2 PSA should be consistent with Level 1 PSA, for which recommendations are given in SSG-3 (Rev. 1) [4]. In particular, the effects induced by hazards or combinations of hazards on the performance shaping factors of operating personnel should be taken into account in a coherent way between Level 1 PSA and Level 2 PSA.

8.7. Specificities in the human reliability analysis for a Level 2 PSA should be taken into account, such as the following:

(a) How human actions are prescribed. Depending on the organization in place to deal with a severe accident, some actions may be taken independently by the plant staff, while others need to be approved or directed by the technical support centre. In the latter case, the technical support centre needs to be fully operational, and good coordination with plant staff is needed.
(b) The lack of specific instructions in severe accident management guidelines as compared with emergency operating procedures. The lack of specific instructions may increase the likelihood of human error, including errors of commission and errors of omission.
(c) How human actions are performed. The potential impact of degraded workplace conditions (in particular, high radiation levels and/or high temperature in certain rooms, potentially degraded or absent lighting) should be taken into account, especially when the necessary action is taken locally, because these conditions might affect human reliability.

8.8. The Appendix provides more detailed information about performing human reliability analysis for a Level 2 PSA. Human reliability analysis in the context of a Level 2 PSA for multi-unit nuclear power plants is covered in Section 14.

8.9. Assessment of human reliability in the context of deploying non-permanent equipment should follow the same principles as in Level 1 PSA human reliability

analysis (see para. 5.110 of SSG-3 (Rev. 1) [4]). In particular, the following should be taken into consideration:

(a) Adverse conditions on the site and nearby (e.g. climatic conditions, road access, radiological conditions due to severe accident progression);
(b) Delay in actions because of simultaneous actions that share the same resources (e.g. equipment, water, human resources) or other actions with higher priority;
(c) The existence of site specific procedures on the use of non-permanent equipment and the conduct of regular exercises on site;
(d) The coordination between the plant organization and external teams if the non-permanent equipment is provided and implemented by external teams.

8.10. Dependencies between operator actions should be assessed and incorporated into the Level 2 PSA model. This includes dependencies between the human actions credited in the Level 2 PSA and those credited in the Level 1 PSA and Level 2 PSA. The degree of dependency can be influenced in particular by the organization and procedures that are implemented at the nuclear power plant, the context of each human action (i.e. preventing core melt or mitigating severe accident progression), the delay between human actions, and whether the human actions are performed by the same people or using the same equipment.

8.11. Potential adverse effects of severe accident management actions should be considered (e.g. as part of the event tree logic). For instance, injecting water into a degraded core may arrest the progression of a severe accident but with potential side effects, including energetic fuel–coolant interaction, fuel shattering and an additional release of steam, hydrogen and radioactive material. The potential phenomena and their effects on scenario and human reliability should be evaluated.

8.12. The results of the Level 2 PSA should be used to identify or improve severe accident management actions, as explained in Section 15.

EQUIPMENT RELIABILITY ANALYSIS

8.13. Equipment reliability, including common cause failures, is usually modelled in a Level 2 PSA using the same techniques as applied in the Level 1 PSA; for example, data analysis and fault tree construction. Reliability models such as fault tree models are usually linked into the accident progression event tree models or included in the extended event trees (i.e. bridge trees; see para. 5.7).

Recommendations related to equipment reliability analysis in Level 2 PSA models are provided in paras 8.14–8.17.

8.14. Assessment of the reliability of equipment credited within the Level 2 PSA should consider the periodic testing and maintenance practices or planned maintenance procedures for installed equipment, as well as the deployment, installation, start-up and operation procedures for non-permanent equipment. Such practices or procedures may differ from those used for the systems and components credited within Level 1 PSA and thus may have an influence on systems reliability.

8.15. The effect of the environmental conditions resulting from a severe accident on the survivability of components and systems credited within the Level 2 PSA model should be assessed. Some components and systems may already be qualified for severe accident conditions. Otherwise (or if severe accident conditions exceed qualification profiles), the survivability assessment should be based on supporting studies or expert judgment. Adverse environmental impacts might include high levels of temperature, pressure, humidity and radiation in the containment and auxiliary buildings. Examples of adverse conditions that could affect equipment reliability are energetic events (e.g. short term temperature and pressure spikes, impulse loadings from detonations or steam explosions) or high radiation levels that impact specific SSCs (e.g. electronic instrumentation, rubber gaskets vulnerable to high radiation levels).

8.16. Repair actions should be credited in a Level 2 PSA only if there is strong justification for their feasibility. It might be possible to credit repair actions if the specific failure mode of the equipment is known for the specific sequence and (a) it is possible to diagnose the failure, (b) the spare parts and repairing personnel are in place, (c) the environmental and work conditions needed for access and for performing the repair are given or can be ensured, and (d) the time window is sufficiently long to credibly assume the possibility for repair, including the time needed to bring spare parts and maintenance personnel to the plant. Reliable data should be used to assess credible probabilities of repairing components and systems. For components that are not reparable after a severe accident but are needed continuously after core melt (e.g. for corium cooling), the failure probability assessment should integrate the mission time for corium cooling. The failure modelling for different time windows could be discretized to consider different consequences as a function of the instant of failure.

8.17. Dependencies relating to system availability should also be taken into account between Level 1 PSA and Level 2 PSA.

8.18. The mission time of each SSC credited in the Level 2 PSA accident progression event tree should be defined in accordance with the role of the SSC during the severe accident progression until the plant reaches a controlled state (see para. 5.5). This mission time is usually derived from the severe accident progression analyses. The SSC mission times for Level 2 PSA that are defined in this way may be different from or the same as those used in Level 1 PSA. The analyst should verify that the reliability data is suitable for use with the defined SSC mission time.

IDENTIFICATION OF SOURCES OF UNCERTAINTY IN RELIABILITY ANALYSIS

Uncertainties in human reliability analysis

8.19. Uncertainties in human reliability analysis should be addressed in the same way as for human reliability analysis credited in Level 1 PSA.

8.20. The analyst should assess human reliability uncertainty on the basis of the uncertainty of the factors affecting human reliability. The factors include, but are not limited to, the duration of the response time window, duration of the human action, environmental conditions, quality of procedural guidance, operator training, and coordination between plant staff and the technical support centre after entering severe accident management guidelines.

8.21. Many human reliability analysis methods use a simplified approach to assessing uncertainties by providing an error factor applicable to different human error probabilities. Sensitivity analyses should be performed to evaluate the range of human reliability affected by the key factors.

Uncertainties in equipment reliability analysis

8.22. Uncertainties in equipment reliability data should be addressed in the same way as for the equipment credited in Level 1 PSA.

8.23. Uncertainties in equipment qualification or survivability with regard to severe accident conditions should be addressed in Level 2 PSA, taking into account areas of uncertainty related to both the evaluation of specific Level 2 PSA environmental conditions and the resilience of the equipment.

9. DEVELOPMENT OF ACCIDENT PROGRESSION EVENT TREES AND QUANTIFICATION OF EVENTS

DEVELOPMENT OF ACCIDENT PROGRESSION EVENT TREES

9.1. For the development of a Level 2 PSA event tree model, two different approaches can be used: an integrated approach or a separated approach, as described in para. 2.6. The Level 2 PSA analyst should be trained in both approaches and the associated computer codes and should be aware of their limitations and the requirements imposed by the use of the approach and the computer codes chosen. More information can be found in Ref. [5].

9.2. In Level 2 PSAs, event trees are used to delineate the sequence of events and severe accident phenomena that challenge containment integrity after the onset of core damage and the successive barriers to a radioactive release. Event trees provide a structured approach to the systematic evaluation of the capability of a plant to cope with severe accidents. The use of event trees in PSA is shown in Fig. 1. Such event trees, termed accident progression event trees in this Safety Guide, include modelling of phenomena, systems actuation or failure, human actions and all effects on the confinement of radioactive products or on radioactive releases to the environment.

STRUCTURE OF ACCIDENT PROGRESSION EVENT TREES AND NODAL QUESTIONS

9.3. Nodal questions (also referred to as top events) in an accident progression event tree should address the events and physical processes that govern accident chronology, plant response to severe accident conditions, the success and failure of SSCs and human provisions intended for severe accidents, challenges to containment integrity and associated barriers to a radioactive release, the physical containment conditions at the time of a radioactive release, and the eventual magnitude of the release to the environment.[23] The nodal questions of the accident progression event tree are specific to plant type, meaning that issues of importance to severe accident behaviour in one type of reactor and/or containment system might not be important to others. The complexity of the accident progression event tree depends on the scope and objectives of the Level 2 PSA.

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[23] Nodal questions may also address issues and actions relating to severe accident management.

9.4. As the list of events and processes can be extensive, accident progression event trees can grow to become rather large with complicated logic models. However, relatively simple logic models can be sufficient for certain applications. Thus, for instance, if the objective of the Level 2 PSA is solely to determine the large early release frequency, and a quantitative assessment of the full range of severe accident source terms is not necessary, smaller accident progression event tree structures can be developed that focus on severe accident sequences with high consequences within the appropriate time frame [25][24]. In any case, the overall structure of the model should be traceable by independent reviewers and manageable by the PSA team. Therefore, in the accident progression event tree structures, a reasonable balance between modelling detail and practical size should be achieved.

9.5. The accident progression event tree structure should be phenomenologically and chronologically consistent, should properly take into account interdependencies among events and/or phenomena, depending on the reactor technology, and should reflect an appropriate level of detail to satisfy the objectives of the Level 2 PSA. Regarding chronology, it is both useful and common practice to divide the accident progression event tree into phases sequential in time, with the transitions between phases representing important changes in the issues that govern accident progression. The following are examples of phases for water cooled reactors:

(a) Phase 1: Immediate response by the plant to the PDS caused by the initiating event during the early period of in-vessel core damage.
(b) Phase 2: Late period of in-vessel core damage up to failure of the reactor pressure vessel.
(c) Phase 3: Reactor pressure vessel rupture and its consequences.
(d) Phase 4: Ex-vessel phenomena and events:
    (i) Phase 4a: Short term ex-vessel phenomena and events.
    (ii) Phase 4b: Long term ex-vessel phenomena and events.

9.6. Phase 1 is the initial period of in-vessel core damage (e.g. fuel rod heating up above the temperature criterion for fuel integrity, generalized oxidation of fuel cladding, melting of startup control rods). Phase 2 typically starts at the time of core melting and relocation to the lower plenum of the reactor pressure vessel (e.g. formation of debris from melting of fuel cladding, reactor pressure vessel

[24] Reference [25] describes all of the technical elements necessary for developing event trees capable of assessing large early release frequency. In some States, the large early release frequency metric is used in regulatory risk informed decision making.

internals and fuel, relocation of debris and melt in the lower plenum of the reactor pressure vessel). Phase 3 starts at the time of reactor pressure vessel failure and covers the phenomena that occur immediately after (e.g. direct containment heating, ex-vessel steam explosion). Phase 4 covers both the short and the long term ex-vessel phenomena and events after the reactor pressure vessel rupture. Phase 4a covers the period up to a few hours after failure of the reactor pressure vessel and addresses immediate ex-vessel molten core behaviour (e.g. stabilization of ex-vessel melt, onset of core–concrete interaction, human actions, equipment behaviour). Phase 4b covers the long term, starting from a few hours after failure of the reactor pressure vessel, and addresses challenges arising from ex-vessel melt behaviour (e.g. pressurization due to the generation of non-condensable gases during core–concrete interaction, combustion phenomena, pressurization due to ongoing steam generation, human actions, equipment behaviour) (see also paras 7.1–7.22 and Section 8). Typical nodal questions for an accident progression event tree used in separated models for a typical pressurized water reactor with a large, dry containment are provided in Table 5. These or similar nodal questions should also form the basis for the accident progression event tree when the integrated model is used. The nodal questions presented in Table 5 are only examples: in practice, nodal questions and their prior dependencies should be precisely developed by the analysts in accordance with the plant specific reactor technology, design and severe accident management strategies.

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
Phase 1: Immediate response of the plant to the PDS caused by the initiating event during the early period of in-vessel core damage			
0	Is the accident induced by a core prompt criticality with immediate consequences for the vessel or the reactor containment?	None	PDS; accident progression
1	Is the containment isolated?	None	PDS

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
2	What is the fraction of the PDS with alternating current power available?	None	PDS
3	What is the mechanical status of sprays in the very early time frame?	None	PDS
4	What is the mechanical status of fans in the very early time frame?	None	PDS
5	Is the reactor coolant system depressurized manually in the very early time frame?	2	Emergency operating procedures
6	Does a temperature induced hot leg failure occur in the very early time frame?	5	Accident progression
7	Does a temperature induced rupture of a steam generator tube occur in the very early time frame?	5, 6	Accident progression
8	Is alternating current power restored or maintained in the very early time frame?	2	PDS
9	Are containment sprays actuated in the very early time frame?	3, 6, 8	Accident progression

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
10	Does hydrogen combustion occur in the very early time frame and what is the induced pressure peak in the containment? Does it impact (a) the fission product releases (resuspension), (b) the containment integrity or (c) specific equipment in the containment (local effects)?	4, 5, 6, 8, 9	Accident progression
11	Does the containment fail in the very early time frame?	1, 10	Accident progression
12	Is containment isolation recovered in the very early time frame?	1, 8	PDS
13	Is the containment filtered vent system actuated in the very early time frame?	1, 10, 11	Accident progression
Phase 2: Late period of in-vessel core damage up to failure of the reactor pressure vessel			
14	Is core damage arrested in the vessel, preventing a failure of the reactor pressure vessel?	5, 6, 7, 8	Design features of the reactor pressure vessel; accident progression
15	Does an energetic fuel–coolant interaction occur and breach the reactor pressure vessel and containment?	5, 6, 7, 14	Accident progression
16	What is the mode of reactor pressure vessel failure and the process of core debris ejection?	5, 6, 7, 14, 15	Accident progression

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
17	Does 'rocketing' of the reactor pressure vessel occur and breach the containment?	16	Accident progression
	Phase 3: Reactor pressure vessel rupture and its consequences		
18	Is the under-vessel region flooded or dry at breach of the reactor pressure vessel?	None	PDS; design
19	What is the mode of under-vessel fuel–coolant interaction following breach of the reactor pressure vessel?	16, 18	Accident progression
20	Does hydrogen combustion and heating up of direct containment occur at breach of the reactor pressure vessel?	4, 8, 9, 10, 14, 16	Accident progression
21	Does the containment fail at failure of the reactor pressure vessel?	1, 11, 13, 15, 16, 19, 20	Accident progression
22	Is the filtered vent system actuated at breach of the reactor pressure vessel?	1, 11, 13, 15, 16, 19, 20, 21	Accident progression
	Phase 4: Ex-vessel phenomena and events		
	Phase 4a: Short term ex-vessel phenomena and events		
23	Is alternating current power restored or maintained in a short time frame?	8	PDS

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
24	Are sprays actuated or are they restored to operate in a short time frame?	9, 23	PDS; accident progression
25	Are fan coolers actuated or are they restored to operate in a short time frame?	4, 23	PDS
26	Is core debris in a coolable configuration outside the vessel?	15, 16, 17, 18, 19	Design features of the core catcher; accident progression
27	Is a containment heat removal system in operation or restored in a short time frame?	1, 10, 23, 24, 26	Accident progression
	Phase 4b: Long term ex-vessel phenomena and events		
28	Is alternating current power restored or maintained in a longer time frame?	23	PDS
29	Are sprays actuated or do they continue to operate in the late time frame?	24, 28	PDS; accident progression
30	Are fan coolers actuated or do they continue to operate in the late time frame?	25, 28	PDS
31	What is the status of fans and containment sprays in the late time frame?	29, 30	Summary type question

TABLE 5. EXAMPLES OF NODAL QUESTIONS AND ASSOCIATED DEPENDENCIES FOR AN ACCIDENT PROGRESSION EVENT TREE FOR A PRESSURIZED WATER REACTOR (cont.)

Question No.	Nodal question	Dependencies with other nodal questions	Technical bases
32	Does hydrogen combustion occur in the late time frame, and what is the induced pressure peak in the containment? (Does it affect (a) the fission product releases (resuspension), (b) the containment integrity, or (c) specific equipment in the containment (local effects)?)	10, 20, 31	Accident progression
33	Is the filter vent system actuated in the late time frame?	1, 10, 11, 13, 15, 19, 20, 21, 26, 28, 31, 32	Accident progression
34	Is a containment heat removal system in operation during the late time frame ?	1, 10, 28, 29, 32	Accident progression
35	Is the integrity of the containment basemat maintained?	11, 12, 21, 22, 26, 33, 34	Accident progression
36	Does containment failure (e.g. slow overpressurization, hydrogen combustion) occur in the late time frame?	1, 10, 11, 13, 15, 19, 20, 21, 26, 32	Accident progression
37	What are the modes of containment failure?	11, 21, 35	Accident progression

QUANTIFICATION OF EVENTS

9.7. The assignment of conditional probabilities to branches of the accident progression event tree (or other modelling associated with a nodal question) should be supported by documented analyses and data to provide a justified representation of the uncertainty in the outcome at each node. Methodology for the estimation of nodal probabilities using both the threshold approach (see

para. 9.12) and convolution approach (see para. 9.13) may be provided in the documentation attached to the Level 2 PSA. Account should be taken of issues that could affect the analyst's ability to predict the progression of severe accidents and assign uncertainties, including limitations in knowledge regarding aspects of severe accident phenomena, model completeness, fidelity and validation of available computer codes, and applicability of available experimental data to full scale reactor conditions. Example methods for dealing with such uncertainties and the use of expert judgement and expert elicitations can be found in Refs [26–33].

