ELEMENTS OF A RADIATION SAFETY PROGRAM FOR ^{125}I PRODUCTION

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ABSTRACT

The development of a radiation safety program for ^{125}I production has been described. The generic elements of a radiation safety program prescribed by the International Atomic Energy Agency have been examined and the special considerations relevant to radiation safety at an ^{125}I production facility have been identified.

Development of four components of a radiation safety program has been described in detail. The four components include: development of action levels for surface contamination, development of air effluent release limits, evaluation of the impact of production activities on the emergency plan and the conduct of a dose optimization review of production activities.

The importance of careful planning and dose budgeting when introducing production activities has been demonstrated. The methodologies for a dose optimization process that resulted in a ninety percent reduction in collective doses are presented.

While the results of the analyses that are presented are specific to radiation safety for ^{125}I production, the general approach to assessment of radiation safety program needs can be applied broadly to designing and implementing radiation safety programs.

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CHAPTER 1: Radiation Safety Program Considerations for ^{125}I Production

Introduction

In this chapter, the critical components of a radiation safety program are identified. The special considerations related to the radiological hazards encountered in production of ^{125}I are discussed. These special considerations are discussed from the perspective of adding an ^{125}I production operation to a facility, such as a research reactor, which is already operated safely with a well established radiation safety program.

Summary of the Radiological Hazards Associated with Production of I-125

Overview of the Production Process

Production of ^{125}I (60.1 d) is achieved through the activation of ^{124}Xe to ^{125}Xe (17.1 h) by neutron capture with subsequent decay of the ^{125}Xe to ^{125}I by electron capture. A brief description of the steps in the patented process employed at the McMaster Nuclear Reactor (MNR) and the radiological hazards associated with each step follows.

1. **Irradiation:** Xenon gas isotopically enriched in ^{124}Xe is placed in the irradiation chamber of a specially designed apparatus known as a "production rig". The production rig is inserted into an irradiation position in the MNR core. During the irradiation phase, ^{124}Xe is activated to ^{125}Xe. Other Xe activation products are also produced, and gas impurities are activated. In addition, some ^{125}I is formed by decay of ^{125}Xe, and some of the ^{125}I is activated to ^{126}I.

   External radiation hazards are well managed through this phase of the operation by utilizing the shielding properties of the pool water. The production rig design incorporates dual containment to minimize the likelihood of releasing the activated gas to the pool water. There is no significant internal hazard presented by this stage of the production.

2. **Cryopumping:** After an optimal irradiation period, the production rig is removed from the core and the gas from the irradiation chamber is cryopumped through a closed system in the rig to a decay chamber. A period of time is allowed for decay of the ^{125}Xe to ^{125}I. The ^{125}I atoms produced plate out on the walls of the decay chamber. By design, the ^{125}I and ^{126}I produced in the irradiation chamber remain plated out there and never enter the decay chamber.

However, it has been found that addition of trace quantities of water to the system will allow the ^{125}I and activated ^{126}I to move from the irradiation chamber to the decay chamber. This highly undesirable occurrence is discussed further in Chapter 3. After the prescribed decay period, the gas is returned to the irradiation chamber by Cryopumping and the production rig is returned to the core for further activation of the ^{124}Xe. The production rig design is such that the decay chamber (now holding the accumulated production of ^{125}I) is above the neutron fluence from the reactor core, so that ^{126}I is not produced in the decay chamber.

There are minimal radiological hazards associated with the Cryopumping phase. It is performed with the decay chamber shielded by over 1m of water and by the concrete of the pool apron. As all of the radioactive material remains inside the containment system of the rig, there is no significant potential for intakes.

3. **Interchange:** After an optimal number of production cycles, the rig is docked at a specially designed station at the pool side. The docking station holds the production rig such that the decay chamber is accessible above the pool while the irradiation chamber and the activated end of the production rig remains shielded below the water. It also allows for a partial, locally ventilated enclosure made from Plexiglas (the "Interchange Box") to be placed around the top of the rig during work. Once inside the Interchange Box, the rig cowling (outer containment) is disassembled to expose the decay chamber and connections. This is shown in Figure 1-1.

   The rig's internal vacuum/piping system is connected to a pool-side pumping apparatus known as the Gas Handling Station (GHS). The connection is made via a long bellows tube protected by a braided hose. This hose and other bellows tubes, necessary due to the difficulty in achieving exact positioning of components that must be connected in the vacuum system, have proven to be important shielding weaknesses. The GHS is used to draw off any non-condensable gasses remaining in the decay chamber after the last cryopump.

   The decay chamber, containing the inventory of ^{125}I produced (typically greater than 1 TBq), can then be isolated and removed from the production rig. It is replaced with another decay chamber and the production rig is reassembled and returned to service.

   There are several radiation hazards associated with Interchange. There is a potential for high radiation fields from the decay chamber if radioactive impurities are still present. This is controlled through limits on the allowable radiation field associated with the production rig cowling before interchange can begin. Multi-TBq quantities of ^{125}I may be present inside the decay chamber. The radiation

fields from this activity are adequately shielded by walls of the decay chamber. However, if a small fraction of the activity becomes unexpectedly mobile within the system and moves to a thin-walled bellows tube, there is a potential for high

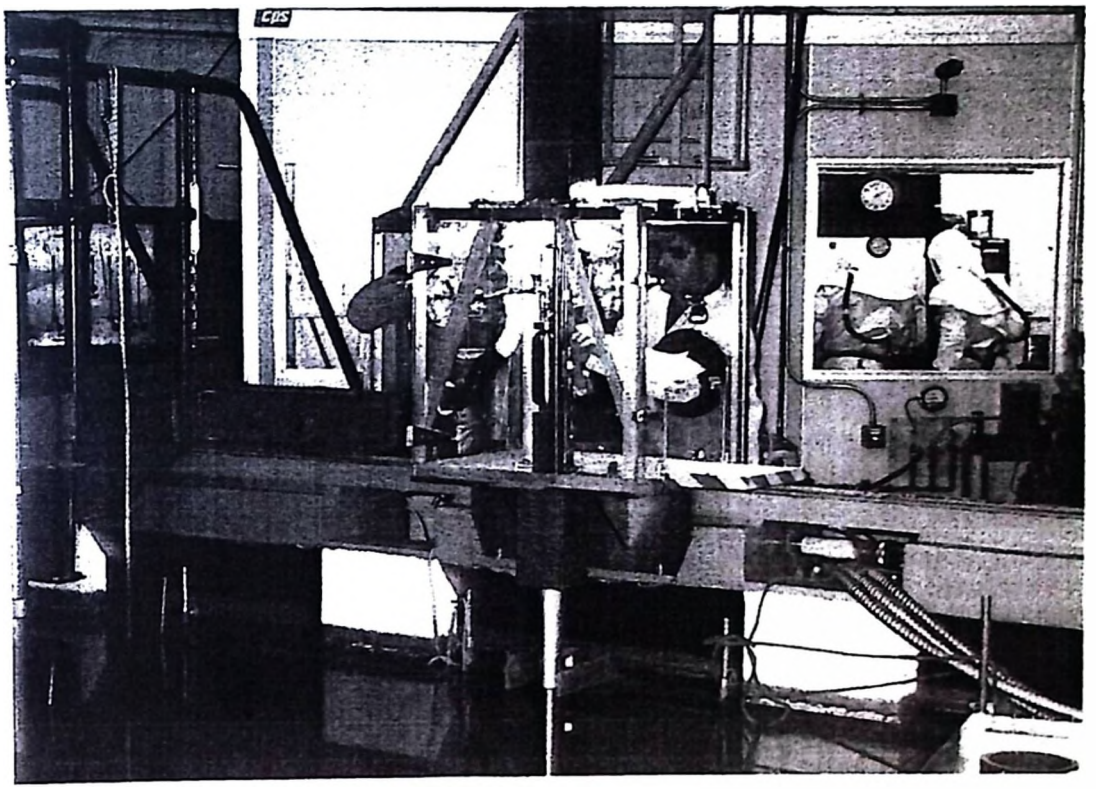


Figure 1-1: Iodine Production Rig in Interchange Position

This figure shows an iodine production rig at the docking station with the interchange box in place and the decay chamber exposed. Note that the irradiation chamber is located at the bottom of the rig which remains underwater. The worker is shielded from the high radiation fields associated with the irradiation chamber and the lower end of the rig by the water and by the pool wall. In the background, two workers wearing air supplied respirators can be seen working in the recovery glove box inside the separately ventilated enclosure.

radiation fields. Personnel are required to wear alarming electronic personal dosimeters during all production activities as a general precaution against unexpectedly high and/or transient radiation fields. During interchange, the piping system of the production rig is opened. This introduces a high potential for contamination spread and intakes. The operation is conducted inside an enclosure that provides local ventilation and some containment for contamination spread. Air supplied respirators, gloves and anti-contamination clothing are worn to protect against inhalation intakes, skin contamination and contamination spread. Surfaces are repeatedly decontaminated with sodium thiosulphate or bisulphite solutions to minimize the airborne release of volatile iodide. Opening the system exposes o-rings which have been shown to accumulate high activities of ^{125}I and as such present a significant hazard for high radiation fields to the extremities.

Significant contamination of pump oil in system vacuum pumps has occurred at McMaster. Pumps must be vented to filtered exhaust to avoid creating a chronic source of airborne contamination.

4. **Recovery:** During the recovery stage, the decay chamber is washed out with a small volume of heated distilled water. This takes the ^{125}I into solution. The high specific activity solution is extracted from the decay chamber into an evacuated vial. The operation is conducted inside of a glovebox, located inside a separately ventilated enclosure (the "Enclosure"). The glovebox is equipped with leaded neoprene gloves to provide shielding from ^{125}I. Quality assurance tests are conducted on the recovered product and it is transferred to inventory. As part of the recovery operation, the entire recovered stock is removed briefly from the glovebox and it is assayed in an ion chamber inside the Enclosure.

   The recovery stage has significant internal and external hazards associated with it. Once removed from the decay chamber, multi-TBq quantities of ^{125}I solution are transferred inside the glovebox. Local shielding of vessels is employed to mitigate the external radiation hazard associated with the stock solutions, and the glovebox gloves provide some protection to the extremities.

