MCMASTER NUCLEAR REACTOR SAFETY ANALYSIS METHODOLOGY OVERVIEW [1, 2]

Wm. J. GARLAND, Professor, Department of Engineering Physics
C. HEYSEL, Director of Nuclear Operations and Facilities
McMaster University
Hamilton, Ontario, Canada

ABSTRACT

A safety analysis methodology overview is presented for the McMaster Nuclear Reactor focusing on meeting the frequency-dose limits derived from IAEA guidelines. The approach was, thus, based on the identification and exploration of event trees and fault trees characteristic of the probabilistic approach. However, since sufficiently accurate failure frequencies could not be established for the manual operations inherent in this research reactor, best estimates were used and a "cut-off" frequency of 10^{-6} events per year was not imposed. Rather, events were nominally followed to fuel failure for the most frequent events in that category.

Typical cell and core codes were used for reactor physics and fission product inventory analysis. In contrast, the simple, single phase thermal-hydraulics and the large uncertainties associated with fission product release and transport did not warrant the use of complex codes; thus, augmented hand calculations proved sufficient.

This practical approach proved successful in the timely preparation of the Safety Analysis Report and helped identify focus areas for operational and technical support staff.

1 INTRODUCTION

The construction and operation of McMaster Nuclear Reactor (MNR) are described in the original 1957 application to the Canadian nuclear regulator [1]. This document was revised in 1972 when five megawatt operation began [2]. Since then, the facility and equipment have been renewed and modified, and operation and usage have expanded and changed. The philosophy of reactor safety analysis has also changed, and there have been many advances in tools and techniques. Thus, it is appropriate that a new report be issued which considers all of these factors. This Safety Analysis Report (SAR) follows the IAEA guidelines as set out in [3], with changes to accommodate the unique features of the MNR facility and to incorporate the fact that MNR is not a new reactor but rather a mature facility with over forty years of operating history and data which can be used to quantify items which could only be estimated for a 'yet-to-be-built' reactor.

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[1] Much of the text of this paper was taken verbatim from the recently prepared Safety Analysis Report, authored by those mentioned in the ACKNOWLEDGMENTS at the end of this paper.

[2] Presented at the IAEA Technical Meeting on Safety Analysis for Research Reactors, Vienna, Austria, June 3-7, 2002.

1.1 MNR Description

The McMaster Nuclear Reactor (MNR) is one of many MTR-type pool reactors. The design and flexible operating capability of this type of reactor has allowed them to be operated safely for over fifty years. Similar reactors are operated in North America, Europe and Asia.. First criticality was achieved on April 4, 1959. The reactor was operated at 1 MW (thermal) until November 1964 and 2 MW (thermal) thereafter. The reactor was operated at 5 MW (thermal) between 1971 and 1974. Operation since then has been at 2 MW.

The reactor has operated with high-enrichment uranium (HEU), eighteen-plate (sixteen fuelled, two dummy) fuel assemblies during its entire operation. At times, ten-plate (all fuelled) HEU fuel assemblies have been used; these will be unavailable after 2004. In 1989, two low enrichment uranium (LEU) assemblies were introduced for evaluation purposes. Conversion to a complete LEU core began in 1999, to be completed by 2006.

McMaster Nuclear Reactor (MNR) is a pool-type research reactor. Plate-type fuel assemblies are used in a grid supported by a bridge spanning a two-section pool. Demineralized light water is used to moderate, cool and shield the reactor. The reactor and the pool are shown in Figure 1. The pool is housed in a reinforced concrete building designed for low air leakage. The adjacent Nuclear Research Building (NRB) contains supporting facilities, including low, medium and high level laboratories, shops and offices. See Figure 2.

Various instrument systems are used to monitor reactor operation. Reactor control is by insertion or removal of neutron-absorbing control rods suspended from control drives mounted on the reactor core bridge. These rods also serve as the emergency shutdown system.

Heat created by the fission process is removed by the primary coolant system, using gravity flow through the core and out of the pool. A pump is used for the return flow to the pool. A hold-up tank, a storage tank, a tube/shell heat exchanger and associated piping make up the remainder of the primary cooling system. The secondary system removes heat from the primary system by conduction through the heat exchanger structure. The other major components of the secondary system are twin cooling towers, a pump and associated piping. The atmosphere is the ultimate destination for the core energy.

