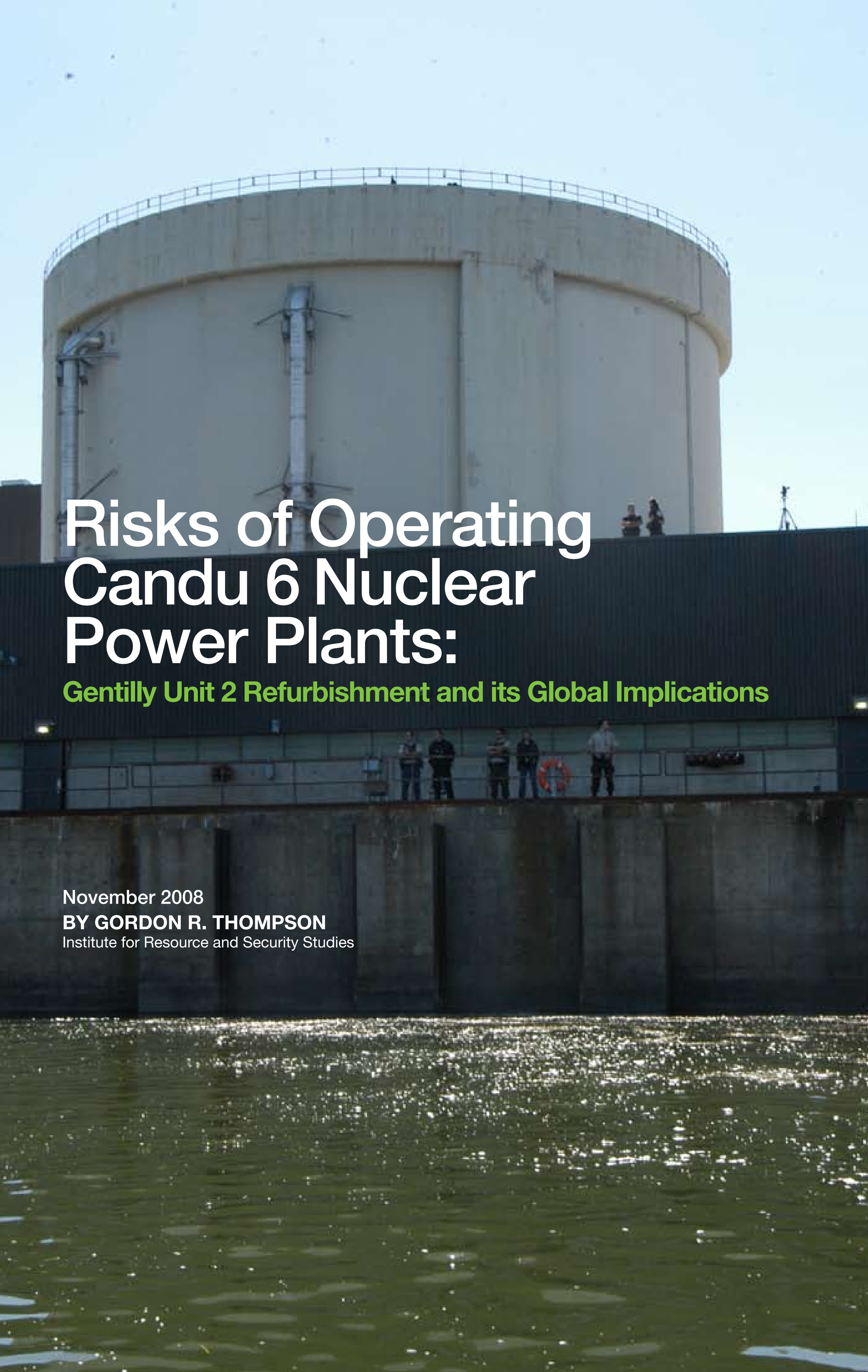

Risks of Operating Candu 6 Nuclear Power Plants:

Gentilly Unit 2 Refurbishment and its Global Implications

November 2008

BY GORDON R. THOMPSON
Institute for Resource and Security Studies

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Prepared under the sponsorship of
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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.