   The iodine solutions are highly volatile. Solutions are extracted and sampled from open containers by pippetting, so extensive internal contamination of the glovebox is inevitable. Drop cloths and decontamination of surfaces with reducing solutions is used to maintain the contamination levels as low as they reasonably can be. The glove box gloves are permeable to iodine and must be replaced from time to time. Elevated airborne ^{125}I contamination in the vicinity of the glovebox results from permeation through the gloves and from releases when materials are brought out from the glovebox. For this reason, the glovebox is located in a

separately ventilated enclosure with charcoal filtration on the exhaust. This arrangement also provides for containment in the event of an error or incident during glovebox operations that could otherwise lead to widespread contamination release. Personnel performing recovery operations routinely wear air supplied respirators, multiple layers of gloves and anti-contamination clothing. The normal airborne concentration in the Enclosure is well under 100 Bq/m^3; the requirement for respirators derives more from preparedness for potential accidents than from a need to protect from chronic exposures.

5. **Dispensing:** During this phase of the operations, stock solutions are accessed inside the glovebox. Customer orders are filled by pippetting the required volume of solution into vials for shipment. The vials are brought out from the glovebox, decontaminated and packaged for shipping.

   The radiological hazards associated with this stage are essentially identical to those encountered during Recovery, and the same precautions are used to manage the hazards.

6. **Decay Chamber Drying:** Following recovery, decay chambers must have residual moisture removed from them prior to being returned to service in a production rig. This is achieved in a specially designed pumping apparatus known as the "Fish Tank". The decay chamber must be removed from the glovebox, decontaminated, and transferred to the Fish Tank. In the Fish Tank, it is connected to a pumping system which draws the residual moisture and ^{125}I activity onto a trap.

   This phase entails contamination spread potential similar to that encountered in the previous steps. The Fish Tank system is highly contaminated. The system must be opened to connect each decay chamber. In addition, the system is designed to move the residual moisture and activity from the decay chamber – high radiation field transients have occurred associated with thin-walled components of the system (thermocouples and bellows tubes). Local shielding and radiation monitoring is employed to control and monitor this hazard.

The contribution of each of the production steps to personnel exposures is discussed further in Chapter 3.

Properties of the Main Radionuclides Contributing to Radiological Hazards.

Three principal radionuclides have been found to contribute to the radiological hazards of the ^{125}I Production Process at McMaster University: ^{125}I, ^{126}I and ^{125}Xe. The radiological properties of each are summarized in Table 1.

Radiological Hazards associated with ^{125}I

As the desired product of the production process, ^{125}I is present in very high activities at the facility. Typically, inventories handled in operations are in the TBq range.

Radiation fields arising from ^{125}I are easily shielded. The steel walls of the vacuum system and chambers containing the inventory are generally sufficient to reduce such fields to easily managed levels. However, care must be taken when thin walled components such as bellows tubes and thermocouple gauges are incorporated into the system. If significant portions of the ^{125}I become unexpectedly mobile, as has happened at McMaster due to introduction of trace quantities of moisture, there is a potential for high radiation fields to be encountered. In addition, in operations where the stock containers are opened, high radiation fields are present above the contain shields and may lead to high extremity exposures.

Radioiodines are readily taken into the body by inhalation, ingestion and adsorption through the skin. For ^{125}I, dose to the skin from iodine contamination can be an important exposure pathway. Exposures arising from ^{125}I contamination are examined in depth in Chapter 2. Chronic low level contamination of working surfaces associated with production has been encountered at McMaster, particularly associated with the glove box gloves due to permeation of the iodine through the gloves. Rigorous decontamination and contamination monitoring practices have been required to control this hazard. Because of the volatility of the iodine combined with the high activities present, chronic airborne contamination can be expected in the workplace. Engineered controls (ventilation and containment) along with administrative practices (frequent decontamination and handling in dual containment) maintain the levels comfortably under the Derived Air Concentration[1] in most instances. Small traces of surface contamination outside the Enclosure have been found to result in rapid increases in the airborne contamination level of the reactor hall.

[1] The Derived Air Concentration (DAC) is that concentration of a radionuclide which, if breathed by Reference Man for 2000 hours, would result in a dose of 20 mSv.

Radiological Hazards associated with ^{126}I

This radionuclide is produced as a byproduct when ^{125}I formed in the irradiation chamber is activated by neutron capture. The activity typically present in a batch of recovered product is in the MBq range, although this is highly variable depending on the condition and moisture loading of the system.

Even in trace quantities, ^{126}I will quickly dominate the external hazard present in a system where the shielding is designed for ^{125}I. The effects of higher than expected recovery of byproduct ^{126}I are reviewed in Chapter 3.

As a surface and air contaminant, ^{126}I will behave identically to ^{125}I. However, because it is typically only present in the part-per-million of ^{125}I range, its contribution to the internal hazard can be disregarded.

Radiological Hazards associated with ^{125}Xe

^{125}Xe is produced as an intermediate step in the production of ^{125}I and is present in the system in TBq quantities. As it is a radioactive noble gas, ^{125}Xe does not pose an intake hazard. However, several energetic photons are emitted in the decay of this radionuclide and this, combined with the large activities present, leads to a significant contribution to the external radiation hazards.

Generally, the process can be designed such that workers are not exposed to significant radiation fields arising from ^{125}Xe. This is done by keeping the decay chamber of the production rig underwater and behind the pool skirt during Cryopumping, for example. However, high radiation fields can be encountered at the gas handling station when drawing off residual ^{125}Xe that has not been removed from the decay chamber by cryompumping. Any accidental movement of the ^{125}Xe through, for example, valving errors or loss of containment, can lead to rapid increases in workplace radiation fields.

Table 1-0-1: Radiological Properties of Principal Radionuclides associated with ^{125}I Production

Property	I-125	I-126	Xe-125
Half Life [3] (d)	60.1	12.9	0.7
Mode of decay [3]	Electron Capture	Beta, Electron Capture	Electron Capture
Progeny [3]	^{125}Te (stable)	^{126}Te (stable) ^{126}Xe (stable)	^{125}I (radioactive)
Main Energetic Electrons Emitted [3]	22 to 35 keV 33 % 0.7 to 4.4 keV 540% (auger and conversion)	289.7 keV β^- 32% 458.5 keV β^- 8.0% 508.4 keV β^+ 3.3%	21.79 keV 22% 155.3 keV 6.1% (conversion electons)
Main Photons Emitted [3]	35.49 keV γ 6.7% 27- 32 keV X rays 140%	666.3 keV γ 33% 388.6 keV γ 34%	188.4 keV γ 55% 243.4 keV γ 29% 453.8 keV γ 4.2% 28.61 keV X ray 54%
Intake Dose Coefficient – ingestion (Sv Bq^{-1}) [4]	1.5E-8	2.9E-8	NA
ALI –ingestion (Bq)	1E6	7E5	NA
Intake Dose Coefficient – inhalation (Sv Bq^{-1}) [4]	1.4E-8 (vapour)	2.6E-8 (vapour)	9.3E-10 (Sv d^{-1} Bq^{-1} m^3) - immersion
ALI –inhalation (Bq)	1E6	8E5	NA
Derived Air Concentration (Bq m^{-3})	417	333	258
Specific Gamma Dose Constant [5] (mSv $h^{-1}MBq^{-1}m^2$)	7.432E-5	1.055E-4	9.622E-5
Tenth Value Layer Pb Shield (cm)[2]	0.00629	2.156	0.8495[3] 3.038[4]
Tenth Value Layer Iron Shield (cm)[2]	0.0336	6.171	3.631[3] 5.319[4]

[2] Approximate tenth value layers determined from running a validated version of the code ®Microshield 5.0. Value given is the tenth value layer for absorbed dose rate in air for a point source with a plane infinite shield over the first three tenth value layers, including the contribution from buildup.

[3] First Tenth Value Layer

[4] Second and subsequent Tenth Value Layers

Radiation Safety Program Elements

The International Atomic Energy Agency (IAEA) has described the major elements required for an effective operational radiation protection program [1]. The major components are identified as: Organization and Management, Personnel Selection and Training, Occupational Radiation Control, Public Radiation Control, Quality Assurance and Emergency Operations. Each program element is broken down into several sub-components. The IAEA program components and sub-components are listed and described briefly in the following sections. Each program component is accompanied by a summary of the special considerations that are required for an ^{125}I production operation.

Organization and Management

This component of the program consists of several sub-components, including: headquarters management, local management, and radiation safety responsibilities of the radiation protection officer, individual worker and radiation safety committee.

Headquarters and local management responsibilities include primarily establishing a high level of commitment to radiation safety and to dose optimization in the operation. This commitment is to be demonstrated and communicated through written policy statements and through support for those responsible for implementation of the radiation safety program. Management must also provide the required resources for radiation safety program implementation.

There is no significant difference in management responsibilities with respect to ^{125}I production radiation safety and those for other aspects of the facility's radiation safety program. As in all production activities, a strong safety culture must be cultivated and the precedence of safety over production must be emphasized. The addition of these operations will necessitate additional specialized facilities and equipment as well as investments in additional radiation safety staff and personnel training. These needs are described further in the following sections.

The responsibilities for radiation safety of managers, the radiation protection officer, individual workers and the radiation safety committee are typically established in a radiation safety program document.

As with any major change to a facility's operations or organization, the radiation safety program document must be carefully reviewed and updated to reflect changes required to encompass the ^{125}I production activities within the program. The changes that are required are those highlighted in the remainder of this Chapter and the following chapters.

Personnel Selection and Training

This program element comprises two components: Qualification and Experience and Post Appointment Training.

Qualifications and experience refers to establishing the minimum entry level requirements for personnel fulfilling various roles within the radiation safety program. It is likely that facilities will benefit from establishing separate qualification standards for personnel whose duties include work with large quantities of ^{125}I. Addition of these activities adds significantly to the complexity of radiation safety tasks. In addition, the qualification standard for the Radiation Protection Officer is likely to be more demanding than might otherwise be required for a research reactor that is not involved in the production of large quantities of volatile radionuclides.

The post-appointment training component describes that training for personnel that provides knowledge of the "risks associated with exposure to radiation of the type and magnitude that could be experienced at the establishment and how such exposure may be controlled". As noted previously, introduction of ^{125}I production activities greatly increases the complexity of a radiation safety program and will necessitate a major expansion of the radiation safety training provided to personnel. While it is likely that only a small sub-set of the staff will be directly involved in production activities, it should be recognized that most personnel in the facility will require additional training because of potential exposure to the hazard and because of changes to performance requirements (e.g. use of different equipment for surface contamination monitoring and recognition of alarms on monitors for airborne ^{125}I contamination). The training program for radiation safety personnel will require extensive expansion as well, in order to encompass knowledge of the different nature of the hazards and the methods used for risk monitoring and management.

Occupational Radiation Control

Dose or Intake Control and Limitation – Technical Measures

This program component includes all physical means of controlling radiation exposure. The examples listed by the IAEA are repeated below, followed by a discussion of the impact of ^{125}I production.

