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A REVIEW OF THE ACCIDENT RISK POSED BY THE PICKERING 'A' NUCLEAR GENERATING STATION

A Report to the
Standing Committee on Energy, Environment
and Natural Resources
of the
Canadian Senate

by

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Abstract

Ontario Power Generation (OPG) seeks to restart the four Pickering 'A' nuclear reactors, which were shut down in 1997. Studies relied upon by OPG suggest that there is about a 1 percent probability of a severe core damage accident at one of these reactors during its future life. However, OPG argues that is highly unlikely that a core damage accident would lead to a large (severe) release of radioactive material to the environment.

Here, the Institute for Resource and Security Studies (IRSS) provides a brief, limited review of the accident risk posed by the Pickering 'A' reactors. IRSS reviews available studies that relate to the Pickering 'A' accident risk, and finds that they do not provide a complete and accurate picture of that risk. Accordingly, IRSS sets forth a recommended process for understanding the Pickering 'A' accident risk.

The scope of IRSS's review was not sufficient to support an independent estimate of the probability of a severe core damage accident or a large release of radioactive material at Pickering 'A'. IRSS's interim judgement is that the probability of a large release is likely to exceed 1 per 100 thousand years, thereby exceeding OPG's own probability limit.

About the Institute for Resource and Security Studies

The Institute for Resource and Security Studies (IRSS) is an independent, non-profit, Massachusetts-based corporation. It was founded in 1984 to conduct technical and policy analysis and public education, with the objective of promoting peace, international security and the sustainable use of natural resources. Projects conducted by IRSS always reflect a concern for practical solutions to resource, environment and security problems, and can range from detailed technical studies to preparing educational materials accessible to the public.

About the author

This review was prepared by Gordon Thompson, who is the executive director of IRSS. Thompson studied and practised engineering in Australia, was based in the UK for the period 1969-1978, and received a DPhil in applied mathematics from Oxford University in 1973. He has been based in the USA since 1979. Dr Thompson has extensive experience in assessing the hazards associated with nuclear facilities, and in identifying alternative designs and modes of operation that can reduce a facility's hazard potential. In addition, Thompson has worked on a range of other subjects related to energy, the environment and international security.

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1. Introduction

The Pickering 'A' Nuclear Generating Station features four CANDU reactors that have been shut down since 1997. Recently, the licensee of these reactors, Ontario Power Generation (OPG), has proposed to return them to service. This proposal has led to renewed interest in the safety of the reactors.

In this report, the Institute for Resource and Security Studies (IRSS) provides a brief review of the accident risk posed by the Pickering 'A' reactors. This review does not purport to be a comprehensive assessment of accident risk. Such an assessment would require considerably greater time and resources than were available for IRSS's review. A recommended process for developing a thorough understanding of the Pickering 'A' accident risk is outlined in this report.

For the purposes of this review, the phrase 'accident risk' refers to the potential for a significant, unplanned release of radioactive material from the Pickering Station to the environment.[1] Such a release could occur to the atmosphere, yielding a radioactive plume which travels downwind. Alternatively, a release could occur into Lake Ontario or into ground or surface waters within or near the site boundary.

An accidental, unplanned release of radioactive material could be initiated by a variety of events. These events could include equipment failures, errors in design, construction or operation, acts of insanity, fires or explosions, and natural forces (e.g., earthquakes). An unplanned release could also be initiated, directly or indirectly, by acts of malice, such as war, terrorism or sabotage.

The Pickering Station has four 'A' reactors and four 'B' reactors. This report focusses on the four 'A' reactors. However, the 'A' and 'B' reactors share safety systems and can interact in various ways. Thus, the accident risk posed by the 'A' reactors must be examined in the context of the Pickering Station as a whole.

Section 2 of this report summarizes the characteristics of the Pickering 'A' reactors. Then, Section 3 discusses the role of accident risk in the proposed restart of the reactors. Section 4 provides some general observations on the assessment of nuclear accident risk, and Section 5 describes available studies that relate to the Pickering 'A' accident risk. In Section 6, those studies are compared with studies performed in the USA. Section 7 provides a critical review of the Pickering 'A' risk studies, and Section 8 outlines a process for developing a thorough understanding of the Pickering 'A' accident risk. Conclusions and

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[1] With this definition, the concept of accident risk does not encompass: (i) routine, planned releases of radioactive material from Pickering; (ii) impacts associated with long-term management, transport and disposal of radioactive wastes generated at Pickering; or (iii) events (e.g., industrial accidents) inside the Pickering Station that do not have the potential to cause a significant release of radioactive material to the environment.

recommendations are set forth in Section 9, and a bibliography is provided in Section 10.

2. Characteristics of the Pickering 'A' reactors

The Pickering Station was built and operated by Ontario Hydro but is now owned and operated by OPG, the successor organization to Ontario Hydro. Pickering features four 'A' reactors and four 'B' reactors. Each of the eight Pickering reactors is a 540 MW(e) CANDU reactor. The Pickering 'A' reactors entered service in the period 1971-1973, and have been shut down since December 1997. The Pickering 'B' reactors entered service in the period 1982-1985, and remain operational.[2]

In Canada, the safety of nuclear facilities is regulated by the Canadian Nuclear Safety Commission (CNSC), which became, in June 2000, the successor organization to the Atomic Energy Control Board (AECB). Hereafter, both organizational names are used, depending upon the time period under discussion.

According to an AECB report published in 1998, the Pickering 'A' reactors were:[3]

> "Placed in a guaranteed shutdown state because the reactor[s] did not meet the regulatory requirement for an enhanced shutdown system. This was followed by a business decision taken by the licensee in 1997 to place the unit[s] in a lay up state."

OPG now proposes to return the Pickering 'A' reactors to service.[4] The first reactor would be returned to service at the end of 2001, and the remaining reactors would be returned at approximately six to eight-month intervals thereafter. Before returning these reactors to service, OPG proposes to make safety-related "improvements" in the reactors, as discussed in Section 3. OPG currently proposes to operate each of the reactors for slightly more than one decade, accumulating a total future operating experience for the Pickering 'A' Station of about 52 reactor-years. If operation of the reactors proves to be economical, it is likely that OPG will seek to extend the reactors' operating lifetimes. Thus, the Pickering 'A' Station might, in practice, accumulate up to 50-100 reactor-years of future operating experience.

General features of CANDU reactors

A CANDU reactor employs natural uranium fuel which is cooled by pressurized heavy water. The fuel is located inside several hundred pressure tubes which are in turn located inside calandria tubes. Surrounding the calandria tubes, confined

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[2] AECB, 1998, page 24.
[3] Ibid.
[4] OPG, 2000.

by a large tank known as the calandria, is a body of heavy water which functions as a neutron moderator. The reactor cooling water passes through steam generators (boilers) in which light water is converted to steam, which drives a turbogenerator. The reactor cooling water itself does not boil.

By contrast, the majority of commercial reactors in the world are light-water reactors (LWRs). In these reactors the fuel is located inside a pressure vessel, and is surrounded by light water which functions as both coolant and moderator. There are two types of LWR: (a) the pressurized-water reactor (PWR), in which the coolant/moderator does not boil during normal operation; and (b) the boiling-water reactor (BWR), in which the coolant/moderator does boil.

Table 1 compares some of the properties of typical PWR, BWR and CANDU reactors. From this table one sees that the CANDU reactor has comparatively large inventories of cooling/moderator water, uranium dioxide and zirconium alloy. Other factors being equal, the larger inventories of water and uranium in the CANDU provide a safety advantage, because their thermal mass reduces the rate of temperature increase of the reactor core if cooling is interrupted. Also, the CANDU's separation of cooling and moderator water can provide a safety advantage. On the other hand, the larger inventory of zirconium alloy in the CANDU provides a safety disadvantage. Zirconium reacts vigorously and exothermically with steam at the temperatures experienced during core damage accidents, and this reaction yields hydrogen gas which can reach explosive concentrations in the air inside the containment.[5]

LWRs have the property that their power level declines if a void forms in the coolant. By contrast, CANDU reactors exhibit an increase in power level if a void forms in the coolant. They share this property with RBMK reactors, one of which experienced a destructive power surge, and a subsequent fire, at the Chernobyl Station in 1986. .

Thus, an accident scenario of particular concern for CANDU reactors involves a loss-of-coolant accident (LOCA) coincident with a failure of the reactor shutdown system. The LOCA can lead to voids in the coolant, causing an increase in reactor power. If the shutdown system is inoperative, the power level can rise dramatically. Calculations show that violent disruption of the reactor core can occur within 4-5 seconds.[6]

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[5] A thorough comparison of the safety of different types of nuclear power plant requires a detailed analysis of many factors. The data provided in Table 1 are not sufficient to support such a comparison. Moreover, the specific design details of each individual plant can be significant for safety. Finally, the quality of maintenance and operation have significant effects on the safety of a particular plant.