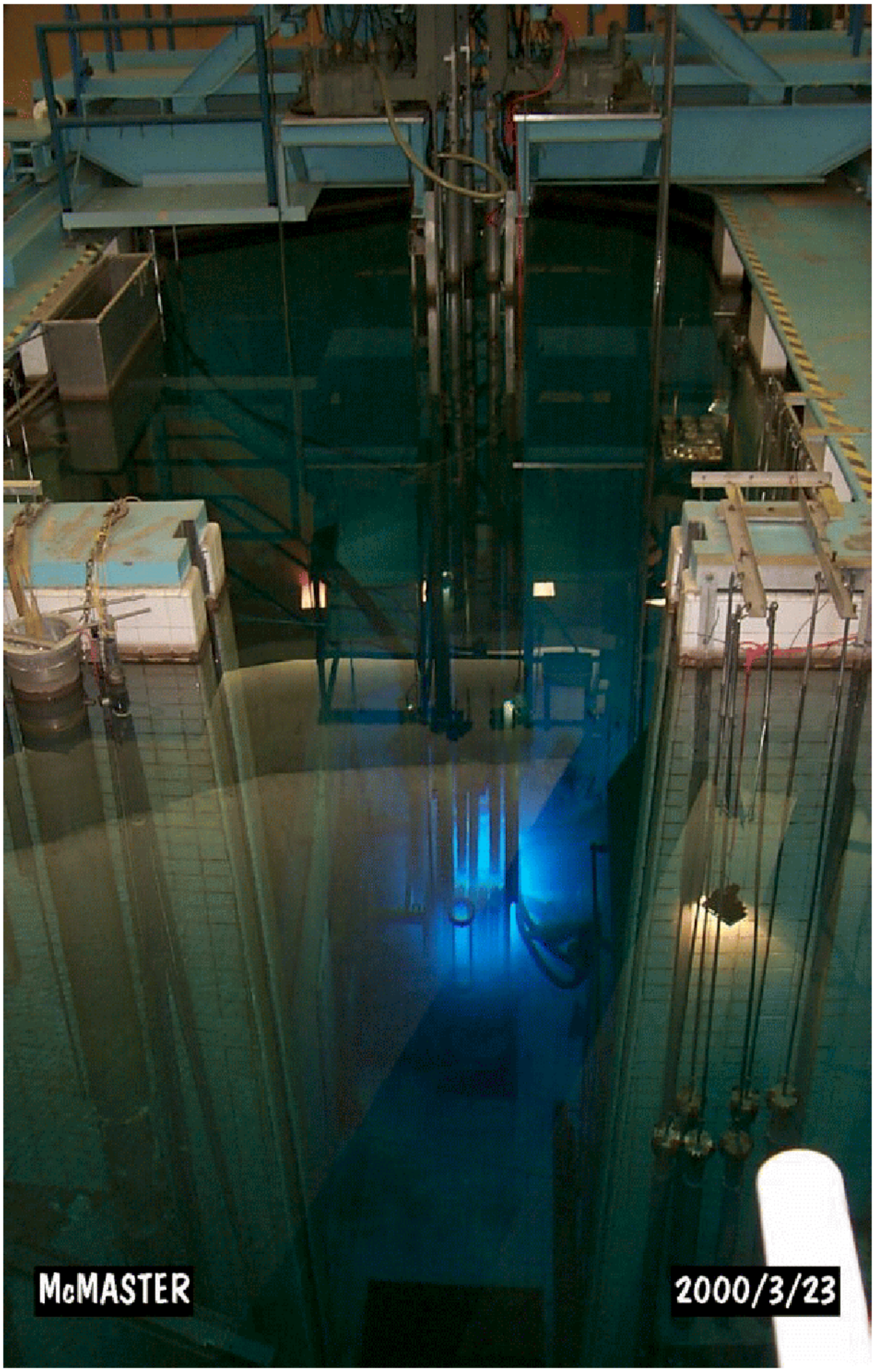


Figure 1: MNR pool and core photo

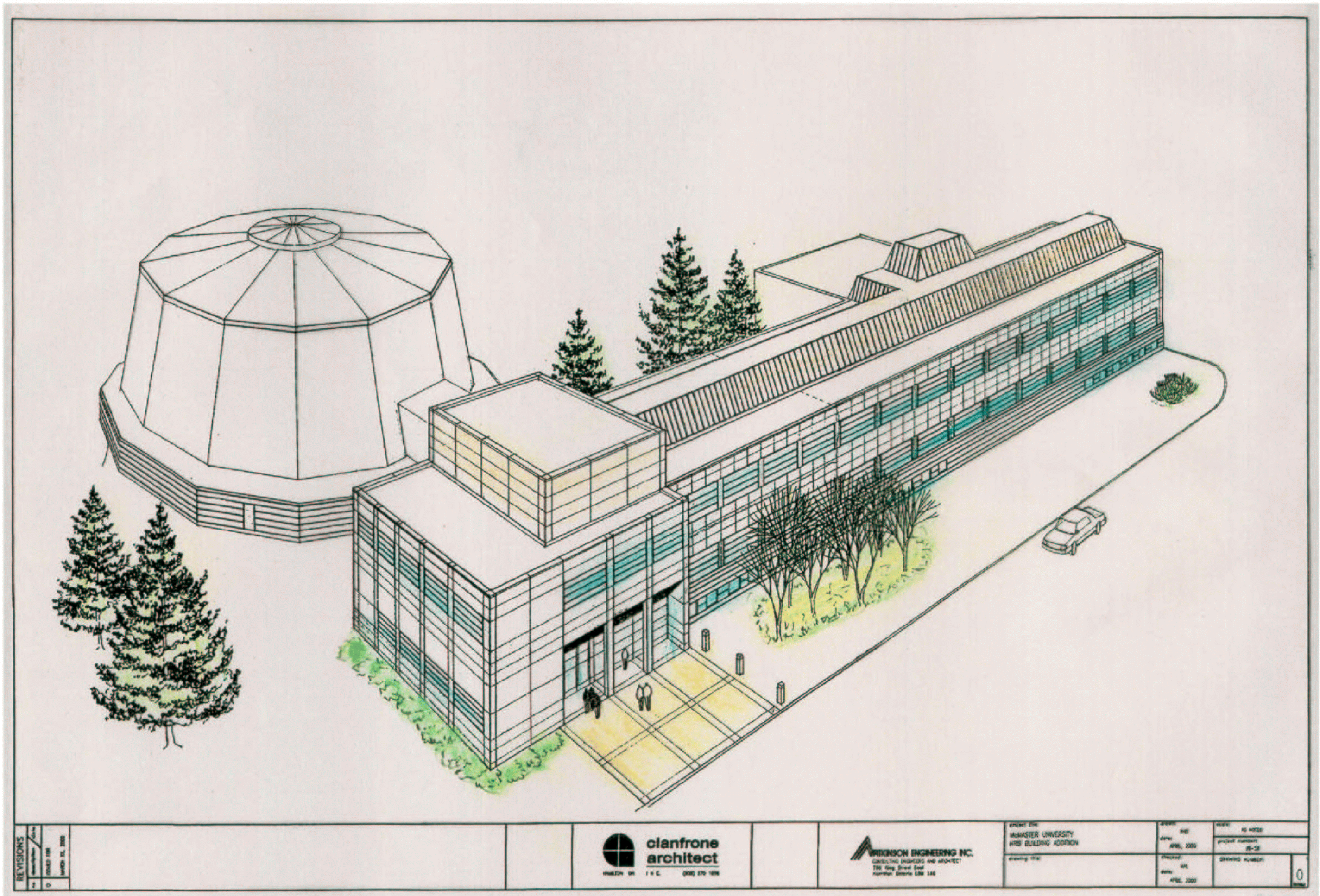


Figure 2: Reactor Building and Nuclear Research Building

2 SAFETY OBJECTIVES AND REQUIREMENTS

2.1 Risk Limitation

The McMaster Nuclear Reactor (MNR) safety philosophy must be based on the reactor being located and operated on the McMaster campus in a densely-populated residential area. The fundamental safety goals for MNR are public and occupational safety, prevention of equipment and facility damage, and safe, efficient operation. More specific goals are achieved by implementing the principles of recognized safety standards in the Canadian regulatory context.

