A Proposed Structural, Risk-Informed Approach to the Periodicity of CANDU-6 Nuclear Containment Integrated Leak Rate Testing

N. Saliba[1], D. Komljenovic[2], L. Chouinard[1], R. Vaillancourt[2], G. Chrétien[2], V. Gocevski[3]
[1] McGill University, Department of Civil Engineering & Applied Mechanics, Montreal, Canada
[2] Hydro-Québec, Gentilly-2 Nuclear Power Plant, Bécancour, Québec, Canada
[3] Hydro-Québec Équipements, Montréal, Québec, Canada

Abstract

As ultimate lines of defense against leakage of large amounts of radioactive material to the environment in case of major reactor accidents, containments have been monitored through well designed periodic tests to ensure their proper performance. Regulatory organizations have imposed types and frequencies of containment tests based on highly-conservative deterministic approaches, and judgments of knowledgeable experts. Recent developments in the perception and methods of risk evaluation have been applied to rationalize the leakage-rate testing frequencies while maintaining risks within acceptable levels, preserving the integrity of containments, and respecting the defense-in-depth philosophy. The objective of this paper is to introduce a proposed risk-informed decision making framework on the periodicity of nuclear containment ILRTs for CANDU-6 nuclear power plants based on five main decision criteria, namely: 1)- the containment structural integrity; 2)- inputs from PSA Level-2; 3)- the requirements of deterministic safety analyses and defense-in-depth concepts; 4)- the obligations under regulatory and standard requirements; and 5)- the return of experience from nuclear containments historic performance. The concepts of dormant reliability and structural fragility will guide the assessment of the containment structural integrity, within the general context of a global containment life cycle management program. This study is oriented towards the requirements of CANDU-6 reactors, in general, and Hydro-Quebec's Gentilly-2 nuclear power plant, in particular. The present article is the first part in a series of papers that will comprehensively detail the proposed research.

1. Introduction

Since the first production of electricity by nuclear energy in 1951, nuclear power plants (NPPs) have been widely operated in many countries around the world. However, nuclear energy has faced oppositions, especially by environmentalists, because of the danger radioactivity can be to the environment when leakage of fission products from the NPP reaches the atmosphere. To cope with radioactive contamination risks, NPPs have been equipped with containment systems properly engineered to prevent and contain large leakage of radioactive material in case of accidents. Special periodic tests monitor the reliability and availability of the containment systems. The challenge which currently faces most of NPPs operators is to cautiously manage the business efficiency of their power plants without compromising their safe operations. In fact, the majority of NPPs which are currently operating were licensed under deterministic regulatory controls. While these conservative controls have ensured safety in plant operation, their economic implications have often kept nuclear power out of the competitive realm. This loss of competitiveness is partially due to the current regulatory controls which include in particular the nuclear containment integrated leakage rate test (ILRT). Recent advances in probabilistic risk assessment models offer defensible foundation for a change to the regulatory approach to safety and monitoring of NPPs, especially the emergent risk-informed approaches. When dealing with the issue of nuclear containment ILRT, the risk-informed approach removes the test frequency from an exclusive technical-specification context to re-locate it to a plant-specific and performance-based program within a global perspective of a life cycle management policy. Once re-located, a combination of probabilistic and deterministic insights is used to assess proposed extensions of ILRT intervals [1]. The ILRT becomes, thus, an integral part of a life cycle containment management program. It cannot be treated and analysed in an isolated manner, thereon.

2. Problem Statement

Nuclear containments are the last lines of defense against leakage of large amounts of radioactive material to the environment in case of major reactor accidents which entail a core damage. They are, therefore, a vital engineered safety feature of a nuclear power plant. Proportionate to their crucial role in safety, nuclear containments have been monitored through well designed periodic tests to ensure the proper performance of the containments and their rate of reliability decrease under the inherent operational and environmental stressors. The rate of leakage of radioactive and fission products to the atmosphere is the metric that governs the success or failure of containment testing. In light of the importance of containment tests to the overall safe operation of a nuclear power plant, codes and regulations that govern the operation of NPPs have defined the nature of tests that must be performed and their frequencies. Regulatory organizations have imposed the types of containment tests and their intervals in the construction/operation licenses of NPPs. A particular emphasis is put on high-pressure integrated-leakage rate tests (ILRT) where the containment is pressurized to its design-accident pressure. Such tests have been carried out in Canada at frequencies of once in three years, nearly, as required by the Canadian standard R-7 [2]. Though beneficial to the safety and reliability monitoring, high-pressure tests of nuclear containments are expensive, and time consuming along the critical path of the NPPs programmed maintenance and refueling shutdowns. Furthermore, over-pressurizing the containment to its design-accident pressure during the ILRT may promote cracking and crack-propagation in the containment structure as a result of the high pressure at which the test is conducted and the relatively important structural loads and deformations induced to the structure during testing. In a report on aging management activities at their Gentilly-2 (G-2) NPP, Hydro-Quebec concluded that the ILRT may be a contributor to the degradation of the pre-stressed concrete containment [3]. The recent developments in the perception and methods of risk evaluation, mainly the evolution of the concept of risk-informed decision making and its implementation in the nuclear safety domain, have convinced the nuclear power regulatory organizations - mainly the U.S. Nuclear Regulatory Commission (NRC) - and industry to relax the ILRT frequencies while maintaining consequent risks within acceptable levels and preserving the integrity of the defense-in-depth philosophy. “On industry-wide generic basis, there is a small risk associated with the extension of the… ILRT surveillance interval from the present to up to 10, 15, or 20 years, provided that the performance bases and defense-in-depth are maintained. There is an obvious benefit to the nuclear power industry in not performing costly, critical-path, time-consuming tests that provide a limited benefit from a risk perspective" [4]. Recently, the NRC authorized ILRT testing intervals of up to 15 years in light of similar conceptual approaches.

The objective of this paper is to introduce a master-plan of a risk-informed decision-making (RIDM) framework on the periodicity of CANDU-6 nuclear containment ILRTs for which such a methodology does not exist. It is based on five main decision criteria, namely, the containment structural integrity, the input from PSA Level-2, the requirements of deterministic safety analyses and defense-in-depth concepts, the obligations under regulatory and standard exigencies, and the return of experience from nuclear containments historic performance. All of these criteria are taken into consideration in a systematic and structured manner for the first time. The concepts of dormant reliability and structural fragility will guide the assessment of the containment structural integrity, within the general context of a global containment life cycle management program. The approach will be validated with operational and historic performance data of Hydro-Quebec's Gentilly-2 nuclear power plant. The present article is the first part in a series of studies that will comprehensively detail the proposed research.

3. Literature Review

Containment is a vital engineered safety feature of a nuclear power plant. It comprises the containment structure, penetrations, dousing system, isolation valves, and airlocks. Nuclear containments engender particular risks of leakage of radioactive material to the environment resulting from core damage or a breach in the containment itself. Commensurate with these risks, nuclear regulatory bodies dedicated detailed requirements for the design features and safety performance-monitoring of nuclear containments, as the following regulations' overview will present.