[6] Foster and Loewenstein, 1992.

Particular features of the Pickering 'A' reactors

The Pickering 'A' reactors were the first commercial CANDU units. As a result, they lacked (or still lack) a number of features that were incorporated into later CANDU units. For example, the Pickering 'A' reactors have one fast-acting shutdown system, designed to be fully effective in 2 seconds, and one slow-acting shutdown system (moderator dump) which requires more than 10 seconds to be fully effective.[7] All subsequent CANDU units have two independent, fast-acting shutdown systems.

Ontario Hydro employed an unusual design concept for all of its nuclear generating stations – Pickering (8 units), Bruce (8 units), and Darlington (4 units). Under this concept, multiple units share the same safety and support systems. Sharing of systems among units does occur at other nuclear generating stations around the world, but generally to a much smaller extent than was chosen by Ontario Hydro. The stations built by Ontario Hydro employ CANDU reactors exclusively, but this is irrelevant to the sharing of systems. CANDU units have been built on a stand-alone basis, and other reactor types -- such as LWRs – could be built with shared safety systems.

At Pickering, all eight reactors share the same containment system, which consists of eight individual reactor buildings, one vacuum building, and one pressure relief duct which connects these buildings. This containment system has the design capacity to absorb 530 GJ of thermal energy, which is approximately the thermal energy (stored energy and radioactive decay heat) available from one reactor during a 1-hour period after shutdown.[8] In other words, the containment system is designed to accommodate one accident at one reactor. In choosing this arrangement, the Ontario Hydro designers were presumably motivated by a desire to reduce capital cost, and were presumably not concerned about the possibility of contemporaneous accidents at two or more reactors.

Table 2 sets forth the volumes and design pressures for the compartments of the Pickering containment system, and provides comparative data for typical PWR and BWR plants. One notes that the Pickering Station's containment has a comparatively large volume, which provides a safety advantage. However, this advantage is offset by Pickering's low design pressure, and by the fact that the pressure relief duct and vacuum building at Pickering are shared by eight reactors but are designed to accommodate an accident at only one reactor.

Another illustration of the Ontario Hydro design concept is provided by the emergency coolant injection system (ECIS) at Pickering. This system was built during construction of the Pickering 'B' Station, because Pickering 'A' did not have an ECIS. The system is designed to provide a flow rate of 640 l/s, to accommodate the largest postulated leak at one reactor. The heart of the system

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[7] Ibid.
[8] Ontario Hydro, 1969.

is a storage tank and pumphouse which can supply any of the eight units at Pickering. The storage tank can provide a flow rate of 760 l/s, and the pumphouse has three pumps which can each provide a flow of 640 l/s.[9] Thus, the ECIS is designed to accommodate one accident at one reactor.

3. Accident risk and Pickering 'A' restart

There is general agreement that Ontario Hydro's nuclear operations have exhibited significant safety-related deficiencies. In 1997, Ontario Hydro commissioned an independent review of its nuclear operations, and the first sentences of the report on this review were:[10]

> "Long standing management, process and equipment problems in Ontario Hydro Nuclear (OHN) plants are well known but have not been aggressively resolved. As a result, the overall performance of OHN is well below the level of performance typically achieved by the best nuclear utilities. Immediate attention is needed to improve performance so that the value of OHN's assets does not depreciate beyond recovery."

The same review gave the Pickering Station a "minimally acceptable" rating, and stated that "the plant material condition has deteriorated". A wide variety of deficiencies in management, maintenance, operation and training at Pickering were identified. An assessment of the Pickering Electrical Distribution System (EDS) concluded:[11]

> "The review found that, in some cases, the ability of the EDS to fulfill its design functions was not assured. In other cases, margins appear to have been reduced to unacceptable levels. System deficiencies are primarily attributed to a failure to account adequately for the cumulative effect of design and operating changes. Ineffective maintenance further reduces confidence in the system's reliability and readiness to function under design basis conditions."

Clearly, deficiencies of this kind raise concerns that the Pickering 'A' reactors will pose an accident risk that is significantly greater than would be the case if the reactors were well maintained and operated. Thus, one of the major factors that should be weighed in determining whether the reactors should be restarted is the extent to which the identified deficiencies have been corrected. Other major factors include: (a) the accident risk posed by the reactors, after correction of the identified deficiencies; and (b) the economics of restarting the reactors.

Safety improvements promised by OPG

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[9] Ibid.
[10] NPAG, 1997.
[11] Ibid.

In its April 2000 environmental assessment report on the restart of the Pickering 'A' reactors, OPG stated that a variety of improvements will be made prior to restart. The cost of these improvements will be about 1 billion dollars. Some of the improvements relate to nuclear safety, while others relate to environmental protection (prevention of oil spills, etc.). Unfortunately, OPG provided only limited information about the promised improvements. The environmental assessment report does not set forth an integrated package of improvements, and does not describe the allocation of resources to particular types of improvement. Thus, for example, one cannot determine how much of the 1 billion dollars will be spent on hardware upgrades, as opposed to 'softer' functions such as training, documentation, and managerial improvements. Nor can one determine the extent to which OPG's promised improvements will affect the risk profile of the reactors.[12]

Additional information on the promised improvements is provided in a November 1999 OPG document.[13] However, this document does not provide a firm commitment to implement all of the listed improvements. In many instances it states that improvements will be made "if possible". Also, this document does not provide any rationale for selecting this particular set of improvements, or any assessment of how the improvements will affect the risk profile of the Pickering 'A' reactors.

One of the safety improvements that OPG is committed to implementing at the Pickering 'A' reactors before restart is a set of enhancements to the fast-acting shutdown system of the reactors. After a decade-long debate that was initiated by the AECB following the 1986 Chernobyl accident, Ontario Hydro agreed to implement these enhancements by the end of 1997. Installation began but was suspended when Ontario Hydro decided to shut down the reactors at the end of 1997. The agreed set of enhancements is a comparatively cheap option, and will not provide a second, independent fast-acting shutdown system. All CANDU reactors other than Pickering 'A' have such a system.

The probability of severe accidents

OPG has established a policy on the use of probabilistic risk assessment (PRA) to assess and manage the risk of accidents at nuclear facilities. As part of this policy, OPG declares that its goal for the average probability of severe core damage is 1 per 100 thousand reactor-years, while its limit for the same probability is 1 per 10 thousand reactor-years. Similarly, OPG's goal for the average probability of a large (severe) release of radioactive material is 1 per 10 million reactor-years, while its limit for the same probability is 1 per 1 million reactor-years.

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[12] OPG, 2000.
[13] OPG, 1999.

The AECB has not formally articulated a similar set of goals and limits. However, a senior AECB official stated at a conference in 1995:[14]

> "Typically, we wish to satisfy ourselves that the probability of significant core damage is less than 10^{-5} [1 per 100 thousand] per year at any station."

If this statement is interpreted as regulatory policy, and if all eight reactors at the Pickering Station become operational, then AECB policy would require (neglecting the difference between 8 and 10) that the average probability of significant core damage at Pickering is less than 1 per 1 million reactor-years. It is not clear if this is a goal or a limit. What seems clear is that the AECB would expect the probability of severe core damage to be at least one order of magnitude lower than OPG's goal, and two orders of magnitude lower than OPG's limit.

As discussed in Section 5, a PRA study performed by Ontario Hydro for Pickering 'A' found a probability of severe core damage of 1.3 per 10 thousand reactor-years.[15] OPG has promised that this probability will be reduced to below OPG's limit of 1 per 10 thousand reactor-years, before the Pickering 'A' reactors are restarted. By itself, however, this is a promise with little content. PRA calculations have many steps, and an adjustment of the final result by a factor of two or more can be achieved by some minor changes in assumptions. Moreover, the uncertainty of the final result is substantially larger than a factor of two.

A meaningful commitment to increased safety would involve hardware and procedural changes that significantly affect the dominant accident sequences for the Pickering 'A' reactors.[16] Also, such a commitment would involve the correction of deficiencies in reactor operations, as discussed above. The manner in which these changes could be reflected in PRA findings is discussed in Section 8.

4. Assessing the risk of nuclear accidents

The present generation of commercial nuclear reactors, including CANDUs and LWRs, derives from prototypes that were built in the 1950s and 1960s. National institutions and concepts for nuclear safety regulation generally date to the same era. Thus, contemporary reactors and safety regulations carry forward the concepts and beliefs that prevailed several decades ago.