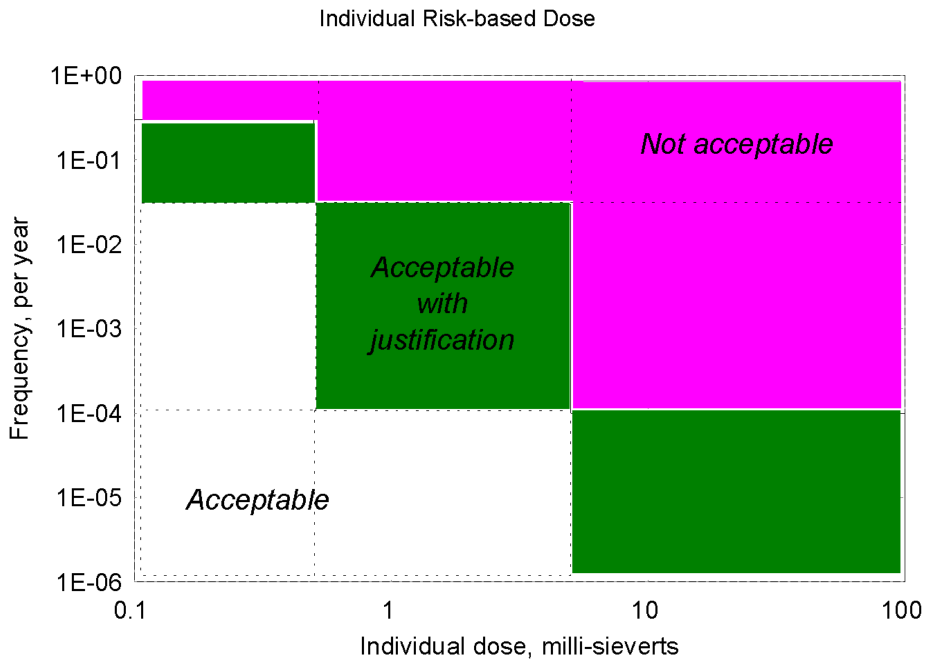
There are no well-defined or universally agreed-to safety rules for small non-power reactors. However, a set of criteria stating safety principles and design criteria for new small reactors was prepared for and presented at an IAEA conference by representatives of small Canadian reactors [4]. The reactors or facilities represented included MAPLE-X10, the proposed SES-10 heating reactor and MNR. These criteria have been used and accepted in Canada [5].

The safety objectives of [4] are applied to MNR using fundamental principles. Acceptance criteria used in this report follow the draft guidelines of the International Atomic Energy Agency. These are risk-based criteria, expressed in terms of individual and population doses as a function of accident frequency. The criteria are illustrated in Figure 3. As shown, the spectrum of accidents is broken into three frequency ranges; for each range there is a dose band spanning about one order of magnitude. Predicted doses below the band (the white regions) are deemed to be acceptable risks; those above the band (the purple / lightly shaded regions) are not. Numerical values of the frequencies and dose limits are given in Table 1.

The green / dark regions represent risks which must be justified for a new reactor, i.e. where there are additional uncertainties because of unproven design. This is hardly the case for MNR; as a mature facility, it is appropriate and necessary to take into account the lessons learned from over forty years of safe operation. Thus, the upper solid lines around the green regions are the appropriate criteria for MNR; these are known as the risk-based dose limits.

Table 1: Frequency-Dose Limits

Frequency, occurrences per reactor-year	Individual Limit, milli-sievert	Public Limit, person-sievert
$f \geq 3\times10^{-1}$	0	0
$3\times10^{-1} > f \geq 3\times10^{-2}$	0.5	1
$3\times10^{-2} > f \geq 1\times10^{-4}$	5	10
$1\times10^{-4} > f \geq 1\times10^{-6}$	100	100



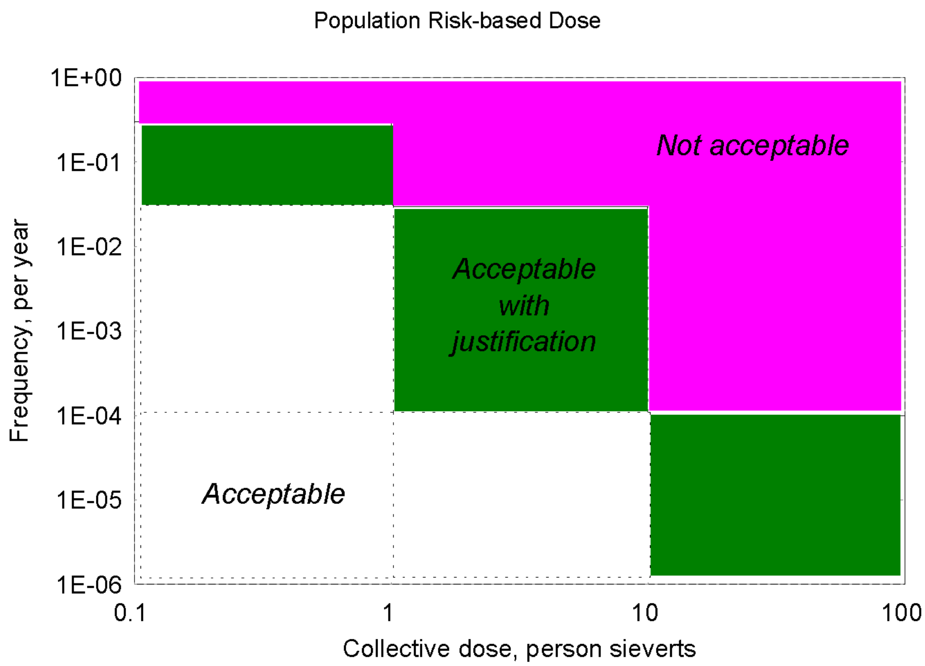


Figure 3: Risk-based dose limits

2.2 Radiological Protection

Risk-based limits do not provide sufficient guidance for controlling or limiting exposures from routine operations. Compliance with the Nuclear Safety and Control Act and Regulations [6] and International Commission on Radiological Protection (ICRP) recommendations [7] provides appropriate radiation protection through specified dose limitation. Dose limits for Nuclear Energy Workers and members of the public are shown in Table 3.

Table 3: Regulatory Dose Limits

	Annual Dose Limits, milli-sieverts	
Organ	Nuclear Energy Worker	Individual Member of the Public
Whole Body	20 mSv (averaged over 5 years); 50 mSv in any single year	1
Lens of the Eye	150	15
Skin	500	50
Extremities	500	50

2.3 Environmental Protection

The only radio-nuclides released from MNR as a result of normal operation or related activities are Ar-41 and I-125. Releases of these radio-nuclides have never been more than a few percent of the Administrative Control limits.