(a) Temporary shielding to augment shielding included in the facility design.

It is possible to design a facility to handle TBq quantities of ^{125}I that is virtually entirely self shielded because the desired radionuclide emits only low energy photons. However, consideration must be given to the potential impact that a small level of radioactive impurities with higher energy photons, such as ^{126}I, can have to doses. This problem is discussed further in Chapter 3. The system should be designed to accommodate temporary additional shielding, should it become necessary. This is

often a difficult change to retro-fit if not addressed in the design stage because of the mass of shielding required. An alternative is to build the entire system in a permanent shielded enclosure, which will greatly reduce the sensitivity of personnel doses to impurities.

(b) Rope barriers or fences and locked doors to control access to sources of radiation and/or areas with high radiation or contamination levels.

Because of the volatility of ^{125}I, it is desirable to enclose the workplace in a room to separate it from the rest of the area. This will enhance the effectiveness of local ventilation (discussed below) and provide a barrier to exposures for those not involved in the production activities. Because ^{125}I represents a challenge with respect to contamination spread and contamination monitoring, it is also desirable to surround the entire ^{125}I handling area within a boundary or "zone". This will provide a reinforcement to administrative practices designed to prevent the uncontrolled spread of ^{125}I contaminated personnel and objects through the rest of the facility.

(c) Use of protective clothing and respiratory protection.

The respirator and personnel protective equipment program for the facility is likely to require significant expansion to address the needs of ^{125}I production. Respirators are not typically required on a frequent basis in light water research reactor environments. Personnel involved in the production of ^{125}I , however, are likely to require routine use of respirators to carry out their duties safely. Negative pressure respirators with combination HEPA[5] and charcoal canisters are adequate for most tasks and should be provided for use in incidents and emergencies. However, positive pressure air supplied respirators should be provided whenever possible for routine use due to their greater protection factor and much greater comfort level for users. At McMaster, a combination of air supplied respirators with tyvek hoods has proven very effective at preventing inhalation intakes.

(d) Decontamination facilities to reduce potential airborne contamination levels and background radiation dose rates.

Extensive surface decontamination has been required at all stages of production at McMaster. Consideration should be given to providing dedicated areas in low enough radiation background for direct contamination monitoring with adequate ventilation rates (such as a fumehood) for personnel to perform special and routine decontamination of equipment. Extensive use of reducing solution, as discussed earlier, has proven effective in minimizing airborne iodine contamination levels.

(e) Local supplementary ventilation facilities, fumehoods and gloveboxes.

[5] HEPA: High Efficiency Particulate Air filtration.

Providing adequate local ventilation is critical to the safe production of ^{125}I. Any part of the process in which the sealed production system or closed vessels are to be open, should be performed inside ventilated containment. The ventilation system should be filtered through activated TEDA[6] impregnated charcoal in order to control releases to the environment. Routine handling of open sources of GBq to TBq quantities of unbound radioiodines should be performed only in gloveboxes with similar filtration on the ventilation. For some facilities, adding charcoal filtration to the main ventilation lines would be cost prohibitive and may lead to unacceptable decreases in flow rates. In these cases, there is further reason to segregate ^{125}I production activities as much as possible to separately ventilated enclosures within the facilities. If there is insufficient draw on the main ventilation lines, closed loop clean-up systems with filtration, which both draw contaminated air from and vent cleaned air to the enclosed work areas, can be used to assist in controlling airborne contamination levels.

Dose or Intake Control and Limitation – Administrative Measures

This program component includes all non-physical means of controlling radiation exposure. The examples listed by the IAEA are repeated below, followed by a discussion of the impact of ^{125}I production.

(a) Appropriately written operating procedures which clearly establish and convey required actions and action levels.

Action levels in this context are limits which are established for quantities that are monitored in the radiation safety program (e.g. personnel dose in a year, surface contamination levels in working areas) at which pre-defined actions must be taken. The actions may be, for example, conducting an investigation for personnel doses exceeding an action level, or cleaning up surface contamination that exceeds an action level. Action levels are set at values which are lower than any corresponding regulatory limit.

It will be necessary to establish a new and distinct set of action levels for ^{125}I. Action levels required would include, for example:

- The ^{125}I surface contamination level acceptable on various surfaces and under various conditions. An example of the development of action levels for surface contamination is provided in Chapter 2.
- The ^{125}I airborne contamination at which respiratory protection will be required and at which an area will be posted as an airborne contamination area. At McMaster, this level is one-tenth of the Derived Air Concentration.

[6] TEDA: triethylenediamine

- The level of ^{125}I in workers thyroids at which investigations must be initiated and/or work restrictions implemented. At McMaster, the level for internal action has been established as 1 kBq thyroid burden for workers. Work restrictions and formal investigations with regulatory reporting are carried out following detection of thyroid burdens of greater than 10 kBq.

(b) Access controls, e.g. administrative control on keys to limit access to radiologically hazardous areas to only qualified personnel.

There are no considerations for access control that are unique to the radiological hazards associated with ^{125}I production. If facilities have not previously had a category of area designation for airborne contamination, this will have to be considered along with the default access restriction that will be implemented for those areas.

(c) Radiation work planning, whereby work likely to result in significant dose is adequately planned and authorized by appropriate levels of supervision.

There are no considerations for radiological work planning that are unique to the radiological hazards associated with ^{125}I production. The high potential for airborne contamination should be recognized in assessing hazards. As with any new process, the work must be carefully reviewed and potential hazards analyzed in order to appropriately manage the hazards.

(d) Classification of work areas.

As noted above, it is likely that facilities will have need of an area designation for airborne contamination areas if that is not already part of their radiation safety program. In addition, consideration should be given to the level of surface contamination at which an area will be declared a "contamination area". At McMaster, the level of ^{125}I contamination that requires such designation is ten times higher than the level for other beta/gamma emitting surface contaminants. The derivation of the surface contamination levels used at McMaster is provided in Chapter 2.

(e) Warning signs at entrances and to show sources of hazard.

Monitoring radiation fields and contamination arising from ^{125}I production requires specialized instrumentation (as distinct from the same hazards arising from common activation and fission products, for example). Signs should therefore be designed to provide a clear indication of the presence of these hazards in a given area.

(f) Mock ups or rehearsals of the work to be done.

This general good practice is applicable to work related to ^{125}I production in the same way that it is for other radiologically hazardous work.

(g) Provision of radiation protection staff to assist and to monitor conditions during hazardous work.

The introduction of ^{125}I production activities in many research reactors is likely to result in greatly increased use of respirators and personnel protective equipment as well as increased complexity of monitoring activities. Consideration should be made for additional radiation protection staff to assist with routine and maintenance tasks.

Surveillance Programs – Individual dose or intake monitoring

This component of occupational radiation control comprises the recording and assessment of internal and external doses.

The response of the external dosimetry system (generally, thermoluminescent dosimeters) should be verified for the low energy photons associated with ^{125}I decay. There is potential for an under response. Similarly, the response for any active dosimeter (such as alarming electronic personnel dosimeters or pocket ionization chambers) must be verified.

Internal dose assessment for ^{125}I can be achieved by urine monitoring or by thyroid counting [2]. Unless a facility already has a bioassay program in which urine samples are routinely collected and analyzed by gamma spectroscopy, it is likely that thyroid monitoring will be more convenient. Thyroid monitoring is also preferred in cases where the time of intake is unknown, because the urinary levels fall rapidly after intake [2]. Intakes of one-tenth of the Annual Limit on Intake can be reliably detected in a routine monitoring program with measurements conducted quarterly [2]. A self-screening program can be established using thin NaI or CsI based contamination monitors which can typically provide in-vivo efficiencies in the order of 5%. An action level should be established for reporting self-screening results to radiation safety personnel for follow-up. At McMaster, personnel involved in ^{125}I production are required to self-screen weekly and an action level of 1000 net counts per minute has been established. This corresponds to a thyroid burden of approximately 300 Bq and an acute intake of approximately 1 kBq [2]. The higher frequency of monitoring increases the likelihood that any intake which does occur will be traceable back to its cause. The appropriate action level for other facilities will depend on the magnitude and frequency of intakes that occur and on the magnitude of intake that necessitates a formal dose assignment. Intakes have been infrequent in the McMaster program, which has allowed for a relatively low action level to be established.

Surveillance Programs – Area monitoring

This radiation safety program element comprises the procedures and equipment required to conduct routine and task based radiation and contamination surveys of the workplace as well as the fixed equipment required to perform area monitoring in the workplace. Exit contamination monitors may also be included in this component of the radiation safety program. Monitoring procedures and equipment must address both routine and emergency conditions that may be encountered in the facility.

Because of the difficulty in detecting the radiation emissions from ^{125}I, facilities embarking on a production program are likely to require an extensive expansion of their instrument inventory and procedures.

Most commercially available radiation survey meters do not have adequate response in the 30 keV photon region required for accurate assessment of radiation fields arising from ^{125}I. Ion chambers or special low-energy versions of gamma survey meters are required.

Most commercially available fixed area gamma radiation monitors will likewise be unable to detect radiation fields arising from ^{125}I. At McMaster, this has been addressed through extensive use of personnel alarming dosimeters with suitable energy response characteristics. However, an area monitor with a better response to low energy photons would be highly desirable.

Typical portable contamination meters used in a research reactor environment will have insufficient sensitivity for ^{125}I contamination. Contamination meters based on thin NaI or CsI crystal detectors will provide adequate efficiencies in the order of 5%. In some applications, a sealed proportional counter with Xe gas added to the counting gas may prove to be useful. Large area detectors with efficiencies of a few per cent are commercially available, but are significantly more costly than the NaI or CsI detectors. They are also more prone to damage during field work. One advantage of the proportional detectors is that they allow users to monitor simultaneously for fission/activation product contamination and ^{125}I contamination.

Exit contamination monitors (hand and foot or whole-body) utilizing Xe filled sealed gas proportional tubes are available. These monitors are the best currently available technology for monitoring of personnel and small items leaving the facility. As noted above, they are capable of simultaneously monitoring for both ^{125}I contamination and normal fission/activation product contamination.