3.1 The American Regulatory Perspective

According to Criterion 16 of Appendix A to regulation 10-CFR-Part-50 [5], "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require". Moreover, General Design Criterion-2 requires the containment to remain functional under the effects of postulated natural phenomena and environmental stressors such as earthquakes, tornadoes, hurricanes, floods, tsunami, temperature and pressure fluctuations, ice, windstorms, etc. This is in addition to stresses imposed on the containment as a result of normal operations of the NPP. In addition to the mechanical stresses and strains generated by transients under normal conditions, the containment is subjected to various types of internal degradation (ageing degradation) depending on inherent characteristics of the materials, and construction methods. The rate and extent of degradation are influenced by sustained environmental conditions such as temperature, humidity, water leakage, etc. The reliability of a containment to perform its intended function under design basic conditions as well as under higher loads due to severe accidents and earthquakes is influenced by the containment's inherent capability and the various stresses and degradation mechanisms that act on it [6]. A reasonable balance is, therefore, preserved among prevention of core damage, prevention of containment failure, and consequence mitigation. Probabilistic safety studies and continuous monitoring procedures are usually conducted to evaluate the annual failure frequency of the defense barriers, as well as the risk of leakage of radioactive material to the environment and the consequences of such leakage on the neighboring population. Because of the role served by the containment as the ultimate barrier against the release of radioactive material, compromising the containment integrity could increase the risk of large early release of radioactive products in case of accidents [7]. Therefore, the basic concept in ensuring the reliability of the containment is to track the degradation of the containment components through periodic inspections, and to check the leak-tight integrity of the containment's pressure-retaining components through periodic leakage rate testing.

Appendix J to regulatory document 10-CFR-Part 50 [5] specifies containment leakage-testing requirements, including the types and frequency of tests. The following types of containment leakage tests are required according to Appendix J: 1)- "Type A" tests, often referred to as ILRTs, aiming at measuring the containment integrated leakage rate; 2)- "Type B" tests whose objective is to measure the leakage rate across each pressure-containing or leakage-limiting boundary for various primary reactor containment penetrations; and 3)- "Type C" tests designed to measure the leakage rates of containment isolation valves. Type B and C tests are referred to as local leakage-rate tests (LLRTs). Additional methods, referred to as on-line monitoring, have been adopted to monitor containment integrity continuously during full-power operation. Appendix J to 10-CFR-Part-50 specifies also a schedule for conducting containment leakage-rate tests (both preoperational and periodic). Preoperational leakage-rate tests are conducted when construction of the reactor containment structure is complete and all operating parts including systems penetrating the containment structure have been installed. Periodic tests are performed during the service life-span of the nuclear power plants on regular basis.

Clause 3.2.3 of standard ANSI/ANS-56.8-2002 of the American Nuclear Society [8] stipulates that periodic "Type A" tests shall be conducted at the first refueling shutdown but not more than three years subsequent to the preoperational test and at intervals not to exceed five years thereafter.

In their report NUREG-1493 [9], the NRC initiated a methodology to reduce and eliminate safety requirements that are marginal but impose an important financial and operating burden on the owners of American nuclear power plants. An interesting conclusion of this methodology relates to the specifications of the containment leakage-rate tests. NUREG-1493 concludes that reviewing the requirements of the aforesaid tests would lightly affect the safety of the containment, but could allow NPPs' operators to respect the NRC safety requirements while allowing a better resource management of their plants. The recommendations and conclusions of NUREG-1493 concerning the requirements of the containment leakage-rate testing can be summarized as follows: 1)- the allowable leakage-rate can be increased by one or more measurement units without significantly affecting the dosage of radioactive material the population will be exposed to in case of accidents; and 2)- the frequency of "Type A" leakage-rate tests can be reduced from three tests in ten years down to one test in ten years, and even once in twenty years. This reduction in high-pressure testing frequency may induce an unimportant increase in radiation risks. However, it can cut down the tests' related cost by 83%, nearly. The California based Electric Power Research Institute (EPRI) built on the findings of NUREG-1493 in two technical reports: TR-104285 [10] and TR-1009325 [11]. These two reports confirmed that the interval between high-pressure containment leakage-rate tests can be increased up to twenty years provided that the required performance criteria and the exigencies of defense-in-depth are sustained. These two technical reports have been updated several times to improve the overall methodology proposed by EPRI [4], [12], [13]. "Over the last several years, the NRC staff has granted one-time extensions of the ILRT interval to 15 years for over 75 operating reactors, and recently approved a methodology that could be used to support a permanent extension to 15 years on a plant-specific basis… the remaining plant life may not be impacted by whether a 15 or 20 year interval is selected… The risk impacts of the ILRT extension are considered to be acceptably small…" [14].

3.2 The Canadian Regulations Standpoint

Canadian standard CSA-N287.7-08 [15] defines the frequency of leakage tests in clause 7.2 stating that "Class containment components shall be leakage-rate tested… at a frequency agreed upon by the owner and the regulatory authority. The leakage-rate test shall be performed at least once every 6 years subsequent to the first in-service use of the containment system". In addition, clause 7.3 requires that on-power leakage-rate test be carried out at least once every two years at negative or reduced positive pressures to demonstrate that the leakage rate is not greater than the maximum allowed. The May 2008 edition of CSA-N287.7 introduced, for the first time, a "performance-based test interval option" opening the door to extending the leakage tests interval to as long as this extension is justified by proper analyses.

Regulatory document R-7 of the Canadian Atomic Energy Control Board [2] describes the requirements for containment systems for CANDU nuclear power plants. It applies to nuclear reactors whose building permits were delivered after January 1st, 1981. Regulatory document R-7 dictates a leakage-rate test at design-accident pressure to be performed once every three years, at least. The frequency of such tests should be brought down to two years would the measured leakage rate be found in excess of the test acceptance leakage criteria. Moreover, R-7 requires other leakage-rate tests carried out at reduced or negative pressures to be performed at a frequency of no less than once per two years. R-7 states that a leakage test at full design pressure shall be carried out at a minimum interval of once every six years in any case.

Recently, the Canadian Nuclear Safety Commission (CNSC) granted Hydro-Quebec an extension until June 30, 2009, to conduct the ILRT periodic test at Gentilly-2. This time extension brought the interval between the 2009 test and the last one performed in 2003 to 5.5 years, nearly [3].

3.3 Risk-Informed Decision Making – A New Look at Safety

While the rationale of most of regulatory provisions for the ILRT periodicity seems to be deeply rooted in deterministic conservative studies and analyses, risk-informed approaches have been developed aiming at coping with the nuclear containment leakage-rate testing frequency under a safe yet less conservative perspective. Kumamoto [16] defined the risk-informed integrated decision making (RIDM) as the satisfaction of safety goals by complementing deterministic analyses with probabilistic approaches resulting in an informed decision that falls within the boundaries of certain acceptable guidelines, and a small increase in risks. Elaborating further on this definition, the International Atomic Energy Agency (IAEA) describes RIDM as an approach that is basically dynamic, fluid, and sensitive to long and short terms feedback which considers, weighs, and integrates complex inputs and insights from traditional deterministic engineering analyses, probabilistic analyses, operational experience, compensating or mitigating measures, or other pertinent considerations [17]. Though not viewed as an exact science [18], RIDM approaches must shape decisions on safety-related matters which comply with the current regulations, in-line with the defense-in-depth philosophy, and secure acceptable levels of redundancy and safety margins. The impacts of such decisions should result in small risk increase which can be monitored by performance measure strategies [19]. Finally, RIDM approaches "… establish requirements that better focus attention on design and operational issues commensurate with their importance to public health and safety... A risk-informed approach enhances a deterministic approach by: 1)- allowing explicit consideration of a broader set of potential challenges to safety; 2)- providing a logical means for prioritizing these challenges based on risk significance, operating experience, and/or engineering judgment; 3)- facilitating consideration of a broader set of resources to defend against these challenges; 4)- explicitly identifying and quantifying sources of uncertainty in the analysis; and 5)- leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions. Where appropriate, a risk-informed approach can also be used to reduce unnecessary conservatism in purely deterministic approaches, or can be used to identify areas with insufficient conservatism in deterministic analyses and provide the bases for safety and/or regulatory decisions that lead to an overall enhancement in safety" [20].

