INFO 0568
(ACNS 20)

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Advisory Committee on Nuclear

Comité consultatif de la sûreté n CA9600099

COMMITTEE REPORTS

Since the 1950's the Atomic Energy Control Board has made use of advisory committees of independent experts to assist it in its decision-making process. In 1979 the Board restructured the organization of these consultative groups, resulting in the creation of two senior level scientific committees charged with providing the Board with independent advice on principles, standards and general practices related to radiation protection and the safety of nuclear facilities. The two committees are the Advisory Committee on Radiological Protection (ACRP), formed in 1979, and the Advisory Committee on Nuclear Safety (ACNS), which was established a year later. Summaries of meetings of the committees are available to the public in the AECB library in Ottawa.

From time to time the committees issue reports which are normally published by the AECB and catalogued within the AECB's public document system. Committee reports, bound with a distinctive cover, carry both a committee-designated reference number, e.g. ACRP-1, and an AECB reference number in the "INFO" series. The reports generally fall into two broad categories: (i) recommendations to the Board on a particular technical topic, and (ii) background studies. Unless specifically stated otherwise, publication by the AECB of a committee report does not imply endorsement by the Board of the content, nor acceptance of any recommendations made therein.

RAPPORTS DES COMITÉS

Depuis les années cinquante, la Commission de contrôle de l'énergie atomique fait appel à des comités consultatifs composés d'experts indépendants pour l'aider dans ses prises de décision. En 1979, la CCEA a restructuré l'organisation de ces groupes de consultation pour former deux comités scientifiques supérieurs chargés de lui fournir des conseils indépendants concernant les principes, les normes et les méthodes générales touchant la radioprotection et la sûreté des installations nucléaires : ce sont le Comité consultatif de la radioprotection (CCRP), formé en 1979, et le Comité consultatif de la sûreté nucléaire (CCSN), établi l'année suivante. Le public peut consulter les comptes rendus des réunions de ces comités à la bibliothèque de la CCEA, à Ottawa.

Les comités rédigent à l'occasion des rapports qui sont normalement publiés par la CCEA et catalogués dans la collection des documents publics de la CCEA. Reliés avec une couverture distincte, les rapports des comités se reconnaissent à leur numéro de référence du comité d'origine (comme CCRP-1) et à leur numéro de référence de la CCEA dans la série «INFO». Ils se divisent habituellement en deux catégories générales : (i) les recommandations présentées à la Commission au sujet d'une question technique particulière; (ii) les études générales. À moins d'indication contraire, la publication par la CCEA du rapport d'un comité ne signifie pas que la Commission approuve le contenu de la publication, ni qu'elle en accepte les recommandations.

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ACNS Advisory Committee on Nuclear Safety

CCSN Comité consultatif de la sûreté nucléaire

ACNS-20

PROPOSED QUANTITATIVE APPROACH TO SAFETY FOR NUCLEAR POWER PLANTS IN CANADA

by the

Advisory Committee on Nuclear Safety

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ABSTRACT

A set of quantitative risk and frequency limits plus required processes is proposed to help ensure that a nuclear power plant in Canada meets the qualitative safety objectives defined in ACNS-2 and in IAEA 75-INSAG-3. As emphasized in this report, risks and hence doses are to be reduced below the limits using ALARA (As Low as Reasonably Achievable, economic and social factors being taken into account) or VIA (value-impact analysis) processes unless, in general, calculated risks and hence doses are below recommended *de minimis* levels. An updated version of ACNS-4, which will be issued as ACNS-21, will incorporate a statement of these limits and objectives as well as assessment criteria and procedures that will facilitate their application. The quantitative approach proposed here is consistent with a growing consensus on the need for, and the elements of, a quantitative approach to risk management of all major activities in an advanced industrial society. The ACNS recommends that the Atomic Energy Control Board adopt the proposed approach as a rational and coherent basis for nuclear power plant safety policy and requirements in Canada.

SOMMAIRE

Un ensemble de limites quantitatives de risque et de fréquence ainsi que de processus requis est proposé en vue d'aider à assurer qu'une centrale nucléaire au Canada satisfait aux objectifs qualitatifs de sûreté définis dans le rapport CCSN-2 de la CCEA et le rapport 75-INSAG-3 de l'AIEA. Tel que souligné dans le présent rapport, les risques, et par conséquent les doses, doivent être réduits au-dessous des limites par l'application du processus ALARA (de l'anglais *As Low As Reasonably Achievable*, c.-à-d. niveau le plus faible qu'il soit raisonnablement possible d'atteindre, compte tenu des facteurs sociaux et économiques) ou du processus VIA (de l'anglais *Value-Impact Analysis*, c.-à-d. analyse valeur-impact), à moins que, de façon générale, les risques calculés, et par conséquent les doses, soient inférieurs aux niveaux *de minimis* recommandés. Une version à jour du rapport CCSN-4, qui sera publiée sous le numéro CCSN-21, contiendra un énoncé de ces limites et objectifs ainsi que des critères d'évaluation et des méthodes qui faciliteront leur application. L'approche quantitative proposée dans le présent document est compatible avec un consensus grandissant en ce qui concerne la nécessité d'une approche quantitative, et les éléments qui la composent, pour la gestion des risques de toutes les activités importantes dans une société hautement industrialisée. Le CCSN recommande que la Commission de contrôle de l'énergie atomique adopte l'approche proposée en tant que base rationnelle et cohérente pour la politique et les exigences relatives à la sûreté des centrales nucléaires au Canada.

EXECUTIVE SUMMARY

The ACNS has reviewed the set of recommended general safety requirements for nuclear power plants embodied in report ACNS-4, published in 1983, in the light of developments since then. This review indicated a need for a basic rationale on which power reactor safety requirements, such as those in ACNS-4, should be established. The ACNS, after considerable study and debate, has now developed such a rationale which is described in this report.

Drawing on the qualitative safety objectives in its report ACNS-2, endorsed by the AECB in 1982, and in a report, 75-INSAG-3, of the International Safety Advisory Group (INSAG) of the International Atomic Energy Agency and taking into account safety and regulatory developments in Canada and elsewhere, the ACNS now proposes a quantitative approach to safety which should ensure that nuclear power plants in Canada meet the qualitative safety objectives of ACNS-2 and 75-INSAG-5.

The proposed approach, illustrated in Figure ES1, consists of:

* quantitative risk or frequency limits that must not be exceeded under any circumstances,

* the use of ALARA (As low as reasonably achievable) and VIA (Value-impact analysis) processes to establish acceptable risks or frequencies below the limit values,

* exemption levels based on de minimis risks below which ALARA or VIA processes are not required.

This approach is to be applied to members of the public, nuclear power plant workers, society and the environment for both normal operation and accident conditions. For members of the public the risks of early as well as delayed fatalities from accidents are to be taken into account. A summary of the elements of the proposed approach is shown in Table ES1.

The quantitative approach proposed here is consistent with a growing consensus on the need for, and the elements of, a quantitative approach to risk management of all major activities in an advanced industrial society.

This approach will serve as the basis for a revision of ACNS-4, which will be published as report ACNS-21. The ACNS recommends that it also serve as a rational basis for Consultative Document C-6, "Requirements for the Safety Analysis of CANDU Nuclear Power Plants", now being revised by the AECB staff, and for AECB policy on severe reactor accidents.