ABOUT THE INSTITUTE FOR RESOURCE AND SECURITY STUDIES

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Gordon R. Thompson is the executive director of IRSS and a research professor at Clark University, Worcester, Massachusetts. He studied and practiced engineering in Australia, and received a doctorate in applied mathematics from Oxford University in 1973, for analyses of plasma undergoing thermonuclear fusion. Dr. Thompson has been based in the USA since 1979. His professional interests encompass a range of technical and policy issues related to sustainability and global human security. He has conducted numerous studies on the environmental and security impacts of nuclear facilities, and on options for reducing those impacts. For example, Dr. Thompson prepared a report in 2000 for the Standing Committee on Energy, Environment and Natural Resources of the Canadian Senate, discussing the accident risk posed by the Pickering A nuclear generating station.

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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]

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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 male-volent 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

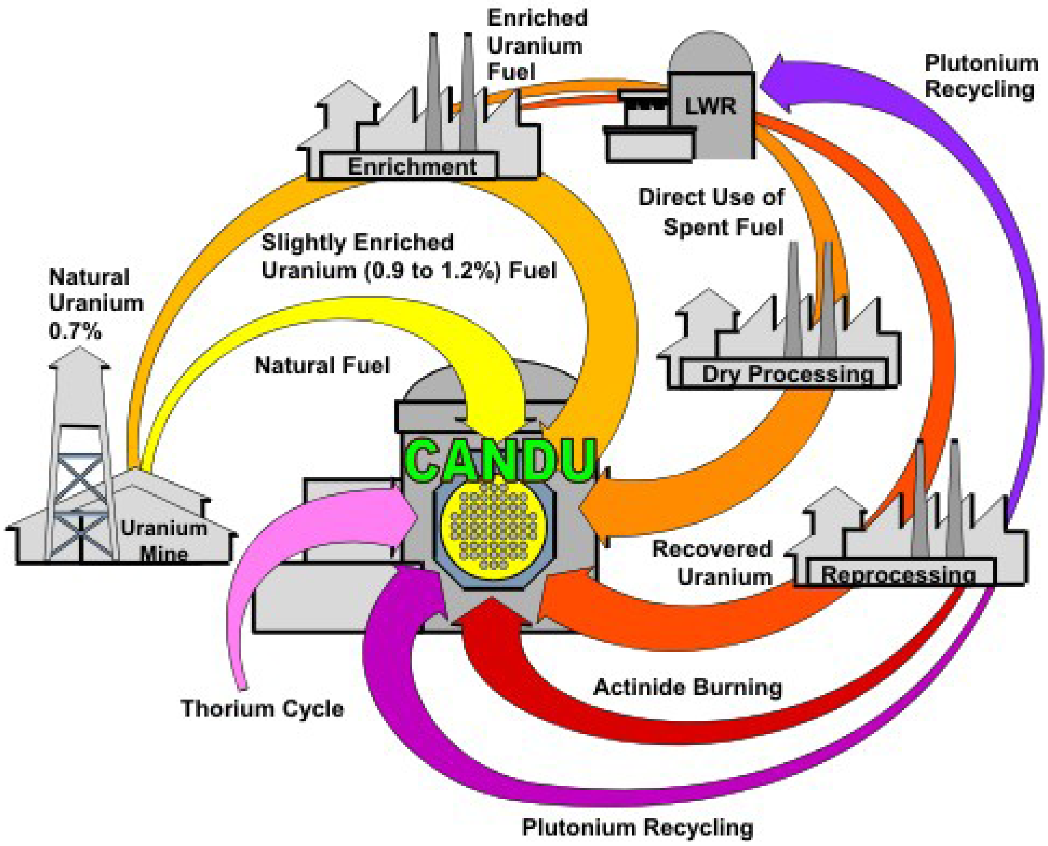
APPLICATION OF CRITERIA	Criteria
Safety performance of the plant during reactor operation (design-basis criteria)	No significant damage of the reactor core or adjacent stored spent fuel in the event of: → Loss of all electrical power (AC & DC), compressed air, other power sources, and normal heat sinks 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