Swipe or indirect check monitoring is normally an important component of a radiation safety program. Task based monitoring of equipment and the work area provides an immediate indication of performance with respect to contamination control. Routine periodic monitoring at established sample points in the workplace provides an important

indication of facility conditions and the long term effectiveness of the contamination control program. Swipe counters deployed in the field are frequently shielded Geiger-Mueller detectors to allow swipes to be counted in elevated backgrounds. Swipe counters with NaI, CsI or Xe filled gas proportional detectors will be required to provide sensitivity for ^{125}I. Counting large numbers of routine swipes is typically performed in a lab environment, often with an automated counting system. Windowless gas proportional counters can be used to monitor swipes for ^{125}I but the efficiency is fairly low – typically about one percent. Counters with these detectors are not widely used in this role. Liquid scintillation counting provides excellent efficiency for ^{125}I and the ability to have samples counted in bulk. The use of a liquid scintillation counter does result in additional expense for the counting medium and sample preparation time.

Continuous air monitoring is a vital component of a radiation safety program for ^{125}I production. Because of the volatility of the iodine encountered in the process and the large quantities used, the potential for sudden excursions in airborne concentrations is high. As with swipe counters, it must be noted that continuous air monitors employed in most facilities are Geiger Mueller tube based and will provide no detection efficiency for ^{125}I. In addition, commercially available activated charcoal canisters are required as the collection medium for ^{125}I detection. Additional or modified continuous air monitors will be required. Experience at McMaster has shown that monitoring of the air in the production area, the occupied area surrounding the production area, the waste storage area and the facility exhaust is necessary to provide a suitable level of surveillance for operations.

Surveillance Programs – Maintenance of records and data

This program component addresses the need to maintain records and data of monitoring conducted, and to review the data from time to time to identify any trends that are occurring. The collection and collation of individual and collective dose data to aid in future decision making is also addressed, and it is recommended that dose budgets be established for operations to ensure that doses are not allowed to increase inadvertently.

There are no considerations for maintenance of records and data that are unique to the radiological hazards associated with ^{125}I production. It is likely that the body of data to be maintained and assessed will increase markedly. Consideration should be given to the means and frequency at which data will be communicated to facility personnel. At McMaster, doses and air monitoring trends are assessed monthly and are communicated formally to facility management and production personnel on a quarterly basis. Immediate feedback occurs when radiation safety personnel note results of concern.

Careful dose tracking and budgeting proved to be important in identifying high unexpected doses that were occurring in the initial stages of production at McMaster. This is discussed in detail in Chapter 3.

Public Radiation Control

Control and limitation of releases to the environment

A crucial component to the control of doses to members of the public is the establishment of limits for releases from the facility. It is specified by the IAEA that these limits should be established by the facility operator but reviewed by the regulatory authority.

In Canada, the limits on releases are known as Derived Release Limits. An example of a Derived Release Limit calculation is provided in Chapter 4.

Location and control of sources in the public domain – Transportation of radioactive material

There are no considerations for transportation of radioactive material that are unique to the radiological hazards associated with ^{125}I production.

Location and control of sources in the public domain – Storage of radioactive wastes.

Radioactive wastes from the ^{125}I production process will consist of two streams – a comparatively low volume stream of activated materials from in-core irradiation and a comparatively large volume of ^{125}I contaminated (or suspect) materials from other parts of the process.

Activated components can be handled in the same manner as other activated waste streams from a reactor. For the relatively larger volume of contaminated and suspect items and material, the opportunity exists to divert that waste through decay in storage techniques. McMaster has excellent success in placing waste in storage for two years prior to processing it. A ventilated waste sorting and monitoring table was designed to allow each waste package and container to be opened, inspected and monitored to confirm it is suitable for release to the non-radioactive waste stream. Inspection of the waste packages was identified as a requirement because of the potential for self shielded items, such as crimped sections of metal piping, to conceal the presence of large quantities of ^{125}I. Small batches are then monitored under large area Xe gas proportional tubes to verify that no significant contamination remains. This also verifies that the material is not cross-contaminated with other longer lived contaminants from the facility. Although this process is somewhat labour intensive, virtually one hundred percent of the waste can be diverted from this stream and it has proven very cost effective.

Surveillance Program

Adequate surveillance of exposure pathways for the public must be maintained. This program component includes source or release monitoring, environmental monitoring, radiation field monitoring and maintenance of records and data.

Undertaking ^{125}I production is likely to result in releases through the air effluent stream. The effluents can be monitored by drawing a sample of the air stream through a charcoal canister. At McMaster, the exhaust is continuously monitored to provide immediate indication of unplanned releases. The sample is removed weekly and is analyzed by gamma spectroscopy. A similar approach is taken with the environmental monitoring program. The only change that has been necessary to the environmental program at McMaster has been the addition of air samplers with charcoal canisters at remote locations.

Quality Assurance

The quality assurance program element includes the documentation and auditing activities required to ensure that the other components of the program are carried out as specified and achieve their intended purpose. There are no considerations for the quality assurance program that are unique to the radiological hazards associated with ^{125}I production. Experience at McMaster has shown that careful quality control monitoring of the product for impurities, particularly ^{126}I, is essential for exposure control.

Emergency Operations

The radiation protection program of the facility must have the capability of effectively responding to accidents and emergencies.

A careful review of a facility's emergency plans must be conducted to ensure that the impacts of initiating ^{125}I production activities are well recognized and are addressed by the plan. Chapter 5 provides an example of the assessment of the impact of "worst case" bounding releases at McMaster and evaluates them with respect to established criteria for implementing the emergency plan.

Summary

The addition of ^{125}I production activities to a facility will require significant revision and expansion of the radiation safety program. The following items should be considered:

- The facility's safety culture program should be reviewed and consideration should be given to the impact of adding a potentially challenging isotope production schedule.
- A qualification standard for ^{125}I production personnel should be established. The radiation safety training program must be expanded to address the nature of hazards associated with the production activities and the new or changed procedures resulting from the production activities (for example, use of new instruments). The impact on training needs of production personnel, radiation safety personnel and other staff should be addressed.
- The proposed production process and the facility needs should be analyzed to determine the best way to provide shielding and ventilation to the production facilities. The shielding assessment should consider the potential impact of impurities in the process. If a permanently shielded facility is not selected, provision to add temporary shielding when required should be included in the design. An enclosed and separately ventilated layer of containment (such as a dedicated room) should be placed around the production activities. The area where ^{125}I production activities will occur should be segregated from other areas to the extent possible and should be designated as a contamination zone. Ventilated glove boxes with lead impregnated gloves should be provided for production activities. Ventilated work areas, such as a fumehood, should be provided in a low background area for decontamination activities. The ventilation for the production areas should be charcoal filtered.
- The respirator and personnel protective equipment program should be expanded to address the needs of the production process. The program should be able to address the routine use of respirators in production tasks. Air supplied respirators should be provided for routine prolonged use.
- An internal dosimetry program, preferably based on thyroid monitoring in most instances, should be implemented for ^{125}I. The external dosimetry should be assessed for response to the low energy photons associated with ^{125}I.
- The radiation safety instrumentation set must be expanded to address the needs of ^{125}I production. In most cases, facilities will require specialized contamination monitoring, airborne contamination monitoring, dose rate and exit monitoring equipment. In addition, the air effluent and environmental monitoring programs must be expanded to include capability to monitor for airborne ^{125}I contamination.

- Action levels for ^{125}I production related parameters should be established. Action levels should be established for dose to production personnel (monthly and annual), for ^{125}I surface contamination levels, for ^{125}I contamination levels in workplace air and air exhaust, for ^{125}I thyroid burden in production workers, for dose rates associated with specified areas/components and for other important parameters monitored in the radiation safety program.
- The radiological classification and posting procedures should be expanded as required. Requirements for airborne contamination areas should be established. Warning signs and postings should be modified to enable indication of the presence as ^{125}I as a component of the hazard when applicable.
- Provision should be made for long term storage of radioactive waste to allow "delay and decay" strategies to be implemented. Facilities and equipment for waste sorting and monitoring should be provided.
- The impact on the facility's emergency plan should be assessed and any additional requirements (personnel, equipment or procedural) addressed.
- The additional responsibilities, action levels, requirements, etcetera described above should be incorporated into the radiation safety program documents and supporting procedures.
- The need for additional radiation safety personnel to implement the radiation safety changes described above should be considered.

References:

[1] International Atomic Energy Agency, Operational Radiation Protection: A Guide to Optimization, IAEA Safety Series No. 101, IAEA, Vienna (1990).

[2] International Commission on Radiological Protection, Individual Monitoring for Internal Exposure of Workers, ICRP 78, Annals of the ICRP Vol. 27 No. 3/4 (1997).

[3] International Commission on Radiological Protection, Radionuclide Transformations: Energy and Intensity of Emissions, ICRP 38, Annals of the ICRP Vol. 11-13 (1983).

[4] International Commission on Radiological Protection, Dose Coefficients for Intakes of Radionuclides by Workers, ICRP 68, Annals of the ICRP Vol. 24 No. 4 (1994).

[5] Schleien, B, The Health Physics and Radiological Health Handbook Revised Edition, Scinta Inc., Silver Spring Md. (1992).

Chapter 2: Derived Surface Contamination Levels for ^{125}I

Introduction

In Chapter 1, the radiation safety program components relevant to occupational radiation control were discussed. Establishing action limits is described as an important component of the administrative measures that form part of the dose and intake control and limitation considerations. This chapter describes in detail the derivation of one such set of limits – the limits on surface contamination for ^{125}I.

The control of surface contamination is an important aspect of an occupational radiation protection program. Extensive effort is often devoted to monitoring for surface contamination and maintaining surface contamination below prescribed limits. Careful selection of the limits is necessary if exposures and radiation safety resources are to be optimized. Selection of limits that are too high will result in unnecessary exposures. Selection of limits that are too low will result in disproportionate effort and resources being devoted to this aspect of the radiation safety program. As there is likely to be some occupational dose resulting from monitoring and decontamination efforts, selection of limits which are too low could also result in unnecessary exposures.

A rational approach to establishing surface contamination limits has been established by the Advisory Committee (ACRP) on Radiation Protection to the former Atomic Energy Control Board of Canada [1]. The ACRP refers to these limits as Derived Working Limits (DWLs).

In the report attached to this chapter, the ACRP's method is described and is applied to determine the DWLs for ^{125}I surface contamination. One significant deviation from the ACRP's recommendations was made in incorporating the guidance of Johnson et al [2] for evaluating dose from ^{125}I retained in the skin, and this deviation is outlined in the report. Surface contamination limits are then selected based on the DWLs and on the objectives and operating experience of the MNR contamination control program. The normal methods of contamination monitoring at MNR are evaluated and the adopted set of contamination limits are expressed in terms of user-observable instrument readouts. Finally, the minimum detectable activities and sensitivities of the various monitoring methods are examined to demonstrate their suitability for detection of contamination at the selected limits.