Reactor operation should be such that radio-nuclide releases during normal operation are less than the Administrative Control Emission Limits [8] listed in Table 4. This notwithstanding, reactor operation shall be such that radio-nuclide releases during normal operation are maintained below the Derived Emission Limits [9] listed in Table 4.

Table 4: Derived Emission Limits for MNR

Isotope	Release Limit, becquerels per year	Administrative Control Level, becquerels per year
Ar-41	2.0×10^{15}	1.6×10^{13}
I-125	1.0×10^{14}	1.0×10^{10}

3 SAFETY ANALYSIS APPROACH

3.1 Overview

In order to show that the consequences of a reactor accident would not exceed the dose limits, analysis must cover a variety of accident scenarios and quantify both frequency and consequence for each scenario. Practical constraints dictate that the accidents be treated broadly, by defining accident categories, with only a few specific accidents in each category examined in detail.

Each accident category covers a group of initiating events which follow similar accident progression paths. These paths are systematically identified by event trees [10]. Branches of the event trees have different probabilities of occurrence, depending mainly on the availability and reliability of mitigating features provided by the design and on the time available for manual actions. The branches also have different consequences, depending mainly on the state of barriers that prevent the release of radioactivity and, again, on the time available for manual actions.

The frequency of equipment failures can be estimated based on approximately 42 years of MNR operating experience. However, there is a problem with accounting for manual operator actions, which are important for many accidents in small reactors. There are no generally accepted criteria or models for operator intervention. Hence, the frequencies for accident sequences that involve the operator actions could be disputed. Similarly, certain non-engineered aspects of the accidents can be difficult to enumerate in terms of frequency (e.g., flow blockage geometry, feedback of void on core power, etc.). Hence, the frequencies of events involving non-engineered processes or phenomena could also be disputed.

A bounding approach could be used which neglects manual operator actions and any phenomena that are difficult to enumerate. However, this approach would significantly distort the risk profile of the facility. The alternative approach used in this report is to consider all relevant events and phenomena using “best-estimates” of frequencies and consequences whenever possible. Invariably, this involves sensitivity analyses and engineering judgements. For some accident phenomena or topics, only conservative parameters are available, or are required to be used by convention and/or precedence. Conservative values are then used for these phenomena and topics, in conjunction with best estimates for the remaining phenomena. This approach, which combines realistic and conservative components, is sometimes called the “best-effort” approach.

In order to provide confidence that important accident sequences were not omitted on the basis of perhaps tenuous frequency estimates, this report does not use the “cut-off” frequency of 10^{-6} events per year beyond which consequences need not be assessed according to the guidelines in [4] and [11]. Instead, the accident sequences are developed up to the first fuel failures, regardless of its frequency, when there is less than twelve hours from the unambiguous indication of the accident and the time of fuel failure. Accidents with frequency less than 10^{-6} per year are called Rare Events. The consequences of these rare events are evaluated.

The objective of safety analyses, in general, is to review design features and operating practices to confirm that pre-defined safety criteria are met. The safety analysis process shown in Figure 4

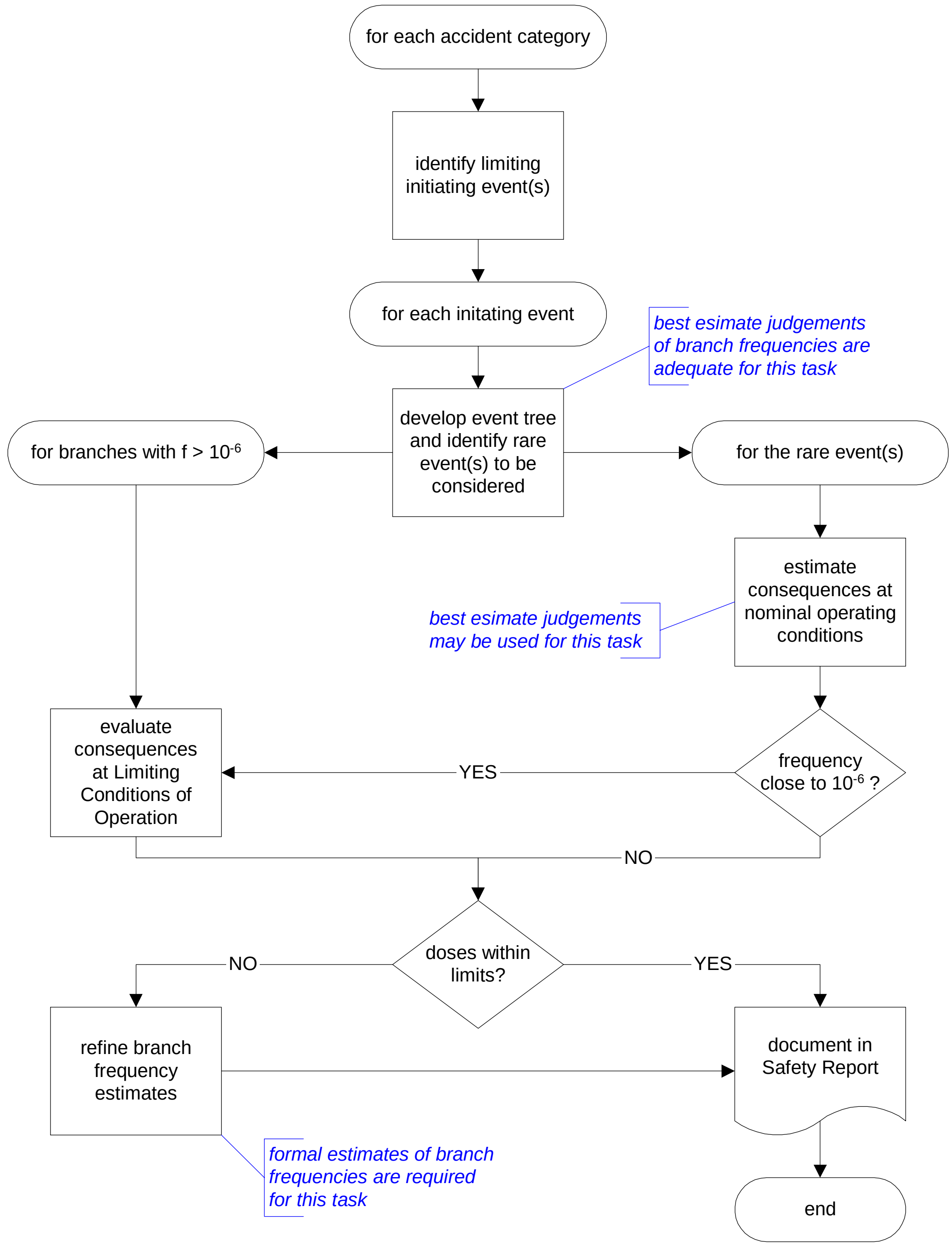


Figure 4: Safety Analysis Process

efficiently accomplishes this.