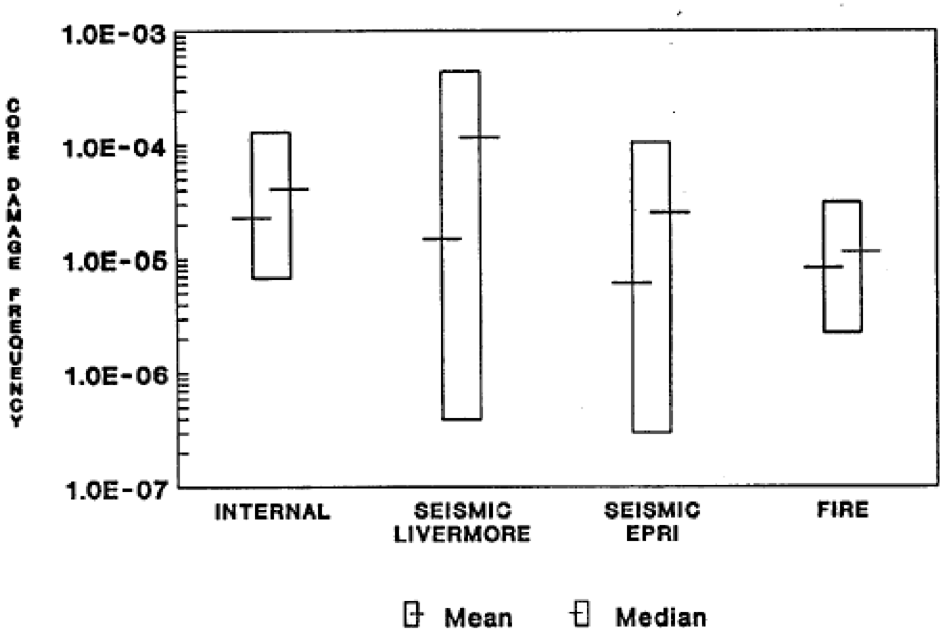
Application of the CANDU Reactor to Various Fuel Cycles, as Envisioned by AECL



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).

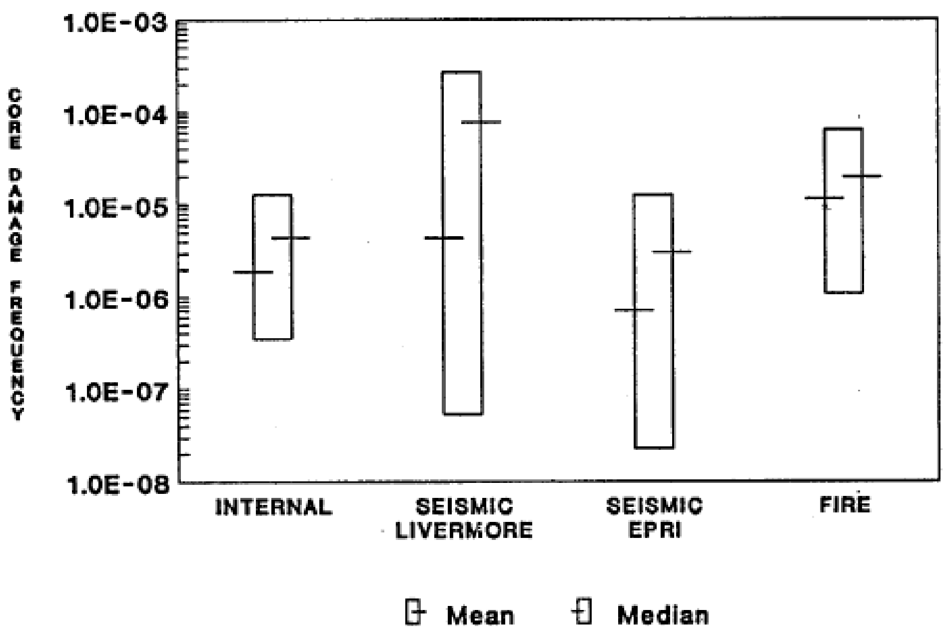
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One is from Lawrence Livermore National Laboratory (Livermore), the other is from the Electric Power Research Institute (EPRI).