In those earlier decades, concern about safety was focussed on preventing 'design-basis' accidents. These are accidents that a nuclear plant is designed to accommodate. They do not include 'severe' accidents, in which the reactor core

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[14] J G Waddington, in AECB, 1995.

[15] The study expressed probability on a per-year basis, but the results are approximately equivalent to probability per reactor-year.

[16] 'Dominant' accident sequences are those which make the greatest contribution to the probability of a specified outcome, such as severe core damage.

suffers severe damage and there is the potential for a large release of radioactive material to the environment. Reactor designs and nuclear safety regulations are, to this day, primarily driven by the intention to accommodate design-basis accidents, including loss-of-coolant accidents (LOCAs). For many purposes, severe accidents are explicitly or implicitly regarded as incredible.

During the 1970s, studies began to show that severe accidents are entirely credible, and can have a surprisingly high probability. The most detailed study of the 1970s was the Reactor Safety Study, published by the US Nuclear Regulatory Commission (NRC) in 1975. A more recent study, published by the NRC in 1990, represents the present state of the art for nuclear reactor risk assessment.[17]

A variety of actual events have further shown that severe accidents are credible. These events include two severe core damage accidents – at Three Mile Island (TMI) in 1979 and Chernobyl in 1986 – and a number of precursor events that were arrested before severe core damage occurred. The TMI accident did not involve a large release of radioactive material to the environment, but the Chernobyl accident did involve such a release. Most of the offsite radiation exposure attributable to the Chernobyl accident is from the isotope cesium-137. About 69 kg of cesium-137 were present in the core of the affected Chernobyl reactor, and about 27 kg were released to the environment.[18] For comparison, each of the eight Pickering reactors contains about 15 kg of cesium-137 during normal operation.[19]

Strengths and weaknesses of PRA

A probabilistic risk assessment for a nuclear reactor can be performed at three levels. At the first level, the PRA estimates the probabilities of specified states of core damage. At the second level, the PRA estimates the probabilities and characteristics of potential releases of radioactive material. The third level is an assessment of the offsite consequences of those releases. In some cases, the PRA is performed solely for 'internal' accident-initiating events (equipment failure and human error), while in other cases the PRA is also performed for 'external' accident-initiating events (e.g., earthquakes, storms, fires and explosions). Acts of insanity or malice (e.g., sabotage, terrorism or war) are not usually considered in PRAs as initiating events, on the grounds that their probability cannot readily be estimated.

PRA methodology can provide a valuable framework for systematically examining a reactor's potential to experience a severe accident. There are, however, two important caveats about the use of PRA.[20] First, a PRA must be

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[17] NRC, 1990.

[18] Thompson, 1998.

[19] Ontario Hydro, 1995. Note that 1 TBq of cesium-137 is equivalent to 0.3 g.

[20] A detailed review of the strengths and weaknesses of PRA is provided in Hirsch et al, 1989.

performed to the highest standards. Second, caution must be exercised in using the findings of a PRA.

Achieving high standards in a PRA requires the use of the best available methodology. It also requires openness and transparency, so that any interested observer can trace the steps in a calculation and identify the assumptions. A thorough review by an independent party is necessary, to confirm that the assumptions and methodology are appropriate. Assumptions about matters such as equipment availability and operator performance should reflect both historical experience and present conditions in the plant. Uncertainties should be explicitly recognized, and carried forward through all stages of calculation.

The findings of a PRA are typically expressed as numerical indicators, such as the probability of severe core damage or the magnitude of a release of radioactive material. These numerical findings must be viewed with caution. They always have substantial uncertainties, but it is common for PRA reports to express findings as a single number, without acknowledgement of the uncertainties. Where uncertainties are acknowledged, they may not be statistically sound or may not have been propagated through all stages of a calculation. PRAs do not consider gross errors in design, construction, maintenance or operation, although these have occurred. Similarly, acts of malice or insanity are not considered in PRAs, although these have occurred at nuclear facilities. One can never guarantee that all significant accident sequences have been identified. Many important phenomena, especially at the second and third levels of PRA analysis, are not well understood. Thus, the ultimate, numerical findings of a PRA can be misleading. A PRA can be a valuable document, but only in the hands of an informed, sceptical reader.

5. Available studies relating to the Pickering 'A' accident risk

An observer interested in the accident risk posed by the Pickering 'A' reactors can consult a wide variety of technical documents. These include many documents that are specific to the CANDU reactor and, within that group, many documents that are specific to the Pickering Station or its 'A' reactors. As one would expect, most of the CANDU-related documents have been prepared in Canada.

Generic studies on CANDU accident risk issues

Over recent decades, the Canadian nuclear industry and the AECB have conducted many generic studies on technical issues that are pertinent to the accident risk posed by CANDU reactors. Some of these issues are relevant only to design-basis accidents, while others are also relevant to severe accidents.[21] The

[21] See, for example: Snell at al, 1988; Rogers, 1984; Kundurpi et al, 1984; Lau et al, 1984; Lee and Knystautas, 1987.

AECB has recently identified a number of generic, technical issues as "open", meaning that they require ongoing investigation by licensees.[22]

There are grounds for debate about whether the generic studies that have been completed to date provide adequate treatment of the issues they address. Generally, the reports on these studies are useful contributions to a scientific debate, because they are open about their assumptions and methodologies. In many cases, the reports are candid about the limitations of the analyses they describe. Openness of this kind is to be expected in an industry that is science-based. Regrettably, some of the site-specific literature that is related to the safety of the Pickering Station is less open.

The Darlington Probabilistic Safety Evaluation

Ontario Hydro's first PRA was performed for the Darlington Station (a 4-unit CANDU station) and published in 1987.[23] This PRA was conducted for internal accident-initiating events only. It was supposedly a third-level PRA, in the sense that it estimated the offsite consequences of accidents. However, no analyses were conducted beyond the first level for accident sequences that involve severe core damage. Thus, for practical purposes, the Darlington PRA was a first-level PRA.

The Darlington PRA found that the probability of severe core damage is 3.8 per 1 million reactor-years.[24] No uncertainty range was provided for this estimate.[25] This author is not aware of any comprehensive, independent review of the Darlington PRA. A limited review was conducted by IRSS and published in 1992, but the available funds were not sufficient to support a comprehensive review. Nevertheless, IRSS's review identified an accident sequence that was not examined in the PRA. Inclusion of just this one sequence, using the methodology and assumptions of the PRA, raised the probability of severe core damage from 3.8 per 1 million reactor-years to 1.5 per 100 thousand reactor-years.[26]

The probability of a large release of radioactive material was not estimated in the Darlington PRA. A subsequent report by Ontario Hydro made a rough estimate that the probability of a large release is 8.2 per 10 million reactor-years.[27] This probability represents 22 percent of the PRA's estimate of the probability of severe core damage.

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[22] AECB, 1998.

[23] Ontario Hydro, 1987.

[24] In the Darlington PRA, accident sequences leading to severe core damage were placed in fuel damage category FDC0.

[25] The Darlington PRA accounted for uncertainty in only one context -- analyses of the offsite consequences of accidents.

[26] IRSS, 1992.

[27] Ontario Hydro, 1993.

The Pickering 'A' safety report

In 1969, Ontario Hydro completed a "safety report" for the Pickering 'A' reactors.[28] This document has subsequently been updated a number of times. Like other reports of this kind that have been prepared in Canada, the USA and elsewhere, the safety report addresses design-basis accidents but not severe accidents. Within these limitations, the Pickering 'A' safety report is informative and well presented.

The Pickering 'A' risk assessment (PARA)

In 1995, Ontario Hydro completed a PRA for the Pickering 'A' Station.[29] Hereafter, this PRA is identified by the acronym PARA. The PARA study was conducted for internal accident-initiating events only. It was supposedly a third-level PRA, in the sense that it estimated the offsite consequences of accidents. In fact, it was primarily a first-level PRA with some limited analyses at the second and third levels. PARA went beyond the Darlington PRA in the sense that it examined the response of containment to accident sequences that involve severe core damage, and it estimated the probability of a large release of radioactive material. However, it did not estimate the magnitudes or other characteristics of large releases, or their offsite consequences. Tables 3 through 5 summarize the major findings of PARA.

Characteristics of the fuel damage categories used in PARA are shown in Table 3. One sees that fuel damage categories FDC1 and FDC2 both involve severe core damage, while categories FDC3 through FDC9 involve lesser amounts of core damage. Fuel damage category FDC2 was estimated to have a relatively high probability -- 1.3 per 10 thousand reactor-years.[30]

For accident sequences leading to severe core damage (fuel damage categories FDC1 and FDC2), PARA identified six consecutive states of core damage. Table 4 summarizes the characteristics of those states. One sees from Table 4 that large fractions of the core inventory of volatile aerosols are estimated to be released into the containment as core damage progresses. For example, Table 4 shows that 60 percent of the core inventory of volatile aerosols will be released into the containment on completion of core damage state CDS2, which can occur within the first few hours of the accident. Cesium-137, discussed in Section 4, above, is one of the volatile aerosols.