FIG. ES1 BASIS OF RECOMMENDED APPROACH

L1: MAXIMUM ACCEPTABLE LEVEL L2: EXEMPTION LEVEL

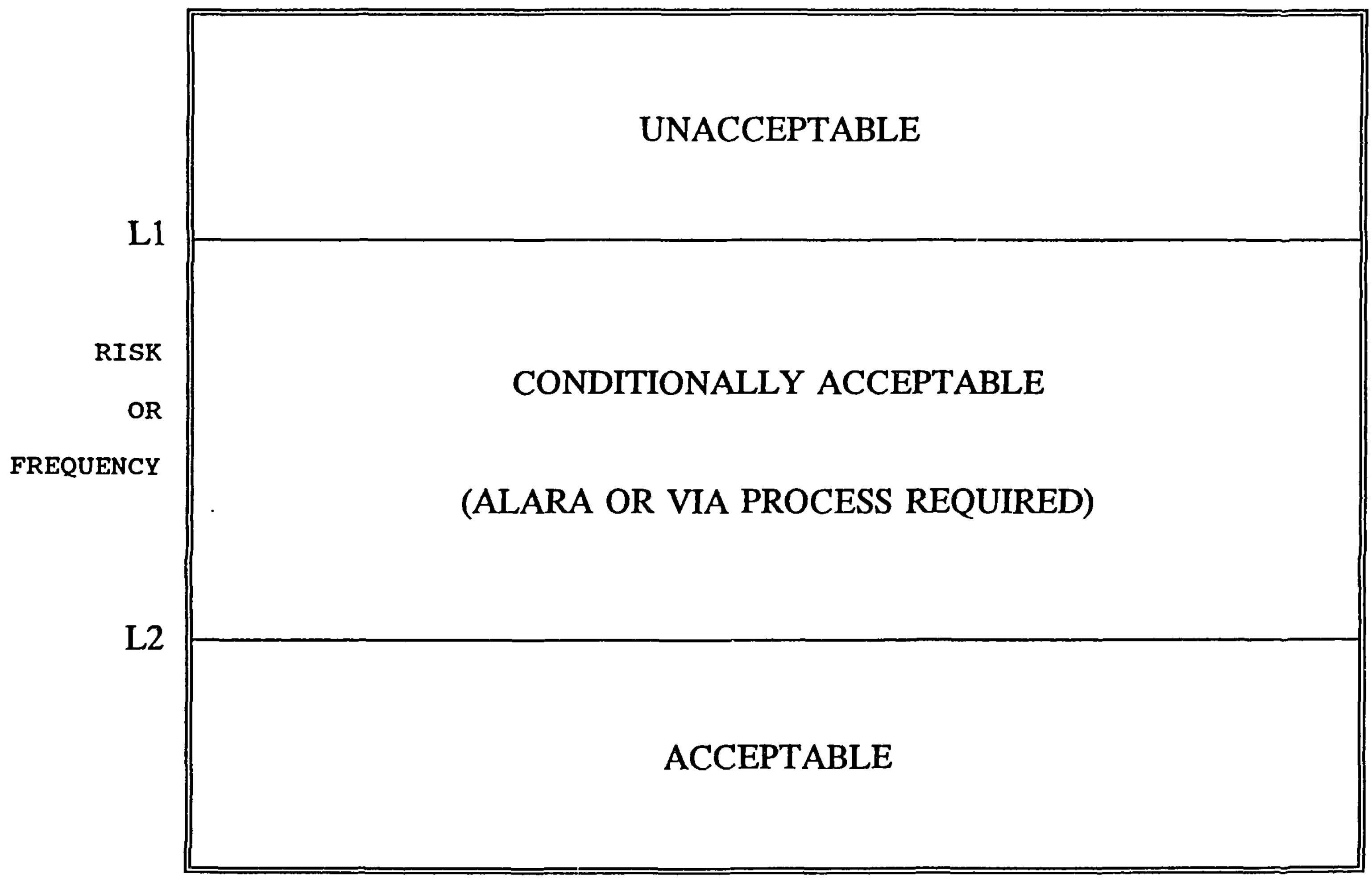

Table ES1

SUMMARY OF RECOMMENDED APPROACH					
Group	Plant Condition	Detriment	Risk or Frequency Limit	Required Process	Risk or Frequency Exemption Level*
Public-critical group	Normal operation	Delayed fatality	5×10^{-5}/site-year	ALARA	5×10^{-7}/site-year
Public-critical group	Accident	Delayed fatality	5×10^{-5}/site-year	VIA	5×10^{-7}/site-year
Public-critical group	Accident	Early fatality	1×10^{-5}/site-year	VIA	5×10^{-7}/site-year
Worker	Normal operation	Delayed fatality	8×10^{-4}/year	ALARA	5×10^{-7}/year
Worker	Accident	Early fatality - all causes	5×10^{-4}/year	VIA	5×10^{-7}/year
Worker	Accident	Early permanent disability - all causes	5×10^{-3}/year	VIA	5×10^{-6}/year
Society and Environment	Severe accident	Large release	10^{-5}/site-year (frequency)	VIA	10^{-6}/site-year (frequency)

* Exemption levels are set equal to *de minimis* levels for the first six groups.

Table of Contents

List of Tables

PROPOSED QUANTITATIVE APPROACH TO SAFETY FOR NUCLEAR POWER PLANTS IN CANADA

1.0 INTRODUCTION

One of the early tasks undertaken by the Advisory Committee on Nuclear Safety (ACNS) after its formation in 1980 was to develop a statement of qualitative general safety objectives for all nuclear activities in Canada in report ACNS-2 [1]. These safety objectives were endorsed by the Atomic Energy Control Board (AECB) at its meeting of April 21, 1982.

Subsequently, the ACNS prepared a set of proposed general safety requirements for nuclear power plants in Canada, published as report ACNS-4 in 1983 [2]. It includes a statement of safety principles and practices developed over the years in Canada. These are mainly deterministic and require a defence-in-depth design. The report also recommends risk-based limits for accident analysis representing the upper bound of acceptability and a procedure for doing the risk-based analyses. The AECB has stated that it intends to use ACNS-4 on a trial basis in the licensing of future power reactors.

The ACNS has reviewed ACNS-4 in the light of developments since it had been published in 1983. This review was prompted mainly by developments in Probabilistic Risk Assessment (PRA)[1] of nuclear power plants, especially the Darlington Probabilistic Safety Evaluation (DPSE) [3,4] and the CANDU-6 PRA [5]. A particular motivation was the comparison by Dinnie of the safety analysis requirements of ACNS-4 with the results of the DPSE analysis which indicated that the approach proposed in ACNS-4 for safety analysis was feasible but suggested certain improvements and clarifications [6]. Other factors which influenced the review and the present proposals include:

- recommendations on safety goals by various bodies [7,8,9]
- requirements for the use of the ALARA process in the AECB regulations [10,11]
- recommendations on *de minimis* dose rates and risks [12]
- new recommendations of the International Commission on Radiological Protection (ICRP) [13]
- developments in value-impact analysis (VIA) for accident conditions [14, 15]

In developing the present proposals, the ACNS has recognized the significant direct and indirect economic and social benefits resulting from the operation of nuclear power plants in Canada [16,17]. It has also recognized that risks to the public and to workers from this operation are very low [18] and, furthermore, are similar to or less than those from alternative commercially feasible methods of generating electricity in Canada [19]. Thus, a major concern of the ACNS is to ensure that appropriate optimized measures are taken in managing the risks of

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[1] Also termed Probabilistic Safety Analysis (PSA) or Probabilistic Safety Evaluation (PSE), although distinctions are sometimes made among these terms.

nuclear power plants so that Canada may continue to receive safely the benefits of nuclear-generated electricity.

The present report:

* outlines qualitative safety objectives for nuclear activities for which there is broad consensual support (Section 2.0);

* draws on these qualitative safety objectives to identify elements that should be incorporated in quantitative safety objectives, in particular a risk[2]-based approach (Section 3.1);

* reviews the development of risk-based safety approaches in Canada and elsewhere (Section 3.2);

* explains the basis for the recommended quantitative safety approach (Section 3.3);

* proposes a quantitative safety approach consistent with the qualitative safety objectives (Sections 3.3.1 to 3.3.6).

* recommends the adoption of the proposed quantitative safety approach as a rational and coherent basis for nuclear power plant safety policy and requirements in Canada (section 4.0).

2.0 QUALITATIVE SAFETY OBJECTIVES

The qualitative general safety objectives described in ACNS-2, which draw on the radiological protection principles of the ICRP [13,20], underlie the regulation of nuclear activities in Canada by the AECB.

These objectives are:

1. Nuclear activities should not lead to unacceptable risks to the workers involved or the general public.

2. For hazards due to ionizing radiation,

   a) all early detrimental effects to individuals should be avoided and the risks of deferred effects (such as consequential development of cancer or production of hereditary defects) should be minimized in accordance with the ALARA principle;

   b) the probability of possible malfunctions that could lead to the escape of radioactive material or the exposure of people to ionizing radiation should be limited to small values, decreasing as the severity of the potential consequences increases so that the likelihood of catastrophic accidents is virtually zero.

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[2] In this report, risk is defined in the technical sense as the probability of some event and its consequences, to permit quantitative assessment. This approach does not preclude the public deciding what "risk" is acceptable in return for the associated benefits. Rather, it permits this decision to be made on a quantitative foundation, adding such other qualitative factors as may be desired.

3. For non-radiological hazards,

   a) the risk to workers and members of the public, from normal operation or practice of the nuclear activity, should be equal to or less than that presented by appropriately comparable industries or activities;

   b) the probability and potential consequences of possible malfunctions that could lead to harm of workers or members of the public should be as low as practicable.

4. The risk to any future generation associated with each nuclear activity should be taken into account and given a priority for prevention not less than that given to risks presented to the current generation.