9.8. The rationale used to develop appropriate probabilities for each branch can sometimes be made more traceable by decomposing the problem into a number of sub-issues in accordance with the governing phenomena [34, 35]. Such assessments may be carried out separately and reported in support documentation of the results that are used in the nodal questions of the accident progression event tree, or they may be an integral part of the accident progression event tree in the form of decomposition event trees that are linked to the headings of the accident progression event tree. The degree to which the assessments are integrated into the quantification of the accident progression event tree is principally dependent on the capabilities of the software being used for quantification of the Level 2 PSA. In this context, linked event trees, fault trees (see, e.g., Ref. [36]), user defined functions and other methods have been used for developing and quantifying accident progression event trees [37].

9.9. Regardless of the approach taken to establish values for the probabilities of events, the process should be traceable so that others can follow and understand the technical rationale, and it should be applied consistently to the full range of events or questions described in the accident progression event tree. The Level 2 PSA model typically involves events of different natures, such as system operation, human action and response of containment or components to severe accident phenomena. Recommendations for assessing human action failure and system equipment failure in the context of a Level 2 PSA are provided in Section 8. Paragraphs 9.10–9.14 provide recommendations on determining probabilities associated with the response of the containment or components to severe accident phenomena.

9.10. Sources of current and relevant information should be used to support the assignment of probabilities. Information used to support the quantification of probabilities can include the following:

(a) Basic physical principle calculations (e.g. cooling water flow rate compared to core decay heat) or phenomenological specific models for treatment of relevant severe accident challenges;
(b) Results and/or insights of supporting deterministic analyses using established computer codes for modelling severe accidents;
(c) Relevant experimental measurements or observations;
(d) Results and/or insights of analyses and findings from studies of similar plants;
(e) Expert elicitation involving independent experts.

9.11. Several methods and tools are available to translate such information into a numerical value for each probability. Two simple tools, the threshold approach and the convolution approach, are briefly described in paras 9.12–9.14. Reference [38] has historically been a key source of information for many Level 2 PSAs. However, the state of knowledge of severe accident phenomena has progressed since that study was published in 1990, thus reducing its usefulness as a reference for modern Level 2 PSA studies, which should reflect the current state of knowledge. A compilation of recent, relevant severe accident phenomena can be found in Refs [5, 39, 40]. Developments in water cooled reactors have taken place in a number of areas, such as the following:

(a) In-vessel steam explosions (alpha mode containment failure) [34];
(b) Direct containment heating [41, 42];
(c) Failure of the lower head of the reactor pressure vessel [43, 44];
(d) Flame acceleration and the transition from deflagration to detonation [45];
(e) Thermally induced steam generator tube rupture and hot leg failure [46];
(f) Recovery of partially degraded cores [47, 48].

Threshold approach

9.12. The threshold approach can be used to estimate the probabilities of events that occur when the predicted accident conditions approach an established limit or criterion. The failure probability is, therefore, a function of how close the parameter is to the failure threshold. The assignment of numerical values is thus indicative of the analyst's confidence in the rigour, applicability and completeness of deterministic predictions of relevant phenomena.

Convolution approach

9.13. In the convolution approach, a higher degree of mathematical rigour is applied to the comparison of how close the parameter of interest (e.g. pressure, temperature) is to the failure threshold (e.g. failure pressure, failure temperature). Both the parameter of interest and the failure threshold are treated as uncertain parameters. Probability density functions representing probability distributions of uncertain parameters are reached on the basis of deterministic analyses and expert judgement, and the overlap and/or interference of two such probability distributions determines the degree of 'belief' in (i.e. the subjective probability for) failure. In this case, the consistency of the resulting probability values is dependent on consistent assignment of distribution parameters (i.e. median values, deviations about the median, choice of distribution type and limits).

9.14. Both the threshold approach and the convolution approach can be applied either independently or in combination in the PSA. In any case, to ensure that probabilities are derived in a consistent manner across the wide range of events and phenomena evaluated in the Level 2 PSA, a set of rules should be developed and included in the PSA documentation. Such rules should include the rationale used to assign particular probabilistic estimates.

GROUPING OF END STATES OF ACCIDENT PROGRESSION EVENT TREES INTO RELEASE CATEGORIES

9.15. Once the end states of the individual accident progression event trees have been identified, they should be grouped into specified release categories. Since this involves the grouping of a large number (typically thousands) of end states of the accident progression event tree into a small number (typically tens) of release categories, a systematic process should be applied to this grouping process. This should normally be done using a computerized tool because of the necessity for efficiently handling a large amount of information. The particular way that this is done depends on the software used for quantification of the accident progression event tree, but it can involve post-processing the end states of the accident progression event tree (cutsets) or including the attributes in the accident progression event tree model and using them in the grouping process.

9.16. The end states of the accident progression event tree grouped in a release category are expected to have similar radioactive release characteristics and off-site consequences. The source term analysis performed for the group therefore

characterizes the entire set of end states within the group and reduces the amount of source term analysis that needs to be performed (see Section 10).

10. SOURCE TERM ANALYSIS FOR SEVERE ACCIDENTS

10.1. This section provides recommendations on release category specification and source term analysis. For a PSA with lower objectives, only the frequency of accidents that would result in a large early release may need to be characterized. The following recommendations can therefore be adapted in accordance with the objectives of the PSA.

10.2. All potential plant specific release paths should be identified in the accident progression event tree and considered in the corresponding end states. For practical reasons, in accordance with Fig. 1, the end states of the accident progression event tree are generally grouped into release categories (with similar properties regarding releases). The source term analysis is then performed only for a representative severe accident scenario of each release category. The preliminary list of representative severe accident scenarios should be based on severe accident scenarios established for identified PDSs (see para. 6.8). The choice of representative scenarios should be justified. It is good practice to perform sensitivity studies for the choice of representative scenarios.

10.3. Source term analysis for severe accidents consists of two subtasks: the definition of release categories and the calculation of radioactive releases for each of these categories. In the first subtask, a set of release categories to be attached to the endpoints of the accident progression event tree sequences is defined. These release categories are each identified by a set of characteristics that impact the amount of radioactive release that will arise for accident progression event tree sequences (see paras 10.5 and 10.6). The process of defining these release categories may be supported by code calculations and may be an iterative process. Some methodologies for Level 2 PSA involve a two-step definition of the release categories, in which an initial set of detailed release categories is analysed and then regrouped into a smaller set of release categories on the basis of similarity

of radioactive release (see para. 10.7).[25] In the second subtask, to calculate the radioactive release for each release category, code calculations are performed for one or more representative sequences for each category. The representative sequences are defined by reference to the accident progression event tree sequences that contribute to each release category and are usually selected after quantification of the accident progression event tree so that sequences can be ranked by frequency to assist the selection process (see Section 9).

10.4. Source term analysis in Level 2 PSA should involve the following steps:

(1) Defining the release categories;
(2) Grouping the end states of the accident progression event tree into the release categories;
(3) Performing source term analysis for the release categories.

SPECIFYING AND GROUPING RELEASE CATEGORIES

10.5. Taking into account the reactor technology, the analyst should consider events that, during accident progression modelled in Level 1 or Level 2 PSA event trees, have a significant influence on the release of radioactive material from the containment, for example:

(a) The mode and time of failure of the reactor core cooling;
(b) The mode and time of failure of the reactor vessel or primary circuit, and the vessel pressure at this time;
(c) The mode and time of failure of the containment, failure location, size and resulting transport pathway to the environment;
(d) The availability of systems (e.g. cooling water) and the efficiency of physical mechanisms for cooling molten core material, considering the depth and composition of ex-vessel core debris;
(e) The availability of credited systems able to reduce radioactive releases (e.g. containment spray system, filtered containment venting system, suppression pool, ice condensers);
(f) The retention mechanisms for radioactive material (e.g. pool scrubbing, retention in pipes, filters).

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[25] In these methodologies, the term 'source term category' may be used to refer to the second, smaller, set of grouped release categories; in most methodologies, however, the terms 'release category' and 'source term category' are synonymous.

TABLE 6. EXAMPLES OF TYPICAL ATTRIBUTES USED FOR THE SPECIFICATION OF RELEASE CATEGORIES FOR WATER COOLED REACTORS

Release attributes	Variations
Time frame of the severe accident in which the containment damage/bypass first occurs	At the onset of core damage (i.e. at the bypass of the containment) Early (i.e. during in-vessel core damage) Intermediate (i.e. immediately following breach of the reactor pressure vessel) Late (i.e. several hours after breach of the reactor pressure vessel)
Pressure of reactor pressure vessel during core damage	High (near nominal) Low (depressurized)
Containment pressure	High Low (depressurized)
Modes or mechanisms of containment leakage (associated with a time frame)	Design basis conditions leakage Beyond design basis conditions leakage Catastrophic rupture of containment Containment bypass: — Loss of coolant accident in interfacing system — Steam generator tube/tubes or header rupture Open containment isolation valves Open material hatch access Basemat penetration
Active engineered features providing capture mechanisms for radioactive material	Sprays Fan coolers Filtered vents Others (e.g. water management, reactor coolant system depressurization)
Passive engineered features providing capture mechanisms for radioactive material	Secondary containments Reactor buildings Suppression pools Overlying water pools Passive containment coolers Ice beds 'Tortuous' release pathways Submerged release pathways Alkaline materials

Typical attributes of Level 2 PSA

TABLE 6. EXAMPLES OF TYPICAL ATTRIBUTES USED FOR THE SPECIFICATION OF RELEASE CATEGORIES FOR WATER COOLED REACTORS (cont.)

Release attributes	Variations	
Phenomena	Molten core–concrete interaction Deposition of aerosols Resuspension of deposed aerosols by energetic phenomena Specificities for chemical process (e.g. iodine, ruthenium)	
Time elapsed since the start of the severe accident[a]	Short (for pressurized water reactor typically less than 2 hours) Medium (for pressurized water reactor typically between 2 and 10 hours) Long (for pressurized water reactor typically more than 10 hours)	Additional attribute for linking to Level 3 PSA
Location of release	Ground level Elevated	
Energy of release	Low (minimal buoyancy in ex-plant atmosphere) Energetic (highly buoyant)	
Release rate	Rapid 'puff' release Slow continuous release Multiple plumes	
Containment failure size	Sizes proposed in square metres	
Source term	Amount and composition of different radioactive nuclides or nuclide groups Duration of the release (e.g. release occurring during number of hours)	

[a] Time in relation to the site specific emergency (evacuation) plan.

10.6. After identification of all events that influence the radioactive release, a set of attributes used to characterize the release categories should be considered. Typical attributes for water cooled reactors are shown in Table 6. The release of radioactive material to the environment is a function of these attributes.

10.7. The set of attributes in Table 6 can be used to specify the set of release categories used for source term analysis in the Level 2 PSA. For some methodologies

for Level 2 PSA, this process may generate a very large initial number of release categories, which should be further grouped into a manageable final set that can be used in the source term analysis through an integral computer code (see paras 10.16–10.21). This grouping process can be less condensed if a mechanistic source term code is used for the source term analysis (see paras 10.22–10.26).

10.8. Some accident scenarios can include several containment failure modes. The analyst should pay attention to the quantification of the frequency of each containment failure individually in order to capture its importance in the global results.

10.9. For slow containment overpressurization sequences, the analyst should distinguish between containment leak and containment break, as a leak might prevent a major containment failure and subsequently limit the amount of released fission products (see para. 7.8).

10.10. The grouping of accident progression sequences in release categories often necessitates the application of some assumptions and simplifications that might introduce additional uncertainties. Special care should be taken to keep track of any assumptions and simplifications so that the additional uncertainties are not overlooked during the uncertainty analysis.

SOURCE TERM ANALYSIS APPROACHES

10.11. In Level 2 PSA, the source term specifies, for a given accident scenario, the amount and composition of radioactive material released from the plant to the environment and the timing, location and kinetic energy of the release. Many plant design features and accident phenomena have been recognized to affect the magnitude and characteristics of source terms for severe accidents. These include fixed plant design characteristics, such as the configuration of the fuel and the control assembly and material composition, core power density and distribution, fuel burnup and concrete composition as well as radioactive decay of radioactive releases. The analyst should be familiar with the specific plant design features (see Section 4) and accident phenomena (see Section 6) for the definition of end states of the accident progression event tree.

10.12. Depending on the reactor technology, a combination of the following approaches could be used for the source term calculations:

(a) Applying one of the integral computer codes described in Annex I for water cooled reactors to a limited number of representative accident scenarios;
(b) Applying a fast-running source term code (see section on dedicated codes in Annex I) to a large number of accident scenarios;
(c) Applying detailed models or bounding estimates of source terms or transposition from another nuclear power plant to obtain preliminary results, for example during the design phase.

The methodology and the justification of the selection of the approach followed should be provided as part of the documentation of the Level 2 PSA.

10.13. The extent to which source term analysis needs to be performed depends on the objectives and intended applications of the Level 2 PSA. If the source term is to be used within Level 3 PSA, the characterization of the source term should be sufficiently detailed to be adequate as an input for Level 3 PSA consequences calculations (see also Refs [49, 50]). The justification of the adequacy should be developed and documented. The analysis of off-site consequences necessitates a detailed characterization of the release of radioactive material (i.e. a quantitative tracking of the core inventory of radioactive material at a detailed level) [51].

10.14. Noble gases, iodine and caesium are often selected as leading indicators of the overall radiological source term. Thus, there are many ways of specifying the attributes of a radiological source term, including that different release categories may have the same source term (i.e. same amount and composition).[26] However, it is important to specify these attributes at the beginning of the Level 2 PSA project (see paras 2.17–2.19).

Source term analysis with a plant specific approach

10.15. Plant specific source term analysis can be performed for each of the release categories. Such an analysis can be conducted using an integral computer code or a fast-running (also referred to as 'dedicated') computer code (see Annex I).

[26] The way the attributes are specified is also influenced by the objectives of the Level 2 PSA, for example, whether or not a Level 3 PSA or part of a Level 3 PSA will be performed.

Source term analysis with an integral computer code

10.16. One option is to use an integral computer code to perform plant specific source term analysis for each of the release categories. This code should be capable of modelling the integrated behaviour of severe accident phenomena such as the thermohydraulic response of the reactor, heat-up of the core, fuel damage and relocation of core material, conditions in the containment and adjacent buildings, release of radioactive material from the fuel and transport of radioactive aerosols and vapour through the reactor coolant system into the containment and subsequently to simulate the source term to the environment.[27]

10.17. In the source term analysis, all of the processes that affect the release and transport of radioactive material inside the containment and in adjacent buildings should be modelled. These processes include the following:

(a) Releases of radioactive material from the fuel during the in-vessel phase;
(b) Retention of radioactive material within the reactor coolant system;
(c) Releases of radioactive material during the ex-vessel phase;
(d) Retention of radioactive material inside the containment and adjacent buildings;
(e) Resuspension, revaporization, condensation and re-entrainment (e.g. as a result of energetic phenomena, chemical reactions or mechanical effects).

10.18. Considering previous processes, in the calculation of the source term and the plant model, the spatial distribution of radionuclides within the reactor coolant circuit and the containment should be estimated, as well as the quantity released to the environment.

10.19. The analysis should be performed for a representative accident scenario in each release category. Sensitivity analyses should be performed to provide confidence that the source terms have been accurately characterized and there is not undue variation in the source term magnitude within each release category.

10.20. Source term calculations with integral computer codes for severe accident progression analysis generally consider group categories of radioactive elements or chemical compounds rather than individual radioisotopes [54, 55].

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[27] In some Level 2 PSAs, parametric source term models have been developed on the basis of calculations performed with codes such as MAAP [52] or MELCOR [53]. This approach enables the uncertainties in the source term parameters to be combined with the integrated process for uncertainty assessment and uncertainty propagation.

This simplification is necessary to reduce the hundreds of radioisotopes generated in nuclear reactor fuel to a reasonable number of groups of radioactive elements that can be tracked by an integrated severe accident computer code. Different group categories have been used in different computer codes. However, most group categories are based on similarities in the physical and chemical properties of the radioactive elements. Group categories also take into account similarities in the chemical affinity of elements to reactions with other radioactive elements and non-radioactive material that they might encounter in transport within the reactor coolant circuit and containment (e.g. steam, hydrogen, structural materials). Typical group categories used in the analysis of releases of radioactive material are shown in Table 7. The source term, therefore, could be expressed in terms of the fraction of the initial core inventory of one or more of these groups of radionuclides. For operating modes relating to refuelling outages, the stage of refuelling (i.e. before or after) and the subsequent mixture of newer and older fuel elements should be considered in the definition of the core inventory. The analyst should be familiar with the composition of radioactive material in the group categories proposed by the integral computer code used.