4. CANDU-6 Nuclear Containment Systems

CANDU-6 nuclear containment systems are designed to restrain radioactive leakage to the environment that exceeds the allowable maximum limits of radiation during events or accidents which engender the emission of high radioactivity inside the reactor building. This task is achieved through, first, limiting the level and duration of high pressurization of the reactor building resulting from eventual accidents, and, second, designing the containment with an appropriate tightness in order to secure minimal radioactive leakage. CANDU-6 containment system consists of passive components (concrete walls, concrete dome, foundation mats, anchoring systems, conduit penetrations, and spent fuel discharge bay, room R2-001), and active components (isolation valves, air locks, and local air coolers). Four containment components can contribute to radioactive leakage in a CANDU-6 system: 1)- the reactor building; 2)- the spherical valves of the spent fuel discharge channels; 3)- the isolation valves; and 4)- the air locks [3].

Hydro-Quebec surveyed the containment testing practices at some nuclear power stations whose reactors are similar to that of G-2 [3]. This survey of some CANDU-6 nuclear reactors operating worldwide revealed that high-pressure containment leakage tests at 124 kPa (Type A) are performed in all of these NPPs. Point Lepreau NPP (New Brunswick – Canada) have performed high-pressure (124-kPa relative) leakage-rate tests at intervals of three years, nearly. The results of the leakage rates observed during these tests are similar to those measured at G-2. However, leakage to room R2-001 of the containment at Point Lepreau has been lower than that observed in G-2. The difference in leakage rates might be caused by the material used during the construction of Point Lepreau (in terms of aggregate silica-alkali reaction observed in G-2 but not in Point Lepreau), and by the cracks the containment of G-2 might have incurred following the Saguenay earthquake in 1988 [3]. As for the low-pressure test, Point Lepreau had conducted such tests until 1992; they have been discontinued thereafter. Embalse NPP (Argentina) have carried out only one high-pressure leakage rate at 124 kPa since the start of their operations. Tests at 42 kPa have been conducted at a frequency of 5 years, based on the technical specifications applicable in 1977. The leakage-rate tests performed at design pressure resulted in a leakage rate of 0.353% vol-RB/day. No data of the leakage rate to room R2-001 is available. Cernavoda Nuclear Power Station NPP (Romania) have tested the leakage-rate at design high pressure during the pre-operational stage in 1998, and conducted three other tests in 1998, 2003, and 2008. Qinshan Power Station (China) conduct high-pressure leakage-rate tests at intervals of five years, and low-pressure (1 kPa) tests every six months. Moreover, the Korean Wolsong NPP operators have performed ILRT at 124 kPa at intervals of five years.

5. Hydro-Quebec Gentilly-2 Nuclear Power Plant

Hydro-Quebec have operated a CANDU-6, pressurized heavy water, nuclear power plant at Gentilly, Province of Quebec, since October 1983. This NPP is referred to as “Gentilly-2” or simply “G-2”. It produces 675 megawatts (constituting 3% of the total electricity produced in Quebec) which supply 74,000 clients, nearly [3].

5.1 Gentilly-2 Containment Overview:

The containment system of G-2 is designed to limit the leakage to the environment of radioactive fission products exceeding allowable limits during an accident through limiting the over-pressurization period of the reactor building (RB) during an accident, and ensuring leakage tightness of the containment enough to minimize radioactive leakage to the environment. The Reactor Building is composed of a concrete containment, machinery anchors, conduit penetrations, and the spent fuel discharge bay (room R2-001). The design of G-2 nuclear containment is done in strict accordance with CANDU-6 specifications, where seven containment sub-systems can contribute to radioactive leakage of fission products to the environment: 1)- containment concrete structure; 2)- equipment anchors; 3)- circular penetrations; 4)- room R2-001; 5)- spherical valves of the spent fuel conduits; 6)- isolation valves; and 7)- air locks [3].

5.2 Leakage-Rate Related Deterministic Analyses:

Comprehensive deterministic analyses have been conducted on the containment of G-2 since 1978. The objective of these analyses has been to define different levels of nuclear containment unavailability under various simulated accident scenarios. Consequently, the following three levels of unavailability were defined relative to the intended role of the containment and in accordance with the operating regulations of G-2 and the requirements of regulatory document R-7 [3]:

- **Level 1**: The containment is unable to obstruct significant leakage of radioactive material to the environment exceeding the allowable limits of 30 rem (thyroid dosage per critical person) in case of single failures. At this level, the containment cannot guarantee the structural integrity of the reactor building under overpressures during severe accidents.
- **Level 2**: The containment has a limited capacity of preventing significant leakage to the environment of radioactive products of 3 to 30 rem (thyroid dosage per critical person) dosages.
- **Level 3**: The containment is able to hamper leakage of radioactive material to the environment. However, its safety margin and redundancy are significantly reduced. The probabilities of occurrence of Level 1 and Level 2 unavailability are increased when Level 3 occurs.

The deterministic analyses in subject correlated also the unavailability levels to the size of cracks or deficient openings in the nuclear containment, and to the overall containment leakage rate, as shown in Table 01 [3]. Moreover, based on the return of experience from the ILRTs conducted on the nuclear containment of G-2, it was possible to determine the contribution of various containment components to the overall leakage rate which differs from the one originally projected in the operating manual. This contribution was found to be 50% from the reactor building (RB), 25% from the air locks, and 25% from the containment valves, in comparison with an original estimate of 37.74%, 18.87%, and 43.30%, respectively. Finally, G-2's comprehensive deterministic analyses determined milestone leakage rates. A measured containment leakage rate of 0.5 %RB vol/day dictates undergoing certain repair actions in order to restrain the leakage from the containment to less than the aforementioned rate; whereas, an unavailability of Level 3 is declared only when the containment leakage rate exceeds 2.5 %RB vol/day. Therefore, 0.5 %RB vol/day has been called the "repair limit". Obviously, it became the decisive criterion which defines the success of an ILRT when the measured containment leakage rate is less than 0.5 %RB vol/day, or its failure when this limit is exceeded [3].