On the international level, a widely recognized statement of qualitative safety objectives for nuclear power plants has been developed by the International Nuclear Safety Advisory Group (INSAG) of the International Atomic Energy Agency (IAEA) [21]. These safety objectives are:

1. General Nuclear Safety Objective.

   To protect individuals, society and the environment by establishing and maintaining in nuclear power plants an effective defence against radiological hazard.

2. Radiation Protection Objective.

   To ensure in normal operation that radiation exposure within the plant and due to any release of radioactive material from the plant is kept as low as reasonably achievable and below prescribed limits, and to ensure mitigation of the extent of radiation exposures due to accidents.

3. Technical Safety Objective.

   To prevent with high confidence accidents in nuclear power plants; to ensure that, for all accidents taken into account in the design of the plant, even those of very low probability, radiological consequences, if any, would be minor; and to ensure that the likelihood of severe accidents with serious radiological consequences is extremely small.

The ACNS has assessed the INSAG safety objectives in the light of ACNS-2 and has concluded that they are quite consistent with those of ACNS-2, taking into account their differences in scope[3] and the discussions of the safety objectives in the two reports [22].

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[3] ACNS-2 applies to all nuclear activities, while 75-INSAG-3 applies to nuclear power plants only; ACNS-2 considers all hazards while the INSAG objectives are confined to radiological hazards; ACNS-2 considers waste management, particularly in objective 4, while 75-INSAG-3 specifically excludes waste management.

3.0 QUANTITATIVE SAFETY REQUIREMENTS

This section first identifies the basic elements of a set of quantitative safety requirements, then reviews the evolution of risk-based safety requirements in Canada and elsewhere and finally describes the recommended safety approach.

3.1 Elements of Quantitative Safety Requirements

While the exact wording of the ACNS-2 qualitative safety objectives may be revised in future to focus their intent more clearly, there is a broad consensus on the principles embodied in them, as there is on the INSAG-3 qualitative safety objectives. Therefore, both of these documents are used here to identify the elements that should be incorporated in the quantitative safety requirements for nuclear power plants in Canada.

These elements are summarized as follows:

a) The safety requirements should be risk based.

b) Risks to members of the public, workers, society and the environment should be considered.

c) Normal operation (including anticipated upsets) and potential accidents should be taken into account.

d) Non-radiological as well as radiological risks should be taken into account.

e) Limits for risks and hence doses which must not be exceeded should be defined.

f) For normal operation, an ALARA process should be used. That is, risks and hence doses should be reduced below the limits to levels as low as reasonably achievable (ALARA), social and economic factors being taken into account.

g) For accidents, risks should be small and should decrease as severity increases. The estimated frequency of severe accidents should be very low.

3.2 The Evolution of Risk-Based Safety Requirements

3.2.1 Regulatory Developments in Canada

Early efforts to develop overall risk-based safety objectives for nuclear power plants were undertaken in Canada by Siddall and Lewis [23], and Laurence [24], as a result of the accident to the NRX reactor in 1952. These efforts eventually led to the adoption by the AECB of a set of requirements, termed the Siting Guide, which set maximum permissible predicted frequencies and individual and collective public doses for two categories of accidents [25], thus incorporating a risk element in the regulatory process.

Later, revisions to the Siting Guide were proposed by an inter-organizational working group (IOWG) which required that the summed predicted frequencies of accidents in six severity categories (measured by the dose to a member of the critical group) should not exceed specified limits, thus implying an overall risk limit [26]. Some of the recommendations of the IOWG were subsequently incorporated in a safety analysis licensing guide, AECB Consultative Document C-6 [27], which stipulated five accident categories with individual dose limits for each but did not explicitly specify frequency limits, thus departing from the risk-based approach previously used in Canada. However, during the application of C-6 on a trial basis (along with the Siting Guide) for the licensing of the Darlington NGS, Ontario Hydro proposed single-event frequency criteria for the categories that were agreed to by the AECB for the licensing of this station.

In 1983 the ACNS issued its report ACNS-4 [2], which proposes six dose categories with limits for the summed frequencies of event sequences in each, similar to those of the IOWG, thus implying a total risk limit. The frequency limits include a "risk-aversion" feature, i.e. the risk limits decrease as the consequences increase, recognizing society's particular concern about large accidents with severe consequences. The risk limits in ACNS-4 are about the same for normal operation and for accidents. Very low frequency event sequences (frequency less than 10^{-7} per reactor-year) are explicitly excluded from the need for consequence analysis since the additional risks of such accident sequences would make an insignificant contribution to the total risk of the plant.

In 1990, Ontario Hydro formally adopted qualitative and quantitative safety goals for its nuclear generating stations [28]. A rigorous comparison of the results of its earlier probabilistic safety evaluation of the Darlington NGS [3] with the quantitative safety goals adopted by Ontario Hydro is not possible since the Darlington PSE did not include consequence analyses for accident sequences in the most severe and least severe categories for radioactive releases from the plant.

Other regulatory developments in Canada include the incorporation of the ALARA principle into the AECB regulations and the publication of recommendations for the application of the ALARA principle in nuclear activities in Canada by the ACNS and the Advisory Committee on Radiological Protection (ACRP) [11]. Also, the ACNS and ACRP have published recommendations on *de minimis* dose rates in Canada [12].

3.2.2 International Regulatory Developments

About the same time as the development of the Siting Guide took place in Canada, other proposals, most notably that of Farmer [29], were made internationally for some form of risk criteria for nuclear power plants, to supplement the deterministic approaches then in use.

An important step towards the use of comprehensive risk-based safety analysis was the Reactor Safety Study conducted for the US Atomic Energy Commission in the mid 1970s [30]. This was the first large-scale application of fault-tree and event-tree techniques to analyze the safety of nuclear power plants. Despite a number of limitations and difficulties, it demonstrated the feasibility of the PRA methodology.

The TMI nuclear plant accident in 1979 focused interest on PRA methodology and inspired efforts to define quantitative safety objectives for nuclear power plants. The US Nuclear Regulatory Commission (USNRC) initiated a program to define comprehensive safety goals for nuclear power plants in the USA which, after several years of work and debate, led to the publication of a regulatory policy for qualitative safety goals and quantitative safety objectives in 1986 [7].

Recently, the USNRC has completed probabilistic safety analyses for five nuclear power plants in the USA [31]. The results show that these existing plants readily meet the USNRC quantitative safety objectives and guidelines.

In other parts of the world, a number of national and international agencies have looked seriously at the possibility of defining quantitative safety goals. The Netherlands in 1985 established risk limits and safety goals within their environmental laws which apply to all potentially hazardous facilities, not just nuclear power plants [9,32]. Also, Argentina has adopted an individual risk criterion in the regulatory process for nuclear power plants [33].

The IAEA has developed comprehensive safety goals (referred to as probabilistic safety criteria) for nuclear power plants [8]. In 1992, the United Kingdom Health and Safety Executive published a revision of its earlier document on "The Tolerability of Risk from Nuclear Power Stations" [34], to provide public information on nuclear power risks and risk management.

Over the past decade, probabilistic safety analyses have been conducted on a number of nuclear power plants throughout the world, in addition to those already discussed, with increasing confidence in the results [35].

3.3 Recommended Safety Approach

Based on reviews of nuclear safety practices in Canada and elsewhere, as summarized in the foregoing discussion, a safety approach is proposed for nuclear power plants in Canada which should ensure that the ACNS and INSAG qualitative safety objectives are met.

In view of the interest being shown by many organizations (for example, USNRC [7], IAEA [8] and the government of the Netherlands [9]) in establishing not only a quantitative dose or risk limit which must not be exceeded but also a more stringent risk goal which must be striven for, the Committee considered the merit of including such a goal in its proposed approach. After careful and lengthy debate, the Committee concluded that the safety approach should not set arbitrary goals but rather should incorporate a process sensitive to the cost of providing safety measures. This approach follows the radiation protection principles of the ICRP [20].

The proposed approach consists of:

* stipulated quantitative risk or frequency limits that must not be exceeded for members of the public, nuclear power plant workers, and society and environment for both normal operation and accident conditions

* the application of ALARA processes for normal operation and value-impact analysis[4] processes for accident conditions to establish acceptable risks and frequencies below the limit risks and frequencies

* exemption levels based on *de minimis* doses and hence risks, below which the ALARA or VIA processes are not required.

The basis of the recommended approach is illustrated in Figure 1. The three-field approach shown is similar to the approaches described in references 8, 9 and 32. The quantitative maximum acceptable levels are based on the new dose-limits recommended by the ICRP [13] and the corresponding fatality risks. While the ICRP recommendations are expressed in terms of dose limits, these dose limits have been based on an evaluation by the ICRP of tolerable risk limits. The ICRP recommendations represent an international consensus by recognized experts on radiological dose limits which has been accepted by all countries with nuclear power programs and which has served as the basis for AECB regulations over the years. The ACNS could identify no more practical basis for the recommended limits for normal operation than the new ICRP recommendations. The risks corresponding to the limits for normal operation were then used as the basis for the proposed limits for accident conditions.