Discussion

Surface contamination can lead to exposures through the following pathways: external irradiation of the skin after contact with contaminated surfaces, inhalation of re-suspended contamination, ingestion of activity transferred to the skin from contaminated surfaces and uptake by the skin leading to systemic distribution and/or irradiation of the skin in which the material resides [1]. When the contamination is present in a controlled area of a nuclear facility, then the exposure is to nuclear energy workers. When contamination is located outside of controlled areas then the exposure may be to workers or members of the public. Each exposure pathway must be examined for a given radionuclide to determine the limiting exposure.

Examination of the way that work is conducted in nuclear facilities has resulted in recommendations that contamination limits must be selected for a variety of surfaces including: personnel skin, personal clothing, protective clothing, controlled area surfaces and uncontrolled surfaces [1]. In addition, it is necessary to identify the appropriate limits for shipments of radioactive material.

Application of the ACRP's recommended approach consisted of evaluating the surface contamination level that would result in the applicable dose limit (i.e. effective dose limit for workers, equivalent dose limit to the skin of workers, effective dose limit to the public, etcetera) on each of the surfaces listed. The result of the analysis was a set of Derived Working Levels (DWLs). The DWLs were then evaluated for inclusion into the MNR radiation safety program.

In selecting the contamination limits, the DWLs were considered as the maximum acceptable level. The objectives of the contamination monitoring program were also considered. The primary objective of the program is:

- To maintain exposures of workers and the public to radioactive materials As Low As Reasonably Achievable (ALARA).

The secondary objectives of the program include:

- Detecting any loss of control resulting from failures of containment or departures from good operating practices.

- Assisting in preventing the spread of contamination from controlled areas. This includes prevention of exposures as well as degradation of background levels in counting areas.

- To provide information for establishing monitoring programs and operational procedures including protective equipment requirements.

The primary objective implies the need for an optimized limit, as opposed to a limit derived from the dose limits. The secondary objectives of detecting loss of control and assisting in preventing spread from controlled areas also lead to the selection of the lowest reasonable limit.

An additional important consideration in selecting the final limits was the operating experience that had been gained at MNR in early production stages, as well as the facility's safety culture with respect to surface contamination. A decision had been made at MNR to maintain the level of beta-gamma emitting surface contamination on generally accessible surfaces in the controlled area at one-tenth the value generally adopted in industry. This decision was made in recognition of the facility layout and operations as well as its role as a teaching and research facility. This philosophy was extended to the limits adopted for ^{125}I. In addition, it had been demonstrated that contamination could be maintained at levels significantly less than the DWLs without undue effort or unnecessary exposures. This experience suggested that surface contamination limits equal to the DWLs would not be optimized limits.

A final consideration in selecting the surface contamination limits for adoption was to evaluate the normal monitoring methods employed at MNR and ensure that they were suitable for detecting contamination at the stated limits. In each case, the detection limits for the current monitoring methods were found to be suitable for the contamination limits that were adopted.

The DWLs and the adopted surface contamination limits are summarized in Table 2-1.

Examination of the DWLs will show that the upper limits acceptable for ^{125}I are significantly higher than the limits routinely adopted for surface contamination in the nuclear industry. The typical nuclear industry limit for beta-gamma emitting contaminants on items being released from controlled areas, for example, is 5 Bq/cm^2. As shown, if the contamination is known to be ^{125}I, then a limit of up to 14 Bq/cm^2 could be adopted.

An important application of the contamination for working surfaces in the controlled area is to define the point at which additional radiological postings, protective measures and access controls must be implemented. An example of this application is shown in Table2-2, which is an excerpt from the MNR radiation safety program document [3]. The table summarizes the contamination levels and the corresponding controls in MNR. Note that, when the contamination is known to be ^{125}I, the limits corresponding to a given area classification are ten times the limits that apply otherwise. The derivation and selection of these values is explained in the attached report.

Table 2-1: I-125 Contamination Limits in MNR

Surface	Applicable DWL (Bq / cm^2)	Adopted Limit for MNR (Bq / cm^2)
Personnel Skin Contamination	130	50
Personal Clothing and Items (Papers, Books carried by hand)	140	50
Protective Clothing	140	50
Normally Accessible Working Areas in MNR (Floors, Walls, Benches, Tools, Equipment, etceteras)	140	5
Areas Posted as Contamination Area	NA	50
Equipment released from MNR	14	5
Outer Surface of "Excepted" Radioactive Shipments	NA	0.5
Outer Surface of Other Than "Excepted" Radioactive Shipments	NA	5

Table 2-2 : Designation and Posting of Areas with Elevated Contamination Levels

Condition	Designation	Requirements
Accessible Contamination $<0.5\ Bq/cm^2$ *	None	No Restriction
Accessible Contamination $>0.5\ Bq/cm^2$ and $<5\ Bq/cm^2$ *	Contamination Area	Post Area with Designation, the contamination level, and entry requirements as determined by Health Physics.
Accessible Contamination $>5\ Bq/cm^2$ *	High Contamination Area	Post Area with Designation, the contamination level, and entry requirements as determined by Health Physics. All work in High Contamination Areas must be reviewed by Health Physics Clean-up of Contamination is to be Supervised by Health Physics. No Access for non-Nuclear Energy Workers
Airborne Contamination > 0.1 DAC	Airborne Contamination Area	Post Area with Designation, the contamination level, and entry requirements as determined by Health Physics. All work in Airborne Contamination Areas must be reviewed by Health Physics No Access for non-Nuclear Energy Workers

* Where the contamination is known to consist only of I-125, the applicable values are ten times higher than those listed

Summary

^{125}I surface contamination limits derived according to the guidance of the ACRP have been presented. The values that are acceptable are significantly higher than the typical nuclear industry limits for beta-gamma emitting surface contamination. Derivation of a radionuclide specific limit is therefore likely to be a worthwhile undertaking for facilities planning to handle large quantities of ^{125}I.

The values suitable for adoption at other facilities will depend on the local conditions and practices, however, the DWLs listed in Table 2-1 should be considered as the upper bound in selecting an optimized limit.

In Chapter 1, a set of instrumentation suitable for use with ^{125}I was described. Standard industry monitoring practices for surface contamination utilizing that instrument set will provide detection limits at or below the DWLs.

Attached Report

The McMaster University Health Physics Department report MNR-99-02, Iodine Contamination Limits in MNR (1999) is attached to this chapter.

The report was prepared solely by the author, using the references cited in the report as guidance. This report was prepared for submission to McMaster oversight committees for MNR and to the federal regulator. The report was used as a basis document to support the MNR radiation safety program document [3] as well as various working procedures. It was also used to establish performance requirements in the selection of fixed exit monitors for the facility.

Minor editorial changes have been made to the report for inclusion in this thesis.

References

[1] Advisory Committee on Radiation Protection, Report on Derived Working Limits for Surface Contamination, ACRP-7, (1993).

[2] Johnson et al, Dose to the Basal Layer of the Skin from ^{125}I Contamination, Radiation Protection Dosimetry, Vol. 20 No. 4 pp. 253-256, (1987).

[3] McMaster Nuclear Reactor, The MNR Radiation Safety Program, HP-9000 (1991)

McMaster University
Health Physics Department

Report MNR-99-02

Iodine-125 Contamination Limits in MNR

Prepared by: D.M. Tucker, Senior Health Physicist Date: 1999 June 1[1]

[1] With slight editorial revisions 2005 November

Report MNR-99-02

Iodine-125 Contamination Monitoring Practices in MNR

Reason for Report

Derived Working Limits for I-125 contamination are determined following the guidance of the Advisory Committee on Radiological Protection (ACRP). The DWLs are evaluated in light of MNR contamination control practices, monitoring capabilities and other applicable limits and conservative limits on I-125 surface contamination for use in the MNR Radiation Safety program are set out.

The standard monitoring practices are described briefly for each surface, and the instrument readouts are determined which correspond to the I-125 contamination limits adopted for MNR.

Objectives of the Contamination Monitoring Program

The primary objective of the Contamination Monitoring Program in MNR is as follows:

- To maintain exposures of workers and the public to radioactive materials As Low As Reasonably Achievable.

Secondary Objectives include:

- Detect any loss of control resulting from failures of containment or departures from good operating practice.
- To assist in preventing the spread of contamination from controlled areas. This includes prevention of exposures as well as degradation of background levels in counting areas.
- To provide information for establishing monitoring programs and operational procedures including protective equipment requirements.

To meet these objectives, contamination limits must be established at the lowest reasonable level considering operational efficiency as well as available instrumentation and monitoring methods

Derived Working Limits for Iodine-125

Derived Working Limits (DWLs) have been calculated following the guidance of the ACRP [2] and the work of Johnson et al [3]. The derivations are shown in Appendix A. These limits are given in Table 1.

The Derived Working Limits are analogous to dose limits. They represent the level at which sustained contamination would be considered barely tolerable. These limits can be compared to the traditional value 4 Bq/cm^2 applied widely in the nuclear industry to unidentified beta/gamma emitters.

Table 1: Derived Working Limits for I-125

Surface	DWL
Personnel Skin Contamination	1.3 E 2
Personal Clothing	1.4 E 2
Protective Clothing	1.4 E 2
Controlled Area Surfaces	1.4 E 2
Uncontrolled Area Surfaces (and equipment leaving a controlled area)	1.4 E 1

MNR Contamination Limits for I-125

In addition to the DWLs, the following limits must be considered.

1. HP-9000, the MNR Radiation Safety Program [1], specifies a limit of 0.5 Bq/cm^2 for beta/gamma emitters for posting as a Contamination Area. An equivalent level for I-125 is required.

   This value is taken as equivalent to the value for a Controlled Area Surface (140 Bq/cm^2). That is, the upper limit for working surfaces in the Controlled Area within the reactor is the same as the level of contamination that requires posting and special precautions associated with a Contamination Area. However, a policy decision has been made to control contamination at an order of magnitude below the traditional industry practice, which would lead to a value of 14 Bq/cm^2. Considering available monitoring techniques and conditions normally encountered in MNR, it is possible to reduce this value further, and so a limit of 5 Bq/cm^2 is chosen. The choice of a value ten times greater than that for beta/gamma emitters is practical and should make it easier for facility personal to remember the number.

2. HP-9000, the MNR Radiation Safety Program [1], specifies a limit of 4 Bq/cm^2 for beta/gamma emitters for posting as a High Contamination Area.

   Extending the rationale given above, a limit of 50 Bq/cm^2 for I-125 is selected.