	Crack Size in RB Only (A, cm^2)	Crack Size Containment (A, cm^2)	Containment Leakage Rate (%RB vol/day)
Level 01	$\geq$ 2.67	$\geq$ 5.33	>> 5%
Level 02	$0.47 \leq A < 2.67$	$0.93 \leq A < 5.33$	> 5%
Level 03	$0.42 \leq A < 0.47$	$0.83 \leq A < 0.93$	= 5%

TABLE 01: Crack/Opening Sizes for Various Unavailability Levels [3]

5.3 Containment Leakage-Rate Testing at Gentilly-2:

Five types of pressurization tests have been performed at G-2 aiming at checking the leakage-tightness of the containment: 1)- High pressure test (ILRT) conducted at the design-accident relative pressure of 124 kPa every 3 years (except the last ILRT conducted in 2009 which was performed at an interval of 5.5 years) during planned reactor shutdowns; 2)- Low pressure test performed at a relative pressure of 3 kPa every 4, 6 or 12 months at power; 3)- Isolation-valves test at a relative pressure of 140 kPa or 35 kPa depending on the valve type; 4)- Air locks pressure test at 125 kPa; and 5)- Test at 140 kPa for the spherical valves of the spent fuel conduits. These types of containment-leakage tests and their frequency have been established in agreement with the requirements of the CNSC and Hydro-Quebec's obligations under the operation license of G-2. The high-pressure ILRT conducted at the design accident relative pressure of 124 kPa is intended to determine and globally quantify the leakage-rates from the concrete structure of the containment, the valves, the anchors, and room R2-001, as well as those that may occur in walls, penetrations, and access points between the reactor building and room R2-001. Two main objectives are sought by the high-pressure test: 1)- prove that the leakage-rate from the containment does not exceed 0.5%RB vol/day (repair criterion); and 2)- assess the overall behavior, problems, and weaknesses of the containment during accidents which lead to its over-pressurization, and propose preventive measures. Clearly, repairs must be initiated when the leakage rate measured during the test exceed the "repair criterion" of 0.5%RB vol/day. Leakage tests are usually conducted on the containment's air locks and valves few days before the ILRT is performed, in order to be assured of their tightness. Accordingly, when the latter are undergone, it is safe to assume that the measured leakage rate originates uniquely from the concrete containment, the internal liner, and the equipment anchors. During the test, the relative pressure inside the reactor building is increased until it reaches 124 kPa. At this stage, deformations of the reactor building are observed: its radius and height are increased by nearly 2.3 mm and 2.7 mm, respectively. Measuring the leakage rate starts thereafter. The reported leakage rate for the reactor building (RB) is conservatively calculated as the measured overall leakage rate minus 95% of the leakage rate from room R2-001 (i.e. $LR_{RB} = LR_{MES} - 95\% \; LR_{R2\text{-}001}$). It is assumed that 95% of the leaked radioactive material to room R2-001 remain trapped inside that room rather than released to the environment [3].

5.4 Preliminary Statistical Analysis of ILRT Results at Gentilly-2:

Table 02 and Figure 01 summarize the results of the high-pressure containment ILRTs conducted at G-2 since 1979 [3]. Browsing through these results, an obvious shift in the total containment leakage-rate can be observed since the ILRT conducted in year 2000 compared to those tests performed before this date. This leakage-rate increase is coupled with the modification in the ILRT procedure implemented since year 2000. The relative pressure inside the RB had been gradually increased to its 124-kPa target during the tests conducted between years 1979 and 1997, observing thus multiple intermediate pressure-stabilization plateaux. In contrast, the relative pressure inside the RB has been sharply raised to the 124-kPa relative test-pressure during the ILRTs carried out since year 2000.

Statistically, the results of the ten ILRTs conducted on G-2's nuclear containment, so far, present a sample of ten data points. However, it is beneficial to divide the results of the tests into two statistical samples, before and after year 2000, relative to the ILRT procedure change mentioned above. For the seven tests conducted before year 2000, the following statistical data can be noted for the leakage-rate from the RB only: 1)- the sample size is equal to 7 data points; 2)- the sample mean and standard deviation are 0.391 %RB vol/day and 0.149 %RB vol/day, respectively; 3)- the RB-only leakage-rate sample mean falls within the limits of the 95% confidence interval on the population mean, i.e. between 0.253 %RB vol/day and 0.529 %RB vol/day (using the Student, *t*, distribution with an unknown population variance); and 3)- a one-sided hypothesis test using the T-statistic at 5% significance-level with a null hypothesis H_0: μ = 0.5 %RB vol/day and an alternative hypothesis H_0: μ < 0.5 %RB vol/day, where μ is the population mean of the leakage-rate from the RB only, leads to the rejection of the null hypothesis. On the other hand, the following statistical facts hold true for the three ILRTs carried out since year 2000: 1)- the sample size shrinks down to 3 data points; 2)- the sample mean RB-only leakage rate increases to 0.444 %RB vol/day while the sample standard deviation decreases to 0.014 %RB vol/day; 3)- the 95% confidence interval on the RB-only leakage-rate is limited by 0.408 %RB vol/day, as lower limit, and 0.480 %RB vol/day as upper limit (using the Student, *t*, distribution with an unknown population variance); and 3)- a one-sided hypothesis test using the T-statistic at 5% significance-level with a null hypothesis H_0: μ = 0.5 %RB vol/day and an alternative hypothesis H_0: μ < 0.5 %RB vol/day, where μ is the population mean of the leakage-rate from the RB only, sustains the rejection of the null hypothesis. Finally, T-tests and F-tests (at 5% significance level) for the difference in populations' means and variances, respectively, result in the rejection of the null hypothesis which claims that the population variances of the RB-only leakage-rates before and after the change in the ILRT pressurization procedure are equal. These two samples may, hence, pertain to two different measured leakage-rate populations.

Reactor Building Leak Rate (% RB vol/day)				
Year	Leak from RB ONLY	Leak RB to R2-001	5% Leak RB to R2-001	Total RB Leak Rate
1979	0.290	-	-	0.290
1981	0.160	-	-	0.160
1985	0.516	0.080	0.004	0.520
1987	0.621	0.190	0.010	0.630
1990	0.370	0.400	0.020	0.390
1993	0.411	0.380	0.019	0.430
1997	0.370	0.400	0.020	0.390
2000	0.448	0.450	0.023	0.470
2003	0.429	0.630	0.032	0.460
2009	0.457	0.822	0.041	0.498

TABLE 02: History of Results of the Containment ILRTs at 124 kPa [3]

These statistical figures confirm that the general tendency of the ILRT results supports an almost stable mean leakage rate from the RB-only. The increase in the leakage rate since 2000 may be explained by the change in the pressurization procedure which accounts for a slightly different response of the RB. Despite this increase, the mean leakage rate from the RB-only remains less than the "repair criterion" set at 0.5 %RB vol/day for G-2 nuclear power plant, regardless of the test over-pressurization procedure. Furthermore, there is an indication that the change in the over-pressurization procedure during the ILRTs led to different populations of the reactor building leakage-rate. This is in addition to confirming that the problems which faced the execution of the ILRTs in the 1980's rendered the pre-procedure-change sample not very representative. An almost stable trend may be observed in the last three successive ILRTs that have been conducted since year 2000, as well as in those undergone between 1990 and 1997. However, the test results show an obvious tendency of increased leakage rate from room R2-001 which, in turn, results in an increase in the overall leakage-rate from the containment during an ILRT at 124-kPa relative design-accident pressure. A detailed qualitative and quantitative study of, and the lessons learnt from, the ILRTs that have been performed on the nuclear containment of G-2 since its commissioning will be described in future related analyses.