The approach recommended in this report stipulates quantitative risk limits for both normal operation and accident conditions. Normal operation and accidents are treated separately because of their different nature and the different means used to monitor their risks and to take remedial action as necessary, as will be discussed later. To ensure that optimized levels of calculated risks below those at the risk limits are achieved, the approach also recommends that, for normal operation, acceptable calculated risks be established by an ALARA process [11] while, for accident conditions, the acceptable calculated risks be established by a VIA process [15].

The approach also recommends that, in general, to avoid unproductive expenditures in investigation and analysis, calculated individual doses or equivalent risks below the *de minimis* levels be accepted without requiring an ALARA or VIA process. The stipulation of this lower limit to the required range for ALARA or VIA processes recognizes that risks at this level are generally accepted as being of no significance to an individual or, in the case of a population, of no significance to society and are already so low that efforts to reduce them further including the cost of analysis are not warranted [12]. Consistent with this approach, i.e., avoiding unproductive expenditures, efforts expended in applying the ALARA or VIA processes should be proportional to the difference between the calculated risks and those at the *de minimis* levels.

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4 Value-impact analysis (VIA) is a general quantitative approach which considers the "values" (or benefits) and the "impacts" (or costs) of preventive and mitigating measures taken to reduce the calculated risks of nuclear reactor accidents. Values include such factors as on-site and off-site health effects averted. Impacts include factors such as development and implementation costs and changes in operating costs. The VIA process weighs values against impacts using various methods taking into account uncertainties [14,15] to establish design and operating procedures to yield an optimized level of calculated risks from reactor accidents.

The approach specifies separate risk limits and required processes for delayed and early fatalities for members of the public as in the approaches of the IAEA [8], the USNRC [7] and Ontario Hydro [28], as well as other organizations. The reasons for separate limits and requirements for early and delayed fatalities are [28]:

a) The total risks of early and delayed fatalities in Canada differ[5].

b) The life-shortening effects for early (i.e., accident) fatalities, typically 30 years, are greater than those for delayed fatalities, typically 10 years, because of the latency periods for cancers.

c) Separate limits and required processes facilitate comparisons with the risks of other industries or activities which may have the potential for early or delayed fatalities, but not both.

In establishing the limits and requirements for a nuclear power site, it is assumed that all of the risk to the public from reactors arises from the radiological hazard. While there are contributions to total public risk from other hazards during operation and under accident conditions, these are negligible compared to the risks from radiological exposure [19]. However, fatality risks to nuclear power plant workers arise from various hazards and this is recognized in the recommendations.

Separate limits and required processes are proposed for:

* normal operating conditions for members of the public

* accident conditions resulting in delayed fatalities for members of the public

* accident conditions resulting in early fatalities for members of the public

* normal operating conditions and rehabilitation periods following accidents for nuclear power plant workers

* accident conditions for nuclear power plant workers

* society and environment (frequency based).

Separate risk limits and processes are not proposed for population or collective doses per se, but collective risks as well as benefits are recognized in the ALARA and VIA processes and societal impacts are taken into account by the recommendations in section 3.3.6 of this report. It is also recommended, as in ACNS-4 [2], that radiological risks to the population for both normal operation and accidents be taken into account in the site selection process.

In recommending the risk-based and frequency-based approach described in this report, the Committee is not abandoning the safety principles and practices (many

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5 The total risk to a member of the public of individual delayed (cancer) fatalities is 2.03×10^{-3} per year and that of early (accident) fatalities is 4.9×10^{-4} per year, based on 1991 data for Canada [36].

of which are deterministic) developed over the years in Canada. These will remain key elements in reactor safety.

The risk from radiological exposure used throughout this report is that of fatality only, rather than the risk of total detriment, i.e., total harm experienced by an exposed group including fatal cancers, non-fatal cancers and serious hereditary defects, as defined by the ICRP [13]. The risk of total detriment from small (stochastic-effect) doses is about 7.3×10^{-2} per Sv for the general population compared to 5.0×10^{-2} per Sv for fatalities and is about 5.6×10^{-2} per Sv for workers compared to 4.0×10^{-2} per Sv for fatalities. Establishing an approach based on maintaining the risks of fatalities at acceptable levels will also ensure that the risks of total detriments are also at acceptable levels. For high (deterministic-effect) doses which could lead to early fatalities, only the fatality risk is relevant [13].

A set of assessment criteria to facilitate the application of the maximum acceptable limits and required safety processes to a nuclear power plant will be incorporated into a revised version of ACNS-4, which will be issued as ACNS-21 [37].

3.3.1 Risk Limit and Required Process for Normal Operating Conditions for Members of the Public

Risk Limit

The calculated risk of fatality to a member of the critical group of the public from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site must not be greater than 5×10^{-5} per site-year.

This risk limit corresponds to an annual dose[6] of 1 mSv based on the latest risk coefficient recommended by the ICRP, 5×10^{-2} per sievert [13]. An annual dose of 1 mSv is that now recommended by the ICRP as the dose limit to a member of the public [13]. It is anticipated that the Atomic Energy Control Regulations will be amended in the near future to reflect the latest ICRP recommendations [10,38]. Consistent with the ICRP approach, this dose, and the corresponding risk, represent the upper limit of tolerability for normal operation.

The risk limit here, and elsewhere in this report, is based on a nuclear power site, rather than a nuclear generating station or a single reactor. This ensures a consistent risk level for members of the public in the vicinity of nuclear power sites which may be occupied in Canada by as many as eight reactors. This requires that, other things being equal, a multi-unit site has lower risk per reactor than a single-unit site. This lower risk is achievable, as would be demonstrated in the site PRA, because of the ready availability of alternative power and water supplies as well as additional skilled workers. Such features should assist in ensuring that a multi-unit site can meet the risk-limit requirements without requiring that the risk levels of individual reactors be impossibly low.

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6 As used in this report, dose means committed effective dose.

Required Process

The calculated risk of fatality to a member of the critical group of the public from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site shall be below the value at the risk limit and be justified by an ALARA process unless the calculated risk is below the de minimis level.

The emission levels of various constantly monitored radioactive groups will be compared to the levels derived from the ALARA process on a periodic basis. The emission levels will fluctuate with time; only when they are consistently high by a certain margin with respect to the derived ALARA levels over several periodic measurements would action be required through an examination of operating procedures or plant design to determine whether improvements were necessary. The specific margins and periods to be used will be defined in the forthcoming ACNS-21 [37].

Before the ALARA principle became part of the AECB regulations, the target for nuclear power plants under normal operating conditions (based on discussions between the AECB and the licensees) was chosen to be less than 1% of the then-existing AECB annual dose limit of 5 mSv for a member of the public. Unfortunately, the "1%" target level has never been defined clearly and several interpretations exist. One interpretation is that the annual dose to a member of the critical group of the public should not exceed 50 μSv from each of air-borne or water-borne pathways [39], or 100 μSv total, if both pathways affect the same critical group. On the other hand, the "1%" target has been interpreted by Ontario Hydro as applying to the derived emission limits (DEL) for each radionuclide grouping, i.e., tritium, noble gases, iodines and particulates for air-borne emissions and tritium and gross beta-gamma for water-borne emissions. Recently, groupings of carbon-14 have been added to each of the air-borne and water-borne categories. Thus, the effective dose to a member of the public, in the unlikely event that each group just meets the 1% target, could be as high as 400 μSv total.

In practice however, the calculated doses to individual members of the public near Canadian nuclear generating stations have, in general, been below these "1%" targets, however interpreted, as shown, for example in Table 1, for 1992.

Recently, the AECB has proposed, in a consultative document, an approach to the requirement to keep all exposures as low as reasonably achievable which establishes a total individual annual dose of 50 μSv as a "de facto" ALARA dose [43].

The ACNS and ACRP have recommended an individual *de minimis* dose rate of 10 μSv per year [12], which is equivalent to a risk of fatality from radiological exposure of 5×10^{-7} per year, based on the latest ICRP recommendation for the risk coefficient for a member of the public. The calculated risk to an individual must consider contributions from the entire nuclear power site so that the exemption level for the ALARA process is defined to be a risk of fatality from radiological exposure of 5×10^{-7} per site-year to a member of the critical group of the public.