PARA identified two ex-plant release categories (EPRC1 and EPRC2) that involve large releases to the environment, and five categories (EPRC3 through EPRC7)

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[28] Ontario Hydro, 1969.
[29] Ontario Hydro, 1995.
[30] PARA presented its probability findings on a per-year basis. For the purposes of the discussion above, PARA's probability findings are regarded as being per reactor-year. See the notes to Tables 3 and 5 for further discussion.

that involve smaller releases. Table 5 summarizes the estimated characteristics of the various ex-plant release categories. One sees that PARA did not estimate the source terms -- magnitudes of radioactive release -- for release categories EPRC1 and EPRC2. Thus, PARA did not fulfil the requirements for a second- or third-level PRA. From Table 5 one sees that PARA estimated the probability of a large release (the sum of the EPRC1 and EPRC2 probabilities) at 6.3 per 1 billion reactor-years.[31] This is a very low probability in the context of nuclear reactor PRAs.

PARA did make an attempt to propagate uncertainty through the steps of its calculations. Thus, for example, it found that the mean probability of severe core damage is 1.3 per 10 thousand reactor-years while the 95th and 5th percentile probabilities are 3.4 per 10 thousand reactor-years and 4.7 per 100 thousand reactor-years, respectively. This is a very narrow uncertainty range, in view of the number of steps in the calculations and the nature of the phenomena involved.

Reviews of PARA

The AECB has reviewed PARA, drawing upon the expertise of AECB staff and consultants. Their review found that PARA is useful for qualitative purposes but that confidence in its numerical results is "low". Other deficiencies were also identified. The review stated:[32]

> "As stated in several reports and presentations to Ontario Hydro and by the independent external contractors, a rework of PARA is deemed necessary in order to correct PARA deficiencies. This rework should be such that it corrects, as much as possible, conservatism and optimism, and produces a model that can be modified and computed quickly by plant staff. It could so be used in plant operation and for regulatory decisions."

Among the qualitative lessons that the AECB learned by reviewing PARA, at least three are important for decisions about reactor restart. First, safety-related systems at Pickering 'A' have a higher degree of interdependence (cross links) than is characteristic of later CANDU reactors, leading to a comparatively high probability of severe core damage.[33] Second, there are many accident sequences in which, following a design-basis initiating event, a single component failure or human error leads to severe core damage. This situation is contrary to the AECB's single failure principle. Third, in many sequences there is a high degree of reliance on operator actions to prevent severe core damage following a design-basis initiating event.

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[31] Comments in the preceding footnote, regarding probabilities per year or per reactor-year, also apply here.

[32] Parsons, 1999.

[33] For example, there is interdependence between moderator cooling and the cooling of emergency coolant injection (ECI) water when the ECI system is used in the recirculatory mode.

Each of these lessons applies to the plant itself, and is a separate issue from the deficiencies in PARA. Thus, the AECB staff requested a two-step process by the licensee: (a) correction of the deficiencies in PARA; followed by (b) use of the reworked PARA to correct safety-related weaknesses in the Pickering 'A' Station. In the words of the AECB:[34]

> "The AECB staff requests that PARA be improved and its results used for correcting Pickering A weaknesses. In particular, correction of the cross links and of the identified single failure non-compliant features should be a prerequisite to any restart of Pickering A."

This author had access to two reviews of specific aspects of PARA, prepared by consultants to the AECB. These reviews were, presumably, reflected in the abovementioned AECB review. One of the consultant reviews examined the fuel damage categories used in PARA.[35] The other, by the same authors, examined PARA's use of ex-plant release categories.[36] Both reports comment on the difficulty of interpreting PARA, and both conclude that the uncertainty in PARA's findings may be substantially greater than is indicated in PARA itself.

Seismic margin assessment for Pickering 'A'

PARA did not consider external initiating events, such as earthquakes. As an alternative, Ontario Hydro conducted a seismic margin assessment for the Pickering 'A' reactors. This was published in 1998.[37] A seismic margin assessment is a simpler and cheaper analysis than a seismic PRA. Accordingly, it provides less information about a plant's potential to experience an earthquake-induced accident.

The Pickering 'A' seismic margin assessment identified a "review level earthquake" with a probability of 1 per 10 thousand years and a peak ground acceleration of 0.235g (this parameter combination being determined at the 84th percentile confidence level). The Pickering 'A' Station was then examined to determine its ability to ride out the review level earthquake without suffering a severe core damage accident.

This analytic approach is more limited than the approach used in a seismic PRA. The seismic PRA approach involves three basic steps. First, a set of 'seismic hazard curves' is developed for the site in question, showing the annual probabilities of earthquakes of differing intensities (ground accelerations), at a range of confidence levels. Second, a set of 'fragility curves' is developed, showing the conditional probabilities of specified outcomes (e.g., severe core

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[34] Parsons, 1999.
[35] Archinoff and Reid, 1996.
[36] Archinoff and Reid, 1998.
[37] Brown et al, 1998.

damage) for earthquakes of differing intensities, at a range of confidence levels. Third, the findings of the first two steps are combined, to show the annual probabilities of the specified outcomes.

In the Pickering 'A' seismic margin assessment, the station's ability to ride out the review level earthquake was determined by examining the components that must function during and after the earthquake if the plant is to follow a "success path" from full-power operation to a safe, shutdown state. These components serve functions such as reactor shutdown, fuel cooling, and containment integrity. The assessment determined that there is high confidence that each needed component would ride out the review level earthquake, thus achieving the success path (and avoiding a severe core damage accident).

Other perspectives on seismic risk

A review of the Pickering 'A' seismic margin assessment was performed by Acres International.[38] The Acres review pointed out that achievement of the success path would require the correct implementation of a variety of operator actions. For example, if the earthquake causes a small LOCA, up to 14 operator actions would be required to remain on the success path. Yet, many components and instruments not in the success path could fail as a result of the earthquake, leading to a large number of alarm indications and a stressful work environment. Thus, correct implementation of the necessary operator actions cannot be assured. To accommodate such concerns, a seismic risk assessment should be integrated with an internal-events PRA. This has not been done for Pickering 'A'.

The Acres review expressed concern that Ontario Hydro's seismic margin assessment did not adequately examine the behavior of the pressure relief duct during the review level earthquake. According to the Acres review, surface waves in the duct could lead to forces and displacements larger than those estimated by Ontario Hydro, potentially causing pounding of expansion joints and threatening the integrity of the emergency coolant injection piping where it enters the reactor building.[39]

An analysis of the seismic fragility of the Pickering Station pressure relief duct was sponsored by the AECB and completed in 1995.[40] That analysis predicted collapse of the duct in the transverse direction at a peak ground acceleration of 0.22g (i.e., for an earthquake less severe than Ontario Hydro's review level earthquake). The effects of a collapsing duct on other safety-related systems, such as emergency coolant injection, were not considered.

Thus, there are grounds to doubt that Ontario Hydro's seismic margin assessment correctly assessed the ability of the Pickering 'A' Station to ride out the review level earthquake. There is also an ongoing debate about the seismic hazard at the Pickering site.[41] Some experts argue that the western Lake Ontario region, and the vicinity of the Pickering site in particular, could have the potential for a more severe earthquake than is generally expected. This minority view deserves careful consideration, because seismic hazard assessment inevitably relies on subjective interpretation of data.[42] Analytic techniques have been developed for combining the subjective interpretations of experts.[43] It is not clear whether such techniques have been used in the context of the Pickering site.

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[38] Acres, 1999.
[39] Ibid.
[40] Ghobarah, 1995.
[41] See, for example: Asmis, 2000; Wallach et al, 1998; AECB, 1995.
[42] J Carl Stepp, in AECB, 1995.
[43] Budnitz et al, 1997.

The significance of fires at Pickering 'A'

Analytic techniques have been developed that allow the consideration of fires as accident-initiating events in PRAs. These techniques have been used in many PRA studies for LWRs. It appears, however, that these techniques have not been applied to Ontario Hydro's CANDU stations. Thus, at Pickering 'A', the contribution of fires to the probability of severe core damage is unknown.