3.2 Accident Categories and Initiating Events

Accident categories are adopted from a recent safety analysis of another Canadian small reactor [5]. The barriers to release of radioactivity (i.e., the fuel and the pool water) are generally similar in all pool reactors, as are the basic reactor characteristics (i.e., a small, enriched core cooled by water at low pressure and temperature).

The accident categories of [5] identify the internal events which must be considered from a safety point of view. By adopting the same categories, this report benefits from the previous investigation and review of the initiators. Specific features of the reactor affect the course of accident progression and the probabilities of individual accident sequences (i.e., the event tree branches). Therefore, the accident sequences within each accident category and their consequences are unique to MNR.

3.2.1 External Events

External events are not evaluated because the low frequencies of these events, in conjunction with the strong containment barrier and the passive heat sink of the reactor pool, reduce the probability of any significant public dose caused by such events to below 10^{-6} per year. In other words, the seismically-qualified pool and containment provide a sufficient assurance of acceptable public consequence for external events.

3.2.2 Internal Events

As discussed, the 10^{-6} frequency criteria is not used as a cut-off for analysis. Instead, the event trees are examined to determine the most likely event which can lead to fuel failure and radiological consequences. Five broad accident categories were identified.

Category 1: Loss of regulation and loss of reactivity control events
- loss of regulation at high power;
- loss of regulation at low power;
- irradiation sample handling accidents on reactor core;
- void substitution in and around reactor core;
- coolant density change in reactor core;
- withdrawal of shim/safety rods;
- fuel handling accidents near reactor core.

Category 2: Fuel power/cooling mismatch events
- valves mis-operation in reactor coolant system;
- flow blockage at power;
- flow blockage in shutdown state.

Category 3: Loss of coolant accidents
- beam tube break caused by a falling load while the tube cover is removed for maintenance of instruments within the tube. This rare accident is closest to the plausible accident threshold of 10^{-6} per year and is evaluated using a best-effort approach in order to explore the depth of defence provided at MNR for this accident category;
- small and large pipe breaks.

Category 4: Failures of safety support systems
- loss of electrical power;
- loss of forced circulation;
- loss of heat sink.

Category 5: Radioactive materials handling events

- criticality safety assessments for fresh fuel storage, spent fuel storage, movement of a single fuel assembly in pool, and movement of the core to the open pool;
- mechanical damage to irradiated fuel.

3.3 Safety Analysis Rules Regarding Operators

With the exception of fast power and/or flow changes which require quick automatic actions, operator response is the ultimate means of accident mitigation. Unfortunately, there are no generally accepted criteria or models for operator error which apply to small, non-power reactors. The following analysis rules are adopted in this report to facilitate a reasonably realistic definition and assessment of accident scenarios which provide ample time for manual actions.

- Manual Shutdown: *The operator is credited with manually shutting down the reactor within five minutes of the first unambiguous annunciation of the accident.*

- Routine Mitigating Actions: *The operator is credited with executing a mitigating action when more than fifteen minutes is available from the first unambiguous annunciation of the accident **AND** the action is a routine one.*

- Complex Mitigating Actions: *The operator is credited with terminating the event when sufficient time is available from the first unambiguous annunciation of the accident to perform the actions **AND** there are no access constraints (e.g., high radiation fields) caused by the event.*

3.4 Supplementary Criteria

Supplementary criteria may be employed to ensure that the dose limits are met without performing an explicit dose calculation. Supplementary criteria are also employed to show that the risk profile trend does not change when low-frequency, rare events are evaluated.

3.4.1 Avoidance of Fuel Failure

Fuel failure is avoided when there are no geometry changes to alter its operating conditions and fission products remain contained within the fuel matrix. The onset of geometry change (i.e. fuel swelling) and the onset of fission product escape from the fuel matrix (i.e. fuel blistering) are both temperature-related. Swelling and blistering occur when the metal matrix “softens” sufficiently for it to creep. The extent of creep depends on the time at elevated temperature. For aluminum alloys and oxide or silicide dispersions in aluminum alloys the creep rates become noticeable at temperature range of 550 ± 100°C. The lower bound of this range (i.e., 450°C) is used as the generic criterion for avoidance of fuel failure.

Note that this conservative criterion is not the onset of fission product release in severe accidents, which use the middle of this range (i.e., 550°C) as the onset of fission product release.

3.4.2 Acceptable Risk Trend

This report examines rare events for which acceptance criteria are not formally defined. The consequences of such rare events are deemed acceptable when the calculated public doses are below the mean trend line in Figure 5. This line accommodates uncertainty in the frequency estimates, which are recognized as being tenuous, particularly for complex operator actions. A target is to show that public doses remain below the lower bound lines in Figure 5, at which condition it is not necessary to revisit the frequency estimates.