3. The Transport packaging of Radioactive Materials Regulations [4] and IAEA guidelines specify a limit of 0.4 Bq/cm^2 for the external surface of Excepted Shipments and 4 Bq/cm^2 for other types of radioactive shipments for "beta and gamma" emitters.

   No Latitude is given in the shipping guidelines to deviate from these values. They are incorporated into the MNR Contamination Limits unchanged.

4. In each case not listed above, the lowest reasonable level must be adopted considering the DWL as an upper limit.

MNR Contamination Limits for I-125 have been selected as outlined above, and are given in Table 2.

Table 2: I-125 Contamination Limits in MNR

Surface	Applicable DWL (Bq / cm^2)	Adopted Limit for MNR (Bq / cm^2)
Personnel Skin Contamination	130	50
Personal Clothing and Items (Papers, Books carried by hand)	140	50
Protective Clothing	140	50
Normally Accessible Working Areas in MNR (Floors, Walls, Benches, Tools, Equipment, etceteras)	140	5
Areas Posted as Contamination Area	NA	50
Equipment released from MNR	14	5
Outer Surface of "Excepted" Radioactive Shipments	NA	0.5
Outer Surface of Other Than "Excepted" Radioactive Shipments	NA	5

Sensitivities of Monitoring Techniques

Several methods are used to monitor I-125 surface contamination in MNR. Each method has been assessed and the instrument reading corresponding to applicable limits has been determined. The derivation of the limits is shown in Appendix B. The results, with instructions to workers encountering I-125 surface contamination, are given in Table B2.

Minimum Detectable Activities (MDAs) have been determined for the monitoring techniques used. For the portable instruments (the Bicron Ratemeter with thin 1 inch NaI combination), the effect of background has been analyzed. The MDAs and the maximum backgrounds for reliable detection of contamination at the MNR limits are presented in Appendix C.

Summary

Contamination Limits for MNR have been set out. In each case, the limit is a small fraction of the applicable DWL and can be detected using existing monitoring techniques. The instrument readings corresponding to these limits and instructions for workers encountering them have been set out and will act as the basis for detailed procedures where required.

The limits for I-125 Surface Contamination will be incorporated into the next revision of HP-9000, the MNR Radiation Safety Program.

References

[1] HP-9000, The MNR Radiation Safety Program

[2] ACRP-7, Report on Derived Working Limits for Surface Contamination, the Advisory Committee on Radiation Protection, 1993

[3] Johnson et al, Dose to the Basal layer of the Skin from ^{125}I Skin Contamination, Radiation Protection Dosimetry, Vol. 20 No. 4 pp. 253-256, 1987

[4] J. Harrison, The Fate of Radioiodine Applied to Human Skin, Health Physics, Vol.9 pp. 993-1000, 1963

[5] CAN/CSA-N288.4-M90, Guidelines for Radiological Monitoring of the Environment, Canadian Standards Association, 1990

[6] ICRP 61, Annual Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations, 1991.

Appendices

A. Derived Working Limits

B. Monitoring Methods and Sensitivities

C. Minimum Detectable Activities

Appendix A: Derived Working Limits for Iodine-125 Contamination in MNR

A methodology for determining Derived Working Levels (DWLs) for surface contamination is described in ACRP-7 [2]. It requires an analysis of the following exposure pathways: external irradiation of skin due to contamination on the surface, ingestion of a fraction of contamination from contaminated skin, uptake of the radionuclide via absorption through the skin and skin dose from a radionuclide retained in the skin. The application of the methodology for ^{125}I is shown below for the first three exposure pathways. Limits are derived based on this methodology for the following surfaces: personnel skin, personal clothing, protective clothing, Controlled Area surfaces, Uncontrolled Area surfaces (and equipment leaving a controlled area).

The general approach in determining the derived limits for surface contamination, as with all derived limits in radiation protection, is to identify an exposure pathway and then to calculate the level of surface contamination that will result in the applicable dose limit via that pathway. The specific dosimetric parameters to be applied in modeling each of the exposure pathways are described below.

The derivations below differ from the guidance of the ACRP in three areas. First, the ACRP notes the significance of the final pathway – dose from a radionuclide retained in skin – for ^{125}I but does not provide specific methodology for deriving a working limit in this case. This exposure pathway has been addressed by Johnson et al [3] and the DWL derived by them for this pathway has been adopted. Second, the method used to derive the working limit for the exposure pathway of uptake through the skin differs slightly as explained in that section. Finally, the derivation of the limit for Uncontrolled Area surfaces varies slightly from the ACRP guidance as explained in that section below.

External Irradiation

For external irradiation, the model for exposure is that the skin becomes contaminated or comes into intimate contact with a contaminated surface, and that exposure to the basal layer of the skin occurs for a period of time, T_c, which varies depending on the contaminated surface. When considering personnel skin, it is assumed that workers would remain contaminated all of the time (8760 hours per year). For personal clothing, it is assumed that the exposure will occur over 16 hours per day of each working day (4000 hours per year). For Controlled Area surfaces and personnel protective equipment, it is assumed that exposure occurs continuously while at work (2000 hours per year). The dose limit of concern in this case is the occupational dose limit for the skin, which is 500 mSv per year. The dose conversion factor (DCF) is the dose rate to the basal layer of the skin from contamination deposited on the surface of the skin.

$$DWL = \frac{H_L}{T_c (DCF)_e} \; Bq \cdot m^{-2}$$

Where

H_L = annual dose limit = 500 mSv y^{-1}

T_c = exposure time

= 8760 hours y^{-1} for skin contamination

= 4000 hours y^{-1} for contamination of personal clothing

= 2000 hours y^{-1} for contamination of protective equipment and controlled area surfaces

$(DCF)_e$ = external dose conversion factor to basal layer of skin

= 1.92×10^{-12} Sv m^2 h^{-1} Bq^{-1} for I – 125

Inhalation

For inhalation, the exposure model applied is that contamination deposited on surfaces becomes resuspended and is inhaled by the worker. A generic resuspension factor of $5x10^{-5}$ is recommended by the ACRP for most conditions. For skin and personnel clothing, it is assumed that the exposure continues during the waking hours of every work day, or 16 hours per day for 250 days per year, mixed between work and rest. The volume of air inhaled by reference man during this time period is taken as 3600 $m^3 y^{-1}$. For Controlled Area surfaces, exposure is assumed to occur continuously through the working year. Reference man is assumed to inhale 2400 m^3 during the working year. The dose limit of concern in this case is that for committed effective dose from occupational exposure. This is represented by the Annual Limit on Intake (ALI) for inhalation in the equation. The ALI for inhalation is the activity which, if inhaled by Reference Man, would result in a committed effective dose of 20 mSv.

$$DWL = \frac{ALI}{I_a R_s} Bq \cdot m^{-2}$$

Where

ALI = Annual Limit on Intake (Occupational)
= 2E6 Bq y^{-1} for I - 125

I_a = inhalation rate ($m^3 y^{-1}$)
= 3600 $m^3 y^{-1}$ for contamination of skin and personal clothing
= 2400 $m^3 y^{-1}$ for contamination of protective equipment and controlled area surfaces

R_s = resuspension factor for particulate material from surfaces
= 5×10^{-5} m^{-1}

Ingestion

For Ingestion, the exposure model assumed is that a fraction of the activity on personnel skin is transferred to food or otherwise accidentally ingested after transfer to items being handled. It is assumed that the transfer would occur only from the hands, which have an area of 0.03 m^2. It is assumed that one tenth of the activity on the hands is accidentally ingested, and that this happens once per working day, or 250 times per year. The dose limit of concern in this case is that for committed effective dose from occupational exposure. This is represented in the equation by the ALI for ingestion. The ALI for ingestion is the activity which, if ingested by reference man, would result in a committed effective dose of 20 mSv.

$$DWL = \frac{ALI}{A_s f_a N_c} Bq \cdot m^{-2}$$

Where

ALI = Annual Limit on Intake (Occupational) by ingestion
= 1E6 Bq y^{-1} for I - 125 [7]

A_s = area of skin being considered - normally taken as the area of the hands
= 0.03 m^2

f_a = fraction of activity on the skin that is ingested
= 0.1

N_c = number of contamination events per year - normally taken as once per working day
= 250

Uptake by Skin

This pathway considers systemic uptake via the skin. Uptake by the skin is a pathway that applies in relatively few cases. Tritium and iodine are examples of radionuclides readily absorbed through the skin. Other radionuclides may be absorbed through the skin if they present in a chemical form that is absorbed through the skin. In this case, the dose limit of concern is that for committed effective dose from occupational exposure (20 mSv per year). It is assumed that the area of the hands (0.03 m^2) is continuously kept contaminated at the level of the DWL, and that 0.1% of the activity present on the skin is taken up per hour (see note below). The exposure time is taken as the entire year, or 8760 hours.

$$DWL = \frac{H_L}{(DCF)A_s f_u N_c} Bq \cdot m^{-2}$$

Where

H_L = Annual Dose Limit

$= 20\ mSv\ y^{-1}$

DCF = Dose Conversion Factor for the critical organ or tissue

$= 1.7 \times 10^{-8}$ Sv/Bq for effective dose due to thyroid irradiation by I-125

f_u = fraction of activity on the skin surface that is taken up into the body through the skin

$= 0.001\ h^{-1}$

N_c = total time of contamination per year

$= 8760\ h$

Note: This derivation varies slightly from that used in ACRP – 7 [2]. The value of f_u is taken from the paper by Harrison [4]. Harrison presents data showing that a fraction of 0.2% of radioiodine applied to the skin is absorbed into the body at the end of two hours. This implies a value of f_u of 0.1% h^{-1}. Harrison goes on to derive a value of 0.008% cm^{-2} h^{-1} by dividing by the area over which the contamination was applied to experimental subjects. However, no basis was provided for expressing the constant as a function of area, and this treatment cannot be applied directly to derivation of a DWL. Use of the value of f_u as described above requires that N_c be taken as the total time the contamination is present on skin in a year, versus the number of contamination incidents as originally defined by the ACRP.

Retention in Skin

Skin dose due to iodine retained in the skin is a significant exposure pathway for ^{125}I. The pathway has been examined by Johnson et al who have derived a DWL of 140 Bq cm^{-2} based on an occupational exposure limit of 500 mSv. The DWL is based on a dose conversion factor (DCF) of 10μGy d^{-1} Bq^{-1} cm^2. The dose conversion factor is dependent on the depth of penetration, the distribution with depth (linear or exponential decrease with depth) and the depth of the basal layer.