10.21. The efficiency with which the groups of radionuclides listed in Table 7 are transported to the environment depends strongly on the chemical form they assume after leaving the core region. Numerous chemical interactions can occur that cause elemental forms of these species to react and form compounds with a wide range of physical properties [56]. Iodine, for example, is widely known to react with caesium to form volatile caesium iodide. However, this is not the only form in which iodine can be transported along the release pathway. Several species listed in Table 7 can be transported in more than one chemical form. Partitioning of the core inventory of reactive species among their possible chemical forms is considered a best practice that might introduce a source of uncertainty in the source term calculations. If performed, the impact of the partitioning in the assessment of source term calculations should be considered via a sensitivity study, for example.

Source term analysis with a fast-running code

10.22. A second option is to use a fast-running computer code to perform plant specific source term analysis with no limitation on the number of calculations. Such a code does not calculate the integrated behaviour of severe accident phenomena (e.g. thermohydraulic response of the reactor, core melt) but calculates only the release of radioactive material from the fuel and transport of radioactive aerosols and vapour through the reactor coolant system into the containment, the behaviour of radioactive material in the containment and the release outside. The information on thermohydraulic response, fuel melt, conditions in containment,

energetic phenomena or accident kinetics are available through the release category attributes and are input data for the fast-running code.

TABLE 7. EXAMPLES OF GROUP CATEGORIES FOR ELEMENTS IN RADIOACTIVE MATERIAL

Group	Elements in group	Representative element in group
Noble gases	Xe, Ne, Ar, Kr, He, Rn	Xe
Halogens (aerosols)	I, Br, Cl, F	I
Halogens (gaseous)	I_2, Br_2, Cl_2, F_2	I_2
Halogens (organic)	ICH_3, $BrCH_3$	ICH_3
Halogens (oxidized)	IO_x, BrO_x	IO_x
Alkali metals	Cs, Rb, Li, Na, K, Fr	Cs
Alkaline earths	Ba, Mg, Ca, Sr, Be, Ra	Ba
Chalcogens	Te, O, S, Se, Po	Te
Refractory metals and platinoids	Ru, Mo[a], Pd, Tc, Rh, Re, Os, Ir, Pt, Au, Ni	Ru
Tetravalents	Ce, Pu, Np, Zr, U[a], Ti, Hf, Th, Pa, C	Ce
Trivalents	La, Al, Sc, Y, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Cm, Bk, Cf	La
More volatile main group	Cd, Hg, Zn, As, Sb, Pb, Tl, Bi	Cd
Less volatile main group	Sn, Ga, Ge, In, Ag	Sn
Boron	B, Si, P	B

[a] Mo and U are represented as separate groups in some models.

10.23. For fast-running codes the grouping of radionuclides is defined in a similar way as for integral codes (see Table 7).

10.24. A fast-running code should be validated, for example by comparison with an integral code or with experimental data.

10.25. Uncertainties in the key parameters related to the accident phenomena mentioned in para. 10.22 should be considered in the fast-running code to allow for the quantification of uncertainties in the source term analysis.

10.26. A fast-running code may be integrated into the accident progression event tree so that the tree quantification includes the source term analysis including uncertainties.

Source term analysis with a simplified approach

10.27. In some cases, a simplified approach can be taken, using the source term analysis from another nuclear power plant whose design and features relevant to severe accident progression are sufficiently similar to the plant being analysed, and for which the results of deterministic analysis are available. When a reference source term analysis is used as a surrogate for plant specific calculations, the following criteria should be met for its use in a Level 2 PSA:

(a) A technical basis should be established to justify that the plant undergoing Level 2 PSA is sufficiently similar to the proposed reference plant. Design features that affect the transport of radioactive material and its retention within the reactor pressure vessel, associated coolant system piping and containment structures and systems should be identified and compared.
(b) It should be ensured that the accident sequence(s) modelled in the reference source term analysis are sufficiently similar to the accident sequences of interest to the Level 2 PSA for the plant under analysis. Differences in the operation of reactor safety systems or containment associated systems can

invalidate the applicability of a reference plant calculation to a particular PDS.[28]

(c) The calculation for the reference plant should be performed using a contemporary model of plant response to severe accident phenomena. Caution should be used in applying reference plant results that are several years old. The state of knowledge and level of sophistication in modelling the progression of severe accidents have evolved significantly in recent years and thus reduced the value of some results available in open literature (e.g. scientific and technical publications).

10.28. The use of the source term analysis from another plant may be helpful to define parametric models or bounding estimates of source terms, especially during the design phase of a new nuclear power plant.

USE OF COMPUTER CODES FOR SOURCE TERM ANALYSIS

10.29. Models and correlations introduced in the computer codes that are used for source term analysis (i.e. integral or fast-running codes) are required to be verified and validated (see Requirement 18 of GSR Part 4 (Rev. 1) [2]).

10.30. The users of the computer code for source term analysis should be trained in the use of the code and be familiar with the phenomena being modelled by the code and the way that they interact, the meaning of the input and output data, and the limitations of the code. Recommendations on the selection of software, approaches and methods are provided in paras 3.16–3.18.

RESULTS OF THE SOURCE TERM ANALYSIS

10.31. The overall results of the source term analysis should be clearly presented and documented. The characteristics of the source terms associated with the release categories should be clearly documented. One way of doing this is to

[28] For example, many calculations of accident sequences involving 'station blackout' for several reactor designs can be found in open literature. However, there are many variations of station blackout, depending on the particular system configuration of a plant. In some cases, sufficient direct current power might be available to operate a small group of components (e.g. relief valves) or systems (e.g. steam driven pumps) in some plants that are not available in other plants. Such differences should be carefully considered before calculated results from the literature are applied to the plant under study.

present the results in the form of a matrix similar to the C matrix described in Section 11, in which the frequency (or the contribution to the total core damage frequency) of each release category is tabulated. An example format for presenting the results of the source term analysis is shown in Table 8 (another example is presented in Ref. [57]).

10.32. The source terms and frequencies of the release categories — the latter obtained as a result of accident progression event tree quantification — should be used to determine the large release frequency or the large early release frequency for comparison with numerical probabilistic safety goals or criteria where they have been set, as described in Section 11. To achieve this, the terms 'large' and 'early' need to have been defined within the Level 2 PSA project. This can be done in a number of ways, as outlined in paras 2.16–2.19 and in Ref. [57].

10.33. An alternative format for displaying the results of the source term analysis is by means of a complementary cumulative distribution function that is based on the frequency of releases greater than X, where X varies from the smallest to the largest calculated quantity of release. To achieve this, the term 'quantity of release' needs to be defined within the Level 2 PSA project; this term might be understood, for example, as the activity of a leading radioisotope or of a group of relevant radioisotopes. The frequency of releases and the magnitude of releases should be considered together for the interpretation of the Level 2 PSA and its applications.

10.34. The insights gained from such a quantitative evaluation of radionuclide releases should be summarized and discussed. The results of the quantitative sensitivity analysis or uncertainty analysis should also be presented and discussed. In particular, for each radioactive material group, the frequency of exceeding a given release quantity should be provided. The results should clearly show the statistical significance of each complementary cumulative distribution function (e.g. mean, median, 95th percentile).

ANALYSIS OF UNCERTAINTIES IN SOURCE TERMS

10.35. Requirement 17 of GSR Part 4 (Rev. 1) [2] states that "**Uncertainty and sensitivity analysis shall be performed and taken into account in the results of the safety analysis and the conclusions drawn from it.**" Uncertainty and sensitivity analyses help to understand how the various modelling options within a code affect the calculated results. In addition to the uncertainties in modelling severe accident phenomena, many chemical and physical processes are still

TABLE 8. EXAMPLE SUMMARY OF SOURCE TERMS FOR WATER COOLED REACTORS

	Release category attributes							Fraction of core inventory[a] to environment[b]			
	Release category	Frequency (a^{-1})	Time release begins	RCS[c] pressure at vessel failure	Mode of containment leakage	Release through auxiliary building	Active attenuation mechanism	Xe	I	Cs	Other
Source terms	1	$i.iE^{-i}$	Early	Low	SGTR[d]	Yes	None	0.995	0.11	0.08	$x.xE^{-x}$
	2	$j.jE^{-j}$	Intermediate	High	Rupture	No	None	0.99	0.14	0.11	$y.yE^{-y}$
	…										
	X	$k.kE^{-k}$	Intermediate	Low	Nominal leakage[e]	Yes	Sprays	0.84	0.04	0.02	$i.iE^{-i}$
	…										
	Y	$m.mE^{-m}$	Late	Low	Rupture	No	Sprays	0.89	0.002	0.001	$j.jE^{-j}$
	…										
	N										

[a] Fraction of core inventory to the environment here refers to the core inventory before the severe accident scenario begins.

[b] These are sample values only.

[c] RCS: reactor coolant system.

[d] SGTR: steam generator tube rupture.

[e] The 'nominal leakage' value is based on normal operating conditions (measured by tests), which might be different from design.

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subject to research. These processes govern the release of radioactive material from fuel, the deposition and retention of radioactive material on reactor internal surfaces and containment surfaces, and radioactive material from scrubbing by containment associated systems. The following are examples of issues that give rise to uncertainties in source term analysis for water cooled reactors:[29]

(a) Uncertainties in core damage processes and containment behaviour (see Sections 6 and 7);
(b) Effects of fuel exposure (burnup) on the release fraction of radioactive material from the fuel matrix;
(c) Chemical forms of volatile and semi-volatile species;
(d) Chemical interactions with fuel, neutron absorbers and structural materials during core degradation;
(e) Deposition rates of radioactive material and aerosols on the surfaces of the reactor coolant circuit;
(f) Deposition of radioactive material in piping and other components in accident sequences with containment bypass;
(g) Release of radioactive material and aerosols during molten core–concrete interaction;
(h) Chemical processes in reactor coolant systems, corium and during molten core–concrete interaction;
(i) Interaction between hydrogen burn or radicals in flame fronts and airborne radioactive material (e.g. possible resuspension of radioactive deposits);
(j) Scrubbing efficiency of aerosols and vapours in suppression pools, ice beds or bubble towers;
(k) Aqueous chemistry of radioactive material captured in water pools;
(l) Revaporization and resuspension of radioactive material from surfaces;
(m) Chemical decomposition of radioactive material aerosols;
(n) Radioactive release into the environment with regard to containment break size, containment leak rate, released fraction of inventory, iodine chemistry and deposition of radioactive material onto containment surfaces.

10.36. Past and ongoing research programmes have made significant progress towards reducing uncertainty in severe accident source terms (see, e.g., Refs [54, 55]). Uncertainties associated with the physical processes involved in core damage and core relocation lead to uncertainty in respect of the release of radioactive material from fuel (see Section 6). Uncertainties associated with

[29] Many of the examples given relate to iodine (and ruthenium) forms and behaviour. Owing to the importance of assessing radiological consequences, reduction of uncertainties is an issue to be addressed in dedicated severe accident research programmes.

the response of the containment to conditions beyond its design basis lead to uncertainty in respect of the driving forces for radioactive material transport along the pathway to the environment. Examples of uncertainties associated with these areas are given in Section 7.

10.37. Up to date knowledge on severe accidents and on fission product behaviour should be used in Level 2 PSA. The assessment of uncertainties can be addressed by carrying out sensitivity studies for the major sources of uncertainty that influence the results of the Level 2 PSA (see paras 11.25 and 11.26). Uncertainty modelling can also be introduced directly into the accident progression event tree (i.e. distribution of probability) for the propagation of uncertainties within the model while it is possible, depending on the PSA tool.

11. QUANTIFICATION OF EVENT TREES AND ANALYSIS OF RESULTS

QUANTIFICATION OF EVENT TREES

11.1. The quantification process consists of calculating the frequencies of the end states of the accident progression event tree. The results of this quantification lead to basic results of the Level 2 PSA, and the basic results can be presented by different groupings of end states (e.g. containment failure modes, type of releases, kinetics, source terms for Level 3 PSA).

11.2. The quantification process depends on the PSA computer codes used for the development of the accident progression event trees. Two categories of PSA computer codes exist: the first based on minimal cutset calculations (Boolean algorithm) and the second based on scenario calculations (chronological algorithm). For both categories, the frequency of the release categories or other groupings are calculated by aggregating the frequencies of all the end states of the accident progression event tree that are assigned to the group, depending on the PSA computer code used.

11.3. The Level 2 PSA quantification process may be accomplished in various formats with either direct or intermediate links between the initiating event sequences and the ultimate Level 2 PSA release categories. For the approach using a combination of small event trees and a large fault tree (the fault tree linking approach [36]), Boolean reduction is carried out by the software for the logic

models developed using event trees and fault trees for each initiating event group. As with the Level 1 PSA, before quantifying the Level 2 PSA, care should be taken to ensure that no logic loops exist in the model. If such loops exist, breaking the loops is a prerequisite for quantification.

11.4. The probabilistic quantification of the Level 2 PSA should be performed using a suitable computer code that has been through a comprehensive process of verification and validation. Most PSA computer codes used for the Level 1 PSA are suitable for Level 2 PSA analysis, as well (i.e. computer codes used for the development of event trees and fault trees). To perform these analyses, the users of the codes should be adequately experienced in PSA modelling, should have a detailed understanding of the severe accident progression process and should understand the limitations of the code.

11.5. The overall results of the quantification of the Level 2 PSA model should include the following:

(a) Release category frequency;
(b) Contributions to the release category frequency arising from each of the Level 1 PSA PDSs (or if directly linked to the Level 1 PSA, the initiating event groups);
(c) Cutsets and cutset frequencies (for the fault tree linking approach) or scenarios and scenario frequencies (for the approach using event trees with boundary conditions);
(d) Contribution of significant accident progression event tree sequences to the release frequencies;
(e) Results of sensitivity studies and uncertainty analysis;
(f) Importance measures (e.g. the risk achievement worth and the risk reduction worth for basic events) that are used for the interpretation of the Level 2 PSA.

11.6. To verify the correctness of the severe accident progression sequence modelling, the sum of the release category frequencies should be validated against the core damage frequency determined from the Level 1 PSA (typically core and/or fuel damage frequency). Justification for any numerical deviations should be given.

11.7. The analyst should check that the accident sequences or cutsets identified by the solution of the Level 1 PSA model are propagated into the Level 2 PSA structure and are appropriately reflected in the release categories. In addition, a check should be made to confirm that the cutsets (or sequences) representing

combinations of initiating events, component failures and severe accident phenomena that are expected to lead to containment failure are included in the list of cutsets (or sequences) generated. The software used for accident progression event tree quantification should be capable of quantifying success branches in the event tree, since success branches are equally important as failed branches in the accident progression event tree. In fact, the conventional meanings of success and failure do not always apply in accident progression event trees: both are alternatives of the process and both may have high probability.

11.8. Taking into consideration the definition of all risk metric terms used to characterize the significance of containment failure and releases that should be defined at the beginning of the Level 2 PSA study (see paras 2.16–2.19), release metrics associated with the containment failure and source term could take the form of an absolute criterion or a relative criterion (e.g. relative to total core damage frequency or large release frequency; see Annex III).

11.9. A check should be made that any post-processing of the Level 2 PSA cutsets (i.e. to remove mutually exclusive events or to introduce recovery actions not included explicitly in the Level 2 PSA model) has produced the correct results. Depending on the PSA code used and the quantification options applied, post-processing of cutsets might introduce inconsistencies in the numerical release frequency results, leading to discrepancies between the total Level 1 and Level 2 PSA results. The PSA analyst should be aware of these possible inconsistencies and should check whether the Level 2 PSA conserves the frequency input from the Level 1 PSA.

11.10. For quantification of the Level 2 PSA, truncation limits (e.g. cut-offs) should be specified to limit the time taken for the analysis. The usual approach is to set a truncation frequency limit so that cutsets with a lower frequency that represent a negligible contribution to risk may be confidently omitted from the final quantification. Justification should be provided that the truncation limit has been set at a sufficiently low level that the overall result from the Level 2 PSA converges and the chosen limit has a negligible impact on the estimated frequency of a large release. The choice of cut-off may vary depending on the application of the Level 2 PSA. Performing a study is a typical way to demonstrate convergence. In a convergence study, the analyst performs quantification at a number of cut-off values to identify a cut-off value at which a stable frequency result is obtained.

ANALYSIS OF RESULTS OF ACCIDENT PROGRESSION EVENT TREES

11.11. Results and insights gained from the quantification of accident progression event trees should be summarized and discussed. The frequency of each release category should be indicated first.