The lack of assessment of the risk implications of fires is especially regrettable because the designers of the Ontario Hydro nuclear stations took an unusual approach to fire suppression. Typical practice in nuclear plant design is to use comparatively small compartments surrounded by rated fire barriers, with systems for fire detection and suppression. The Ontario Hydro stations employ a different approach. Thus:[44]

> "...OHN [Ontario Hydro Nuclear] CANDU units are characterized by expansive openness, the design intent of which was to facilitate access for manual fire-fighting which is the prevailing method of fire suppression, containment and control. This is not a design concept that would necessarily be repeated in the future if fire protection was a dominant consideration."
>
> and
>
> "Suppression system coverage within the OHN units is minimal. Essentially the entire plant fire-fighting strategy is predicated on manual response to fires."
>
> and
>
> "One example from that plant [Bruce 'A'] observed by the Team was the 693' elevation Main Office Area, which sits immediately above Main Control Room. This room contains large combustible loads consisting of plastics, fabric-covered dividers, carpet flooring, and paper. It has no fire/smoke detection or automatic suppression system and is not inhabited beyond normal hours of (07:00 to 17:00 hrs)."

The factors affecting fire risk at the Ontario Hydro stations are not limited to plant design. There has been a general lack of attention to the control of fire risk. Thus:[45]

> "There is one overall theme within this FPFI [fire protection functional inspection] -- that there is a lack of understanding of the level of risk to nuclear safety that is posed by fire at OHN's units. The Team feels that fire

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[44] Ontario Hydro, 1997.
[45] Ibid.

> is probably now (and should not be) a very significant contributor to nuclear risk."

On this limited but compelling evidence, it appears that fires could make a significant contribution to the probability of severe core damage at Pickering 'A'. Reliance on manual response to fires could create numerous, comparatively high-probability scenarios for the disabling of safety-related systems. These scenarios could be systematically examined using proven PRA techniques.

6. A comparison with studies performed in the USA

Section 5 has described a variety of studies that are relevant to the accident risk posed by the Pickering 'A' reactors. Two of these studies -- PARA and the seismic margin assessment -- provide direct estimates of risk. Those estimates can usefully be compared with the findings of similar studies performed in the United States for PWR or BWR plants. The comparison must be performed cautiously, because the reactor designs are different and each plant has unique risk characteristics. Nevertheless, there are enough common features that the comparison can be instructive.

Individual plant examinations

The licensee for each commercial nuclear reactor in the USA is obliged to conduct an individual plant examination (IPE) for the reactor. Minimal requirements for an IPE are similar to those for a second-level PRA performed for internal initiating events. All probability estimates in IPEs are for internal initiating events only. Findings from the IPEs are published, and a summary of the findings has been published by the NRC.[46]

For PWRs, the IPE-estimated (central estimate) probabilities of severe core damage range from about 4 per 1 million reactor-years to 4 per 10 thousand reactor-years. For BWRs, the probabilities of severe core damage generally range from 1 per 1 million reactor-years to 1 per 10 thousand reactor-years.[47] The IPE-estimated probabilities of a significant, early release of radioactivity to the environment generally range from 1 per 100 million reactor-years to 3 per 100 thousand reactor years.[48]

External-events PRAs

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[46] NRC, 1997.

[47] The IPEs for two BWRs showed a severe core damage probability of 1 per 10 million reactor-years. All other IPEs showed a severe core damage probability above 1 per 1 million reactor-years.

[48] The IPEs for two BWRs showed an early release probability of 1 per 10 billion reactor-years. All other IPEs showed an early release probability above 1 per 100 million reactor-years.

In the United States, PRAs have been performed for external initiating events that include earthquakes, fires, internal or external floods, high winds, and lightning. A compilation of findings from some of these PRAs shows that the estimated probability of severe core damage from external initiating events is often comparable to, or greater than, the estimated probability of severe core damage from internal initiating events.[49] For these PRAs, earthquakes and fires are typically the major contributors to the core damage probability from external events.

In 1990, the NRC completed a research program in PRA development that involved the performance of third-level, external-events PRAs for five US nuclear plants.[50] These PRAs found that fires and earthquakes make a contribution to the probability of severe core damage that is comparable to the probability attributable to internal events. In its research program, the NRC devoted considerable effort to identifying uncertainties and propagating them through the calculations. The uncertainty distributions for the ultimate findings (e.g., probability of severe core damage) are broad, typically spanning one or more orders of magnitude.

Lessons for Pickering 'A'

IPE and PRA findings in the USA are compatible with PARA findings about the probability of severe core damage at Pickering 'A', for internal initiating events. However, the US findings suggest that PARA has substantially under-estimated the probability of a large release of radioactive material. Also, the US findings suggest that external initiating events at Pickering 'A' might contribute a severe core damage probability comparable to that from internal events.

[49] Chen, 1993.
[50] NRC, 1990.

7. A critical review of the Pickering 'A' studies

Sections 5 and 6 summarize a variety of studies that are relevant to the Pickering 'A' accident risk. Of these studies, two provide direct findings about accident risk. These two studies are: (a) PARA; and (b) the seismic margin assessment. Here, in Section 7, these two studies are critically reviewed, drawing upon the other literature discussed in Sections 5 and 6.

PARA and severe core damage

PARA is the only available direct source of information on the probability of severe core damage at Pickering 'A', for internal initiating events. In evaluating the findings of PARA, one must ask two broad questions. First, does PARA provide high-quality analysis? Second, does the analysis account for actual conditions at Pickering?

The quality of the analysis in PARA has been reviewed by the AECB.[51] That review found a variety of deficiencies, and concluded that confidence in PARA's numerical results is "low". Among the deficiencies which the AECB found in PARA were: (a) use of a mixture of conservative and optimistic assumptions; (b) no modelling of common-cause failure events; (c) the documentation is difficult to understand; (d) limited use of event trees; (e) several initiating events were not analysed by an event tree; and (f) little uncertainty analysis and no sensitivity analysis. However, in spite of these deficiencies, the AECB found (see Section 5) that PARA provided qualitative insights into the risk posed by the Pickering 'A' reactors. These qualitative insights are consistent with PARA's finding of a comparatively high probability of severe core damage.

Whatever the merit of PARA as a piece of analysis, it is clear that the PARA findings to date are for a 'theoretical' station, not for the actual Pickering 'A' Station. Models in PARA were developed in the 1988-1991 period, and most of the assumptions about component reliability reflect an Ontario Hydro database covering the period through 1986.[52] The numerous deficiencies in nuclear operations that were identified by Ontario Hydro in 1997 (see Section 3) were not reflected in PARA.

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[51] Parsons, 1999.
[52] Ibid.

PARA and large releases

PARA is the only available direct source of information on the probability of a large release of radioactive material from Pickering 'A', for internal initiating events. As shown in Table 5, PARA finds a very low probability of such a release -- 6.3 per 1 billion reactor-years. This low probability contrasts with a comparatively high PARA estimate for the probability of severe core damage -- 1.3 per 10 thousand reactor-years. In effect, PARA finds that 0.005 percent of severe core damage accidents would involve a large release. This finding implies two important claims about the Pickering containment: (a) the containment is sufficiently robust to retain its integrity during almost any severe core damage accident; and (b) the probability of a substantial pre-existing opening in the containment (e.g., an open hatch) is very small. Both claims are incompatible with the findings of many PRAs that have been performed for PWRs and BWRs.

It is instructive at this point to consider, in simplified terms, the factors that will determine the magnitude of a release to the atmosphere, given the occurrence of a severe core damage accident. First, larger openings in the containment imply faster, larger releases. Second, earlier openings in the containment imply larger releases (because aerosols released from the damaged reactor fuel will settle out in the containment over time). Third, shorter release paths imply larger releases.

The first and second of these factors, viewed in the context of hydrogen explosions, provides insight into PARA's claim that large releases are improbable. A hydrogen explosion is a mechanism that could potentially create a substantial opening in the Pickering containment. Table 1 shows that the Pickering reactors contain a large inventory of zirconium which, during a severe core damage accident, will react with steam to produce copious amounts of hydrogen. Table 2 shows that the Pickering containment has a low design pressure. Thus, one can postulate that a hydrogen explosion early in the accident sequence will create a large opening in the containment, leading to a large release of radioactive material. PARA, however, finds otherwise.

PARA rules out the possibility of a hydrogen explosion early in the accident sequence, but allows for the occurrence of a hydrogen explosion at a later stage. However, even at that later stage, and assuming that the hydrogen igniters in the containment are inoperative, PARA assumes a probability of only 5 percent that an opening in the containment will occur.[53] Thus, PARA finds that the creation of an opening in the containment through a hydrogen explosion is an improbable event which, if it occurs, will occur late in the accident sequence. This is a highly significant finding in the context of the Pickering 'A' risk profile, and deserves thorough review.

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[53] Electrically-powered hydrogen igniters are located inside the containment, to burn off hydrogen before it reaches an explosive concentration.