Table 1

Calculated Total Annual Effective Doses to
Members of Critical Groups at Canadian NGS, 1992

Station	Total Annual Effective Dose, μSv
Pickering A & B	23
Bruce A & B	6.4
Darlington	2.8
Gentilly	18.5
Point Lepreau	1.3
Derived from references 40, 41, 42	

The exemption-level risk is thus 1% of that at the risk limit. It is pointed out that this risk and the corresponding annual dose are 20% of the de facto ALARA level proposed by AECB [43]. As stated in section 3.3, efforts expended in applying the ALARA process should be proportional to the difference between the calculated risk at any stage of the process and the *de minimis* level.

3.3.2 Risk Limit and Required Process for Delayed Fatalities under Accident Conditions for Members of the Public

Risk Limit

The calculated risk of delayed fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site must not be greater than 5 x 10^{-5} per site-year.

The ICRP, in recommending a dose limit, and thus a fatality risk limit, for members of the public from normal operation, implicitly acknowledged an additional risk from accidents. Recognizing this, and lacking any other basis for establishing the risk limit for delayed fatality to a member of the critical group from accidents, the ACNS arbitrarily set it equal to that for fatality from normal operation. This approach is similar to that proposed in ACNS-4 [2] and is consistent with that recommended by the IAEA and the Nuclear Energy Agency of OECD [35].

The total recommended risk limit for delayed fatalities to the public from both normal operation and from all accidents for a nuclear power site is thus 10^{-4} per site year.

A probabilistic risk analysis of the nuclear power site would be required to calculate the risk. The analysis is to exclude the risk of earthquakes, since their frequency, magnitude and consequences cannot be predicted with sufficient reliability.[7] Deterministic means, as in current practice [3,21,44] are to be

[7] The comprehensive PRA study undertaken by the USNRC, NUREG-1150 [31], did not determine public risk from earthquakes at nuclear power plants in the USA, although it did estimate core damage frequencies for seismic events at two of the five nuclear power plants studied.

used to ensure that the risks to the public from seismic events are negligibly small. Protection of the public against earthquakes is provided by requiring that design and construction follow well-established standards for seismic events [45]. Confidence that the deterministic means provide adequate protection rests primarily on sound seismic design and engineering judgment, supported by seismic safety margin analyses [46] and confirmed by plant experience in actual earthquakes, for example at the Perry Nuclear Power Plant in Ohio[8] [47].

The ACNS recommends that, in addition to the deterministic design approach to protect the public from earthquakes, the issue of this risk should be addressed in the site selection process, as recommended in ACNS-4 [2].

<u>Required Process</u>

The calculated risk of delayed fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site shall be below the value at the risk limit and be justified by a VIA process unless the calculated risk is below the de minimis level.

As for normal operation, the exemption-level annual risk of delayed fatality from accidents to a member of the critical group below which a VIA process is not required is 5×10^{-7} per site-year, which is equal to the recommended *de minimis* individual annual risk [12]. Thus, it maintains for the exemption levels the balance between risks of delayed fatalities from normal operation and from accidents proposed for the risk limits. As for the risk limit, the risk from earthquakes need not be considered in this analysis.

Site designs exhibiting annual risks from accidents of delayed fatalities to a member of the critical group in the range of 5×10^{-7} to 5×10^{-5} would be considered acceptable only if the risk were justified by a value-impact analysis. As in the application of an ALARA process for normal operations, efforts expended in applying the VIA process in this case should be proportional to the difference between the calculated risk at any stage of the process and the *de minimis* exemption level.

While the recommended maximum acceptable risk of 5×10^{-5} and exemption-level risk of 5×10^{-7} are the same for normal operation and for accident conditions, the actual risk for accident conditions will not necessarily be the same as that for normal operation, since each is established by an independent process. However, these processes together should yield an optimum overall risk for the nuclear power site.

Since it is not possible to measure directly the risk from accidents as it is to measure emissions and hence risks during normal operations, it would be required that a procedure be in place for regularly updating the PRA throughout the life of the nuclear power site based on reliability data obtained from system testing and in-service inspection and on results of research on safety issues. An indication that the risk level had increased would then prompt a review to determine

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[8] In 1986, a magnitude 5 earthquake with its epi-centre 17 km from the Perry NPP resulted in a peak ground acceleration and a frequency response spectrum which exceeded design values, but no structural damage occurred at the plant [47].

what action, if any, should be taken. In this case, the impacts considered in any VIA process would also include such factors as occupational exposures during and after implementation and replacement power costs and health effects [15].

3.3.3 Risk Limit and Required Process for Early Fatalities under Accident Conditions for Members of the Public

Risk Limit

The calculated risk of early fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site must not be greater than 10^{-5} per site-year.

Required Process

The calculated risk of early fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site shall be below the value at the risk limit and be justified by a VIA process unless the calculated risk is below the de minimis level.

The ACNS recommends that maximum acceptable and exemption-level risks for early fatalities to members of the public resulting from accidents at nuclear power sites be specified for the reasons given in section 3.3. Such early-fatality risk limits and exemption levels have not been explicitly stated by the AECB up to now. Again, the risk from earthquakes need not be considered in the analyses. The early-fatality risk limit is set below the delayed-fatality limit because the life-shortening effects of early fatalities are greater than those for delayed fatalities and because of the greater public concern which would arise from immediate fatalities following an accident at a nuclear power site than would result from predictions of fatalities in the future.

The calculated total risk of early fatality from radiological exposure due to accidents will be established by the site probabilistic risk assessment. The acceptability of this calculated risk will be established following the same procedure used to establish the acceptability of the calculated risk of delayed public fatalities due to accidents. That is, as a criterion of unconditional acceptability, the calculated risk of early public fatalities should be below 5×10^{-7} per site-year. Should the plant PRA show that this criterion is not met, the major contributors to the risk must be identified, methods of risk reduction of these contributors proposed and value-impact analysis applied to establish an optimized risk level. Once again, efforts expended in applying the VIA process in this case should be proportional to the difference between the calculated risk at any stage of the process and the *de minimis* exemption level.

3.3.4 Risk Limit and Required Process for Normal Operating Conditions for Nuclear Power Plant Workers

The ACNS and INSAG qualitative safety objectives require that nuclear power plant workers be adequately protected from radiological hazards and the ACNS objectives require that they be protected against other hazards also.

Safety criteria have been recommended by the ICRP [13] for exposures of workers to radiological hazards and workplace safety standards have been established in general in all industrialized countries.

Rehabilitation of a nuclear power site following an accident represents a controlled and monitored situation and thus should be treated in the same manner as normal operation.

Based on these criteria and standards, risk limits are proposed here for nuclear power plant workers. As before, ALARA and VIA processes are to be used to reduce risks below those at the risk limits.

Risk Limit

The calculated average risk of fatality to a nuclear power plant worker from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site, and during any rehabilitation that follows a reactor accident must not be greater than 8 x 10^{-4} per year.

The above risk limit is the annual risk corresponding to a dose of 20 mSv per year received by a nuclear power plant worker. This dose limit over a period of five years is that now recommended by the ICRP for a radiation worker [13], which is expected to be adopted by the AECB [10,38], and the corresponding risk is based on the latest cancer-fatality risk coefficient for a worker, 4 x 10^{-2} per sievert, recommended by the ICRP [13]. The higher annual radiological fatality risk limit for normal operation for workers, 8 x 10^{-4}, than for members of the public, 5 x 10^{-5}, is justified on the basis that the exposure of workers to the risk is voluntary and compensated for, at least to some extent, and monitored so that over-exposures will be detected and prompt remedial action taken. It should be noted that the ICRP recommendations also permit a total dose to a worker in any given year of 50 mSv, corresponding to a fatality risk of 2 x 10^{-3}.

The actual record of worker exposure in Canadian nuclear power plants in recent years can be compared to these dose limits. From 1985 to 1992 inclusive, there were a total of 14 workers over-exposed at all Canadian nuclear power plants, or an average of fewer than two per year, based on the permitted one-year limit of 50 mSv, which was also the permitted average limit during this period [48]. The maximum worker exposure in this period was 129 mSv [49].

Required Process

The calculated risk of fatality to a nuclear power plant worker from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site, and during any rehabilitation that follows a reactor accident shall be below the value at the risk limit and be justified by an ALARA process unless the calculated risk is below the de minimis level.

In applying an ALARA process to establish doses and corresponding risks to nuclear power plant workers, the occupational dose performances achieved in nuclear generating stations in Canada in recent years (Table 2) should be taken into account. From Table 2, the highest of the average individual occupational doses to radiation workers at such stations has been about 7 mSv per year, while the lowest average dose has been about 2 mSv per year. These doses are equivalent to cancer fatality risks of about 2.8 x 10^{-4} and 8 x 10^{-5} per year, respectively. As Table 2 indicates, there has been a downward trend of the average dose with time.