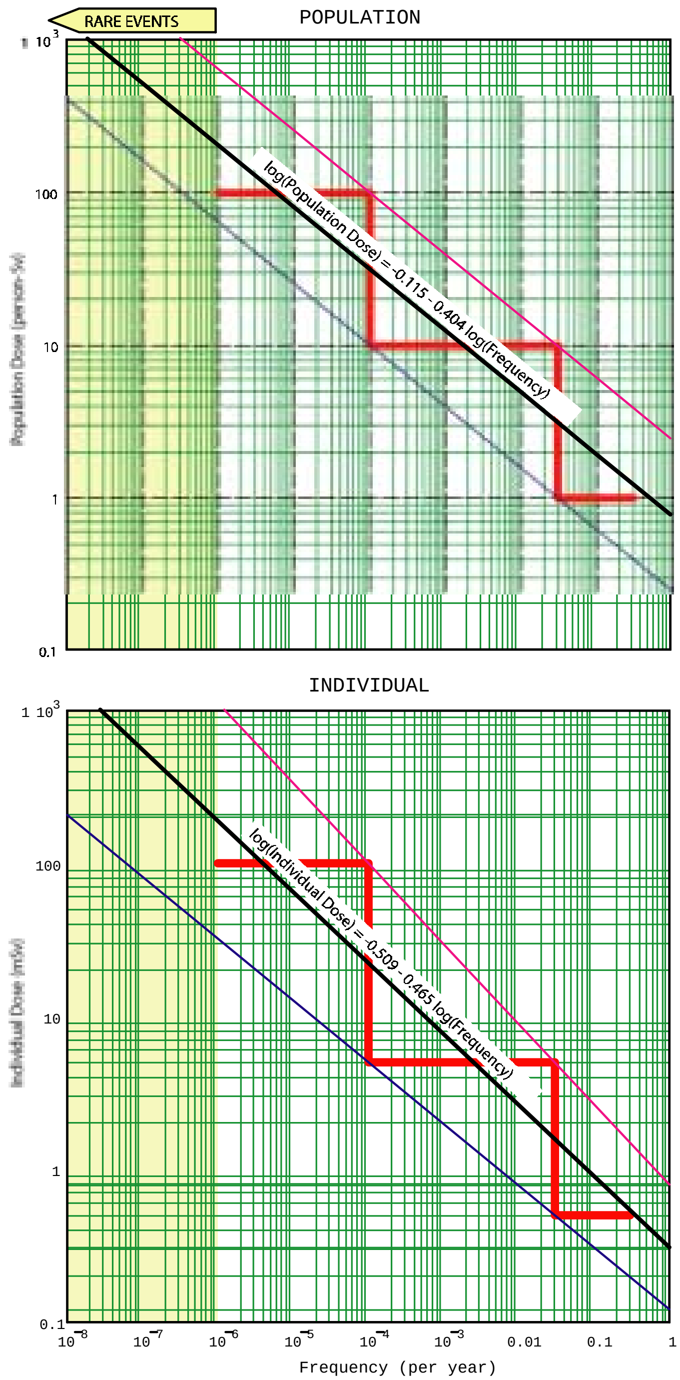


Figure 5: Projection of Dose Limits to Rare Events

4 ANALYSIS METHODOLOGY

4.1 Physics

Reactor physics analysis requires multiple analysis stages at different levels of physical complexity and with appropriate choice of methodology. A major complicating factor for McMaster Nuclear Reactor (MNR) analyses is a strong negative void reactivity feedback on reactor power combined with a buoyancy effect on the gravity-driven flow through the core.

As long as boiling is avoided, fuel and water/moderator properties can be accurately enumerated by mathematical simulations. If the coolant starts to boil in any parallel flow path, a complex interplay between static head, buoyancy and local pressure arises, which alters the coolant/moderator conditions in the affected fuel. At low pressures relevant to MNR, the specific volume of vapour is about 900 times larger than that of liquid. The large density difference between liquid and vapour significantly alters buoyancy forces and local pressure, reducing the gravity-driven flow and accelerating the voiding. A strong negative void reactivity feedback on power then reduces the heat load to coolant/moderator, inducing flow and power oscillations. There are no computer codes that reliably describe transient, oscillatory flow in multiple, parallel flow paths with simultaneous feedback on power. Hence, simulations are not feasible in a low pressure boiling system at power as is the case with MNR. For these latter conditions, the analyses rely on extensive experimental database available for Material Test Reactors (MTRs) similar to MNR [12].

4.1.1 Cell Modelling

Static calculations are performed using the standard approach of lattice cell calculations followed by whole-core analysis. Small sections of the system, *i.e.*, “cells”, are modelled in fine geometric and spectral detail [13] . The rest of the system is accounted for by the appropriate choice of boundary conditions. Once all of the relevant cells have been modelled, the characteristic parameters, *i.e.*, average cross-sections, of each cell are incorporated in a broader model of the system.

The transport theory code WIMS-AECL [14][15][16] was used. The version of WIMS-AECL and the cross-section library used are:

- WIMS-AECL Release WL2-4Z 1996-01-03
- WIMS ENDF/B-V LIBRARY VERSION 1.4W

The library is based on the ENDF/B-V data file, and contains cross-sections for 168 nuclides, including 30 fissile isotopes and 45 fission products, in 89 energy groups. The energy group structure is divided into 42 thermal groups, 23 resonance groups and 24 fast groups.

WIMS-AECL generates burnup-dependent cross-sections for the different materials in the model as well as cell-averaged values. The geometry for plate fuel is limited to 1D-infinite slab. For non-fuel regions, a 2-D annular super-cell geometry was used.

4.1.2 Static Core Modelling

The second stage, full core modelling, uses a diffusion theory code. The geometry and spectral details of the core model are more coarse than used in the prior transport theory based cell models.

For MNR core calculations, the code 3DDT [17] was used. 3DDT is a finite-difference multi-group diffusion theory code. It has 1D, 2D and 3D Cartesian geometry capabilities. The code requires multigroup macroscopic cross-section data such as that generated by WIMS-AECL as input, together with geometry specifications.

The front-end code MAPDDT [18] was used to prepare input for 3DDT. The versions of MAPDDT and 3DDT used in this analysis are:

- MAPDDT Release V1-1D_Beta LD= 1996-03
- 3DDT Release V1-1C LD= 1995-08-16

For the MNR core model, each active fuel region (roughly 8 cm by 8cm by 60 cm) was sub-divided into a 6 x 6 x 30 mesh. The complete 3D core model of MNR used 82 x 60 x 56 mesh zones and 8-energy groups. The WIMS-AECL/3DDT models were validated against the IAEA calculational benchmark for MTR reactors [19][20] and flux wire measurements [21].

4.1.3 Transient Modelling

For transient calculations, point kinetics was used for MNR. This is a widely accepted analysis method for MTR type reactors such as MNR. For a tightly coupled system such as MNR spatial characteristics may be determined via static calculations, *e.g.*, power-peaking factors. The temporal behaviour of the system is then investigated with point kinetics.

Transient analysis of MNR was conducted using in-house codes with kinetics input data from literature [22]. These codes were used to determine general trends in dynamic behaviour of the MNR core as well as for parametric analysis with respect to the MNR Startup Transient [23]. The codes were validated against data [24] and are consistent with results in the literature for a Startup Transient.