A thorough review of the hydrogen explosion issue was beyond the scope of this report. However, Table 6 presents a scoping calculation that is instructive. In this calculation, which could represent a Pickering 'A' electrical blackout, hydrogen generated from four severely damaged reactor cores is uniformly distributed throughout the Pickering containment (including the Pickering 'B' reactor buildings).[54] The calculation shows that the average hydrogen concentration would be in the range where a destructive explosion (potentially a detonation) could occur.[55]

In an actual accident, the concentrations of hydrogen, steam and air in containment would be non-uniform, and would vary over time. Production of hydrogen through the steam-zirconium reaction could be substantially less than is assumed in Table 6. Yet, non-uniformities in concentrations could offset this effect, so that hydrogen concentrations reach the explosive (possibly detonable) range in parts of the containment. Although simple, the calculation presented in Table 6 shows that the potential for a destructive hydrogen explosion cannot be readily dismissed. This potential has a major influence on the risk profile of the Pickering 'A' reactors, and requires a more rigorous analysis than PARA has provided.

Accidents involving more than one reactor

Because reactors at Pickering share safety systems, a thorough PRA must consider the potential for multi-reactor accidents. Contrary to this requirement, PARA does not examine the implications of multi-reactor accidents for containment integrity, despite identifying accident sequences (such as station blackout) that could lead to multi-reactor accidents. A multi-reactor accident could generate a copious amount of hydrogen, as discussed above. In addition, it could generate an amount of steam substantially greater than the amount which the Pickering containment is designed to accommodate. Both phenomena could challenge the integrity of the containment.

Earthquake-initiated accidents

The seismic margin assessment for Pickering 'A' concluded that the reactors could be safely shut down following a review level earthquake. By contrast, other studies (see Section 5) suggest that the reactors could suffer severe core damage following this earthquake, and/or gross containment failure (e.g., collapse of the pressure relief duct).

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[54] Station blackout sequences, or other accident sequences involving inoperative hydrogen igniters, account for a comparatively small fraction of the severe core damage probability identified by PARA. Nevertheless, their collective probability substantially exceeds PARA's estimate for the probability of a large release -- 6.3 per 1 billion reactor-years.

[55] A variety of factors determine the explosive potential of hydrogen-air-steam mixtures. See, for example, Lee and Knystautas, 1989.

Seismic hazard curves typically cover a wide range of probabilities, perhaps from 1 per 10 years to 1 per 10 million years. The peak ground acceleration typically increases with decreasing probability. Thus, Ontario Hydro's review level earthquake (peak ground acceleration of 0.235g, probability of 1 per 10 thousand years) represents a point on a seismic hazard curve. At some lower probability (such as 1 per 100 thousand years), a typical curve would predict an earthquake with a peak ground acceleration exceeding 0.235g. No seismic hazard curve is available for the Pickering site, but it is plausible that such a curve could show an increase in peak ground acceleration by a factor of two or more as probability declines from 1 per 10 thousand years to 1 per 100 thousand years.

If the Pickering 'A' reactors are vulnerable during an earthquake at the 1 per 10 thousand year probability level, they will be significantly more so during an earthquake at the 1 per 100 thousand year probability level. A seismic PRA can systematically explore this effect. Also, it can show the risk implications of various experts' estimates of seismic hazard. A seismic margin assessment cannot provide this depth of information about risk.

Other external initiating events

The role of fires as accident-initiating events is discussed in Section 5. While the contribution of fires to the probability of severe core damage at Pickering 'A' is unknown, there is reason to suspect that this contribution is significant. Well-established PRA techniques could determine this contribution. Note that fires might cause multi-reactor accidents.

A thorough, external-events PRA would examine the risk implications of earthquakes, fires and other potentially significant sources of threat (e.g., aircraft crash). Under current practice, such a PRA will not consider acts of insanity or malice. Yet, the potential for such acts cannot be ignored. Moreover, an act of insanity or malice might simultaneously initiate a severe core damage accident and threaten the integrity of the containment.

Interactions between the Pickering 'A' reactors and other Pickering facilities

A variety of safety-related interactions between the 'A' and 'B' reactors at Pickering are possible. Since these reactors share safety-related systems, they are mutually vulnerable to failures in those systems. Also, an accident at one reactor could expose other reactors to steam, hydrogen and radioactive material, potentially initiating or exacerbating accidents at those reactors. A systematic, site-level PRA could examine all such interactions. It could also examine the potential for non-reactor accidents (e.g., accidents involving spent fuel) that could lead to a significant release of radioactive material.

Summary

PARA and the seismic margin assessment provide an incomplete and in some respects misleading picture of the accident risk posed by the Pickering 'A' reactors. PARA provides useful qualitative insights into the reactors' potential to experience severe core damage, but its estimate of the probability of a large release of radioactive material is not credible. IRSS's interim judgement is that the probability of a large release from Pickering 'A' is likely to exceed 1 per 100 thousand years. Internal events, earthquakes and fires would all contribute to this probability, perhaps at roughly comparable levels.

8. A process for understanding the Pickering 'A' accident risk

Studies published to date do not provide a complete and credible picture of the accident risk posed by the Pickering 'A' Station, either now or after the improvements promised by OPG. To correct this inadequate state of understanding, two consecutive stages of action are recommended by IRSS. First, an analytic model should be developed, whereby the accident risk can be determined for any given condition of the Station. Second, this model should be applied to the Station assuming : (a) its present condition; and (b) its condition after the implementation of the improvements promised by OPG.

These two stages of action would be similar to the two-step process called for by the AECB staff in its review of PARA.[56] However, the recommended analytic model would have a wider scope than the reworked PARA that was requested by the AECB staff. Also, the AECB staff made a general request that the results of the reworked PARA be used to correct weaknesses in Pickering 'A'. Since that request was made, OPG has proposed specific improvements costing about 1 billion dollars. The risk implications of those improvements should be examined by using the analytic model.

The new analytic model would be a 'full-scope' third-level PRA. It would consider internal and external initiating events, would evaluate offsite consequences for all significant accident sequences, would consider potential interactions between the Pickering 'A' and 'B' Stations, and would examine the potential for non-reactor (e.g., spent fuel) accidents at Pickering. This would not be a cheap model to develop, but its cost would be a small fraction of 1 billion dollars. It would allow an informed debate about the risk implications of restarting the Pickering 'A' reactors.

The new PRA could draw upon previous studies. Its internal-events component could be a reworked PARA, as requested by the AECB staff.[57] This component would be integrated with an entirely new external-events PRA. Additional modules would cover interactions between the Pickering 'A' and 'B' Stations, and non-reactor accidents.

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[56] Parsons, 1999.

[57] Ibid.

The environmental assessment for Pickering 'A' restart

The two stages of action, as recommended by IRSS, should be factored into the environmental assessment for Pickering 'A' restart. At present, the environmental assessment process is not well informed about accident risk. The CNSC staff have repeated OPG claims without subjecting them to critical review. As an example, the CNSC staff have recently stated:[58]

> "A severe reactor accident with loss of containment is not a realistic scenario, and not appropriate for the purposes of the environmental assessment."

That statement is not supported by technical analysis, and is contradicted by an extensive body of evidence which is presented in this report.

9. Conclusions & recommendations

This review has led IRSS to the following major conclusions:

C1. OPG's claims about the accident risk posed by the Pickering 'A' Station rest upon analyses (PARA, seismic margin assessment) that apply to a 'theoretical' station which does not have operational deficiencies. In practice, the Pickering Station has experienced many operational deficiencies.

C2. PARA (the Pickering 'A' risk assessment) estimated the probability of a severe core damage accident to be 1.3 per 10 thousand reactor-years, which exceeds OPG's probability goal by an order of magnitude. Although confidence in PARA's numerical findings is low, this probability estimate is consistent with other studies that consider 'internal' accident-initiating events. This estimate implies that the probability of a severe core damage accident at Pickering 'A' over the future life of the restarted reactors would be about one percent.

C3. A seismic margin assessment has concluded that the Pickering 'A' reactors could be safely shut down following an earthquake with a probability of 1 per 10 thousand years. By contrast, other studies suggest that the reactors could suffer severe core damage and/or containment failure following such an earthquake. In addition, there is scientific dispute about the severity of earthquakes that could be experienced at Pickering.

C4. Analytic techniques are available for estimating the probabilities of reactor accidents due to earthquakes, fires or other 'external' accident-initiating events. These techniques have not been used for Pickering 'A'.

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[58] CNSC, 2000.

C5. PARA estimated the probability of a large (severe) release of radioactive material from Pickering 'A' to the environment to be 6.3 per 1 billion reactor-years. This low probability is not credible. The actual probability of a large release is likely to exceed 1 per 100 thousand years, thereby exceeding OPG's own probability limit. Additional investigations are needed to determine the probability. A release could potentially involve all eight reactors at Pickering, leading to substantial offsite consequences.