Since nuclear power plant workers are also exposed to a variety of non-radiological risks, the ALARA process should include consideration of these risks along with the radiological risks, as recommended in report AC-2 [11] (particularly sections 4.3 and 8.3). As before, efforts expended in applying the ALARA process should be proportional to the difference between the estimated risk level and the *de minimis* risk level.

The *de minimis* fatality risk level for a worker is assumed to be the same as that for a member of the public, i.e., 5×10^{-7} per year.

Table 2

Average Individual Occupational Doses to Radiation Workers at Canadian Nuclear Generating Stations

	Average Effective Dose, mSv/a	Source Reference No.
Ontario Hydro (Average for all NGS, 1980-84)	7	18
Pickering A & B (1987)	4.5	50
Bruce A (1987)	3.7	50
Bruce B (1987)	2.0	50
All Canadian NGS (1989)	3.2	51
All Canadian NGS (1992)	2.5	52
All Ontario Hydro NGS (1993)	2.3	53

3.3.5 Risk Limits and Required Process for Accident Conditions for Nuclear Power Plant Workers

Risk Limits

The calculated total risk of early fatality to a nuclear power plant worker from all hazards due to accidents must not be greater than 5×10^{-4} per year, and the calculated total risk of early permanent disability to the same worker from all hazards due to accidents must not be greater than 5×10^{-3} per year.

The proposed early fatality annual risk limit for nuclear power plant workers from all hazards, i.e., non-radiological as well as radiological, is set at a level[9], 5×10^{-4}, which is comparable to the actual annual fatality risks to workers in such industries as mining (6.56×10^{-4}), transportation (3×10^{-4}), and construction (2.66×10^{-4}) as listed in Table 3. As in the case of normal operation, the early fatality accident risk limit for nuclear power plant workers, 5×10^{-4}, is greater than that for a member of the public, 1×10^{-5}, but the ratio of these risk limits is higher for accidents, 50, than for normal operation, 16, since the accident limit for workers includes the risks from all hazards while the normal operation limit for workers is based on radiological hazards only. The limits for the public are based on radiological hazards only for both normal operation and accidents.

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[9] Equivalent to 25 fatalities per 100 million hours worked, assuming an average of 2000 working hours per year.

The importance of taking appropriate action to prevent disabling injuries as well as fatalities to workers was emphasized in the Ontario Nuclear Safety Review on the safety of Ontario's nuclear reactors [18]. The proposed risk limit for permanent disabilities to nuclear plant workers, 5 x 10^{-3} per year, or 25 per 10 million hours worked, based on 2000 working hours per year, is comparable to the actual risks of permanent disabilities in the industries cited above [50].

Under accident conditions at a nuclear power site, urgent corrective action by plant workers may be necessary to prevent a large increase in the scale of an accident or to save a life. The corrective action may result in doses above the limit for normal operation to one or more plant workers. For limitations of doses under such conditions see references 13, 54 and 55.

Table 3

Fatality Rates for Major Industries in Ontario: 1981-1984

Industry	Fatality Rate for 100 million hours worked	Equivalent Annual Risk x 10^5
Forestry	40	80
Mining, Quarrying, Oil Wells	32.8	65.6
Transportation, Communication, and Other Utilities	15	30
Agriculture	13.5	27
Construction	13.3	26.6
Public Administration	3.8	7.6
Manufacturing	2.5	5
Services	1.3	2.6
Trade	1	2

Source: Adapted from Reference 18, Ontario Hydro Submission, Table 7-4.

Required Process

The calculated risks of early fatality and of early permanent disability to a nuclear power plant worker from all hazards under accident conditions shall be below the risk limits and be justified by a VIA process unless the calculated risks are below the corresponding de minimis levels. It shall also be ensured that a sound safety culture exists.

When applying a VIA process as described in section 3.3 in this case, the actual performance to date in Canadian nuclear generating stations and in Canadian industries that are considered safe should be taken into account.

Considering that there have been no at-work fatalities in almost 221 million hours of nuclear operations at Ontario Hydro to the end of 1993 [56], and assuming that fatalities would follow a Poisson probability density distribution, the observed performance would be consistent with a fatality risk of about 9.1 x 10^{-6} per year[10], less than 2% of the recommended risk limit in 3.3.5.

Data for all Ontario manufacturing industries give an average of 2.5 fatalities per 100 million hours worked over the period 1980 to 1986 [50], equivalent to a fatality risk of 5 x 10^{-5} per year, 10% of the proposed occupational risk limit. The fatality rate for the Ontario manufacturing industry of 2.5 fatalities per 100 million hours worked is low compared with that for many other industries in Ontario as can be seen in Table 3.

For permanent disabilities, the actual performance of Ontario Hydro Nuclear over the nine years between 1985 to 1993 inclusive resulted in an average of 3.5 x 10^{-5} permanent disabilities per year [57], compared to 3.28 x 10^{-3} for all Ontario manufacturing industries from 1980 to 1986 [50]. Although no *de minimis* risk level has been defined for early permanent disabilities, it is proposed that the *de minimis* level, and hence exemption-level, risk for early permanent disabilities be 5 x 10^{-6} per year, based on the ratio of the risk limit for early permanent disabilities to that for early fatalities.

While cost-benefit or value-impact analysis is important for assessing safety, other means are necessary to achieve it. For example, it is expected that some form of Institutional Quality Assurance (IQA) [58] will be in place to contribute to a positive and effective safety culture.

3.3.6 Frequency Limit and Required Process for Society and Environment

Most organizations whose publications were examined for this report have not defined a quantitative societal safety objective per se[11]. An exception is the U.S. Atomic Industrial Forum, which defined a societal safety goal of less than one public fatality per gigawatt-year of electrical energy produced [59]. There is much uncertainty in the calculation of such a parameter, but estimates for Canada indicate a public fatality risk from normal operation of from 0.06 to 0.29 public fatalities per GWa [19], based on the latest ICRP risk coefficient.

Nevertheless, most of these organizations have defined a large-release objective which then acts as a surrogate for a societal safety objective. Obviously, if the frequency of a large release is limited, then the frequency of severe impacts on society, such as large-scale evacuations, significant losses of livelihood over large regions as well as physical harm and psychological stress to large numbers of people, will also be limited.

In the same manner, a frequency limit for a large release serves as a surrogate for an environmental limit in that it will also set a maximum frequency for

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[10] This probability is obtained by equating the probabilities of observing zero and one fatality in 221 million hours worked, assuming a Poisson distribution and an average of 2000 working hours per year.

[11] Although the delayed fatality safety objective of the USNRC was initially intended to be a societal or collective safety objective, it is now essentially an individual safety goal [7,60].

large-scale impacts on the environment. As pointed out by the ICRP, limits on doses to the public will serve also to protect the environment [13, paragraph 16].

The necessity of defining a large-release objective, in addition to a safety objective for an individual member of the public, has been demonstrated by Dinnie [61] in a 1989 paper. He showed that while safety criteria focused on the individual will, if met, protect people near a nuclear generating station from the risk of fatality, they may not adequately protect against accidents having other significant societal consequences. He calculated that the Ontario Hydro individual safety goal for delayed public fatalities (1×10^{-5} per year) could permit a Chernobyl-type accident with a frequency of 1 per 1000 years[12]. A similar conclusion was reached by Whipple and Starr in assessing the USNRC safety goals in the light of the Chernobyl accident [62]. Considering his result, Dinnie recommended a societal safety objective based on the loss of livelihood suffered from a large accidental release of radioactivity from a nuclear power site.

The definition of a large release has caused some difficulties for the organizations proposing or adopting such a safety objective. For example, the USNRC has not defined it at all [7], while Ontario Hydro has defined it as one requiring the evacuation of large numbers of the public for long times, without defining precisely "large numbers" or "long times" [28], and the IAEA simply lists a number of possible definitions of a large release [8].

A large release cannot occur from a power reactor unless an accident results in severe core damage causing or coincident with a serious impairment of containment including containment by-pass. Therefore, it is proposed here to set a total frequency limit for such accidents. Since the methodology used in PRA studies which will be needed to apply the earlier risk limits and processes proposed here already involves the grouping of accident sequences according to consequences, i.e., according to core damage and containment impairment, this approach to a societal and environmental safety limit may be easier to apply than others proposed[13]. As before, the contribution of seismic events need not be considered in this analysis.

Considering the foregoing, the following societal and environmental frequency limit and required process are proposed.

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12 Dinnie used a radiation risk coefficient of 1×10^{-2} per sievert. Using the recent ICRP recommendation of 5×10^{-2} per sievert for the public risk coefficient, Dinnie's calculated frequency would become 1 in 5000 years, i.e., 2×10^{-4} per year, still a rather high frequency for such a significant accident.