11.12. In addition, useful information and insights gained through Level 2 PSA quantification should also be indicated. For this purpose, results can be presented in various formats. They are often tabulated in the form of a containment performance matrix ('C matrix') [58], which is a concise way of comparing the relative likelihood of the various outcomes of the accident progression event trees. The C matrix identifies the PDS frequency and the conditional probabilities that a release category can be realized, given a specific PDS (see Eq. (1)). An example layout for the content of a C matrix is presented in Table 9. Uncertainty analysis leads to alternative sets of values of the elements of the C matrix:

$$R(n) = \sum_{m=1}^{M} F(m)\ C(m, n) \tag{1}$$

where

$R(n)$	is the release frequency;
M	is the total number of PDSs;
$F(m)$	is the PDS frequency;
$C(m, n)$	is the conditional probability that a release category n can be realized, given a PDS m.

11.13. For each PDS, the main minimal cutsets could also be indicated. A variant of the C matrix is to identify the conditional probability that a release category can be realized, given the families of PDSs for the list of initiating events (see Eq. (2)). This might not be sufficient, so additional useful information may be presented, such as the release category frequencies for each plant operational state, or the distribution of the different causes of containment failure for specific release categories.

TABLE 9. CONTAINMENT PERFORMANCE MATRIX (C MATRIX)

PDS	Release category							PDS frequency
	1	2	...	n	...	...	N	
1	C(1,1)	C(1,2)	...	C(1,n)	...	...	C(1,N)	F(1)
2	C(2,1)	C(2,2)	...	C(2,n)	...	...	C(2,N)	F(2)
3	C(3,1)	C(3,2)	...	C(3,n)	...	...	C(3,N)	F(3)
...	...	...	...	..	...	...	...	...
m	C(m,1)	C(m,2)	...	C(m,n)	...	...	C(m,N)	F(m)
...	...	...	...	..	...	...	...	...
M	C(M,1)	C(M,2)	...	C(M,n)	...	...	C(M,N)	F(M)
Release category frequency	R(1)	R(2)	...	R(3)	...	...	R(N)	R

$$R(n) = \sum_{m=1}^{M} \sum_{j=1}^{J} A(j)\, B(j, m)\, C(m, n) \qquad (2)$$

where

R(n) is the release frequency;

M is the total number of PDSs;

J is the total number of initiating events;

C(m, n) is the conditional probability that a release category *n* can be realized, given the families of PDSs, *B (j, m)*, for the list of initiating events *A(j)*.

11.14. The major contributors to each release category of interest should be identified and explained. This generally concerns large releases, but this approach can also be extended to any other consequence deemed necessary. The root causes of variations in the conditional probability of each examined consequence among the various PDSs should be explored and explained.

11.15. As discussed above, the frequencies and uncertainties associated with each release category can be determined by combining the results of the

Level 1 PSA (frequencies of occurrence of the various PDSs and their associated uncertainties) with the conditional probabilities of various containment failure modes and/or release modes and their associated uncertainties resulting from quantification of the accident progression event tree. The contribution of each release category to the total release frequency should also be tabulated to allow for identification of major contributors to the total release frequency.

11.16. For each of the selected release categories or related group of release categories, one representative accident sequence should be selected for which a source term is estimated on the basis of results obtained from plant specific calculations employing an appropriate computer code for estimating source terms for severe accidents (see Section 10 and Annex I) or past analyses from Level 2 PSAs of representative plants. When using representative plant analyses for releases, care should be taken to take into account plant differences in core fission product inventory (typically associated with fuel design, core power and operational history, and containment failure modes and failure pressures). Considerations regarding the acceptability of source terms from representative plant specific PSAs should be documented.

11.17. The selection of the representative accident sequence should be governed by its frequency and consequence dominance within the release category. Alternatively, source terms can be estimated for each accident sequence that contributes to a particular release category. An intermediate approach is sometimes taken, in which calculations are performed for the dominant accident sequence and an alternative accident sequence in each release category. In addition, for release categories that result from potentially uncertain mechanisms (e.g. steam explosion, direct containment heating) for which trustworthy models might not be readily available for the code used, code calculations could be augmented by simple analyses and expert judgement.

IMPORTANCE, UNCERTAINTY AND SENSITIVITY ANALYSES

Importance analysis

11.18. Importance measures for basic events, groups of basic events, credited systems and groups of initiating events, among others, should be calculated and used to interpret the results of the Level 2 PSA. Importance metrics are typically focused on contributions to containment failure frequency, large early release frequency and large release frequency, but other potential end states may be of interest. These metrics may be more specific or may encompass more than one

operating mode or operational state. Importance measures typically include: (a) Fussell–Vesely importance; (b) risk reduction worth; (c) risk achievement worth and (d) Birnbaum importance. The various importance measures provide a perspective on which basic events and other systems and events contribute most to the current estimate of risk (Fussell–Vesely importance, risk reduction worth), which contribute most to maintaining the level of safety (risk achievement worth) and for which basic events the results are most sensitive (Birnbaum importance).

11.19. Importance analysis should identify contributions and impact on the risk of reactor operating modes, main SSC failures, actions from operating personnel, internal and external hazards, and mitigation strategies.

Types of uncertainty

11.20. Since Level 2 PSA analysts use probabilities in the accident progression event trees to reflect confidence that particular choices of modelling parameters or event outcomes are the correct ones, the treatment of uncertainties is one of the most important aspects of Level 2 PSA.

11.21. Paragraphs 11.22–11.25 provide recommendations on meeting Requirement 17 of GSR Part 4 (Rev. 1) [2] on uncertainty and sensitivity analysis for Level 2 PSA (issues giving rise to uncertainties are presented in Table 4 and para. 10.35). Recommendations on uncertainties related to systems and operator actions are provided in SSG-3 (Rev. 1) [4].

11.22. Uncertainty arises in a Level 2 PSA analysis as a result of several factors. The analyst should consider the following sources of uncertainty or should develop, use and justify an alternative source, as appropriate:

(a) *Incompleteness uncertainty.* The overall aim of a Level 2 PSA is to assess the possible scenarios (sequences of events) that can lead to releases of radionuclides; mainly those scenarios modelled in the Level 1 PSA. However, there is no guarantee that this process can ever be complete and that all possible scenarios have been identified and properly assessed. This potential lack of completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify. It is not possible to address this type of uncertainty explicitly. However, extensive peer review can reduce this type of uncertainty, for example by verifying the adequacy of the sequence consisting of cutsets, correctness of the input parameters and assumption of human errors. Sensitivity analyses, including bounding analyses, may be employed to provide estimates regarding the

significance of the uncertainty, so the Level 2 PSA should ensure that those sensitivity analyses are performed and reviewed (see paras 3.23–3.29).

(b) *Loss of detail due to aggregation.* Loss of detail can occur at several points in the Level 2 PSA. Grouping accident sequences or cutsets from the Level 1 PSA into PDSs for input into the Level 2 PSA for practical reasons also introduces uncertainties due to the resulting loss of some modelling detail. Further, the process of grouping (binning; see para. 1.9) accident sequences on the basis of key sequence attributes introduces bias and uncertainty, as the attributes used by the analyst to group accident sequences need to be broad. In Level 2 PSA, these grouped sequences can vary in the timing of the initial fuel damage, the impact of the event progression on the containment, and the magnitude and content of the containment fission product release, among other things. The grouping process often has two competing goals: optimizing the number of detailed severe accident progression analyses to be performed, and providing a sufficient level of detail for Level 2 PSA insights or for the characterization of the Level 3 PSA inputs. The impact of the grouping process on uncertainty is difficult or impossible to quantify precisely; however, this grouping process is intended to maintain the key features of the release category such that more refined analysis or subdivision will not impact Level 2 PSA insights. In practice, the grouping process is intended to be performed conservatively. As more detailed resolution regarding severe accident phenomenology becomes available, and as increases in computing resources allow increasing levels of detail to be captured in the PSA, this uncertainty is expected to diminish. Sensitivity analyses may be used to assess the extent of these modelling uncertainties on the plant response. To mitigate these uncertainties, the grouping within a Level 2 PSA should be justified and well documented.

(c) *Modelling uncertainty.* This arises due to a lack of complete knowledge concerning the severe accident phenomenology, limitations related to the reproducibility of real severe accident conditions by research experiments, and appropriateness of the methods, models, assumptions and approximations used in assessment of those processes and the individual analysis tasks that support a Level 2 PSA. Modelling uncertainties are formally addressed as part of the uncertainty treatment in the Level 2 PSA, with consideration of severe accident progression analysis (see paras 6.24–6.27), containment integrity analysis (see paras 7.23–7.29), human and equipment reliability analysis (see paras 8.19–8.23), development of accident progression event trees (see paras 9.7–9.11) and source term analysis (see paras 10.35–10.37).

(d) *Parameter uncertainty.* This arises due to the uncertainties associated with the values of the fundamental parameters used in the quantification of the

Level 2 PSA, such as equipment failure rates and frequencies of initiating event sequences. Parameter uncertainty may also arise from uncertainty in the physical attributes of the parameters used to model the containment challenge and containment response. This is the type of uncertainty that is usually addressed by an uncertainty analysis through specifying uncertainty distributions for all of the parameters and performing sensitivity studies or propagating them through the analysis. Recommendations on addressing parameter uncertainties within Level 2 PSA are provided in paras 11.24–11.26.

11.23. The Level 2 PSA analysts should identify the dominant sources of uncertainty in the analysis and should quantitatively characterize the effects of these uncertainties on the baseline (point estimate) results. Characterization of the effects of these uncertainties is typically supported using two methods: uncertainty analysis and sensitivity analysis.

Uncertainty analysis

11.24. Uncertainty analysis examines a range of alternative models or parameter values, assigns each model or value a probability distribution and generates a distribution of the results, within which the baseline results represent one possible outcome. Each result within the full distribution is accompanied by a (subjective) probability representing the degree of belief in that result. Cumulative probability levels for the results can be calculated (e.g. the 5th, 50th and 95th percentiles represent 5%, 50% and 95% probabilities, respectively, and the 'true' result is below the respective level for which each of these probabilities is stated). In general, the process of quantification and propagation of uncertainties in the Level 2 PSA can be divided into four principal steps that the analyst should take, as follows:

(a) *Specification of the scope of the uncertainty analysis.* The sources of uncertainty in a Level 2 PSA are numerous, and it is impractical to address all of them quantitatively. Experience in performing uncertainty studies for limited aspects of severe accident phenomena suggests that the effects of uncertainties from some sources are larger and more dominant than the effects of uncertainties from other sources. In an integral sense, then, the aggregate uncertainty in Level 2 PSA results can be estimated by selecting the dominant sources of uncertainty and treating them in detail. Reference [33] provides information on an evaluation of uncertainties in relation to severe accidents and Level 2 PSA.

(b) *Characterization and/or evaluation of uncertainty issues.* After the definition of the scope of the analysis, the second step is to identify the range of values of uncertain parameters. Each value within the range of values that the uncertain parameter can take on is associated with a probability, thereby creating a probability density function or probability distribution. In many cases, such density functions or probability distributions have been determined in the assessment of probabilities for branch points in the accident progression event tree. Judgements reflected in the probability distributions for each parameter should be supported by data, analyses and consideration of the published literature and should be considered for peer review as part of the independent verification (see paras 3.24–3.29).

(c) *Propagation of uncertainties.* Propagation of uncertainties is usually done by simulation methods based on either simple (Monte Carlo) random sampling or stratified (Latin hypercube) sampling procedures. Additional details can be found in Refs [25, 26, 59–66].

(d) *Display and interpretation of results.* The results of the uncertainty analysis should be carefully evaluated to strengthen the conclusions of the Level 2 PSA. In modern PSAs that include a quantitative assessment and propagation of uncertainties, the results are displayed using histograms, probability density functions, cumulative distribution functions and tabular formats showing the various quantiles of the calculated uncertainties, together with the estimates of the mean and median of the probability distributions [25, 26]. Regression analysis techniques can also be applied to assess the importance of particular uncertain issues in the PSA (see, e.g., Ref. [33]). Correlation coefficients of dependent variables with respect to uncertain issues or phenomena can provide insights into their importance.

Sensitivity analysis

11.25. Sensitivity analysis is used to measure the extent to which results would change if alternative models, hypotheses or values of input parameters were selected (and thus provides an evaluation of uncertainty in respect of a particular issue or a particular group of related issues at a time). Sensitivity analysis, specific to parameters, events and/or phenomena may be used to supplement a more comprehensive uncertainty analysis. Sensitivity analysis is a useful tool to guide the selection of dominant sources of uncertainty. Example areas of uncertainty related to the progression of severe accidents are listed in Table 4.

11.26. If a sensitivity analysis is used as a surrogate for a comprehensive uncertainty analysis, metrics should be developed to indicate the influence of alternative models or parameter values on the results of the Level 2 PSA.

12. DOCUMENTATION OF LEVEL 2 PSA: PRESENTATION AND INTERPRETATION OF RESULTS

OBJECTIVES AND CONTENT OF DOCUMENTATION

12.1. Requirement 20 of GSR Part 4 (Rev. 1) [2] states that "**The results and findings of the safety assessment shall be documented.**" General recommendations related to the documentation of PSA results are provided in paras 3.17–3.19 of SSG-3 (Rev. 1) [4] and are not repeated here.

12.2. The results of internal reviews, audits and peer reviews related to the Level 2 PSA study should be documented and made available for consultation, either as part of the Level 2 PSA report or as part of internal reports.

12.3. The documentation of the Level 2 PSA report should provide sufficient information to satisfy the objectives of the study and to support the needs of the users of the Level 2 PSA.

12.4. To support maintaining a living PSA, in line with Requirements 12 and 24 of GSR Part 4 (Rev. 1) [2] (see para. 2.20), the documentation of the Level 2 PSA should also facilitate its subsequent refinement, updating and maintenance in the light of changes to plant configuration, technical advances in severe accident progression analysis, integration of new topics, use of improved models, broadening of the scope of the PSA in question and its use for alternative applications.

12.5. The documentation of the Level 2 PSA should explicitly present the assumptions, exclusions, limitations and features considered in the Level 2 PSA study for extending and interpreting the Level 2 PSA results, as this information is also of critical importance to users.

12.6. A fully auditable trail of calculations, including intermediate analyses, rationales for probabilistic estimates, assumptions and supporting calculations should be provided in the documentation, either as appendices or as internal reports. This is very important for reconstructing and updating each detail of the analysis in the future and for facilitating the independent review of the Level 2 PSA.

12.7. Conclusions should be distinct and should not only reflect the main general results but should emphasize the conclusions drawn from the analysis of uncertainties associated with phenomena, models and databases, and the supporting analyses. The effect of underlying assumptions, uncertainties and

conservatisms in the analyses and methods on the results of the Level 2 PSA should be demonstrated through the presentation of the results of sensitivity studies.

12.8. If screening criteria have been applied to eliminate accident sequences with low frequencies of occurrence from further analysis, for example, from the output of the Level 1 PSA or in the definition of PDSs, then an estimate of the contribution of the truncations should be made and should be presented with the final Level 2 PSA results.

12.9. The Level 2 PSA report should clearly document important findings of the Level 2 PSA, including the following:

(a) Plant specific design or operational vulnerabilities identified;
(b) Key operator actions for mitigating severe accidents;
(c) Potential benefits of various engineered safety features;
(d) Areas for possible improvement in operations or hardware for the plant and in particular for the containment;
(e) Significant accident scenarios contributing to a large release or any other applicable risk metrics (see paras 2.16–2.19).

12.10. The results of the PSA may be compared with probabilistic safety criteria for Level 2 PSA, if these have been set. Available probabilistic safety criteria and/or goals vary considerably among States, but the most common risk metrics for Level 2 PSA include criteria and/or goals for the frequency of a large early release and the maximum tolerable frequency of releases of various magnitudes (see paras 2.16–2.19 and Annex III). While the threshold for large early release frequency represents a point estimate frequency for a particular unacceptable release, the maximum tolerable frequency of releases of various magnitudes expands this concept across the full range of possible releases.

12.11. Some parts of the documentation may be intended for use within the operating organization, while other parts of the documentation may be intended for wider external use. Some users (e.g. the public) might primarily use the summary report of the PSA, while others (e.g. regulatory bodies) might use the full PSA documentation, including the computer model. The nature and the amount of information for inclusion in the documentation for external use compared with that intended for in-house support documentation should be established in accordance with the policies and process defined by the operating organization. This decision making process may include the PSA team and the project management team for the Level 2 PSA.

ORGANIZATION OF THE DOCUMENTATION

12.12. Recommendations on the organization and preparation of documentation for PSA are provided in paras 3.20–3.24 of SSG-3 (Rev. 1) [4] and are equally applicable to Level 2 PSA. This section provides specific recommendations on documenting the results and findings of a Level 2 PSA. The Level 2 PSA documentation should be divided into three major parts, namely:

(1) Summary report;
(2) Main report;
(3) Appendices to the main report.