C6. OPG has promised to make improvements at the Pickering 'A' Station that will correct some of the Station's past operational deficiencies and will reduce the probability of severe core damage. OPG has not provided an evaluation of the effects of the promised improvements on the Station's risk profile.

C7. The CNSC staff have recently stated that a severe core damage accident at Pickering 'A', leading to a large release of radioactive material, "is not a realistic scenario, and not appropriate for the purposes of the environmental assessment". This statement is not supported by technical analysis, appears to be based on an uncritical acceptance of OPG claims, and is contradicted by evidence presented in this report.

C8. Conclusions C1 through C7 show that OPG and the CNSC have not established the basis for an informed public debate about the risk implications of restarting the Pickering 'A' reactors.

Drawing from these conclusions, IRSS offers the following recommendations for action by OPG, the CNSC, and the Senate Standing Committee on Energy, Environment and Natural Resources:

R1. PARA should be reworked by OPG to meet the specifications set forth by the AECB staff. The reworked PARA should employ data that reflect the actual condition of the Pickering Station.

R2. An 'external-events' PRA should be prepared by OPG in conjunction with the reworking of PARA. The external-events PRA and the reworked PARA should be integrated as a 'full-scope' third-level PRA. The new, integrated PRA should consider potential interactions between the Pickering 'A' and 'B' Stations.

R3. The new, integrated PRA should (unlike PARA) evaluate the source terms and offsite consequences for all significant accident sequences. Uncertainties should be identified and propagated through the calculations. Calculations regarding containment behavior and source term phenomena should be accompanied by sensitivity analyses.

R4. The safety improvements promised by OPG should be evaluated using the new PRA, to determine the implications of these improvements for the risk profile of the Pickering Station.

R5. The CNSC should arrange for a thorough, independent, open review of the new PRA, and of its use to evaluate the risk implications of OPG's promised safety improvements. These studies should then be reworked, to incorporate the findings of the review.

R6. After completion of the tasks described in R5, the CNSC should incorporate the complete findings of the new PRA, and the evaluation of OPG's promised safety improvements, into the environmental assessment for the proposed restart of the Pickering 'A' reactors.

R7. Completion by OPG and the CNSC of the tasks specified in recommendations R1 through R6 should be a prerequisite for any decision on restarting the Pickering 'A' reactors.

R8. The Senate Standing Committee should seek to ensure that OPG and the CNSC perform all future analyses with the openness and thoroughness that should characterize a science-based industry whose facilities can cause severe offsite harm.

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Goordon Thompson, "Severe Accident Potential of CANDU Reactors", a consultant's report published in Hare et al, 1988.

(Wallach et al, 1998)
J L Wallach, A A Mohajer and R L Thomas, "Linear zones, seismicity, and the possibility of a major earthquake in the intraplate western Lake Ontario area of eastern North America", Can. J. Earth Sci., volume 35, pp 762-786, 1998.

(All quantities in kg/MWt)

	PWR (Zion)	BWR (Grand Gulf)	CANDU (Pickering B)
Reactor Coolant System Water Inventory	76.1	85.5	69.0[b]
Moderator Water Inventory	---	---	152.1[b]
UO_2 in Core	30.3	43.4	60.0
Zirconium Alloy	6.2	20.7	23.2[c]

Notes

(a) Adapted from Exhibit 1 of Thompson, 1988.
(b) Heavy water.
(c) Zirconium alloy in fuel bundles (5.9 kg/MWt), in pressure tubes (12.5 kg/MWt), and in calandria tubes (4.8 kg/MWt).

TABLE 1

WATER INVENTORIES AND CORE MASSES:
TYPICAL PWR, BWR AND CANDU PLANTS

	Ratio of Free Volume to Reactor Power (m^3/MWt)	Design Pressure (psig)
BWR, Mark I (Peach Bottom)		
drywell	1.4	56
wetwell	1.0	56
PWR, Large Dry Containment (Zion)		
containment	23.6	47
CANDU (Pickering B)		
reactor building	29.1	6
pressure relief duct[b]	17.7	6
vacuum building	46.9	-15/+0

Notes

(a) Adapted from Exhibit 6 of Thompson, 1988.
(b) This duct connects eight reactor buildings to the single vacuum building.

TABLE 2

CONTAINMENT VOLUMES AND DESIGN PRESSURES: TYPICAL BWR, PWR AND CANDU PLANTS

Event Sequences Involving Severe Core Damage
(i.e., the core loses its structural integrity)

FDC1
(Estimated Frequency: 5.0 per 10 million years)

Fuel damage category FDC1 involves rapid loss of the structural integrity of the core. Event sequences in this category involve a rapid reactivity excursion (at least 1 mk/s) followed by failure to shut down the reactor. The reactivity excursion could arise from a large loss-of-coolant accident (LOCA).

FDC2
(Estimated Frequency: 1.3 per 10 thousand years)

Fuel damage category FDC2 involves loss of the structural integrity of the core, but over a longer period than applies for FDC1. The onset of significant core damage might occur within 0.6-36 hours after accident initiation. Event sequences in this category involve failure to remove decay heat from the core.

Notes

(a) All items of information presented here, including estimated accident frequencies, are from Ontario Hydro, 1995.
(b) The usual practice in nuclear risk assessment is to show estimated accident frequencies per reactor-year (RY). Ontario Hydro has presented its findings on a per-year basis. Most of the event sequences addressed by Ontario Hydro involve only one reactor. For these sequences, the two presentations are identical. However, some event sequences involve more than one reactor. In those cases, the per-year presentation is the correct one.

TABLE 3 (page 1 of 3)

FUEL DAMAGE CATEGORIES IDENTIFIED IN
ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT

Event Sequences Not Involving Severe Core Damage
(i.e., the core retains its structural integrity)

FDC3
(Estimated Frequency: 7.4 per 1 million years)

Fuel damage category FDC3 contains those event sequences in which high pressure emergency coolant injection (HPECI) fails after a small (LOCA2) or large (LOCA3 or LOCA4) LOCA. The core retains its structural integrity. Moderator cooling is successfully provided (otherwise an FDC2 event sequence would result). Event sequences involving HPECI failure after a very small (LOCA1) LOCA are in category FDC4.

FDC4
(Estimated Frequency: 1.2 per 100 thousand years)

Fuel damage category FDC4 contains three types of event sequence: (i) sequences involving HPECI failure after a very small LOCA; (ii) sequences involving failure of main and auxiliary boiler feedwater supply, the shutdown cooling system and emergency boiler feedwater supply; and (iii) failure of emergency coolant injection in the recirculatory mode, after a LOCA of size LOCA2 or greater, but with successful continuation of moderator cooling.

FDC5
(Estimated Frequency: 4.0 per 100 thousand years)

Fuel damage category FDC5 contains event sequences involving failure of emergency coolant injection in the recirculatory mode, after a very small (LOCA1) LOCA, but with successful continuation of moderator cooling.

TABLE 3 (page 2 of 3)

FUEL DAMAGE CATEGORIES IDENTIFIED IN
ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT

<u>FDC6</u>
(Estimated Frequency: 2.3 per 100 thousand years)

Fuel damage category FDC6 contains event sequences in which: (i) rapid fuel damage occurs in some fuel channels due to flow stagnation in those channels after a large LOCA, despite successful operation of HPECI; or (ii) equivalent damage occurs through other sequences. Moderator cooling is successfully provided in all sequences. Event sequences in the second category include failure to shut down the reactor following a transient or a slow (less than 1 mk/s) insertion of positive reactivity.

<u>FDC7</u>
(Estimated Frequency: 7.5 per 10 thousand years)

Fuel damage category FDC7 contains event sequences in which the fuel bundles in one channel experience overheating, while the remainder of the core is well cooled. An increase in containment pressure occurs in all sequences.

<u>FDC8</u>
(Estimated Frequency: 7.0 per 1 thousand years)

Fuel damage category FDC8 contains event sequences similar to those in FDC7, except that no increase in containment pressure occurs.

<u>FDC9</u>
(Estimated Frequency: 2.6 per 100 years)

Fuel damage category FDC9 contains event sequences in which HPECI is initiated but no fuel damage occurs.

TABLE 3 (page 3 of 3)

FUEL DAMAGE CATEGORIES IDENTIFIED IN ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT

(The following core damage states (CDSs) are consecutive.)

<u>CDS1: Fuel Heats Up</u>

Fuel heats up within the fuel channels due to a lack of cooling. Moderator (heavy water) leaves the calandria due to evaporation, leakage, dumping or expulsion. In FDC1 event sequences, CDS1 begins at the time of accident initiation. In FDC2 event sequences, CDS1 begins 0.6-36 hours after accident initiation. The estimated cumulative release of fission products from the core into the containment on completion of CDS1 includes 25 percent of noble gases and 0.25 percent of volatile aerosols.