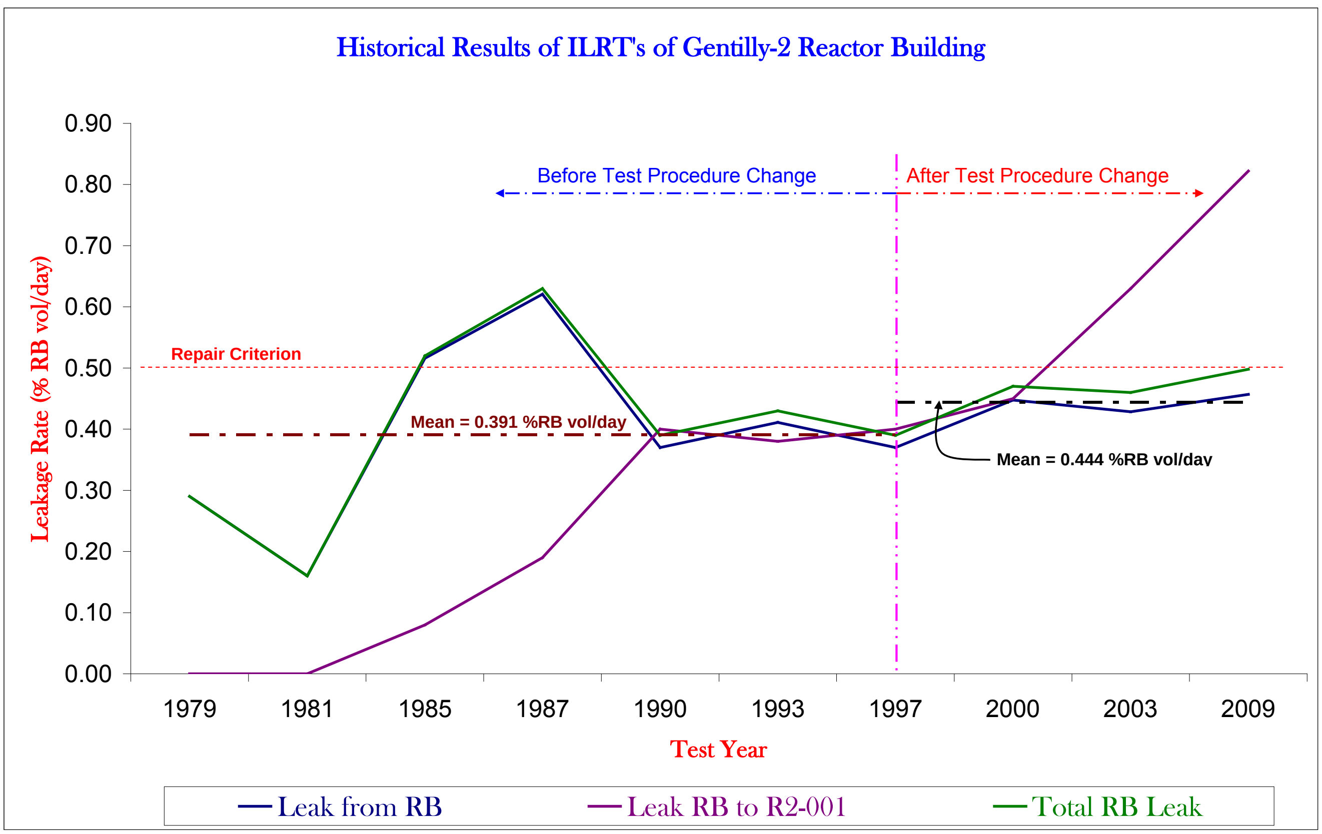

Figure 01: Historical Data of Containment ILRTs' Results

5.5 Leakage Rate Tests at Low Pressure in Gentilly-2:

Low-pressure tests, at 3-kPa (relative), have been performed in G-2 since 1989. They have been conducted after major repairs or maintenance periods to ascertain the absence of any openings (valves, rupture disks, etc.) left open after a prolonged shutdown of the power plant, to verify the non-existence of openings in the containment whose diameters exceed 2.5 cm under 3-kPa pressure, and to predict, if possible, the tendency of the containment leakage rate under 124-kPa relative pressure. This test was performed at a frequency of once every six months from 1989 to 2009. This frequency will be relaxed to one performance per twelve months effective year 2010 as a result of a special authorization recently granted to Hydro-Quebec by the Canadian Nuclear Safety Commission (CNSC). Recall that the total leakage rate is defined as the sum of radioactive material leaked outside the containment from three different sources: the reactor building, the air locks, and the containment valves. It is measured in percent of the reactor-building volume per day (%RB vol/day).

5.6 Numerical Model of G-2's Containment Structure:

At G-2, ILRTs have been a useful tool in monitoring the safe performance of the containment as well as in predicting its behavior under core-damage or any other type of accidents which may result in large release of radioactive products. Despite their efficiency, Hydro-Quebec sought to complement these tests by developing numerical models of the containment concrete structure. The development of these numerical models aimed at analyzing the effects of the following stressors on the structural integrity of the concrete containment : 1)- the significant effects of ageing of the containment concrete structures exposed to severe climatic conditions (like in Canada); and 2)- the impact of the on-going aggregate alkali-silica reaction (ASR) on the integrity of the containment concrete, especially that tests have indicated that aggregates used in concreting G-2's containment are slightly reactive. These are in addition to numerically investigating the compounded effects of ageing and ASR on the integrity of the containment concrete. The numerical model in subject is a non-linear finite element model for both static and dynamic analyses. The investigation included the evaluation of various aging/degradation mechanisms that are likely to affect the future performance of the containment concrete structure. The numerical simulations incorporate a transient thermal analysis during the period of construction, a series of static analyses including the effects of the ASR, as well as dynamic analyses simulating seismic events and other low-frequency accidental impacts. The numerical model of G-2 containment structure is viewed as an integral component of the performance review of the global life cycle management program of the nuclear containment. The model was calibrated with the results of ILRTs. It could be used to predict the structural response of the containment concrete during ILRTs and in-between successive ILRTs. The numerical model developed by Hydro-Quebec is qualified to Canadian standard N286.7, and might be unique in its kind among CANDU-6 reactor operators [3].

5.7 Probabilistic Safety Assessment (PSA) Level-2:

Probabilistic Safety Assessment (PSA) of a nuclear power plant provides a comprehensive, structured approach to identifying failure scenarios and deriving numerical estimates of the risks. It evaluates the level of safety of NPPs playing, thus, an important role in any risk informed decision-making process. PSA Level-2 uses acceptance criteria in the form of containment failure or large release frequency (LRF). These acceptance criteria are: 1)- maximum allowable limit of 10^{-5} per year LRF; and 2)- plant safety goal target of 10^{-6} per year LRF [22]. The U.S. nuclear power industry and the NRC have developed an approach to evaluating the impact of reducing the ILRT frequency on safety (in terms of eventual increase in radioactive dose to population) for PWR/BWR reactors. In this approach, two specific accident classes are added to the PSA Level-2 model to represent events in which the containment has either a small or a large pre-existing leakage. For dose assessment purposes, these accident classes are assigned a leakage rate equivalent to 10 times and 100 times the maximum allowable leakage rate of the containment, respectively. The frequency of each leakage class is determined by multiplying the frequency of accident sequences affected by the ILRT extension (i.e. the intact containment frequency) by the conditional probability of a small or a large leak; the intact containment frequency is then reduced by that amount. The leakage probability values are based on ILRT test data developed through two industry surveys plus additional leak rate data from 35 recently completed ILRTs in the U.S. The model assumes that the frequency of pre-existing leakage increases linearly with the test interval, and quantifies the impact of the increased test interval on three risk metrics: 1)- large release frequency; 2)- 50-mile population dose; and 3)- conditional containment failure probability [14]. No similar approach exits for CANDU-6 reactors. Therefore, there is an urge to establish methods to account for the effect of the ILRT frequency change on safety of CANDU-6 reactors' operations, given that the ILRT periodicity is not directly credited (i.e. modeled) in the PSA Level-2 [3].