13 It should be noted that the Advisory Committee on Reactor Safeguards in the USA has proposed the following definition of a large release: "A large release is any release from an event involving severe core damage, reactor coolant system pressure boundary failure and early failure or significant by-pass of the containment." [63] This definition is very close to the definition proposed here.

Frequency Limit

The calculated total frequency of accidents that result in severe core damage with resulting or coincident serious containment impairment in a nuclear power plant must not be greater than 10^{-5} per site-year.

Required Process

The calculated total frequency of accidents that result in severe core damage with resulting or coincident serious containment impairment in a nuclear power plant must be less than 10^{-6} per site-year unless a VIA process permits a higher frequency.

Since the criteria for this group are based on frequencies rather than risks, as in the previous groups, there is no link here to ICRP risk levels to establish the limits nor to the Advisory Committee *de minimis* levels to establish the exemption levels. The frequencies selected are based on a consideration of public tolerance for the frequency of accidents with significant societal or environmental consequences as reflected in the consensus that has developed amongst various national and international bodies concerned with reactor safety.

The IAEA, in proposing probabilistic safety criteria for nuclear power plants, has reviewed the proposals of the USNRC [7] and the UK Health and Safety Executive [34], as well as an objective adopted in France. Noting the consensus among these organizations, the IAEA [8] has proposed that the "design target" for the frequency of a large off-site release should be 10^{-6} per reactor-year, consistent with the consensus. Figure 1 of the IAEA report shows that the "design target" corresponds to line L2 in Figure 1 of this report, i.e., the exemption level frequency. The Nuclear Energy Agency of the OECD has also noted the growing international consensus on a "target" frequency, defined in the same way as the exemption level frequency [35, Table V], for major releases for radioactive materials from a nuclear power plant. It too proposes a value of 10^{-6} per plant year, which it notes is consistent with 75-INSAG-3 [21]. Therefore, the ACNS has adopted an exemption-level summed frequency for all event sequences that result in severe core damage with resulting or coincident serious containment impairment of 10^{-6} per site-year. Earthquakes need not be considered in this evaluation for the same reasons as those discussed in section 3.3.2 of this report.

It should be noted that the frequencies for large releases cited above are expressed in terms of frequency per reactor-year or per plant-year, whereas the exemption-level frequency specified in the required process is based on a site-year. The more stringent level proposed here reflects the existence of more than one reactor at certain sites in Canada, and the objective of setting the same exemption-level risk for society and the environment around any nuclear power site.

Because of concerns about accidents that could have significant social impacts, the calculated frequency limit for such accidents is set at 10^{-5} per site-year, giving only one order-of-magnitude range between the exemption-level and the maximum acceptable level in this category rather than the larger ranges in the other categories.

The definition of the criteria used to define "severe core damage with resulting or coincident serious containment impairment" will be provided in the forthcoming lower-tier document, ACNS-21 [37], since these criteria must be consistent with the consequence groupings for accident analysis which will be proposed there.

Should the PRA show that the exemption-level criterion is not met, methods of reduction of this frequency must be proposed and a VIA process applied to establish an optimized frequency level. Again, should the periodic up-dating of the PRA indicate that the calculated frequency of severe core damage with resulting or coincident serious containment impairment had increased, then a review must be undertaken using value-impact analysis to determine any action required. Efforts expended in applying the VIA process should be proportional to the difference between the calculated frequency and the exemption level frequency.

An earlier draft of the USNRC safety goals contained core-damage and containment impairment goals [60], while the proposed IAEA safety goals incorporate a core-damage safety criterion [8].

In addition to the Frequency Limit and Required Process, the ACNS expects the continuation of the existing requirement that there be emergency measures procedures adequate to prevent or minimize disastrous health consequences to the public in the event that such a low probability accident were to occur.

4.0 CONCLUSION AND SUMMARY OF RECOMMENDED SAFETY APPROACH

The quantitative approach to reactor safety proposed in this report incorporates the elements, identified in section 3.1, required to meet the principles embodied in the qualitative safety objectives of ACNS-2 [1], as endorsed by the Atomic Energy Control board, as well as those of the internationally recognized IAEA document 75-INSAG-3 [21].

The ACNS recommends that the Atomic Energy Control Board adopt the quantitative risk-based approach proposed here as a rational and coherent basis for reactor safety requirements, in particular for Consultative Document C-6, "Requirements for the Safety Analysis of CANDU Nuclear Power Plants" now being revised by the AECB staff [64]. The quantitative approach proposed here is consistent with a growing consensus on the need for, and the elements of, a quantitative approach to risk management of all major activities in an advanced industrial society, such as exemplified in references 9, 65 and 66. In the judgment of the ACNS, it is essential for the Atomic Energy Control Board, together with other regulatory agencies, to demonstrate to the public, workers, governments and industry that a rational, defensible approach to risk management underlies the regulations that it issues. A quotation from a recent public letter from the Advisory Committee on Reactor Safeguards to the Chairman of the US Nuclear Regulatory Commission is relevant here: "... the Commission is now engaged in a long-term metamorphosis into a risk-based regulatory agency, meaning that regulatory decisions will be, in the end, justified in terms of their expected impact on the health and safety of the public and the workers." [67]

The proposed approach incorporates recommendations related to severe accidents in nuclear power plants. Thus, adoption of this approach would also provide a rational policy basis for the regulatory treatment of severe reactor accidents, the subject of a recent AECB workshop [68], which would furthermore be consistent with the basis for other aspects of reactor safety.

For convenience, the essentials of the proposed quantitative approach to nuclear power plant safety are listed below and a tabular summary is given in Table 4.

Risk Limit and Required Process for Normal Operating Conditions for Members of the Public

Risk Limit

The calculated risk of fatality to a member of the critical group of the public from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site must not be greater than 5 x 10^{-5} per site-year.

Required Process

The calculated risk of fatality to a member of the critical group of the public from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site shall be below the value at the risk limit and be justified by an ALARA process unless the calculated risk is below the de minimis level.

Risk Limit and Required Process for Delayed Fatalities under Accident Conditions for Members of the Public

Risk Limit

The calculated risk of delayed fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site must not be greater than 5 x 10^{-5} per site-year.

Required Process

The calculated risk of delayed fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site shall be below the value at the risk limit and be justified by a VIA process unless the calculated risk is below the de minimis level.

Risk Limit and Required Process for Early Fatalities under Accident Conditions for Members of the Public

Risk Limit

The calculated risk of early fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site must not be greater than 10^{-5} per site-year.

Required Process

The calculated risk of early fatality to a member of the critical group of the public from radiological exposure due to accidents at a nuclear power site shall be below the value at the risk limit and be justified by a VIA process unless the calculated risk is below the de minimis level.

Risk Limit and Required Process for Normal Operating Conditions for Nuclear Power Plant Workers

Risk Limit

The calculated average risk of fatality to a nuclear power plant worker from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site, and during any rehabilitation that follows a reactor accident must not be greater than 8×10^{-4} per year.

Required Process

The calculated risk of fatality to a nuclear power plant worker from radiological exposure due to normal operations, including anticipated upsets, at a nuclear power site, and during any rehabilitation that follows a reactor accident shall be below the value at the risk limit and be justified by an ALARA process unless the calculated risk is below the de minimis level.

Risk Limits and Required Process for Accident Conditions for Nuclear Power Plant Workers

Risk Limits

The calculated total risk of early fatality to a nuclear power plant worker from all hazards due to accidents must not be greater than 5×10^{-4} per year, and the calculated total risk of early permanent disability to the same worker from all hazards due to accidents must not be greater than 5×10^{-3} per year.

Required Process

The calculated risks of early fatality and of early permanent disability to a nuclear power plant worker from all hazards under accident conditions shall be below the risk limits and be justified by a VIA process unless the calculated risks are below the corresponding de minimis levels. It shall also be ensured that a sound safety culture exists.

Frequency Limit and Required Process for Society and Environment

Frequency Limit

The calculated total frequency of accidents that result in severe core damage with resulting or coincident serious containment impairment in a nuclear power plant must not be greater than 10^{-5} per site-year.

Required Process

The calculated total frequency of accidents that result in severe core damage with resulting or coincident serious containment impairment in a nuclear power plant must be less than 10^{-6} per site-year unless a VIA process permits a higher frequency.