(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).

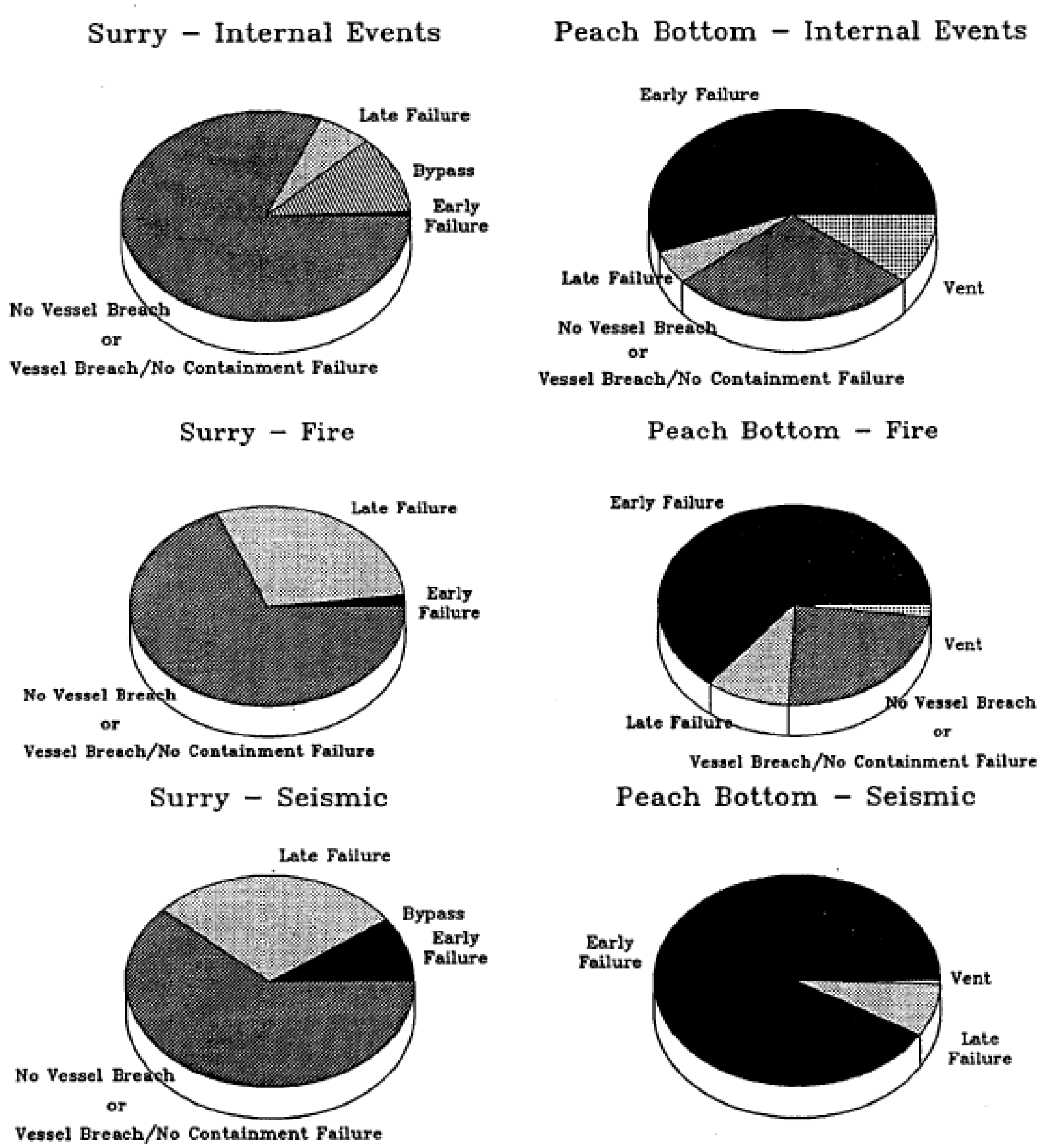
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One is from Lawrence Livermore National Laboratory (Livermore), the other is from the Electric Power Research Institute (EPRI).