6. Proposed Methodology

The literature review presented above emphasized on the importance of the structural integrity of the containment in maintaining safe operation of a nuclear power plant. The role of a soundly operating nuclear containment is crucial in mitigating the risks of large early release of radioactive products to the atmosphere in case of major accidents, such as core damage. The design and durability of the containment structure is, therefore, of primordial importance to the safety of a nuclear power plant and its neighborhood. Combining these two factors leads to the importance of the containment fragility in assessing its safety. Another crucial factor contributing to the safety of a nuclear containment is its reliability, or its ability to successfully perform the task it was intended for under normal design conditions or design-accident stresses. The literature review reinforces the importance of periodic leakage-rate testing in monitoring the reliability of the containment system. Though providing excellent tools to control and check the leakage progress through the containment, these tests may produce potentially damaging effects in the structure of the containment by causing cracks and promoting crack propagation. Furthermore, concrete aging is considered as a deteriorating agent which may adversely affect the containment structure, its material properties, and, obviously, its reliability.

In another synopsis, the literature review reveals the importance of risk-informed decision techniques in shaping the current regulations that govern NPPs, in general, and the frequency of the ILRTs, in particular. While merging the outcomes of deterministic analyses with those of probabilistic assessment, risk-informed decision approaches rely on past experience and historic performance of the containment to enlighten the final decision. A risk-informed decision-making process can be used to optimize regulatory requirements while maintaining a very small increase in risk which is not significant to the public safety. Risk-informed approaches have allowed the nuclear power industry and regulatory organizations to extend the testing intervals between successive ILRTs up to 10 or 15 years (with some works allowing even 20 years), reducing heavily the costs of periodic containment testing at the expense of a minimal increase in the incurred risk. However, there are no such approaches that apply to CANDU-6 reactors.

Therefore, a methodology for evaluating the periodicity of high-pressure containment tests (ILRTs) of nuclear power plants equipped with CANDU-6 reactors is proposed, aiming at reducing the frequency of ILRTs without inducing a significant increase in the risk of radioactive leakage to the public. The proposed methodology is a risk-informed decision making process whose target is to optimize the frequency of containments ILRTs at design accident relative pressure. Its is based on five main decision-making criteria: 1)- containment structural integrity; 2)- input from Level-2 probabilistic safety assessment; 3)- requirements of deterministic analyses and defense-in-depth philosophy; 4)- obligations under regulatory standards; and 5)- return of experience and lessons learnt from the past performance of a nuclear containment. Moreover, the proposed methodology will be validated using design and operating conditions of Gentilly-2 nuclear power plant. The inputs required in some of its components, namely reliability and fragility related matters, may use synthetic and simulated data generated by the numerical model developed by Hydro-Quebec (ref. Section 5.6). Therefore, the proposed methodology will achieve its objectives out of the following five decision criteria, within an adequate containment life cycle management program. Its flowchart is presented in Figure 02, hereafter.

6.1 Containment Structural Integrity

Experience has shown that the structural integrity of concrete containment structures starts to degrade, earlier and faster than initially hypothesized at the design stage, since they are subjected to several kinds of environmental and operating stressors. Moreover, prolonged exposure of containments' concrete to irradiation may negatively impact its properties. Nuclear safety may be compromised, thus. "Concrete structures are essentially passive components under normal operating conditions, but play a key role in mitigating the impact of extreme or abnormal operating and environmental events. Structural components are somewhat… the limiting factor for plant life since they are mostly irreplaceable…[they] are subject to time-dependent variation of material properties under the influence of environmental stressors and ageing factors that may impact their ability to perform its safety function. As NPPs age, assurance need to be provided that the capacity of [containment] concrete structures to mitigate extreme events has not deteriorated unacceptably… Since time-dependent changes to structures and potential future challenges are random, structural reliability theory constitutes a useful approach for safety evaluation of [containment] concrete structure integrity". It is important to identify what degradation methods are most pertinent to containment safety, and the way to correlate in-service inspection and performance monitoring data into the structural fragility of nuclear containments [23].

In light of this importance and for the purpose of the proposed methodology, the structural integrity and reliability of the nuclear containment are the primordial factor which restrains the leakage of radioactive material to the environment in case of severe accidents. Integrating the containment structural integrity into an overall RIDM framework for ILRT frequency is a novel approach since it is not fully embedded within the U.S. RIDM analyses relative to ILRT frequency [14]. The following three criteria will be examined in the objective of evaluating the containment structural integrity. Input from the numerical model mentioned in Section 5.6 could be used in developing these three criteria.

6.1.1 Containment Dormant Reliability Evaluation

As implied by the literature review, nuclear containments are designed to withstand loads from over-pressurizing the reactor building in the event of severe accidents like core damage. While not enduring the design-accident pressure loads, containments are subject to operating loads and other stressors (such as environmental loads, concrete aging, material deterioration like the aggregates' alkali-silica reaction, etc.) which erode their capacities, reduce their reliability, and increase their failure rate with time. These conditions best fit the definition of dormant reliability [24]. In fact, a containment can be considered in a dormant state during the interval between two successive exposures to its design-accident loads, in other words during high-pressure ILRTs (at 124-kPa relative) or accidents like core damage. In evaluating the dormant reliability of G-2 pre-stressed concrete containment, two models will be adopted, in principal: the Duane model, and the Gompertz model. These models were chosen because: 1)- they have the ability to predict future system reliability; 2)- they can model dormant reliability; and 3)- their parameters are relatively easy to calculate, saving thus on the simulation computer time [24]. Monte Carlo simulation techniques will be used, then. The purpose of Monte Carlo experiments is to generate sets of dormant reliability data. The end result is to establish test-frequency versus dormant-reliability curves. These curves will be used in assessing the various reliability losses for various leakage-rate testing frequency scenarios.

6.1.2 Containment Fragility Evaluation under Seismic and ILRT Induced Loads

The structural fragility of the containment will be determined for three stressors that may affect the structural integrity of the pre-stressed concrete containment, and promote cracking development and possible leak paths: 1)- the high-pressure ILRT at 124-kPa; 2)- the concrete ageing and environmental deterioration modes; and 3)- the seismic loads especially with regard to the revised seismic hazard of Gentilly area imposed by the latest seismic provisions of the Canadian design codes. The nature of fragility analysis is highly dependent on the data and structural design of the system being analyzed. However, a preliminary evaluation may recommend the use of the provisions of NUREG/CR-6920 [7] in developing the fragility curves of the high-pressure leakage-test and concrete aging stressors, while the methodology of EPRI's technical report no. 1002988 may be implemented in establishing the fragility curves under seismic loads [25].