Table 4

SUMMARY OF RECOMMENDED APPROACH					
Group	Plant Condition	Detriment	Risk or Frequency Limit	Required Process	Risk or Frequency Exemption Level*
Public-critical group	Normal operation	Delayed fatality	5×10^{-5}/site-year	ALARA	5×10^{-7}/site-year
Public-critical group	Accident	Delayed fatality	5×10^{-5}/site-year	VIA	5×10^{-7}/site-year
Public-critical group	Accident	Early fatality	1×10^{-5}/site-year	VIA	5×10^{-7}/site-year
Worker	Normal operation	Delayed fatality	8×10^{-4}/year	ALARA	5×10^{-7}/year
Worker	Accident	Early fatality - all causes	5×10^{-4}/year	VIA	5×10^{-7}/year
Worker	Accident	Early permanent disability - all causes	5×10^{-3}/year	VIA	5×10^{-6}/year
Society and Environment	Severe accident	Large release	10^{-5}/site-year (frequency)	VIA	10^{-6}/site-year (frequency)

* Exemption levels are set equal to *de minimis* levels for the first six groups.

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GLOSSARY

Absorbed Dose - The amount of radiation energy deposited per unit mass of tissue.

Accident - An event or sequence of events leading to fuel damage which results in a significant release of radioactive materials from the fuel beyond those identified in the license as anticipated upsets.

ALARA - The process by which all radiological exposures are kept As Low As Reasonably Achievable, economic and social factors being taken into account.

Anticipated Upsets - All operations deviating from normal conditions, as identified in the license, beyond specified limits and conditions that may be expected to occur one or more times during the life of a nuclear power plant and which do not lead to accident initiating events.

Committed Dose - The total absorbed dose received from a radioactive substance in the body during 50 years following the intake of this substance.

Critical Group - A fairly homogeneous group of people whose location, age, habits, diet, etc., cause them to receive an average dose from a given source (or a given combination of sources) of radiation which is greater than the average dose received by any other group exposed to that source or combination of sources.

de minimis Dose or Risk - A level of dose or risk generally accepted as being of no significance to an individual or to society.

Deterministic Method - A prescriptive method of safety design or analysis that explicitly excludes the use of probabilistic arguments to set safety requirements.

Early Fatalities - Fatalities resulting from high radiation doses delivered at high dose rates occurring within one year of exposure. The effects of such doses are called "deterministic effects" and occur only if certain threshold doses are exceeded.

Effective Dose - The weighted sum of all the equivalent doses in all the tissues and organs of the body (formerly known as effective dose equivalent). The weighting factors ensure that the detriment is equal whether or not the whole body is irradiated uniformly.

Equivalent Dose - A quantity that expresses all radiation doses on a common biological scale. It is the product obtained by multiplying the absorbed dose in tissue by a quality factor to account for the different potential for injury of different types of radiation.

Event Sequence - A progression of events following an initiating event which describes possible responses of a nuclear power plant in a logical sequence and which may result in a release of radioactive materials.

Exemption Level - A level of dose or risk below which normal operation is acceptable without an ALARA process. A level of calculated dose, risk or frequency for accident conditions below which a reactor design is acceptable without a VIA process.

Late Fatalities - Fatalities occurring at random one or more years after exposure to low radiation doses at low dose rates. The effects of such doses are called "stochastic" effects and are assumed to have no threshold with their probability of occurrence being proportional to the effective dose.

Normal Operation - The operation of a nuclear power site within specified limits and conditions and under anticipated upsets, including start-up, power operation, shutdown, maintenance and testing.

Nuclear Power Site - A site used to produce electricity from nuclear fission. It may be composed of one or more nuclear power reactors affecting the same critical group. The reactors at a nuclear power site may comprise one or two Nuclear Generating Stations (NGS), e.g., Pickering A-NGS (4 reactors) and Pickering B-NGS (4 reactors) comprise one nuclear power site.

Probabilistic Method - A method of safety design or analysis that takes into account the probabilities of various events.

Radiation Worker - Any person who in the course of his or her work is likely to receive a dose of ionizing radiation in excess of the dose limits specified for the public in the Atomic Energy Regulations, as amended.

Risk - the product of the probability of the occurrence of an event and the consequences of the event.

Risk Limit - A level of risk above which normal operation is unacceptable under any circumstances. A level of calculated risk or frequency for accident conditions above which the reactor design is unacceptable.

Safety Culture - That assembly of characteristics and attitudes in organizations and individuals which establishes as an overriding priority that nuclear site safety issues receive the attention warranted by their significance.

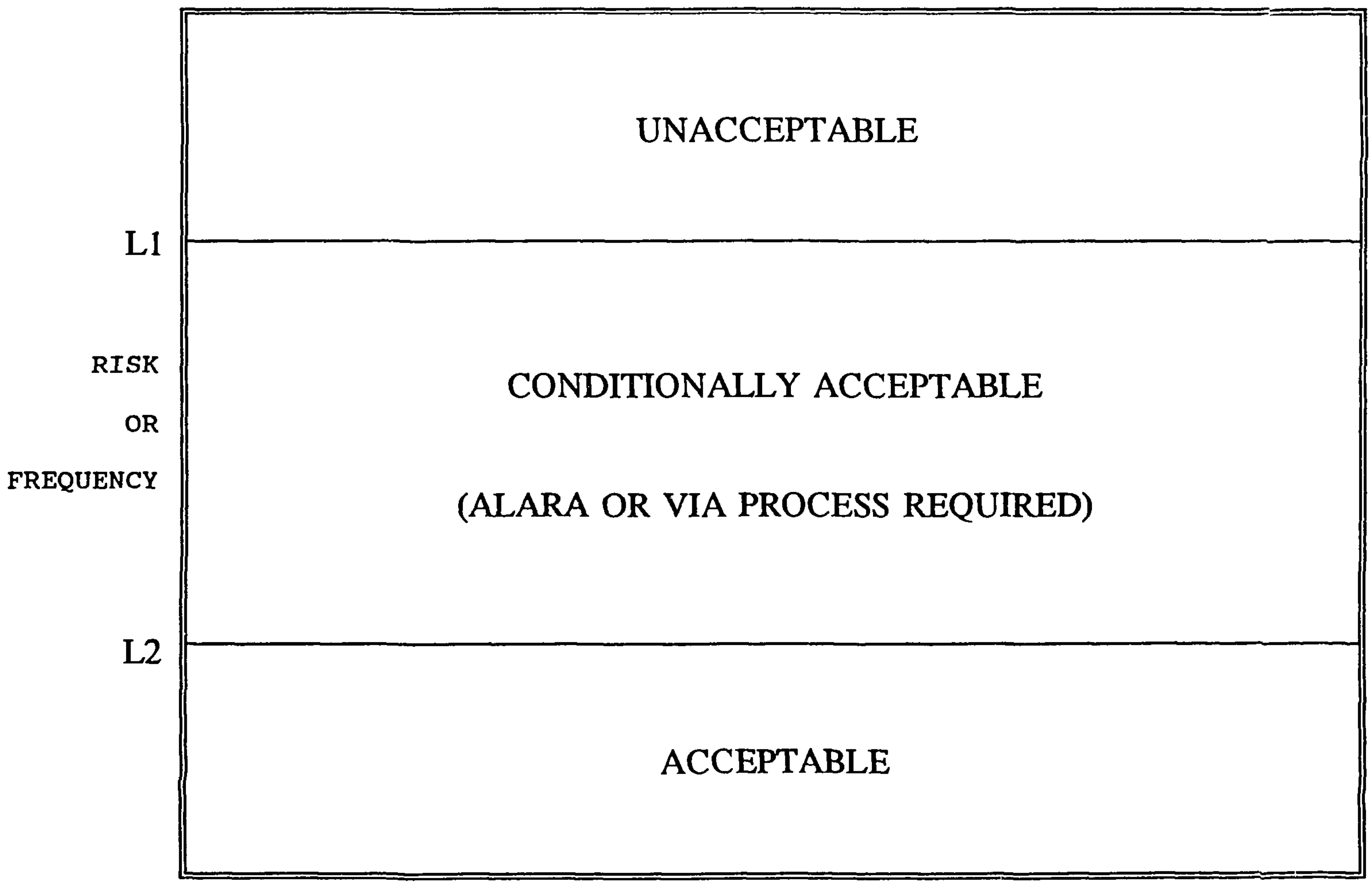
Serious Containment Impairment - A state of the containment of a nuclear power plant which would permit a significant release of radioactive substances from containment.

Severe Core Damage - Loss of reactor core structural integrity, e.g. by fuel melting, following failure of heat removal or power excursion which results in significant releases of radioactive substances from the fuel.

Value-Impact Analysis (VIA) - A cost-benefit analysis to establish the extent to which accident impacts or risks may be optimized, taking into account economic and social factors. For an example of the application of this process, see references 14 and 15.

FIG. 1 BASIS OF RECOMMENDED APPROACH

L1: MAXIMUM ACCEPTABLE LEVEL   L2: EXEMPTION LEVEL



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