Uncontrolled Areas

For Uncontrolled Areas, ACRP recommends that the application of the equation for the ingestion pathway be modified as follows:

$$DWL_{UA-I} = \frac{ALI}{A_s f_a N_c} Bq \cdot m^{-2}$$

Where

ALI = Annual Limit on Intake (Occupational) by ingestion / 20 (ratio of occupational to public dose limit)
= 1E6 Bq y^{-1} for I - 125 for ingestion

A_s = averaging area for contamination measurements (nominally set to 0.01 m^2)
= 0.01 m^2

f_a = fraction of activity on surface that is ingested
= 0.1

N_c = number of contamination events per year - normally taken as once per working day
= 250

In addition, it is recommended that each of the exposure pathways be considered and that a scaling factor of 1/20 be applied to the derived limit for a Controlled Area. The scaling factor of 1/20 is only valid for exposure pathways where the DWL is limited by effective dose (inhalation, ingestion, and skin uptake). For the pathways where the DWL is limited by skin dose (external exposure and retention in skin), the relevant scaling factor is 1/10, which is the ratio of the public skin dose limit to the occupational skin dose limit.

Table A-3: Derived Working Limits for I-125 Contamination (Bq cm^{-2})

Surface	External Exposure	Inhalation	Ingestion	Skin Uptake	Retention in Skin
Personnel Skin Contamination	3.0 E 3	1.1 E 3	1.3 E 2	4.5 E 2	1.4 E 2
Personal Clothing	6.5 E 3	1.1 E 3	NA	4.5 E 2	1.4 E 2
Protective Clothing	1.3 E 4	1.7 E 3	NA	4.5 E 2	1.4 E 2
Controlled Area Surfaces	1.3 E 4	1.7 E 3	NA	4.5 E 2	1.4 E 2
Uncontrolled Area Surfaces (and equipment leaving a controlled area)	1.3 E 3	8.3 E 1	2.0 E 1	2.2 E 1	1.4 E 1

Appendix B: Monitoring Methods and Sensitivities

Direct and indirect monitoring for ^{125}I in MNR is performed with a variety of techniques and instrument combinations. The various methods used for each surface, and the result corresponding to the applicable MNR Contamination Limit, are summarized in Table B1.

Direct Check:

Direct checks of surfaces are performed with Bicron Surveyor (or equivalent) ratemeters with a thin 1" NaI detector and with a Berthold LB-122 (Xe) Gas Proportional contamination monitor. The expected net count rate is given by:

$$R\ (cpm) = [U]\ (Bq \cdot cm^{-2}) \bullet A_d\ (cm^2) \bullet \varepsilon\ (s^{-1} \cdot Bq^{-1}) \bullet \left[\frac{60\ s}{min}\right]$$

Where

R = The count rate in counts per minute

[U] = the activity per unit area on the surface in Bq cm^{-2}

A_d = the area of the detector in cm^2

ε = the efficiency of the detector

The parameters used are as follows:

Table B1: Parameters for Instruments Used in Direct Checks

Instrument	Detector Area (cm^2)	Detector Efficiency (%)
Bicron ratemeter with thin 1" NaI	5	10
Berthold LB-122 (Xe) Gas Proportional	200	1

The Berthold contamination monitor also has a user-selectable mode that provides direct readout in Bq/cm2 averaged over the 200 cm^2 area of the detector with pre-programmed efficiency.

Indirect (Swipe) Check:

Indirect, or swipe, checks are performed over a nominal 300 cm^2 area and are counted either using the Bicron ratemeter with thin 1" NaI detector or in the liquid scintillation counter. The expected count rate is given by:

$$R\ (cpm) = [U]\ (Bq \cdot cm^{-2}) \bullet A_s\ (cm^2) \bullet f \bullet \varepsilon\ (s^{-1} \cdot Bq^{-1}) \bullet \left[\frac{60\ s}{min}\right]$$

Where

R = The count rate in counts per minute

[U] = the activity per unit area on the surface in Bq cm^{-2}

A_s = the area of the swipe in cm^2

f = the fraction of activity on the surface transferred to the swipe (the swipe efficiency)

ε = the efficiency of the detector

When operated in scalar (count integration) mode, the expected number of counts is given by:

$$[C] = [U](Bq \cdot cm^{-2}) \bullet A_s\,(cm^2) \bullet f \bullet \varepsilon\,(s^{-1} \cdot Bq^{-1}) \bullet t \bullet \left[\frac{60\,s}{min}\right]$$

Where

$[C]$ = The expected number of counts

$[U]$ = the activity per unit area on the surface in $Bq\,cm^{-2}$

A_s = the area of the swipe in cm^2

f = the fraction of activity on the surface transferred to the swipe (the swipe efficiency)

ε = the efficiency of the detector

t = the counting time in minutes

The swipe efficiency has been taken as 10% for all swipes. Normal practice for swipes collected during and after tasks is to use a swipe dampened with sodium bisulphate reducing agent. This has been shown operationally to increase swipe efficiency. No credit is taken here for this known but unquantified increase. The weekly swipe checks performed by Health Physics and counted in the liquid scintillation counter are collected with dry swipes.

The efficiency of the NaI detector is taken as 10%, as above for Direct Checks, and the efficiency of the liquid scintillation counter is taken as 100%.

Results of the calculations and corresponding instructions to workers are provided in Table B2.

Table B2: Monitoring Results Corresponding to MNR I-125 Contamination Limits

Surface	MNR Contamination Limit (Bq / cm²)	Monitoring Method	Net Instrument Reading Corresponding to Limit	Comments / Actions
Personnel Skin Contamination	50	Direct Check with Bicron Ratemeter with 1" thin NaI	1500 cpm	Report skin contamination above this level to Health Physics immediately upon detection.
		Direct Check with Berthold LB-122 Contamination Meter	100 cps [cps mode] 50 Bq/□[2] [I-125 Bq/cm² mode]	Decontamination of any detectable contamination (even below this level) must be attempted by washing with mild soap and luke warm water. Inform Health Physics that the contamination has occurred. If decontamination is incomplete, report to Health Physics prior to departing.
Personal Clothing and Items (Papers, Books carried by hand)	50	Direct Check with Bicron Ratemeter with 1" thin NaI	1500 cpm	Report contamination above this level to Health Physics immediately upon detection.
		Direct Check with Berthold LB-122 Contamination Meter	80 cps [cps mode] 40 Bq/□ [I-125 Bq/cm² mode]	Item to be bagged for decay in storage or decontaminated. Health Physics must be consulted prior to removing any item with detectable contamination (even below this level).
Protective Clothing	50	Direct Check with Bicron Ratemeter with 1" thin NaI	1500 cpm	Decontaminate or dispose as appropriate.
		Direct Check with Berthold LB-122 Contamination Meter	100 cps [cps m 50 Bq/□ [I-125 Bq/cm² m	

[2] N.B. This is a representation of the display on the instrument screen.

Surface	MNR Contamination Limit (Bq / cm²)	Monitoring Method	Net Instrument Reading Corresponding to Limit	Comments / Actions
Normally Accessible Working Areas in MNR (Floors, Walls, Benches, Tools, Equipment, etceteras)	5	Swipe check counted with Bicron Ratemeter with 1" thin NaI	900 cpm	Decontaminate Area / Item. If immediate decontamination of an area is not possible, cordon off area and post as "Contamination Area". Notify Health Physics. If immediate decontamination of an item is not possible, place item in containment and notify Health Physics. Note: Any detectable contamination must be removed to the extent possible. This level is an upper limit for planning and interpreting monitoring.
		Swipe check counted on Liquid Scintillation Counter	9000 cpm	
Areas Posted as Contamination Area	50	Swipe check counted with Bicron Ratemeter with 1" thin NaI	9000 cpm	Decontaminate Area / Item immediately. If immediate decontamination of an area is not possible, cordon off area and post as "High Contamination Area". Notify Health Physics. If immediate decontamination of an item is not possible, place item in containment and notify Health Physics. Note: Any detectable contamination must be removed to the extent possible. This level is an upper limit for planning and interpreting monitoring.
		Swipe check counted on Liquid Scintillation Counter	90000 cpm	

Surface	MNR Contamination Limit (Bq / cm²)	Monitoring Method	Net Instrument Reading Corresponding to Limit	Comments / Actions
Equipment released from MNR	5	Direct Check with Bicron Ratemeter with 1" thin NaI	150 cpm	Decontaminate Item. Health Physics must be consulted prior to removing any item with detectable contamination (even below this level).
		Swipe check counted with Bicron Ratemeter with 1" thin NaI	900 cpm	
		Swipe check counted on Liquid Scintillation Counter	9000 cpm	
Outer Surface of "Excepted" Radioactive Shipments	0.4	Swipe check counted on Liquid Scintillation Counter	720 cpm	Decontaminate or repackage item. Health Physics must be consulted prior to removing any item with detectable contamination (even below this level).
		Swipe check counted with Bicron Ratemeter with 1" thin NaI	72 cpm	
Outer Surface of Other Than "Excepted" Radioactive Shipments	4	Swipe check counted on Liquid Scintillation Counter	7200 cpm	Decontaminate or repackage item. Health Physics must be consulted prior to removing any item with detectable contamination (even below this level).
		Swipe check counted with Bicron Ratemeter with 1" thin NaI	720 cpm	

Appendix C: Minimum Detectable Activities

The Critical Level, Detection Limit and MDA for paired observations are given by [5]:

$$L_c = 2.33\,\sigma_B$$

$$L_D = (2.71 + 4.65\,\sigma_B)$$

$$MDA = L_D / \varepsilon$$

Where

L_c = the critical level at the 95% confidence level

L_D = the lower limit of detection at the 95% confidence level

MDA = the Minimum Detectable Activity corresponding to the L_D

ε = the efficiency of the detector

Table C1: Minimum Detectable Activities for I-125

Method[1]	Typical Background	Critical Level	MDA (Bq)	MDA (Bq/cm^2)
Direct Check with a Bicron Ratemeter with 1" NaI detector operated in scalar mode for 1 minute count.	200 counts in one minute	234 gross counts in one minute	11.4	2.3
Swipe Check counted with a Bicron Ratemeter with 1" NaI detector operated in scalar mode for 1 minute count and 300 cm^2 swipe area.	200 counts in one minute	234 gross counts in one minute	11.4	0.4
Swipe check counted for two minutes in the Liquid Scintillation Counter	50 counts per minute (100 counts in two minutes)	62 gross counts per minute (124 counts in two minutes)	0.4	0.01

[1] Efficiencies, swipe efficiencies and swipe areas are as specified in Appendix B.