<u>CDS2: Core Disassembly</u>

This core damage state begins when the first fuel channel disassembles and ends when the calandria vessel fails and starts releasing the core materials into the calandria vault. In FDC1 event sequences, CDS2 begins 0.25 hours after accident initiation. In FDC2 event sequences, CDS2 begins 1.3-38 hours after accident initiation. The estimated cumulative release of fission products from the core into the containment on completion of CDS2 includes 65 percent of noble gases and 60 percent of volatile aerosols.

<u>CDS3: Calandria Vessel Failure</u>

This core damage state begins when the calandria vessel fails and releases predominantly solid debris into the calandria vault. It ends when all debris is displaced, or when core-concrete interaction commences in the calandria vault. In FDC1 event sequences, CDS3 begins 1.1-5.6 hours after accident initiation. In FDC2 event sequences, CDS3 begins 2.7-43 hours after accident initiation. The estimated cumulative release of fission products from the core into the containment on completion of CDS3 includes 100 percent of noble gases and 90 percent of volatile aerosols.

Note: All information presented here is from Ontario Hydro, 1995.

TABLE 4 (page 1 of 2)

CORE DAMAGE STATES IDENTIFIED IN ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT, FOR EVENT SEQUENCES INVOLVING SEVERE CORE DAMAGE

<u>CDS4: Initial Corium-Concrete Interaction</u>

At the end of CDS3, the corium in the calandria vault reaches its melting temperature and all subsequent decay power is assumed to go into the latent heat of melting of corium. It is assumed that the dump tank's supporting structure collapses through erosion when 50 percent of the corium becomes molten, and this ends CDS4 (see below for exceptions). In FDC1 event sequences, CDS4 begins 1.9-7.9 hours after accident initiation. In FDC2 event sequences, CDS4 begins 4-54 hours after accident initiation. The estimated cumulative release of fission products from the core into the containment on completion of CDS4 includes 100 percent of noble gases and volatile aerosols.

<u>CDS5: Molten Debris Quenching</u>

If the dump tank contains water when it supporting structure collapses, the molten debris in the calandria vault is quenched, and reheats only after the water is evaporated. If there is no water in the dump tank when its supporting structure collapses, then the corium-concrete interaction started in CDS4 continues, and this accident does not pass through states CDS5 and CDS6. In FDC1 event sequences, CDS5 begins 2.2-9 hours after accident initiation. In FDC2 event sequences, CDS5 begins 5.5-64 hours after accident initiation. The estimated cumulative release of fission products from the core into the containment on completion of CDS5 includes 100 percent of noble gases and volatile aerosols.

<u>CDS6: Final Corium-Concrete Interaction</u>

After quenching, the mixture of corium and dump tank debris may remain solid, or may re-melt and subsequently freeze after a further period of interaction with concrete in the calandria vault. In FDC1 event sequences, CDS6 begins 14-29 hours after accident initiation. In FDC2 event sequences, CDS6 begins 22-88 hours after accident initiation. The estimated cumulative release of fission products from the core into the containment on completion of CDS6 includes 100 percent of noble gases and volatile aerosols.

TABLE 4 (page 2 of 2)

**CORE DAMAGE STATES IDENTIFIED IN
ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT,
FOR EVENT SEQUENCES INVOLVING SEVERE CORE DAMAGE**

Large Releases

EPRC1
(Estimated Frequency: 4.0 per 10 billion years)
(Estimated Source Term: not estimated by Ontario Hydro)

Ex-plant release category EPRC1 involves a large, unfiltered release from containment in the period 0-24 hours after accident initiation. The release occurs through a pre-existing opening in the containment envelope.

EPRC2
(Estimated Frequency: 5.9 per 1 billion years)
(Estimated Source Term: not estimated by Ontario Hydro)

Ex-plant release category EPRC2 involves a large, unfiltered release from containment in the period 6-24 hours after accident initiation. Many of the event sequences in EPRC2 are similar to those in EPRC1 except that there is no pre-existing opening in the containment envelope.

Notes

(a) All items of information presented here, including estimated accident frequencies, are from Ontario Hydro, 1995.
(b) The usual practice in nuclear risk assessment is to show estimated accident frequencies per reactor-year (RY). Ontario Hydro has presented its findings on a per-year basis. Most of the event sequences addressed by Ontario Hydro involve only one reactor. For these sequences, the two presentations are identical. However, some event sequences involve more than one reactor. In those cases, the per-year presentation is the correct one.

TABLE 5 (page 1 of 3)

EX-PLANT RELEASE CATEGORIES IDENTIFIED IN ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT

Smaller Releases

EPRC3
(Estimated Frequency: 9.4 per 100 million years)
(Estimated Source Term: 100 percent of noble gases; 0.2 percent of volatile aerosols; lesser amounts of other species)

Ex-plant release category EPRC3 involves an unfiltered release from containment in the period 1 day to 1 month after accident initiation. Many of the event sequences in EPRC3 involve a late containment failure due to a hydrogen explosion.

EPRC4
(Estimated Frequency: 2.2 per 100 million years)
(Estimated Source Term: 100 percent of noble gases; 0.7 percent of volatile aerosols; lesser amounts of other species)

Ex-plant release category EPRC4 involves a release from containment in the period 0-6 hours after accident initiation. The release occurs through a pre-existing opening in the containment envelope. Many of the event sequences in EPRC4 are similar to those in EPRC1, but the release is smaller because some of the containment subsystems are operational, mitigating the driving forces and filtering the release.

EPRC5
(Estimated Frequency: 1.8 per 100 million years)
(Estimated Source Term: 100 percent of noble gases; 0.1 percent of volatile aerosols; no release of other species)

Ex-plant release category EPRC5 involves a release pathway through the heat transport system to the external environment, bypassing the containment. Typical release pathways involve failure of steam generator tubes, or blowback through the emergency coolant injection system.

TABLE 5 (page 2 of 3)

EX-PLANT RELEASE CATEGORIES IDENTIFIED IN ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT

Smaller Releases (continued)

EPRC6
(Estimated Frequency: 2.6 per 10 million years)
(Estimated Source Term: 100 percent of noble gases; 0.03 percent of volatile aerosols; lesser amounts of other species)

Ex-plant release category EPRC6 contains event sequences in which the release occurs early, within a few seconds of the initiation of the accident, and is of short duration. A representative event sequence involves a large LOCA with subsequent failure to shut down the reactor, resulting in a large power excursion, but with all containment subsystems available.

EPRC7
(Estimated Frequency: 1.3 per 10 thousand years)
(Estimated Source Term: 100 percent of noble gases; no release of other species)

Ex-plant release category EPRC7 contains event sequences in which severe core damage occurs with the containment intact and all its subsystems available.

TABLE 5 (page 3 of 3)

EX-PLANT RELEASE CATEGORIES IDENTIFIED IN ONTARIO HYDRO'S PICKERING 'A' RISK ASSESSMENT

<u>Free Volume Inside Containment Envelope</u>: (8 reactor buildings @ 51,000 m^3, 1 pressure relief duct @ 31,000 m^3, 1 vacuum building @ 82,000 m^3)	521,000 m^3
<u>Zirconium Alloy Available for Zirc-Water Reaction:</u> (4 reactors, each having a zirconium alloy inventory of 10.3 Mg (fuel), 21.9 Mg (pressure tubes) and 8.5 Mg (calandria tubes))	163 Mg
<u>Potential Production of Hydrogen</u> (H_2 or D_2): (assuming complete reaction of 163 Mg of zirconium)	3,580 kg-mole
<u>Average Mole Fraction of Hydrogen in Containment:</u> (assuming dispersal of 3,580 kg-mole of hydrogen throughout the containment envelope, pressure of 1 atmosphere, temperature of 60 degrees C)	19 percent
<u>Average Mole Fraction of Steam in Containment:</u> (assumptions as above)	20 percent
<u>Average Mole Fraction of Air in Containment:</u> (assumptions as above)	61 percent

Notes

(a) Differences in the properties of normal and heavy steam are neglected.
(b) The underlying accident could be a Pickering 'A' blackout, leading to severe core damage at all four Pickering 'A' reactors, with hydrogen igniters unavailable.
(c) This calculation is highly simplified, and does not purport to represent actual containment conditions.

TABLE 6

A SIMPLIFIED, ILLUSTRATIVE CALCULATION OF THE POTENTIAL FOR HYDROGEN ACCUMULATION IN THE PICKERING CONTAINMENT FOLLOWING SEVERE CORE DAMAGE