12.13. The summary report should be designed to provide an overview of motivations, objectives, scope, assumptions, results and conclusions of the Level 2 PSA and potential impacts on plant design, operation and maintenance. The summary report is generally aimed at a wide audience of reactor safety specialists and should be adequate for high level review.

12.14. An outline of the main report should also be provided in the summary report to guide reviewers to sections where additional details and supporting analyses related to severe accident progression, human performance, equipment reliability and containment integrity analyses are included. The summary report should provide a comprehensive overview of the entire Level 2 PSA study. It should be independently reviewed by individual task leaders and/or analysts for correctness and consistency.

12.15. The summary report of a Level 2 PSA should include a subsection on the structure of the report, which should present concise descriptions of the contents of the sections of the main report and of the individual appendices. The relation between various parts of the Level 2 PSA should also be included in this subsection of the summary report.

12.16. The main report should give a clear and traceable presentation of the complete Level 2 PSA study, including the following:

(a) A description of the plant at the moment of the study;
(b) The objectives, scope and approach of the study;
(c) The methods and data used;
(d) The grouping of accident sequences from the Level 1 PSA, the PDSs considered as well as the screening criteria for the final set of PDSs;

(e) The assumptions and results related to severe accident progression regarding the modelling of phenomena, the containment strength analysis, the human and equipment reliability modelling and the accident progression event tree;
(f) The results and the conclusions of the Level 2 PSA study documenting the source term, the risk metrics, the uncertainties, and the sensitivity and importance analyses.

12.17. The main report, together with its appendices, should be designed to do the following:

(a) Support technical review of the Level 2 PSA;
(b) Communicate key detailed information to interested users;
(c) Permit the efficient and varied application of the Level 2 PSA models and results;
(d) Facilitate the updating of the models, data and results in order to support the continued safety management of the plant.

12.18. The appendices should contain detailed data, records of engineering computations and detailed models. The appendices should be structured so as to correspond directly to the sections and subsections of the main Level 2 PSA report, as far as possible.

12.19. Reporting of the rationale and analyses employed for a Level 2 PSA should include information on the methods used, the PSA process, and the insights and conclusions drawn, presented in a logical manner. The report should be compiled in such a way that it facilitates review activities, including peer review, and provides a structured entry route to detailed supporting material.

12.20. Sample outlines of the contents of the summary report and main report for a Level 2 PSA are provided in Annex II.

COMMUNICATION OF RESULTS

12.21. Paragraph 5.9 of GSR Part 4 (Rev. 1) [2] states:

> "Consideration shall also be given to ways in which results and insights from the safety assessment may best be communicated to a wide range of interested parties, including the designers, the operating organization, the regulatory body and other professionals. Communication of the results from the safety assessment to interested parties shall be commensurate

with the possible radiation risks arising from the facility or activity and the complexity of the models and tools used."

12.22. For communication of the Level 2 PSA results, the target audience and the amount of information and data provided should be taken into consideration. The various reports listed in para. 12.12 are suitable sources of information that could be communicated, depending on the target audience.

12.23. As reports on Level 2 PSA might contain sensitive information, security considerations should be taken into account for the communication of such reports. Guidance on security aspects related to information classification, sharing and disclosure is provided in IAEA Nuclear Security Series No. 23-G, Security of Nuclear Information [67].

13. LEVEL 2 PSA FOR A SPENT FUEL POOL

13.1. Interest in the risk related to accidents occurring in the SFP increased after the accident at the Fukushima Daiichi nuclear power plant [68, 69]. A Level 2 PSA for the SFP might not be necessary, given the generally low frequency, the long time frames for proceeding to fuel damage and the potential limited mitigation capabilities once fuel damage has occurred in the SFP. Depending on the pool's location and plant design specifics, available capabilities to prevent off-site releases in case of fuel damage at a SFP could be limited. In particular, para. 6.68 of SSR-2/1 (Rev. 1) [3] states (footnote omitted):

> "For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site."

13.2. The recommendations provided in this section are focused on the development of Level 2 PSA when the SFP is located inside a building capable of ensuring the confinement function in severe accident conditions (see para. 6.20). If this is not the case, the practice during Level 2 PSA has been to consider that accidents involving damage of fuel stored in the SFP lead directly to large radioactive releases. In addition, an analysis can be performed with the objective

of substantiating the capabilities for crediting some fission product retention in buildings or water sources in severe accident conditions.

13.3. In principle, the Level 2 PSA for the SFP is based on the same methodology as the Level 2 PSA for the reactor outlined in Sections 5–11. Accordingly, the general process for conducting a Level 2 PSA for the reactor can be adapted for the SFP, with additional consideration of the specific aspects addressed in this section.

13.4. The goal of performing a Level 2 PSA for the SFP should be clearly defined; usually this goal is similar to the goal of a Level 2 PSA for the reactor. A Level 2 PSA for the SFP can be performed independently or in combination with a Level 2 PSA for the reactor, depending on the specific needs and applications for developing the Level 2 PSA. Further, the definition of the undesired end states in the Level 1 PSA and the location of the SFP (e.g. inside the reactor containment, outside the reactor containment but inside the reactor building, outside the reactor building) can determine the analysis needs for a Level 2 PSA for the SFP. For example, the location of the SFP determines whether an accident progression event tree needs be developed or whether other factors that could reduce the source term could be taken into consideration. The source term could be reduced by the possibility of closing the reactor containment and the availability of the ventilation system and SFP cooling system.

INTERFACE WITH LEVEL 1 PSA FOR A SPENT FUEL POOL

13.5. As when performing a Level 2 PSA for a reactor, PDSs specific to the SFP can be considered in the development of a Level 2 PSA for the SFP. Factors specific to SFP analysis include items that influence the accident progression and source term, such as: time since last core unloading, the fuel loading in the pool (e.g. number of fuel assemblies, fuel burnup, fuel loading pattern), pool configuration (i.e. whether the SFP is isolated from or interconnected to the reactor or to the SFPs in other unit(s)), water inventory of the pool and water leak rate.

13.6. The undesired end states (e.g. uncovering of fuel stored or handled in the SFP, boiling of the pool water) defined in Level 1 PSA for the SFP, as described in paras 10.2–10.6 of SSG-3 (Rev. 1) [4], should also be addressed in the Level 2 PSA.

13.7. If the Level 2 PSA for the SFP and the Level 2 PSA for the reactor are integrated in the same model, the PDSs should include the status of both the reactor and the SFP for events that can affect both. Reactor accident sequences might impact the SFP; for example, reactor containment venting might accelerate the

boiling of the water in the SFP if the SFP is located inside the reactor containment. In addition, reactor accident sequences that do not result in Level 1 reactor core damage events might impact the mitigatory actions for the SFP accidents and therefore might need to be considered for inclusion in the PDSs.

SEVERE ACCIDENT PROGRESSION ANALYSIS OF FUEL STORED IN THE SPENT FUEL POOL

13.8. To support Level 2 PSA for an SFP (if necessary; see para. 13.2), deterministic analyses should be performed to analyse severe accident progression in the SFP using one or more computer codes capable of modelling the accident progression and severe accident phenomena in the SFP. Severe accident phenomena to be considered in this analysis include heat transfer within the pool and to the fuel racks and surrounding walls (and to the ground, for SFPs below ground level), fuel and cladding behaviour (e.g. fuel burnup, decay heat, cladding behaviour), fuel assembly and rack degradation including interactions (e.g. zirconium clad and water reaction, hydrogen generation, zirconium fire, corium–concrete interaction), and fission product transport. The calculations made should provide information on the fraction of the fuel assemblies that would be damaged, depending on the fuel assemblies' arrangement, burnup and storage time in the SFP.

13.9. The boundary conditions should be defined in accordance with the PDS. In particular, care should be taken in defining in the calculations the amount of fuel that is normally replaced during a refuelling outage and during a full core unload (if a full core unload is prescribed by the operating procedures and is within the scope of the plant operational states included in the PSA).

13.10. Severe accident management measures for prevention of fuel damage in the SFP as well as mitigation, such as reflooding of the SFP or the use of mitigative spray, should be considered in the severe accident progression analysis.

13.11. The following accident progression parameters should be considered for the SFP:

(a) Time to boiling;
(b) Time to fuel uncovering;
(c) Time to fuel damage;
(d) Time to spent fuel structure breach;
(e) Time to penetration of concrete around the SFP (if credited);

(f) Source term magnitude and timing.

13.12. Depending on the plant configuration (e.g. SFP inside the reactor containment, SFP outside the reactor containment but inside the reactor building, SFP outside the reactor building), severe accident progression analysis should consider the interactions between the reactor and the SFP, as a reactor accident might have an impact on or induce an SFP accident and vice versa. From this analysis, some additional accident scenarios (involving both the reactor and the SFP) could be incorporated into the Level 2 PSA if not already considered in the Level 1 PSA, such as the following:

(a) Common events that have an impact on the reactor and the SFP simultaneously (e.g. station blackout) or consequential failure during accident progression, which affects the safety functions for the other source;
(b) Impact of the accident management strategies for the reactor on the SFP (e.g. if the SFP is located inside the reactor containment, actuation of the filtered containment venting system leads to more intensive water boiling in the SFP);
(c) Hydrogen release that could result in deflagration or detonation events that lead to the failure of structures or electrical and/or mechanical equipment;
(d) Fission product releases that inhibit or preclude access to the areas needed for local manual actions;
(e) Effects of heat radiation from the damaged SFP on the containment structure or steel liner;
(f) Subsequent independent failure of installed equipment that ensures safety functions for both the reactor and the SFP (e.g. residual heat removal), which might force the operators to decide where to prioritize the use of any remaining non-permanent equipment.

13.13. If the SFP is located inside the reactor containment, accident progression analysis should address the impact of a combined reactor and SFP accident on conditions inside the reactor containment (e.g. pressure, temperature, corium spreading, inflammable gas).

13.14. Hydrogen generation should be considered for loss of heat removal scenarios that cause the evaporation of a large amount of water from the SFP. For SFPs located inside the containment, the capacity of the existing severe accident management arrangements for the reactor core (e.g. passive autocatalytic recombiners, filtered containment venting systems) should be verified to ensure that they are sufficient to cope with the hydrogen additionally generated from the damaged fuel stored in the SFP.

13.15. Generated hydrogen might leak into rooms adjacent or connected to the SFP. The retention of hydrogen in the rooms on the release path, which might lead to combustion and damage to the rooms, should be justified.

13.16. For fuel utilizing zirconium as fuel cladding, the risk of zirconium fire and its propagation under dry conditions in the SFP should be considered.

13.17. In general, SFP criticality is not likely owing to the amount of fissile material in the SFP, as well as its geometrical configuration and the presence of neutron absorbing material. Nevertheless, the issues of criticality in SFP should be addressed in the Level 2 PSA documentation.

13.18. For an SFP located inside a building capable of ensuring the confinement function, the overpressure created by steam and heat should be considered when crediting retention of fission products inside the building.

ANALYSIS OF ACCIDENTS DURING FUEL TRANSFER OPERATIONS BETWEEN THE REACTOR AND THE SPENT FUEL POOL

13.19. The consequences of accidents during fuel transfer operations between the reactor and the SFP, if not already screened out, should be considered for dedicated analysis in the Level 2 PSA. Typical accidents to be considered are related to fuel uncovering due to the loss of the spent fuel cooling system (e.g. as a result of a station blackout or an external hazard such as a seismic event).

ACCIDENT PROGRESSION EVENT TREE FOR A SPENT FUEL POOL

13.20. If the SFP is located inside the containment, the accident progression event tree can be developed similarly as for the reactor Level 2 PSA (see Section 9). Nevertheless, the analyst should introduce the dependencies between the reactor and the SFP (e.g. systems, human actions, containment response) in the accident progression event tree. In comparison with a stand-alone Level 1 PSA for an SFP, some additional scenarios of SFP accidents induced by a reactor core melt accident may be introduced. For an SFP outside the containment, the accident progression event trees for the Level 2 PSA can be simple event trees, depending on the extent to which mitigation strategies are credited.

13.21. The harsh environments (e.g. high temperature, humidity and radiation levels) expected to be present around the SFP should be carefully considered

when crediting local operator actions in the Level 2 PSA. The effect of these harsh environments on the survivability of mitigative equipment such as hoses, on the fittings and nozzles present around the SFP and on the availability of the instrumentation system (e.g. SFP water level) should also be considered in the Level 2 PSA.

SOURCE TERM AND RELEASE CATEGORIES FOR A SPENT FUEL POOL

13.22. The source term calculations for an SFP Level 2 PSA can be performed similarly to those performed for the reactor Level 2 PSA (see Section 10).

13.23. Release paths from the SFP to the environment that are to be considered in the Level 2 PSA depend on the location of the pool. If the SFP is located inside the reactor containment, the potential release paths to the environment are almost the same as for reactor core melt accidents. However, additional release paths should also be reviewed (e.g. after penetrating the concrete wall or bottom of the SFP, the molten debris could penetrate the containment wall, leading to another type of late containment failure). If the SFP is located outside the reactor containment, the potential release paths to the environment depend on plant specific designs, such as ventilation systems, building doors, roof under thermal impact, and size of rooms on the path.

13.24. In the case of a severe accident in the reactor concurrent with significant fuel damage in the SFP, the source term evaluation should consider the different timing of radioactive releases from those two sources.

13.25. The deposition of aerosols in the buildings in the release paths could mitigate the environmental impact and should be considered.

13.26. A dedicated source term analysis should be performed for the SFP and for accidents involving the transfer of fuel assemblies between the reactor and the SFP on the basis of the burnup of the fuel elements. The fuel inventory of the SFP at potential accident times should be analysed considering the history of refuelling and the subsequent mixture of newer and older fuel elements.

13.27. Similarly to in the reactor Level 2 PSA, release categories can be defined for the SFP in order to group similar accident sequences based on the magnitude and timing of the release and to calculate an associated frequency of release.

QUANTIFICATION OF EVENT TREES AND ANALYSIS OF RESULTS FOR A SPENT FUEL POOL

13.28. The recommendations provided in Section 11 are also applicable to the SFP Level 2 PSA. In addition, the Level 2 PSA models for the reactor and for the SFP should be integrated so as to correctly model the dependencies of shared systems. This is particularly important for the initiating events affecting both the reactor and SFP simultaneously, and for a further Level 2 PSA study (in particular for plants with the SFP inside the containment).

14. LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.1. Paragraphs 14.2–14.29 provide recommendations on the development of Level 2 PSA for sites where multiple units are located, suitable for use where national regulatory requirements compel such studies. Given the complexity of models and the high level of uncertainty, the development of such Level 2 PSAs is not yet common practice among States. However, it can be useful to capture risks relevant to the whole site as well as dependencies among units from the Level 2 PSA perspective if these risks and dependencies were not already addressed in the development of the PSA model for each single unit. Therefore, the recommendations in this section are intended to harmonize the development of such studies among States. More information on Member States' experience, practical case studies and guidance on PSA for multi-unit nuclear power plants are provided in Refs [70, 71].

OBJECTIVE OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.2. The objective of the development of Level 2 PSA for multi-unit nuclear power plants is to complement the Level 2 PSAs for each unit on the site, with regard to topics that might have not been fully addressed, such as equipment and systems that are shared or common to multiple units.

SCOPE OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.3. In developing the Level 2 PSA for multi-unit nuclear power plants, the analyst should identify a list of topics that have not been fully addressed in the single unit Level 2 PSA, such as the following:

(a) Hazards affecting the whole site, especially the implications for the severe accident mitigation system, equipment reliability and for human resources;
(b) Correlated or shared SSCs and resources among different units;
(c) Plant operational states of each unit on the same site (see Section 5 for considerations of operational states other than full power);
(d) Impact on the other units of consequences induced by a unit with a severe accident (e.g. fuel melt accidents happening in another unit).

The selection of topics should not lead to excessive complexity in the development of the Level 2 PSA for multi-unit nuclear power plants.

14.4. The analyst should justify whether or not those topics are introduced in the multi-unit Level 2 PSA on the basis of a screening process. A Level 2 PSA for a multi-unit nuclear power plant should consider the reactor units (e.g. power and/or research reactors) and the SFPs on the site. When a single reactor unit includes multiple reactor modules, the Level 2 PSA for that reactor unit should address interactions and dependencies among the reactor modules in the same way that reactor units are addressed in the Level 2 PSA for a multi-unit nuclear power plant. For the purpose of this Safety Guide, reactor modules are understood as nuclear reactors that are in the same reactor unit, in accordance with Ref. [72].

PREREQUISITES OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.5. The recommendations provided in paras 4.11–4.17 related to plant familiarization for the development of a single unit Level 2 PSA are also applicable for the development of the Level 2 PSA for multi-unit nuclear power plants.

14.6. The availability of a multi-unit Level 1 PSA model for a multi-unit site is a prerequisite for the development of Level 2 PSA for a multi-unit site.