The codes use point kinetics with explicit spatial characteristics such as axial importance weighting; reactivity feedback is modelled with simple heat transfer models for fuel and coolant geometry.

4.1.4 Fission Product Modelling

Fission product inventories were calculated using the SCALE suite of codes [25]. This primarily means the driver module SAS2H. The modules BONAMI and NITAWL perform resonance self-shielding calculations; their output and user-specified geometry is input to the one-dimensional transport code XSEDRN, which produces homogenized cross-sections. These are in turn input to ORIGEN with power history data for depletion calculations. Cross-sections are updated during the calculation cycles to reflect changes in nuclide densities and neutron spectrum.

The basic data library for the calculations is called “27BURNLIB”. It contains 27-energy group neutron and gamma data for common materials as well as a large amount of information on fission products. It is reported to be the most suitable database for general depletion problems [26].

Fuel plate models for XSEDRN were based on 1-D infinite slab approximations, much like those used in the WIMS-AECL cell models. Version 4.3 of the code package and cross-section library was used for this work.

4.2 Thermal Hydraulics

4.2.1 Reactor Coolant System

Thermal-hydraulics play an important role in most reactor accidents. A major complicating factor for MNR analyses is a strong void feedback on the gravity-driven flow through the core combined with an strong negative void reactivity feedback on the reactor power.

As long as boiling is avoided, the flow is governed mainly by hydraulic characteristics of flow paths between the reactor pool and the hold-up tank. In the relevant range of temperatures (i.e., between about 30°C and saturation temperature), properties of liquid water do not change significantly. Hence, the flow is governed by hydraulic characteristics of parallel flow paths, which are invariably input parameters in any model. Thermal-hydraulic system codes (such as CATHENA, [27]) as well as hand calculations [28] can all predict the single-phase flow and the resulting heat transfer, subject to correct inputs of hydraulic resistance.

If the coolant starts to boil in any parallel flow path, a complex interplay of static head, buoyancy and local pressure affects the flow, while a strong negative void reactivity feedback affects the power. The thermal-hydraulic system codes [27] capture experimentally observed flow patterns (i.e., flow oscillations, [12]), but details are sensitive to nodalization and other modelling factors. Hence, similar to the reactor physic analysis, simulations are not feasible in a boiling system at power for MNR. For these latter conditions, the thermal-hydraulic analyses also rely on the extensive experimental database available for reactors closely similar to MNR [12].

4.2.2 Hand Calculations

Documented hand calculations are used to delineate the combinations of measurable process parameters (i.e., total core flow, core power and core configuration) up to the onset of boiling in any flow path. This onset of boiling represents the boundary of the Safe Operating Envelope. The Safe Operating Envelope (SOE)[28] makes allowances for process parameter uncertainties and is used to ascertain that the fuel does not fail in plausible accidents (i.e., accidents with frequency $\geq 10^{-6}$ per year). The actual operating limits were set at 2/3 of the SOE.

The calculations are documented in MathCad 2000 or 2001 worksheets and are printed in Portable Document Format (PDF) to facilitate revision control and review. Water and steam properties used in these calculations are generated by WaterSteamPro functions for MathCad, which are certified to reproduce the values from the standard reference data (GSSSD R-776-98) and the IAPWS Industrial Formulation 1997 for the thermodynamic properties of water and steam.

4.2.3 CATHENA Simulations

The original intent was to refine the existing MNR model [29], qualify it against MNR test data [30][31] and use it to define the SOE boundaries. It has turned out that these tasks were too time-consuming for the qualified model to be available for the SAR [32]; hence, hand calculations are employed for the SOE as described above. CATHENA simulations are used for sensitivity analyses of conditions when the coolant flow through the assembly or flow channel is predefined.

4.2.4 Containment

The containment thermal-hydraulics governs the pressure in an isolated containment envelope, which is essentially a closed volume with a fixed amount of air. If the temperature in this closed volume increases, the specific volume of air increases and the containment pressure rises. If the temperature decreases, the containment pressure decreases.

The temperature in the sealed containment envelope is governed by the Air Cooling Units (ACU). There are several such units, but the ACU of the reactor hall (i.e., Reactor Hall Unit) dominates the average containment temperature. The air stream of this ACU is directed across the reactor pool and collects a heat load from the reactor pool. The analysis thus cannot treat the containment as a uniformly heated volume when the ACU is in service.

No transient calculations of isolated containment temperature and pressure are performed for accidents during which the fuel remains submerged in water (i.e., for most accidents). This state is within the realm of operating experience. With the core submerged, the containment heat load amounts to a few kilowatts. The ACU maintains an approximately constant temperature, or the temperature can only decrease.

Hand calculations of temperature and pressure transients are performed for a rare loss of coolant accident that uncovers the reactor core. These calculations are documented in MathCad 2000 or 2001 worksheets and printed in Portable Document Format (PDF) to facilitate revision control and review. Results show that maintaining the constant temperature in the containment is the limiting post-accident state in terms of radioactivity leakage into the environment. This means that the simple heat balance, which ignores the heat transfer between the containment atmosphere and the structures, is sufficient to illustrate the trends. In other words, transient containment temperature and pressure calculations are only used to confirm behaviour trends; the limiting condition (i.e., the automatic control of temperature) does not require a calculation.

Not having to evaluate a heat transfer between the containment atmosphere and the structures considerably simplifies the analysis. This heat transfer is very complex for the modest steaming rates relevant to severe loss of coolant accidents and it cannot be reliably evaluated without a major effort to set up and qualify a three dimensional model. ACU failures can be postulated, but these failures combined with a rare uncovering of the core have too low a frequency to warrant analyses.

4.3 Fission Products Release and Transport

Releases of fission products from the fuel and their subsequent transport to the containment atmosphere and, ultimately, to the outside environment are evaluated for rare events that do not involve extreme, prompt-critical power excursions. All plausible events (i.e., those with accident frequencies $\geq 10^{-6}$ per year) avoid fuel failure, so the topics of release and transport do not come into play. Destructive criticality accidents have always been recognized to be dangerous, so the reactor design and the operating practices provide multiple features to prevent these accidents from occurring. Consequently, a combination of events that would lead to an extreme power excursion is of an extremely low probability.