6.1.3 Containment Cumulative Time at Risk under ILRTs

The literature review has shown that ILRTs conducted at the design-accident pressure may induce damages to the containment structure in the form of cracking or crack propagation [3]. Therefore, the ILRT can be viewed as a risk by itself characterized by a well defined frequency and conditions of occurrence unlike any other risk the containment might cope with. It is obvious that the extent of the consequence, i.e. the containment test-induced damage, increases with the increase in exposure to the risk. Therefore, the duration of the ILRT can be considered as “Time at Risk” for the nuclear containment out of the remaining lifespan of the NPP. Recall that the projected service life of G-2 expires in year 2035 rendering its remaining life-span (between 2009 and 2035) equal to 26 years. The proposed methodology will treat the containment time-at-risk, defined above, as one of the risk-informed decision factors towards determining the ILRT frequency in terms of its effect on the structural integrity of the containment.

6.2 PSA Level-2 Input

The proposed methodology will consider Probabilistic Safety Assessment (PSA) Level-2 as one of the decision criteria on the periodicity of ILRT at design-accident pressure (124 kPa). PSA Level-2 is currently under development at G-2. Its results will be used in the proposed methodology to guide an informed decision on the periodicity of the containment leakage test at 124-kPa relative pressure. The input from Level-2 analyses required in this methodology pertains to the frequency and magnitude of ILRT failure, eventual increase in population radioactive dose, and risks of large release frequency increase. This is in addition to the significance of results from performance monitoring tools such as the low pressure containment test at 3-kPa and periodic inspections [3]. To conform with the general philosophy of PSA Level -2, risk-informed decision making, and the related regulatory guidelines, the ILRT frequency calculated in accordance with the proposed methodology must yield an acceptable level of eventual increase in the large release frequency (LRF) within a total respect of the requirements of the defense-in-depth philosophy. Therefore, once the ILRT periodicity is determined, and in-light of the input from the PSA Level-2, the change in leakage risk and LRF is estimated. This evaluation will be based on EPRI’s risk-informed framework [13], after adapting it to the requirements of CANDU-6 reactors. This framework assesses the risk impact of the extended testing interval by: 1)- quantifying the baseline risk in terms of frequency per reactor year; 2)- estimating the risk impact for the interval extension cases, in terms of population dose rate and percentile change in population dose rate; 3)- determining the risk impact in terms of the change in LRF and the change in the conditional containment failure probability (CCFP); and 4)- evaluating the sensitivity of the results to assumptions. Mathematically, EPRI’s framework can be formulated as follows:

RISK = Probability x Consequence

Δ LRF = (Δ ILRT Failure Probability) x CDF
Δ Population Dose = (Δ ILRT Failure Probability) x Population Dose
CCFP = 1 – (Intact CDF / Total CDF)

Because of the variation in probabilities resulting from using different statistical methods, estimates, and/or assumptions, determining the probability of containment leakage is a candidate for expert elicitation. Expert elicitation is performed also to reduce excess conservatisms in the ILRT data, assess the impact of more realistic ILRT failure values on delta LRF, and evaluate the sensitivity of the primary analysis results [10], [11], [12], [13]. Recall that this expert elicitation exercise is exclusive to this study, without any further intention of imposing such an endeavour on any ILRT frequency modification approach on other regulatory tests.

6.3 Containment Historic Performance and Return of Experience

"Performance Monitoring" is an important feedback towards a risk-informed decision. At G-2 nuclear power plant, the performance of the pre-stressed nuclear containment has been monitored by high-pressure tests at 124-kPa (ILRT) every 3 to 5 years, and by more-frequently-conducted low pressure tests at 3-kPa. While the leakage-rate obtained from the low-pressure tests has been nearly constant, increases in the leakage-rate to the environment were observed during the ILRTs conducted in years 2000, 2003 and 2009. In addition, the leakage-rate from the reactor building to room R2-001 increased during the high-pressure tests of years 2000, 2003 and 2009. This matter is discussed in more details in Section 5.4 above. Under a broader scope, the literature review revealed that the U.S. Nuclear Regulatory Commission (NRC) authorized testing-intervals which are largely beyond those allowed by their Canadian counterparts (10 or 15 years). In both cases, pre-stressed concrete containments were involved. Because of the important impact historic performance information bears on a risk-informed decision, the proposed methodology will examine the following three past-performance disparities in order to clarify and assess the significance of their contribution to the containment historic-performance knowledge: 1)- Gentilly-2 ILRT results for the tests done in 1997 (in terms of the longer than usual duration of the test); 2)- the problematic continuous increase in the leakage-rate to the spent fuel discharge bay (room R2-001); and 3)- the comparison between CANDU-6 containments with the containments of light-water reactors in the U.S.A. that qualified for ILRT frequency reduction to once every 10 or 15 years.

6.4 Regulatory and Standard Requirements

Of essential pertinence to the proposed methodology is the compliance of any defined ILRT periodicity with the essence of the regulatory and standard requirements that govern the nuclear power industry. This decision criterion will mostly relate to both relevant CNSC regulatory requirements and CSA respective standards.

6.5 Deterministic Analyses, and Defense-in-Depth Requirements

The literature review presented aspects of the deterministic analyses carried out at G-2 relative to the levels of containment unavailability and ILRT success criteria, as discussed in Section 5.2 above. These factors will constitute one of the main factors in the risk-informed decision making process of the proposed methodology. This is in addition to examining the impact of the ILRT frequency resulting from the proposed methodology on the redundancy and defense-in-depth requirements of the containment, where no compromise should be tolerated.

6.6 ILRT Periodicity Risk-Informed Decision Model

Sections 6.1 through 6.5 as well as Figure 02 present the decision criteria the proposed research methodology will adopt in order to determine the ILRT periodicity and the possibility of reducing its frequency below once per 3 to 6 years.

Consequently, the proposed research plan suggests the following risk-informed decision model which combines the five decision factors exposed above:

$$\begin{aligned}\text{ILRT Interval} = \ & [\ C_1 \times (\text{ Containment Structural Integrity })\ ] + \\ & [\ C_2 \times (\text{ PSA Level-2 Analyses })\ ] + \\ & [\ C_3 \times (\text{ Deterministic Analyses and Defense-in-Depth Requirements })\ ] + \\ & [\ C_4 \times (\text{ Regulatory and Standard Requirements })\ ] + \\ & [\ C_5 \times (\text{ Containment Return of Experience })\ ]\end{aligned}$$

C_i are weight coefficients which scale the impact of each criterion on the whole decision on the frequency of the high-pressure integrated leakage-rate test of the containment, with $-1 \leq C_i \leq +1$. These coefficients are additive, in case they favor the ILRT interval extension, or subtractive, when they penalize or refrain that extension. Owing to the lack of relevant historic data, weight coefficients C_i will be determined by expert elicitation.

6.7 Expert Elicitation

The proposed methodology will perform an expert-elicitation exercise as an integral component. The expert elicitation will serve five main purposes: 1)- determining the criterion-weight coefficients, C_i, used in the risk-informed decision model presented in Section 6.6 of this study; 2)- estimating the probability of pre-existing small and large leaks in the containment; 3)- reducing any eventual excessive conservatism in the results of the proposed methodology, especially in calculating the risk impact of an extended testing-interval, e.g. in terms of change in

LRF; 4)- evaluating the impact of the overall uncertainty engendered by the proposed methodology on the determined ILRT frequency; and 5)- examining whether the ILRT periodicity resulting from the proposed methodology does not breach any regulatory requirement, defense-in-depth principles, and general safe-operation regulations of the nuclear power plant. This is in addition to exploring the eventuality of any other factors that affect the periodicity of the ILRT and its resulting effects on the containment and operation safety that may have been overlooked by the current methodology. The proposed expert-elicitation will be conducted in conformance with the guideline of Delphi structured expert-elicitation method [26], [27]. It will take place in two stages as shown in Figure 02. In its first stage, the expert elicitation exercise will determine the criterion weight factors, C_i, required in the risk-informed decision model of Section 6.6; whereas, all remaining four purposes will be addressed in the second stage. As a pre-requisite of expert-elicitation, an expert qualification/selection scale will be established. The projected scale will describe qualitatively and, if possible, assess quantitatively the required skills which qualify a person as an expert in the nuclear containment and nuclear power plants risk assessment, and whose expertise can serve the objectives targeted by the proposed methodology. Depending on the outcome of expert elicitation, the ILRT periodicity determined through the proposed methodology will be accepted, relaxed, or rejected.