RISK METRICS FOR LEVEL 2 PSA FOR A MULTI-UNIT NUCLEAR POWER PLANT

14.7. The risk metrics used in PSA for a single unit site (e.g. large release frequency) should be used to the greatest extent possible to express the risk profile in the context of multi-unit nuclear power plants for corresponding decision making (see paras 2.17–2.19). These risk metrics could be adapted to specific multi-unit risk metrics, for example the conditional probability of large releases from several reactors, on the basis of the large releases from one reactor of a unit on a multi-unit site.

INTERFACE BETWEEN LEVEL 1 PSA AND LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.8. The Level 2 PSA for a multi-unit nuclear power plant begins when one unit is affected by fuel damage. The Level 1–Level 2 PSA interface should transfer the information on those units considered into the Level 2 PSA.

14.9. In principle, the PDS methodology for a single unit PSA, as described in Section 5, can be applied to PSAs for multi-unit nuclear power plants. The attributes for PDSs have to be adapted so as to represent all units and to limit the complexity of the model. The extent of the adaptation of the PDSs should depend on the identified topics of interest.

14.10. PDSs for accidents involving multiple units should group accident sequences coming from a Level 1 PSA for multi-unit nuclear power plants that are equivalent in terms of the risk of release considering the parallel evolution of all units on site (e.g. kinetics of accident progression, availability of mitigating systems).

ACCIDENT PROGRESSION AND CONTAINMENT ANALYSIS IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.11. Existing accident progression and containment analysis studies for single unit Level 2 PSAs (as described in Section 6) should, as far as possible, be used as the basis for such analysis in the context of a Level 2 PSA for a multi-unit nuclear power plant. Additional accident progression analyses may be needed, depending on the differences in reactor technologies and designs on the site and the identified topics of interest.

14.12. The same general techniques and tools can be applied to perform the multi-unit accident progression analysis, with due consideration of the availability of mitigating systems, shared systems and the ability of operators to perform actions.

14.13. The major cause of differences from analysis for single units is the potential for correlated phenomenological factors to influence the accident progression. These factors include, but are not limited to, the various severe accident phenomena discussed in Sections 6–8. The way in which these factors combine in an event involving multiple units could influence the timing and magnitude of the generation of volatile fission products, containment failures and releases from containments.

HUMAN AND EQUIPMENT RELIABILITY ANALYSIS IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.14. The recommendations provided in Section 8 should be considered applicable for Level 2 PSA for multi-unit nuclear power plants with regard to human and equipment reliability analysis, since the operator actions undertaken and credited within single unit PSA still apply to a multi-unit assessment. Specific aspects of Level 2 PSA for multiple units (e.g. recovery post-core damage, ensuring containment performance) necessitate the consideration of the state of other units on the site in terms of available resources and feasibility of specific Level 2 actions.

14.15. For plants with multiple units, further to the interactions between the units (both positive and negative, from a risk point of view) considered in Level 1 PSA from the perspective of the unit under consideration, the shared systems between units and the degree of shared responsibility in the operator response to accidents should be considered in the Level 2 PSA.

14.16. The dominant consideration for human reliability analysis purposes is whether the broader organization of the site depends on common human teams to provide some of the responses necessary in accident mitigation (e.g. a single common fire brigade for the whole site). In such instances, the ability of those responses to occur in parallel on multiple units should be considered.

14.17. Sites with multiple units of the same design are more likely to have common facilities, such as a common main control room, power systems, switchyards and cooling water intakes. Multiple units with different designs

might also share systems and teams, however. In the event of concurrent events at multiple units, the impact on operator actions might have implications for Level 1 PSA for multi-unit sites that should be considered for Level 2 PSA for multi-unit sites.

ACCIDENT PROGRESSION EVENT TREE FOR LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.18. The development of an accident progression event tree for Level 2 PSA for multi-unit nuclear power plants should be based on the recommendations provided for single unit plants in Section 9.

14.19. The degree of interconnectedness between units should be taken into account. Consideration should also be given to the sharing of SSCs between units as well as the potential consequences of one unit on the undisturbed operation or mitigation of accident conditions on another unit.

14.20. Since the number of units can add significant complexity and size to the accident progression event tree, consideration should be given to simplifying the single unit models before combining them for multi-unit plants, with account taken of major risk contributors. Since each Level 1 sequence results in multiple Level 2 sequences by definition, it is prudent to simplify them where possible. Methods to simplify the modelling could include the justified removal of low risk initiating event contributors, a focus on initiating events that could affect several units at the site at the same time (e.g. total loss of external power supply, total loss of ultimate heat sink, external flooding, earthquake) and the grouping of similar Level 1 sequences under a single PDS and/or grouping release categories to capture the generic representation of an accident sequence.[30]

SOURCE TERM AND RELEASE CATEGORIES IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.21. The recommendations provided in Section 10 on source term analysis in single unit Level 2 PSA should also be considered for source term analysis in Level 2 PSA for multi-unit nuclear power plants, since they generally apply the same assessment of source terms and the resulting release category assignments.

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[30] For example, for a three unit site, grouping the end states into those for one, two and three units resulting in releases, as opposed to modelling each possible combination of units.

Much of the existing framework for release categories may be used as is, with consideration given to where interconnectedness and dependencies for multi-unit Level 2 PSA may arise (as captured in the interface with multi-unit Level 1 PSA).

14.22. In the analysis of source terms and release categories in Level 2 PSA for multi-unit nuclear power plants, account should be taken of small and early releases from different sources at the site, as these releases might become relevant when aggregated for the calculation of the large release or large early release risk metrics for the site.

14.23. It may be desirable to simplify the release categories once multiple units are considered, to avoid an exponential increase in the potential combinations with limited benefits in terms of obtaining risk insights from the multi-unit Level 2 PSA.

14.24. It may be justifiable to further reduce the total number of release categories for multi-unit Level 2 PSA by considering the relative contribution of the individual release category on a specific basis, such as release magnitude, classes of release category in terms of their activity (i.e. Bq), frequency and/or timing. For example, if a particular binary combination is overwhelmingly driven by the contribution of one release category, engineering judgement may be applied to consider if an additional level of detail is needed for the purposes of Level 2 PSA for multi-unit nuclear power plants in comparison to a single unit analysis.

14.25. The addition of SFP accidents or shutdown states to the scope of a Level 2 PSA for multi-unit nuclear power plants might also further complicate the source term analysis for multi-unit PSA due to the further increase in combinations of outcomes (even beyond those discussed in para. 13.28). By similar reasoning, screening and simplification of those outage states and radioactive sources can be applied.

14.26. The aggregation and simplified combination of release categories for various nuclear installations on a given site can be further assessed on the basis of the commonality or differences in the definitions of releases from various nuclear installations. While a reactor and SFP may have differing characteristics of core or fuel damage, it is easier to define releases from either nuclear installation as exceeding a defined release magnitude and facilitate easier grouping (binning) of contributors to a single release category.

QUANTIFICATION OF EVENT TREES AND ANALYSIS OF RESULTS IN LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.27. The integration and quantification process for Level 2 PSA for multi-unit nuclear power plants should be based on the approach used in the single unit Level 2 PSA (see Section 11). In the case of coupling PSA models from different units into a single PSA model, the major concern is the additional complexity resulting from the additional event tree end states, release categories and combinations. Quantification is expected to involve additional consolidation and screening to include a manageable set of inputs for Level 2 scenarios that take into account the effect of multiple units undergoing Level 1 and Level 2 aspects. Depending on the topics of interest identified (see para. 14.3), an approach based on the post-processing of the single unit Level 2 PSA results could be sufficient to obtain relevant insights.

14.28. The treatment of sensitivity and uncertainties does not bear any major methodological differences to Level 2 PSA for single unit nuclear power plants (see paras 11.24–11.26), but it should be expected that a manageable addition for specific multi-unit impacts via sensitivity cases for phenomenology and modelling uncertainties will be necessary to appropriately characterize the state of knowledge for multi-unit Level 2 PSA.

DOCUMENTATION OF LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS

14.29. The recommendations provided in Section 12 should be considered applicable to a Level 2 PSA for multi-unit nuclear power plants, with no additional areas that need to be addressed. However, additional details and discussion regarding the characteristics of releases, the phenomenology that might arise due to accidents involving multiple units at a nuclear power plant, and the resulting sensitivity and uncertainty in the source terms and release categorization should be well documented.

15. USE AND APPLICATIONS OF LEVEL 2 PSA

15.1. This section provides recommendations on meeting Requirement 23 of GSR Part 4 (Rev. 1) [2] on use of the safety assessment for Level 2 PSA. PSA has

been applied in the design and operation of nuclear power plants in many States to complement results obtained by traditional methods of safety assessment. Many PSA applications use the results of Level 1 PSA (see SSG-3 (Rev. 1) [4]), often in combination with Level 2 PSA results. The following list includes some examples of applications of Level 2 PSA (these applications are not in use in every State):

(a) Comparison of results of the Level 2 PSA with probabilistic goals or criteria to determine if the overall level of safety of the plant is adequate;
(b) Evaluation of plant design to identify potential vulnerabilities in the mitigation of severe accidents;
(c) Development of severe accident management guidelines that can be applied following core damage;
(d) Use of the source terms to provide an input into the development of emergency preparedness and response arrangements;
(e) Use of the source terms and frequencies to determine off-site consequences (Level 3 PSA);
(f) Prioritization of research relating to severe accident issues;
(g) Development of a list of severe accident scenarios to be addressed in the nuclear power plant design.

SCOPE AND LEVEL OF DETAIL OF LEVEL 2 PSA FOR VARIOUS USES AND APPLICATIONS

15.2. The scope and the level of detail of the Level 2 PSA should be consistent with its intended uses or applications. For example, the scope and the level of detail of a Level 2 PSA intended to provide an estimate of the large release frequency or the large early release frequency and to provide insights into the potential failure modes of the containment is different from the scope of a Level 2 PSA intended to provide an input to emergency preparedness and response or to a Level 3 PSA. In the calculation of large release frequencies or large early release frequencies, there is a need to identify accident sequences and their frequencies where the release would be categorized as 'large'. However, for the purposes of emergency preparedness and response, the release characteristics (including the associated source terms) and, to a limited extent, the frequencies associated with the occurrence of such releases, need to be specified more accurately. For Level 3 PSA development, both the source terms and their frequencies need to be specified more accurately. In addition, the level of detail of the PSA needs to be greater if the Level 2 PSA model is intended for use in a risk monitor. Recommendations on the risk monitor application in relation to PSA are presented in SSG-3 (Rev. 1) [4].

These recommendations are valid for the SSCs added in the Level 2 PSA model and are therefore not repeated in this Safety Guide.

15.3. The scope of the Level 2 PSA should be commensurate with its intended uses and applications (see also para. 2.7) and is based on the scope of the Level 1 PSA. A full scope Level 2 PSA is most suitable for a large number of uses and applications, with due consideration given to the uncertainties in key parameters and limited strength of knowledge on some data and assumptions that could impact the PSA results and insights. A full scope Level 2 PSA can be developed only on the basis of a full scope Level 1 PSA, as defined in para. 2.2 of SSG-3 (Rev. 1) [4].

15.4. If risk insights are to be derived from a Level 2 PSA that is not full scope (e.g. not all initiating events and hazards considered), this should be recognized when applying the insights from the PSA.

15.5. A full scope PSA ensures that the insights from the PSA relating to the risk significance of accident sequences, SSCs, human errors and common cause failures are derived from a comprehensive, integrated model of the nuclear power plant. If the Level 2 PSA is based on a Level 1 PSA that has a more limited scope or details, these limitations need to be taken into account in the application of the Level 2 PSA.

USE OF LEVEL 2 PSA THROUGHOUT THE LIFETIME OF THE NUCLEAR POWER PLANT

15.6. Level 2 PSAs should be actively maintained and periodically updated, taking into account changes in plant design and operational practices as well as feedback from experience and advances in technology that might compromise the validity of the PSA (see also paras 2.20–2.23). This updating should take account of changes in the provisions made and the guidance provided for severe accident management, updates to the severe accident progression analysis performed to support the Level 2 PSA model, and research results that provide a better understanding of the phenomena that occur during a severe accident.

15.7. The Level 2 PSA should provide one of the inputs into design evaluation throughout the lifetime of a nuclear power plant. It should be used during the design process for a new plant to determine whether adequate features for the mitigation of severe accidents are being incorporated into the design and should be updated throughout the construction and operational stages of the lifetime of the plant.

15.8. The Level 2 PSA should also provide an input into the development of the severe accident management guidelines, which should be available when the plant goes into operation.

RISK INFORMED APPROACH TO LEVEL 2 PSA

15.9. The aim of applying a risk informed approach is to ensure that a balanced approach is taken when making decisions on safety issues by considering probabilistic risk insights with any other relevant factors in an integrated manner [73, 74].

15.10. In any of the applications of Level 2 PSA, the insights from the PSA should be used as part of the process of risk informed decision making on issues related to the prevention and mitigation of severe accidents at the plant, taking into account all relevant factors, such as:

(a) Any mandatory requirements that relate to the PSA application under consideration (e.g. legal requirements or regulations);
(b) The insights from deterministic safety analysis (e.g. whether the provisions of defence in depth requirements are being met; whether there are adequate safety margins; whether lower level requirements such as the provision of sufficient levels of redundancy and diversity in the SSCs that perform safety functions are being met; whether the equipment in the plant has been qualified to a sufficient level that it can withstand the harsh environmental conditions that follow initiating events);
(c) Any other applicable insights or information (e.g. a cost–benefit analysis, details of the remaining lifetime of the plant, inspection findings, operating experience, doses to workers that would arise if necessary changes were made to the plant hardware, environmental protection concerns).

COMPARISON OF LEVEL 2 PSA WITH PROBABILISTIC SAFETY GOALS OR CRITERIA

15.11. The overall results of the Level 2 PSA should be compared with the probabilistic safety goals or criteria (if these have been specified; see paras 2.17–2.19). The aim should be to determine whether the risk criteria or goals have been met or whether additional features for prevention or mitigation of accidents need to be provided.

15.12. This comparison should take account of the results of the sensitivity analyses that have been performed and the uncertainties inherent in the Level 2 PSA. The sensitivity analyses and the uncertainty analyses should be used to indicate the degree of confidence in meeting the criterion or target and the likelihood that it will be exceeded.

15.13. Probabilistic criteria for a large off-site release of radioactive material needing a short term off-site response were proposed in Ref. [10].[31] Several States have also set similar numerical values, which have generally been defined as objectives or targets (see Annex III).

15.14. In addition, for future nuclear power plants, rather than defining probabilistic criteria, Ref. [10] states that the objective should be as follows:

> "[T]he practical elimination of accident sequences that could lead to large early radioactive releases, whereas severe accidents that could imply late containment failure would be considered in the design process with realistic assumptions and best estimate analysis so that their consequences would necessitate only protective measures limited in area and in time."

LEVEL 2 PSA FOR DESIGN EVALUATION

15.15. The Level 2 PSA should be used to perform a safety evaluation of the plant design. The aim should be to gain insights into how severe accidents progress, identify plant specific vulnerabilities and provide an input into the consideration of whether improvements need to be made to the design of the plant.

Identification of plant vulnerabilities

15.16. The use of Level 2 PSA for design evaluation is very similar to that for Level 1 PSA, as described in paras 12.19–12.46 of SSG-3 (Rev. 1) [4]. As well as calculating the overall value of the large release frequency or large early release frequency, the computer codes used to develop the Level 2 PSA model and to quantify it provide a range of other information including the following:

(a) The frequency of each of the release categories;

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[31] According to Ref. [10], the objective for large off-site releases needing short term off-site response is 1×10^{-5} per reactor-year for existing plants.

(b) The possible combinations of failures (cutsets) that contribute to each of the release categories;
(c) The importance functions for systems, components and other basic events included in the PSA model (this depends on the computer code used for the development of the Level 2 PSA but could include the Fussell–Vesely importance, the risk achievement worth, the risk reduction worth and the Birnbaum importance).

15.17. The information provided by the Level 2 PSA should be used to identify weaknesses in the features provided for the prevention and mitigation of severe accidents. This information could include the following:

(a) The significant failure modes of the primary circuit and the containment;
(b) The dominant phenomena that lead to (early or late) containment failure;
(c) The SSCs that have the highest importance for large release frequency or large early release frequency.

15.18. Consideration should be given to making improvements to the safety features provided for the prevention or mitigation of severe accidents in order to reduce contributions to the overall risk of sequences with the highest risk significance. The improvements considered should include the provision of additional protective systems and features for mitigating the consequences of a severe accident. This could involve incorporating such additional protective systems and features into a new design or backfitting them into an existing plant.

15.19. The results of the Level 2 PSA should be used as a resource for determining whether adequate provisions for defence in depth have been made. For example, the PSA could provide a basis for determining whether severe accident management measures and guidelines fully address the fourth level of defence in depth as defined in para. 2.13 of SSR-2/1 (Rev. 1) [3].

Comparison of design options

15.20. When design improvements are being considered with regard to severe accident management measures, a range of options are often available. The Level 2 PSA may be used to provide an input into the comparison of these options in accordance with paras 2.24–2.36.