4.3.1 Fission Product Release under Gradual Heat-up

During normal operation, fission products are retained within the matrix of the metal-alloy fuel meat. This is different from the ceramic UO_2 fuel used in the power reactors, where considerable fractions of the more volatile fission products migrate to the space between UO_2 grains and to the voids between the fuel pellets and their metal cladding. Because fission products within the fuel meat are immobile, they do not interact with each other until an accident alters the conditions of the fuel matrix to permit their motion. Therefore, metal plate fuel contains the fission products in their elemental form, while the ceramic pellet fuel contains both elements and chemical compounds. The chemical form of the fission products defines their volatility.

4.3.2 Onset of Release

Aluminum alloys become plastic (i.e., they creep at some noticeable rate) somewhere between 450° and 600°C. The Al-3%Mg alloy used in the reference fuel is at the upper end of this range. As the cladding and the fuel matrix deform by creep, trapped fission product gases and vapours can expand, causing the fuel to swell. Gas bubbles expanding near the fuel surface form "blisters" which can burst and release their contents into the surrounding environment.

The reference fuel is made of uranium silicide (U_3Si_2) grains dispersed in Al-3%Mg alloy. This fuel can blister when held for some time at a temperature just below the melting point of the alloy [33][34]. Although U_3Si_2 has melting point of 1665°C [35], it slowly interacts with Al at temperatures above about 620°C [33]. Presumably, this interaction disrupts the U_3Si_2 grain to free some of the trapped fission product gases. The gas then expands within the plastic aluminum metal surrounding the U_3Si_2 grains. This process is similar to bubbles trapped directly in the metal (see the preceding paragraph). However, fuel swelling and blistering can only occur slowly, in conjunction with the interaction process. Hence, for the reference fuel, blistering may only be relevant to "ramp and hold" temperature conditions below the melting point, which are often used in experiments, but which are unlikely to be achieved in an accident. In a temperature excursion, the aluminum would liquefy before there could be any swelling and blistering.

The onset of fission product release is taken to be the middle of the temperature range which covers all the fuel types (i.e., 550° ± 100°C). At 550°C, aluminum and its alloys are sufficiently plastic to permit the expansion of trapped gases and vapours, although the reference fuel would need a somewhat higher temperature for the interaction. The low end of this temperature range (i.e., 450°C) is used as a conservative criterion for avoidance of fuel failure.

Based on the boiling points of fission product elements, “blister” release can only involve noble gases and possibly halogens. Radioactivity release below the fuel melting temperature is invariably small [36], because only the fission products in the immediate proximity of the fuel surface can form blisters when the gas bubbles expand. The amount released by blisters bursting is burnup-dependent (i.e., high burnup fuel will have more gas bubbles near the surface than low burnup fuel).

When the blistering temperature of 550°C is reached, it is assumed that 1 percent of the noble gas inventory and 0.1 percent of the iodine inventory in the affected fuel are instantaneously released. The release fraction of noble gases can vary with burnup. The assumed value is a judgement based on release rates reported below the melting point of aluminum for a variety of plate fuels [36]. Halogens are typically not released by the blistering process (i.e., the bubbles are too small to perforate the softened cladding), but a conservative approach accounts for the possibility of some iodine release. Less than 0.1 percent iodine release during blistering is reported in the literature [37]. The boiling point of caesium is sufficiently high for it not to be volatile when blistering occurs.

4.4 Radiological Consequences

Limiting doses are quantified only for members of the public off-campus. Doses to the ”on-campus public” and operators are not calculated for the following reasons:

- Analyses show that there are no credible accidents[3] that would release any significant radioactivity from the MNR before emergency measures could be implemented. These emergency measures ensure that all members of the public within the campus boundary are either evacuated or sheltered in isolated buildings[4] until an orderly and controlled evacuation is executed. Sheltered people avoid doses due to inhalation of radioactive substances as well as ground-shine doses resulting from a prolonged stay in a contaminated environment[5]. These people would receive only a fraction of the dose taken in by an exposed individual on the boundary Hence, no explicit analyses of doses to sheltered people are performed here.

- Doses to the operating staff cannot be evaluated deterministically. The staff is trained to evacuate the facility after putting it in a safe state, should an internal radiation hazard arise. However, the only conservative assumption that can be made is that the operator does not evacuate. This assumption invariably leads to large operator doses for rare events with extensive fuel failures.

Individual doses are evaluated at a public roadway immediately adjacent to the campus boundary

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3 Including rare accidents with frequencies well below 10^{-6} per annum.

4 An isolated building has ventilation air turned off and all penetrations closed.

5 It is presumed that measures will be taken to prevent any significant ground-shine exposure (e.g., evacuation by an uncontaminated route) according to actual post-accident conditions.

and at the edge of private property on the other side of the public roadway. These two separate locations are examined because the complex topology around the MNR (i.e., the surrounding buildings) could locally affect radioactivity dispersion. A person is assumed to remain on the roadway for 24 hours, starting from the onset of radioactive release into the environment. Another person is assumed to stay outside at the closest edge of private property for 1 month.

A stable wind towards the critical individual is assumed to prevail for the first day after the onset of release into the environment. For the first hour (the short-term release period), wind direction and speed are constant. For the next 23 hours (the prolonged release period), wind speed remains constant but a meander occurs. Average weather with changing wind direction is assumed to prevail from 1 day to 1 month (the long term release period). The same weather assumptions are used for the population dose estimates. The wind is assumed to blow only into the affected sector for 1 day and is variable thereafter. These weather assumptions follow the guidelines of Canadian standard [38].

The atmospheric dispersion of radioactivity is evaluated using the PEAR computer code [39] with a plume model option per Canadian standard guidelines [38]. The code estimates public doses using built-in dose conversion factors.

5 COMMENTS AND CONCLUSIONS

Standard deterministic analysis codes, simple hand calculations and decades of experimental data were used in the framework of probabilistic based event trees. This proved to be a practical and effective methodology for the safety analysis of the McMaster Nuclear Reactor. The extensive manual operations that are typical of the research environment precludes an accurate estimate of failure rates but the overall framework of the probabilistic approach proved very useful in enabling a meaningful piece-wise decomposition of the safety analysis task. But, as important and the results of the analysis are, perhaps the biggest benefit to the overall safety of the reactor was the <u>engagement in the process</u> of formulating the methodology and in doing the analysis. This freshens the appreciation of the importance of empowerment of the individual and of the importance of a good safety culture.

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