6.8 Cost Benefit Feasibility and Analysis of the Resulting ILRT Frequency

The final step in the proposed methodology entails a cost benefit analysis prorated to the calculated ILRT frequency through the seven steps above. The aim of the cost-benefit analysis is to justify: 1)- the implementation of the new ILRT periodicity by providing a solid backup business case; and 2)- the economic counterbalance of an acceptable increase in risks resulting from the new ILRT frequency versus an alleviation of the operating and testing costs of the NPP. The projected cost-benefit analysis will establish various cash flow scenarios corresponding to various ILRT frequencies, including the currently-implemented periodicity under the present regulatory requirements, subject to pre-defined interest and inflation rates. The analysis will backup the new ILRT frequency by comparing the net present values (NPV) from various cash flow scenarios.

7. Conclusion

Nuclear containments are vital engineered features of nuclear power plants. Commensurate with their important role as the last lines of defense to obstruct leakage of radioactive products to the environment beyond allowable safe limits, containments have been subjected to strict leakage-rate testing and monitoring. These requirements have been often based on conservative deterministic analyses, imposing hence important financial burden on nuclear power plant operators. The recent trends in risk analysis and risk informed decision making techniques seem to have paved the way for both nuclear power industry and regulatory agents to relax the frequency of ILRTs, among other imposed regulatory safety measures, while cautiously preserving the safety of operations and of the public. This study introduced a proposed methodology aiming at determining the periodicity of nuclear containment ILRT using a structural, risk-informed based approach. The study is mainly oriented towards operational and performance history of CANDU-6 reactors, in general, and Hydro-Quebec's Gentill-2 nuclear power plant, in particular. This paper is the first part in a series of articles which will comprehensively present the proposed methodology.

List of Acronyms

ASR	Aggregate Alkali-Silica Reaction within Concrete
CANDU	CANada Deuterium Uranium
CCFP	Conditional Containment Failure Probability
CNSC	Canadian Nuclear Safety Commission
CDF	Core Damage Frequency
EPRI	Electric Power Research Institute – USA
G-2	Hydro-Quebec's Nuclear Power Plant Gentilly-2
IAEA	International Atomic Energy Agency
ILRT	Integrated Leakage Rate Test
LRF[1]	Large Release Frequency (term mainly used in Canada)
LLRT	Local Leakage Rate Test
LWR	Light Water Reactors
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission of the U.S.A.
PHW	Pressurized Heavy Water
PRA	Probabilistic Risk Assessment
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RIDM	Risk-Informed Decision Making

1 LRF is defined as the sum of frequencies of all event sequences that can lead to such a release that may require long term relocation of the local population [22].

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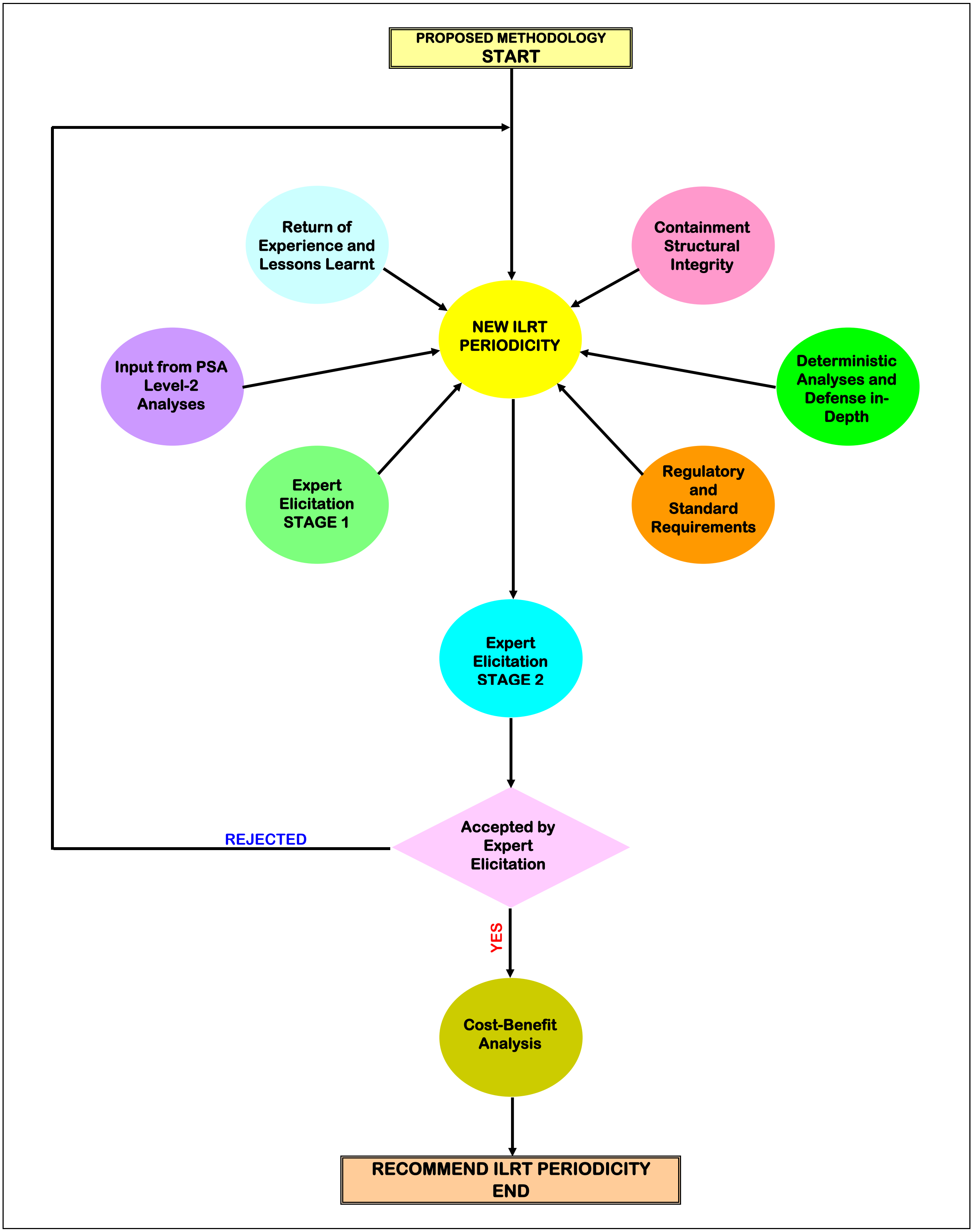

Figure 02: Flowchart of the Proposed ILRT Frequency Determination Methodology

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