15.21. The Level 2 PSA should be used to compare the benefits in terms of risk reduction and balance of the design from the incorporation of these additional systems and features. The way that this comparison is done depends on the

complexity of the design options being considered but could range from producing a revised PSA model to post-processing the cutsets to take account of simpler changes, or even to performing sensitivity studies that relate to the design options. In doing this comparison, it needs to be recognized that a design change might impact a whole sequence of events modelled in the accident progression event tree, or even change the basis for evaluation of some nodes of the accident progression event tree. A design change might also affect the Level 1 PSA. Competing impacts should be recognized and taken into account in the evaluation of the design change. As an example, a modification to the spray system may benefit the control of steam pressurization but might lead to combustible conditions in some time frames or even lead to concerns about containment underpressure.

USE OF LEVEL 2 PSA IN THE DEVELOPMENT OF SEVERE ACCIDENT MANAGEMENT GUIDELINES

15.22. The Level 2 PSA should be used as a basis for evaluating the measures in place and the actions that can be taken to mitigate the effects of a severe accident after core damage has occurred. The aim of such measures and actions should be to arrest the progression of the severe accident or mitigate its consequences by preventing the accident from leading to failure of the containment or the reactor pressure vessel (for in-vessel melt retention strategy) and controlling the transport and release of radioactive material with the aim of minimizing off-site consequences. Examples of mitigatory actions that could be taken for pressurized water reactors include the following:

(a) Opening the pressurizer relief valves in order to reduce the reactor coolant system pressure and thus avoid molten core material being ejected from the reactor pressure vessel under high pressure;
(b) Adding water to the containment by any available means after the molten core has exited from the reactor coolant system so as to provide a cooling mechanism.

15.23. The results of the Level 2 PSA should be used to determine the effectiveness of the severe accident management measures that are described in the severe accident management guidelines or procedures, regardless of whether they have been specified using the Level 2 PSA or by another method.

15.24. In developing a Level 2 PSA, the uncertainties associated with the probability of phenomena occurring in the course of a severe accident should be recognized. The interrelations between these phenomena should also be noted, as

an accident management measure aimed at mitigating a particular phenomenon might make another phenomenon more likely. Examples of such interrelations for pressurized water reactors include the following:

(a) Depressurization of the primary circuit may prevent high pressure melt ejection but might increase the probability of an in-vessel steam explosion;
(b) Introducing water into the containment may provide a cooling medium for molten core material after it has come out of the reactor pressure vessel but might increase the probability of an ex-vessel steam explosion;
(c) Operation of the containment sprays may provide a means of removing heat and radioactive material from the containment atmosphere but might increase the flammability of the containment atmosphere by condensing steam.

15.25. These interdependencies between the various phenomena that can occur during a severe accident should be identified using the Level 2 PSA and should be taken into account in the development of the severe accident management guidelines. Updates of the Level 2 PSA and updates of the severe accident management guidelines should be performed in an iterative manner to facilitate the progressive optimization of the severe accident management guidelines.

PRIORITIZATION OF RESEARCH ACTIVITIES ON SEVERE ACCIDENTS

15.26. Level 2 PSA models the complicated and highly interrelated phenomena that occur after a severe accident. Although there has been a considerable amount of research into these phenomena, there is still a lack of knowledge in some areas, which leads to a significant level of uncertainty in the predictions of the Level 2 PSA.

15.27. The Level 2 PSA should provide a basis for the identification and prioritization of research activities. Such research activities should focus on identifying areas of research that can contribute most to improving knowledge in order to reduce the uncertainty in the highest risk significance parameters or phenomena.

INPUT FOR LEVEL 3 PSA

15.28. The source terms and frequencies derived in the Level 2 PSA can be used as the starting point for determining the off-site consequences that can result from a radioactive release from the plant. Such off-site consequences include health effects to members of the public and a range of consequences, including contamination of land, water and food, evacuation, and permanent relocation.

15.29. If a Level 2 PSA is interfaced with a Level 3 PSA, consideration should be given to revisiting the release category definitions to ensure that all of the information needed for the Level 3 PSA is available in the Level 2 PSA end states.

15.30. The source terms and frequencies derived in the Level 2 PSA should be used as the starting point for the Level 3 PSA performed to address the off-site consequences that could arise from a severe accident at the plant. The scope of the Level 2 PSA to be used for this purpose should include a detailed model of the dispersion of radioactive material and its release from the plant.

EMERGENCY PREPAREDNESS AND RESPONSE

15.31. The source terms and, to a limited extent, their frequencies derived in the Level 2 PSA, along with projections of the off-site dose as a function of distance, may be used as inputs into the development of off-site emergency preparedness and response arrangements. One or more reference accidents can be defined and used in this process.

15.32. For a Level 2 PSA that is to be used for emergency preparedness and response, the releases considered should be accurately specified in terms of isotopic composition, amount and timing of radioactive material released (i.e. source terms), as well as in terms of relevant additional attributes (see Table 6).

15.33. Requirement 4 of IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [75] states that "**The government shall ensure that a hazard assessment is performed to provide a basis for a graded approach in preparedness and response for a nuclear or radiological emergency.**" With a view to meeting this requirement and Requirement 23 of GSR Part 4 (Rev. 1) [2] on the use of the safety assessment as part of the integrated risk informed approach, the source terms and, to a limited extent, their frequencies derived in the Level 2 PSA can be used as an input to determine the emergency planning zones and emergency planning distances.

OTHER PSA APPLICATIONS

15.34. The Level 2 PSA should be used in combination with the Level 1 PSA results for a number of applications, as described in SSG-3 (Rev. 1) [4] for the Level 1 PSA. The use of Level 1 and Level 2 PSAs in combination provides additional insights to those obtained solely from the Level 1 PSA, since the relative importance of SSCs is normally different for Level 2 PSA results (e.g. large release frequency, large early release frequency) than for Level 1 PSA results (e.g. core damage frequency).

Appendix

CONSIDERATIONS FOR HUMAN RELIABILITY ANALYSIS IN A LEVEL 2 PSA

A.1. Significant differences in context exist for nuclear power plant operators performing tasks before and after core damage to protect the plant and the public. These differences significantly impact the factors to be considered when performing human reliability analysis for a Level 2 PSA. Example factors include command and control, coordination between the organizations involved in responding to the event, training of the nuclear power plant staff and the other organizations in responding to the events, specific procedural instructions, predictability of scenario progression, degraded plant instrumentation and the effect of the environment on task performance. A large amount of experience has been accumulated in performing human reliability analysis for Level 1 PSA, but much less for Level 2 PSA. This appendix discusses the differences in context and their implications in performing human analysis assessment in Level 2 PSA.

A.2. When core damage is imminent or occurring, the operator enters the severe accident management guidelines to mitigate the event. Significant changes to the command and control occur after entering the severe accident management guidelines to various degrees depending on the organization. These changes include the transfer of decision making authority from the main control room to the technical support centre (or at least the inclusion of the technical support centre in the decision making process), the supersedence of the emergency operating procedures with the severe accident management guidelines and a shift in event response focus from preventing core damage to preventing and reducing the release of radioactive material to the environment. In this new command and control structure, the technical support centre may decide or propose the mitigation strategies. Operators in the main control room implement the mitigation strategies and coordinate their implementation. Depending on the organization, after entering severe accident management guidelines, this practice may differ from Level 1 PSA, where operators in the main control room make almost all decisions, even if in some organizations, the technical support centre may also recommend specific actions in accident conditions.

A.3. Multiple organizations may participate in responding to severe accidents. For example, the local fire brigade and companies with contracts with the plant for emergency response could come on to the site to support event mitigation. The government and police could evacuate residents. Ambulances could

come onto the site to pick up injured workers. Organizations that have not participated in routine emergency response exercises with the plant could have issues in communication and coordination. Matters such as whether the contact information and contracts are up to date, whether the communication equipment is compatible, and whether the line of command is clear might have a significant effect on human reliability. For example, the technical support centre might delay issuing instructions for a controlled radioactive release if the line of command is not clear; as a result, the release could impact evacuation.

A.4. Most nuclear power plants routinely use plant specific, full scope plant simulators to train their operating personnel to prevent core damage. Most simulators cannot simulate prolonged post core damage phenomena. As a result, operator training on Level 2 scenarios mainly relies on classroom training that provides only high level guidance. Operator reliability in knowing the plant status can be significantly affected if instrumentation is not available or not reliable. For example, if a core crust shields the molten core from cooling water, the instrumentation might not provide information for the operator to evaluate the effectiveness of the cooling approach or information on whether the reactor vessel has remained intact. In general, severe accident mitigation involves the implementation of specific sensors, qualified for severe accident conditions, in order to improve the success of mitigatory actions. Because of the significant uncertainty in scenario progression, the severe accident management guidelines might be written such that operating personnel can more easily deviate from them than from the emergency operating procedures. In this case, the practice could be prone to errors of commission. The modelling of errors of commission therefore deserves careful consideration.

A.5. Environmental factors such as high temperatures, high levels of radiation, seismic aftershocks and blockage of routes for equipment transportation can affect task performance, as was the case at the Fukushima Daiichi and Daini nuclear power plants after the earthquake and tsunami event in 2011. After the Fukushima Daiichi accident, many nuclear power plants began to use non-permanent equipment stored outside the site to mitigate similar events. The transport and operation of such non-permanent equipment takes place in an open environment that is susceptible to the impacts of adverse conditions. In addition, an adverse environment could affect the availability and accuracy of instrumentation needed for operators to make decisions.

A.6. Paragraphs A.2–A.5 relate to factors in severe accidents that affect the cognitive functions of operating personnel in detecting plant information, understanding the plant status, making event mitigation decisions and

implementing mitigation strategies. In addition, there are two significant differences between human reliability analysis practices in Level 2 PSA and Level 1 PSA. The first difference relates to the repair of unavailable components. Both emergency operating procedures and severe accident management guidelines may instruct operating personnel to repair unavailable components for event mitigation. While repair actions are hardly credited for Level 1 PSA, the longer time frame for Level 2 PSA makes repairing components an option. Operator interventions to mitigate severe accidents could still be needed even a few days after the core damage. The human reliability analysis community needs more data to credibly assess the probabilities of repairing unavailable components in Level 2 PSA. The second difference is about modelling task dependency. For example, a common practice of Level 2 PSA is to start the sequence with the PDS. Each PDS consists of many Level 1 scenarios. This practice simplifies the modelling of Level 2 PSA but it might have an influence on performing detailed human reliability analysis. If a scenario is started from a PDS instead of an initiating event, depending on the level of detail of the PDS defined, important scenario specific information might not be carried to Level 2 PSA for detailed human reliability analysis. The missing information, such as initiating events, failures of the SSCs, the availability of electric power and previous operator errors, may be important for dependency analysis. One way of reducing the effects on PSA results is to analyse the dominant minimal cutsets specific to the PDS to identify their representative event sequences from the initiating events to model dependency effects.

A.7. A few human reliability analysis methods and processes have been developed with Level 2 PSA in their scopes. These include the expanded Human and Organizational Reliability Analysis in Accident Management (HORAAM) method [76, 77], Human Action Modelling Standardized Tool for Editing and Recording (HAMSTER) [78], the Assessment Method for the Performance of Safety Operations (MERMOS) [79], Integrated Human Event Analysis for Event and Condition Assessments (IDHEAS-ECA) method [80], and expert judgment using a human reliability assessment process [81]. In addition, Ref. [82] provides guidance on applying the existing human reliability analysis methods to assess the human reliability of performing tasks in extreme conditions.

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Annex I

COMPUTER CODES FOR SIMULATION OF SEVERE ACCIDENTS FOR WATER COOLED REACTORS

I–1. Severe accident phenomena are complex and have many interdependencies that can be realistically examined using complex computer codes. This annex provides insights into the types of code typically used in Level 2 probabilistic safety assessments (PSAs) and a brief description of their areas of application. The most common codes are presented in Refs [I–1] to [I–32]. Severe accident phenomena encountered in a light water reactor are presented in Fig. I–1.

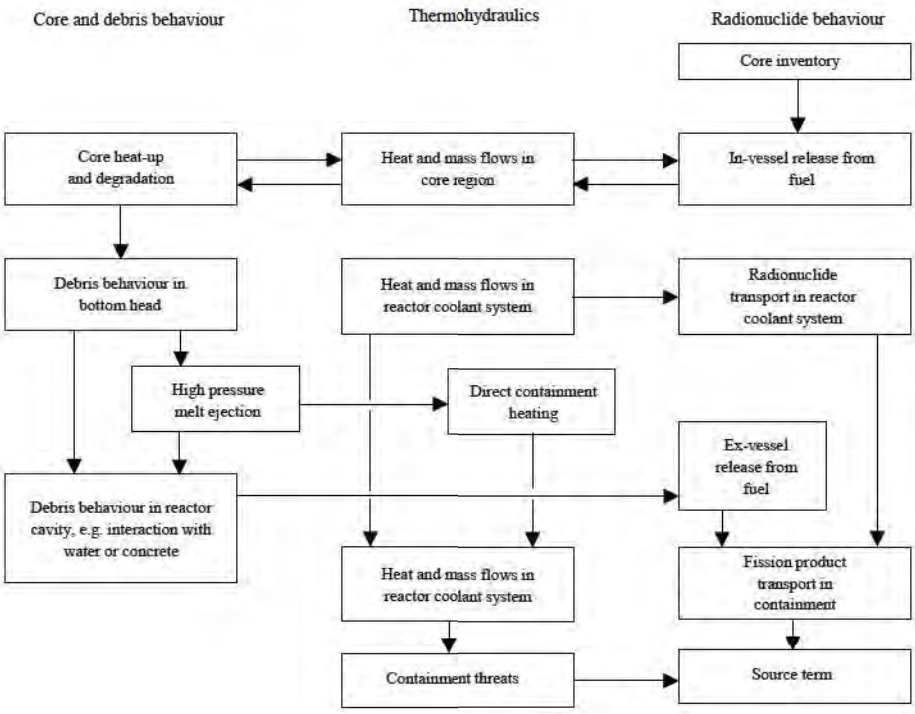

FIG. I–1. Severe accident phenomena encountered in light water reactors.

TYPES OF CODE USED FOR LEVEL 2 PSA

I–2. The codes that model the physical response of the core, the reactor coolant system and the containment to severe accidents can be divided into three types, in accordance with their capabilities and intended use: mechanistic codes (see paras I–4 to I–5), integral codes (see paras I–6 to I–7) and dedicated codes (see paras I–8 to I–9).

I–3. In the past, separate codes, each dealing with a particular phase or aspect of severe accident behaviour, were coupled in a suite with an interfacing facility for the transfer of information between the codes. However, for routine PSA applications, it is desirable to have automatic transfer of information between the elements of a code suite, as manual transfer is slow and can lead to the introduction of errors. A more integrated and modular approach has tended to be adopted in the newer generation of severe accident codes. As such, integral codes are able to model the feedback between different phenomena that could be missed if a specific mechanistic model alone is used.

Mechanistic codes

I–4. Mechanistic codes (also referred to as 'detailed codes') calculate governing phenomena with best estimate models based on first principles, with computational resources being of secondary importance. Mechanistic codes are typically used in research to design and analyse severe accident experiments and to simulate, as accurately as possible, the phenomena and the behaviour of the nuclear power plant structures, systems and components during a severe accident. Once validated against appropriate experimental conditions and tests (i.e. integral or simplified tests), mechanistic codes are also used to establish benchmarks for integral codes or to define simplified models in integral codes. Codes of this type span a wide range of technical disciplines, from the behaviour of damaged fuel to the release of radioactive material, and from transport to hydrogen mixing and combustion processes. The main features of selected mechanistic codes are briefly described in Ref. [I–33].

I–5. Mechanistic codes generally provide an evaluation at a higher level of detail than needed for most Level 2 PSAs. Nevertheless, their application is occasionally necessary under special circumstances, such as when particular issues are unusually important to severe accident behaviour in a unique plant design.

Integral codes

I–6. Integral codes that are designed for routine application in PSA generally use simplified models of certain phenomena so that calculations can be completed relatively quickly (within hours or at most a few days with the current computing technology). As they are relatively fast-running, integral codes can be used to evaluate plant response to many different accident sequences or can be run several times for the same accident sequence to support sensitivity and uncertainty analysis. To ensure that the overall execution time of the code is reasonable, the modelling approach to some phenomena (e.g. fuel damage, melting) is simpler than the approaches used in mechanistic codes. Whereas in a mechanistic code, models might be used to evaluate explicitly the individual effects of several damage mechanisms within fuel rods, the same effects might be evaluated in a simpler and composite manner in integral codes.[1] The extent to which such simplifications properly reflect important characteristics of the actual governing phenomena is determined by comparison of the calculated results with experimental data and with the results of parallel calculations performed with mechanistic codes. Examples of such comparisons are found in Refs [I–34, I–35]. The main features of selected integral codes are briefly described in Ref. [I–33].