(d) CDFs are not estimated for external initiating events other than earthquakes and fires.

(e) Malevolent acts are not considered.

Figure 3-3

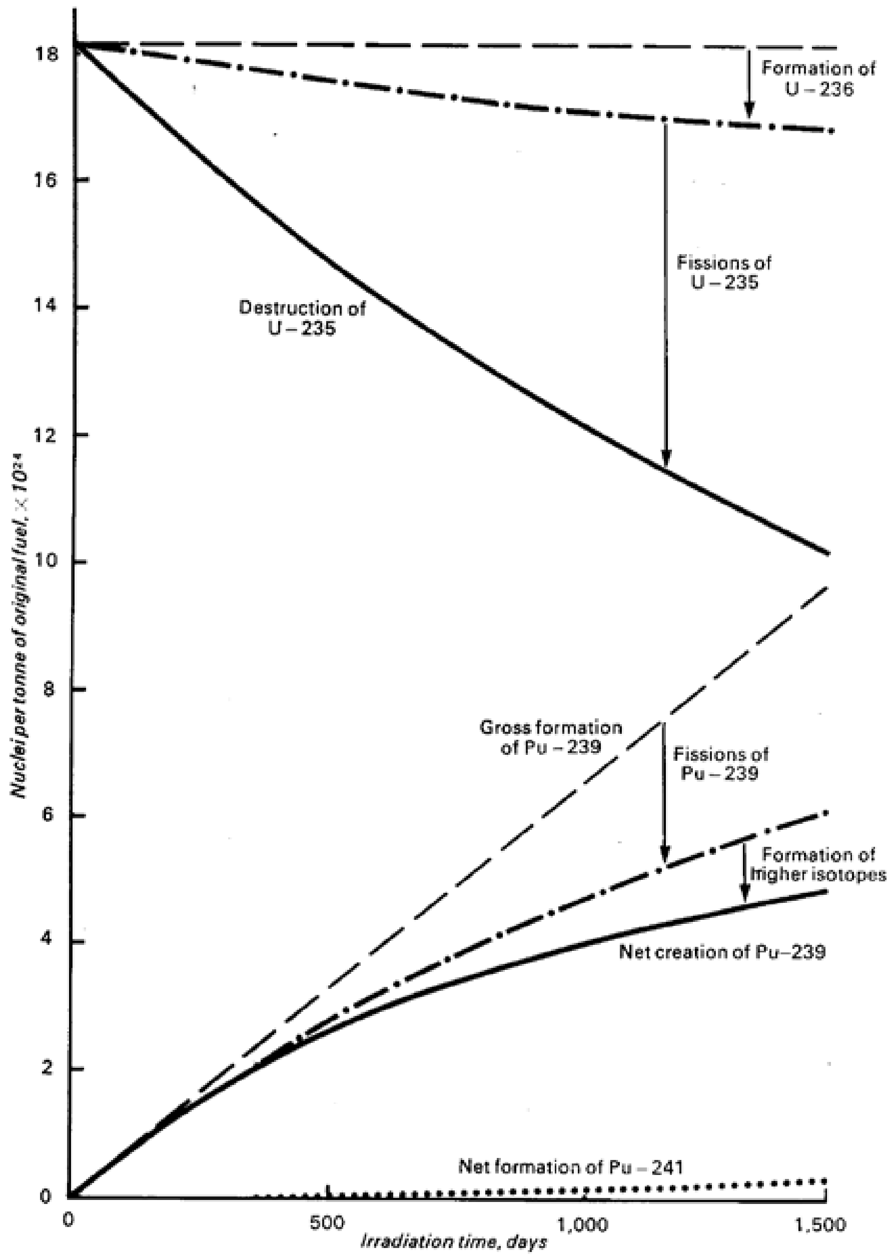
Conditional Probability of Containment Failure Following a Core-Damage Accident at a Surry PWR or Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



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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