I–7. Integral codes model the physical response of the entire plant to postulated severe accidents from the initiating event through to the release of radioactive material to the environment. Examples of phenomena and processes modelled by such codes include the following:

(a) Thermohydraulic processes in the reactor coolant system, the containment structure and/or the structures, systems and components ensuring the confinement function;
(b) Degradation of core cooling, fuel heat-up, cladding oxidation, fuel degradation (loss of fuel geometry), and melting and relocation of core material;
(c) Heat-up of the reactor pressure vessel lower head from relocated fuel material and the thermal and mechanical loading and failure of the reactor pressure vessel lower head;
(d) Transfer of core material from the reactor pressure vessel to the containment cavity;

[1] For example, for cladding failures, mechanical codes might evaluate thermomechanical interactions between the swollen fuel pellet and bounding cladding, and local ballooning at weak points in the cladding, whereas such failures might be represented by specifying an effective cladding failure temperature in integral codes.

(e) Thermochemical interactions between molten core debris and concrete on the containment floor and resulting generation of aerosols;
(f) In-vessel and ex-vessel production of combustible gases (e.g. hydrogen, carbon monoxide), transport and combustion;
(g) Radioactive material release (aerosol and vapour), transport and deposition;
(h) Behaviour of radioactive aerosols in the reactor containment, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere, such as particle agglomeration and gravitational settling;
(i) Impact of engineered safety features on thermohydraulic and radionuclide behaviour.

Dedicated codes

I–8. Dedicated codes and algorithms (also referred to as 'fast-running codes') provide rough estimates of parameters for specific PSA applications, such as estimation of the radiological source term [I–36] or of containment loads accompanying high pressure melt ejection [I–37]. Such codes and algorithms are generally used to establish the primary technical basis when more runs are needed than can be reasonably handled, even by contemporary PSA codes. Dedicated codes are based on simple parametric models that interpolate between fixed points, for which calculations with a more complicated code have already been performed, to determine the values of the parameters. The use of such codes is reasonable for the generation of uncertainty values, but the parameters used in the codes, as well as the results produced by them, have to be calibrated by more detailed calculations or experimental data.

I–9. Depending on the objective, dedicated codes could be based on a fast-running algorithm or a complex calculation to describe the single phenomenon under study. They can be used for different reactor technologies such as water cooled reactors and high temperature gas cooled reactors [I–31, I–32].

VALIDATION OF A CODE

I–10. The verification and validation of computer codes is crucial for enhancing confidence in their application. Achieving validation of severe accident codes is very difficult, as the extreme conditions that occur in a severe accident and the scale of the physical geometry are difficult to replicate in experiments. To conduct the process of validation, a validation matrix involving many simulations is generally needed. Caution needs to be applied if code validation has been performed by varying the values of user supplied parameters until a

reasonable fit to experimental data is achieved; at best, this is an indirect experimental measurement of the parameter values and not an independent validation of the code.

USE OF CODES

I–11. Deterministic accident analysis codes need to be designed so that a Level 2 PSA analyst familiar with general accident phenomena can run them reliably without needing to have the same detailed knowledge as a specialist using the code. In order for the code calculations to be meaningfully incorporated into the framework of a Level 2 PSA, the analyst needs to have a sound knowledge of the reactor systems and a reasonable knowledge of the following:

(a) The phenomena addressed in a code and their modelling approach and limitations;
(b) The meaning of the input variables and the range of validity for the selected code;
(c) The meaning of the output variables.

I–12. The user needs to have a sound knowledge of the strengths and limitations of the code. Moreover, the code may not be used out of the range of situations and conditions for which it has been designed.

CODES DEVELOPED SPECIFICALLY FOR LEVEL 2 PSA

I–13. Codes for simulating fault trees and event trees and other simulation codes that are typically used for Level 1 PSAs are also often used for Level 2 PSAs. In many cases, such codes have been adapted or enhanced to address certain unique needs of Level 2 PSA applications, such as the solution of logic models with large event probabilities, and enhanced capabilities or more diverse methods for addressing uncertainties. For separate Level 1 and Level 2 PSA modelling, there are several codes designed specifically for the modelling of Level 2 aspects (e.g. accident progression after core damage). Such codes have special features, such as user defined functions that calculate the severe accident event probabilities. Codes that have been specifically developed for accident progression event tree analysis are generally very well qualified for phenomenological issues in Level 2 PSA but may need to be adapted to model the behaviour of systems.

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Annex II

SAMPLE DOCUMENTATION FOR A LEVEL 2 PSA STUDY

II–1. Given the great number of uncertainties associated with the performance of Level 2 first use in Annex II, standardized contents of the reports presenting Level 2 probabilistic safety assessment (PSA), studies allow for a more effective peer review process. A sample outline of the contents of the summary report and the main report are presented below.

SAMPLE CONTENTS OF THE SUMMARY REPORT FOR A LEVEL 2 PSA STUDY

S1. Introduction

S2. Overview of the objectives and justification for the Level 2 PSA study

S3. Overview of the approach

S4. Containment failure modes and their likelihoods

S5. Radiological source terms and their frequencies (complementary cumulative distribution functions)

S6. Summary of plant vulnerabilities to severe accidents, interpretation of results

S7. Conclusions and recommendations

S8. Possible risk reduction measures

S9. Outline of the main report

SAMPLE CONTENTS OF THE MAIN REPORT FOR A LEVEL 2 PSA STUDY

M1. Introduction

M1.1 Background

M1.2 Objectives

M1.3 Scope of the Level 2 PSA study

M1.4 Project organization and management

M1.5 Composition of the study team

M1.6 Overview of the approach

M1.7 Structure of the report

M2. Description of the design of the plant and the containment

M2.1 Plant and containment design features affecting severe accidents

M2.2 Operational characteristics

M2.3 Description of plant modifications and containment system modifications (if any)

M3. Interface with Level 1 PSA

M3.1 Grouping of accident sequences and specification of attributes

M3.2 Plant damage states for internal initiating events at full power and associated uncertainties

M3.3 Plant damage states for internal and external hazards at full power and associated uncertainties

M3.4 Plant damage states for other plant operational states and associated uncertainties

M4. Analysis of the containment's structural performance

M4.1 Description of the structural design and failure modes of the containment

M4.2 Approach for structural analysis

M4.3 Structural response and fragility results

M4.4 Summary of uncertainties and/or fragility curves for containment performance

M4.5 Impact of external hazards

M5. Accident progression and containment analysis

M5.1 Severe accident progression analysis

M5.1.1 Scope of analysis

M5.1.2 Method of analysis (e.g. codes, models)

M5.1.3 Summary of point estimate results for plant damage states analysed

M5.2 Accident progression event trees

M5.2.1 Accident progression event tree structure

M5.2.2 Operating procedures and recovery

M5.2.3 Accident progression event tree quantification process

M5.2.4 Grouping (binning) of accident progression event tree end states

M5.2.5 Treatment of uncertainties

M5.2.6 Results

M5.2.6.1 Point estimate containment performance matrix (C matrix)

M5.2.6.2 Uncertainties in failure probabilities

M5.2.6.3 Interpretation of results

M6. Accident source terms

M6.1 Grouping of radionuclides

M6.2 Method of analysis (e.g. codes, models)

M6.3 Summary of point estimate results for plant damage states analysed

M6.4 Treatment of uncertainties

M6.5 Results

M6.5.1 Point estimate source term characteristics

M6.5.2 Uncertainties in source term characteristics

M6.5.3 Interpretation of results

M7. Sensitivity and importance analyses

M7.1 Identification of sensitivity issues

M7.2 Results of sensitivity analysis

M7.3 Importance ranking of issues, systems and components

M8. Conclusions

M8.1 Key insights on characteristics of severe accidents and containment response

M8.2 Design features and inherent mitigation benefits

M8.3 Conclusions relative to PSA objectives

APPENDICES

A1. Basis for containment structural fragilities

A2. Basis for accident progression event tree quantification

A3. Results of deterministic severe accident progression analyses

A3.1 Containment loads

A3.2 Accident source terms

A4. Basis for probability distribution and ranges of uncertain parameters

A5. Detailed results of uncertainty analysis and/or sensitivity analysis

II–2. The successful performance of Level 2 PSA needs careful planning of relevant activities as part of the project management. In addition, the performance of planned activities for Level 2 PSA depends on several factors such as the level of expertise, the preparation of the team, the scope of the Level 2 PSA, the regulatory process for review and approval, as well as resources involved. A Level 2 PSA takes from one year to several years to implement. Figure II–1 shows an example plan of Level 2 PSA related activities. The timing estimates are based on the assumption that the team performing the PSA is well trained.

FIG. II-1. Example plan of activities for performance of a Level 2 PSA.

Annex III

EXAMPLES OF COMMON RISK METRICS IN LEVEL 2 PSA

III–1. Large release frequency and large early release frequency are the most common measures of risk used in Level 2 probabilistic safety assessment (PSA). In many States, numerical values of this type are used as probabilistic safety goals or criteria. For example, Level 2 PSA risk metrics for large early release frequency should provide information on the frequency of release, the main radionuclides in each release category and the notion of the time of the release. The large release frequency should be used as an integral indicator of the risk profile covering early and late radioactive releases. Level 2 PSA risk metrics for large release frequency should provide information on both the frequency of release and the main radionuclides in each release category, integrated over a specified period of time.

III–2. Tables III–1 and III–2 provide examples of some Member States' values and definitions of large release frequency and large early release frequency, respectively. The examples are based on the original regulatory texts (and direct translation into English, as appropriate) and the regulatory framework for safety in each Member State is to be taken into account when viewing the tables.

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
Bulgaria	[III–1]	Large releases shall mean releases of radioactive material to the environment, which necessitate off-site protective actions to be implemented for protecting people and their application cannot be limited in terms of times and areas.	Accidents with nuclear fuel melting, resulting in early or large radioactive releases to the environment shall be practically eliminated.
Canada	[III–2]	For operating nuclear power plants (consistent with INSAG-12 [III–3]): Large release frequency (a release of more than 100 TBq of Cs-137)	$< 1\times10^{-5}$
		For new nuclear power plants: Large release frequency (a release of more than 100 TBq of Cs-137 or requiring long term relocation of the population)	$< 1\times10^{-6}$
		Small release frequency (a release of more than 1000 TBq of I-131 or requiring temporary evacuation of the local population)	$< 1\times10^{-5}$
Czech Republic	[III–4]	More than 1% of the initial amount of Cs-137 of the core inventory released to the environment	
Finland	[III–5]	100 TBq of Cs-137	$< 5\times10^{-7}$

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
France	[III–6]	Primarily for new nuclear power plant designs: Protective measures for the public should be very limited in terms of extension and duration, meaning no permanent relocation, no evacuation needed outside of the immediate vicinity of the plant site, neither sheltering nor long-term restriction of food consumption outside the vicinity of the plant site. Consequently, these accidents should not lead to either contamination of large areas or long-term environmental pollution.	[No quantitative value]
Japan	[III–7]	Taking into account the TEPCO's Fukushima Daiichi nuclear power plant accident, it is necessary to incorporate the viewpoint of the environmental contamination by radioactive materials into safety goal, and to keep the impact on the environment as low as possible if accidents occur. The frequency of accidents in which the release of Cs-137 exceeds 100 TBq should be reduced to no more than once in one million reactor-years except for those caused by terrorist attacks.	
Russian Federation	[III–8]	The release of radioactive substances into the environment during an accident at a nuclear power plant when, in the case of exceeding established criteria for radiation doses, it is necessary to implement measures to protect the population within the initial stage of the accident (up to 10 days) on the border of the protective actions planning zone and outside it. It should be noted that the established frequency of release is not a safety goal; it is a safety target.	$< 1 \times 10^{-7}$

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
Slovakia	[III–9]	> 1% of Cs-137 released from the core inventory	
Switzerland	[III–10]	> 200 TBq of Cs-137 per calendar year	
Ukraine	[III–11]	Large release is defined as requiring public evacuation at the boundary of the protection area.	For existing plants: Criterion: $< 1\times10^{-5}$ Goal: $< 1\times10^{-6}$; For new plants: Criterion: $< 1\times10^{-6}$ Goal: $< 1\times10^{-7}$
United States of America	[III–12], [III–13]	The United States Nuclear Regulatory Commission does not use large release frequency as a safety goal. For new reactor design certification reviews, the Nuclear Regulatory Commission has defined a core damage frequency goal and a conditional containment failure probability goal, complemented by a deterministic containment performance goal. The Nuclear Regulatory Commission uses the large release frequency metric for new reactors of 1×10^{-5} as a screening criterion to inform the staff whether new reactor design applicants are meeting the Commission's expectations for a higher standard of severe accident safety performance and increased margin before exceeding safety limits.	

TABLE III–1. DEFINITIONS OF LARGE RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State Reference	Large release frequency risk metrics definition	Safety goal frequency (reactor-years)
	The Nuclear Regulatory Commission has not approved a formal definition of a large release or a large release frequency. One informal definition for large release frequency is the frequency of an unmitigated release of airborne fission products from the containment to the environment that is of sufficient magnitude to cause severe health effects, regardless of its timing. New reactors transition from large release frequency to large early release frequency metric at or before initial fuel load and discontinue regulatory use of large release frequency thereafter.	

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
Bulgaria	[III–1]	Large releases refers to releases of radioactive material to the environment that necessitate off-site protective actions, which are to be implemented to protect people and the application of which cannot be limited in terms of times and areas. Early releases refers to radioactive releases to the environment that would require off-site emergency measures for protection of the public but that are rendered impossible due to insufficient time to implement them.	Accidents with nuclear fuel melting, resulting in early or large radioactive releases to the environment will be practically eliminated.
Czech Republic	[III–4]	More than 1% of the initial amount of Cs-137 of the core inventory released to the environment within 10 hours after the beginning of the severe accident (temperature of cladding = 1200°C).	
Finland	[III–14]	The accident sequences, in which the containment function fails or is lost in the early phase of a severe accident, have only a small contribution to the reactor core damage frequency. 'Early' means that there is no time to implement the warning and protective measures prior to the release. An exact number of hours has not been defined, but warning and protection are typically estimated to take approximately four hours after the rescue department receives information on the need to take shelter. The objective is that protective measures are not needed in a situation in which there would practically be no time to implement them.	

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
France	[III-6]	The objective of the design is that event sequences that could lead to large releases with kinetics that might not allow the timely implementation of the measures necessary for the protection of populations should be physically impossible and, if not, very unlikely to happen with a high degree of confidence.	
Hungary	[III–15]	For operating nuclear power plants: a radioactive release in the case of which urgent precautionary measures are required off the site but no sufficient time is available for their introduction	$< 1\times10^{-5}$ for operating nuclear power plants (1×10^{-6} target)
	[III–16]	For new nuclear power plants: (a) Urgent protective measures are required beyond a distance of 800 m from the nuclear reactor OR (b) There is a need for any kind of temporary action (i.e. the temporary evacuation of the population) beyond a distance of 3 km from the nuclear reactor OR (c) There is a need for any kind of subsequent protective measure (i.e. the final resettlement of the population) beyond a distance of 800 m from the nuclear reactor OR (d) There is a need for any long term restriction on food consumption.	$< 1\times10^{-6}$ for new nuclear power plants
Korea, Republic of	[III–17, III–18]	The frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the surrounding population is such that there is a potential for early health effects.	$< 1\times10^{-5}$ for operating nuclear power plants $< 1\times10^{-6}$ for new nuclear power plants

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
Pakistan	[III–19]	The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated.	$< 1\times10^{-6}$
Russian Federation	[III–20]	Decisions on measures to protect the population in the event of a major radiation accident with radioactive contamination of the territory are made on the basis of a comparison of the predicted dose that is prevented by the protective measure and the levels of contamination over a period of 10 days.	The term large early release frequency is not defined in the Russian Federation.
Slovakia	[III–21]	More than 1% of Cs-137 released from the core inventory to the environment within 10 hours after the beginning of the initiating event.	$< 1\times10^{-5}$ for operating nuclear power plants $< 1\times10^{-6}$ for new nuclear power plants
Switzerland	[III–22]	The large early release frequency is the expected number of events per calendar year with a release of more than 2×10^{15} Bq of I-131 per calendar year within the first 10 hours after core damage.	$< 1\times10^{-5}$ for operating nuclear power plants $< 1\times10^{-6}$ for new nuclear power plants

TABLE III–2. DEFINITIONS OF LARGE EARLY RELEASE FREQUENCY RISK METRICS AND SAFETY GOAL FREQUENCIES AS USED IN THE REGULATIONS OF SOME MEMBER STATES (cont.)

Member State	Reference	Large early release frequency risk metrics definition	Safety goal frequency (reactor-years)
United States of America	[III–23]	The large early release frequency is defined as the sum of the frequencies of those accidents leading to rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is the potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events and loss of containment isolation.	$< 1\times10^{-5}$

Additional information on risk metrics used in other countries can be found in Ref. [III–24].

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24-05216E-T

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International Atomic Energy Agency
Vienna, 2025
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ISBN 978-92-0-137424-0