The Bicron ratemeter/NaI detector is a portable instrument. One such instrument is kept in a low background area for exit monitoring and low background swipe counting. Other instruments of this type are frequently used in work areas to monitor contamination during and after tasks. The effect of background on detectable contamination levels has been assessed for both scalar mode and ratemeter mode and is shown in Figures C1 and C2 respectively. In assessing the impact of background on detectable activity in the ratemeter mode, it has been assumed that a trained user can detect an increase of 33% over the background level. For example, while monitoring in a background of 10 000 cpm, the detectable activity is that which would lead to a gross count rate of 13 333 cpm. This requires a steady background, which is consistent with the manner in which the unit is operated in these circumstances: the instrument is left in a fixed location and objects, hands, smears etceteras are brought near to the face of the detector.

The performance of the instrument in operational monitoring, as shown in Figures C1 and C2, is highly satisfactory. The maximum acceptable background for detection of 5 Bq/cm^2 and 50 Bq/cm^2 in each mode is shown in Table C2.

Table C2: Maximum Background for Detection of Contamination Using Bicron with Thin NaI Detector

Mode	Limit	Maximum Background
Ratemeter Mode - Direct Check	5	450
	50	4 500
Ratemeter Mode – Swipe Check	5	2 700
	50	27 000
Scalar Mode – Direct Check	5	1 000
	50	100 000
Scalar Mode – Swipe Check	5	36 000
	50	3 600 000

Figure C1: Minimum Detectable Activity for 1 Minute Counts on Bicron Ratemeter with thin NaI vs Background Count Rate

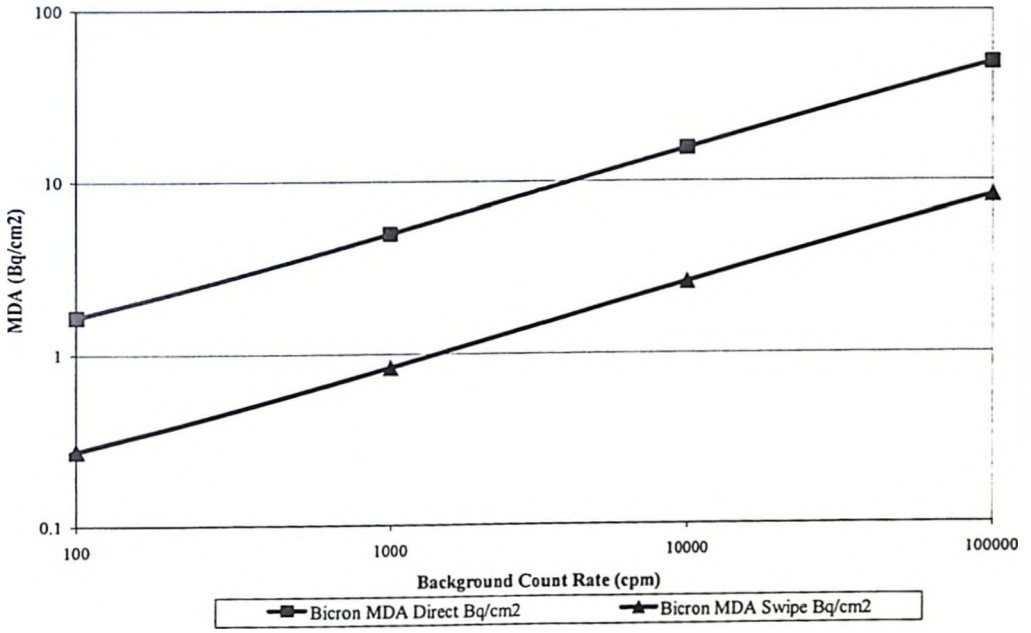
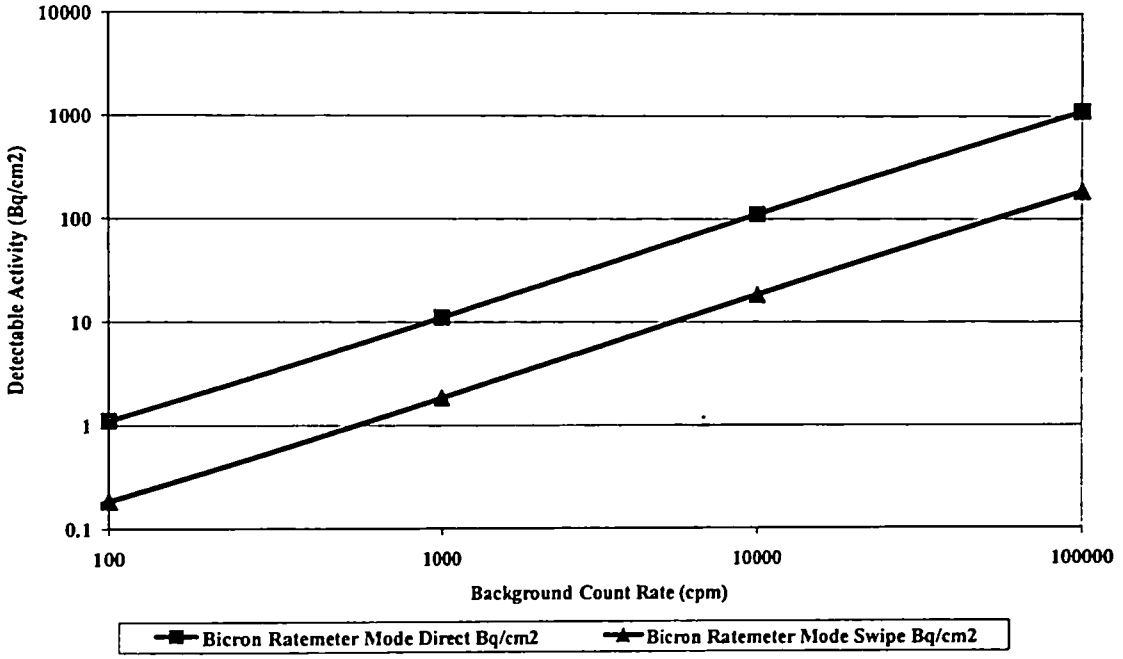


Figure C2: Detectable Activity vs Background for Bicron with NaI in Ratemeter Mode
(Based on Ability to Recognize 33% Change from Background)



Chapter 3: An ALARA Review of an ^{125}I Production Process

Introduction

The International Commission on Radiological Protection (ICRP) is the organization which provides definitive guidance on radiation safety practices. Canadian regulations are based primarily upon ICRP recommendations, as are those of most nations.

The ICRP Framework for Radiological Protection includes three fundamental components: Justification, Optimization and Dose Limits[1]. Justification is the first necessary step in approval of a practice and entails verifying that the proposed practice will result in more good than harm to society. For practices which are justified, it is necessary to optimize the net benefit of the practice. This is done by ensuring that the radiological exposures associated with the practice are maintained As Low As Reasonably Achievable, social and economic factors being taken into consideration (ALARA). Finally, regardless of the outcome of the optimization process, the practice must be designed and conducted such that doses to individuals (workers and members of the public) are maintained below corresponding dose limits. Dose limitation is intended to ensure that no individual or group of individuals is disproportionately disadvantaged by a practice.

The importance of optimization of practices, as embodied in the ALARA principle, is a central dogma of occupational radiation safety. The ALARA principle arises from the adoption of the linear-no-threshold model of dose response for stochastic effects in making protection decisions. This model postulates that every increment of exposure carries with it a proportional increment in risk for stochastic effects of radiation. Thus, doses must be maintained at the lowest levels they can be to avoid needless detriment. However, the ALARA principle incorporates the concept of "reasonableness" in establishing the degree to which doses must be reduced. The two main considerations generally applied in identifying what is reasonable are the absolute magnitude of investment required to achieve a given safety advantage (often in the context of the monetary investments made in society for other similar safety gains) and the concept of diminishing returns whereby increasingly greater investments are typically required to achieve increasingly smaller gains.

The report attached to this chapter documents an ALARA review that was conducted of ^{125}I operation at MNR.

Discussion

^{125}I production practices were developed and carried out in prototype scale at MNR throughout the 1990s. As part of the preparation for a transition to production scale activities, dose estimates were made [2]. The predicted doses were very low, consistent with those experienced during prototype operations. As commercial production was initiated, doses accumulated by workers were compared with the projections and were found to be startlingly higher than expected. A summary of the predicted and actual doses is shown in Table 3-1, taken from the attached report.

Table 3-1: Predicted [2] and Actual I-125 Production Related Doses

		Predicted		Actual			
		Calculated	Estimated from trials	1996	1997	1998 Q3	2002
A	Iodine Personnel Dose per Unit Activity	0.62 µSv	0.82 µSv				
B	Iodine Personnel Dose per Year	0.62 mSv	0.82 mSv	4.9 mSv	15.5 mSv	NA	16.8 mSv
C	Iodine Production Dose per Unit Activity	0.62 µSv	0.82 µSv	12 µSv	20 µSv	43 µSv	7 µSv

Notes: The "calculated" and "estimated from trials" values in Row A and B are from the original report.[2]
The calculated and "estimated from trials" values in Row C are inferred from the report. [2]
The calculated doses are based on calculation and do not include contribution from the elevated background in MNR [2]
The "estimated from trials" doses are based on extrapolation of the results of two test irradiations. [2]
The actual doses are based on TLD results for production personnel in the dosimetry periods indicated.
The Iodine Production Dose is the collective dose to staff that is attributable to production.

As shown in the Table, the doses were approximately an order of magnitude (or more) higher than anticipated. The need for immediate action to identify and mitigate the source of these higher than expected exposures was clear. The process and findings are described in detail in the attached report and are summarized briefly below.

A team of operations and radiation safety personnel were deployed to address the problem. The significance of the problem was clearly identified. A commitment by facility and radiation safety management to rectify the situation was broadly communicated. Immediate actions included deploying electronic personal dosimeters to

production personnel and implementing a dose accounting program. Production staff were provided with additional training on dose awareness and radiation survey techniques and requirements for additional surveys were incorporated into their procedures. Both of these actions provided immediate protection enhancements for production staff while also increasing the process monitoring data available and thus the likelihood that abnormal situations would be identified. The badge change frequency was also increased to provide timelier feedback on the impact of any changes.

Findings occurred generally in two phases. In the first phase, several sources of high dose rates were identified and mitigated. Most of these sources were production components that were not providing adequate shielding for ^{125}I such as thin walled bellows tubes and thermocouple gauges on the vacuum systems. ^{125}I mobility in the system was found to be higher than expected and large quantities of the product were accumulating on the walls of these components and/or were passing through them, resulting in high chronic and transient dose rates. These sources were identified and eliminated within the first few months of the assessment.

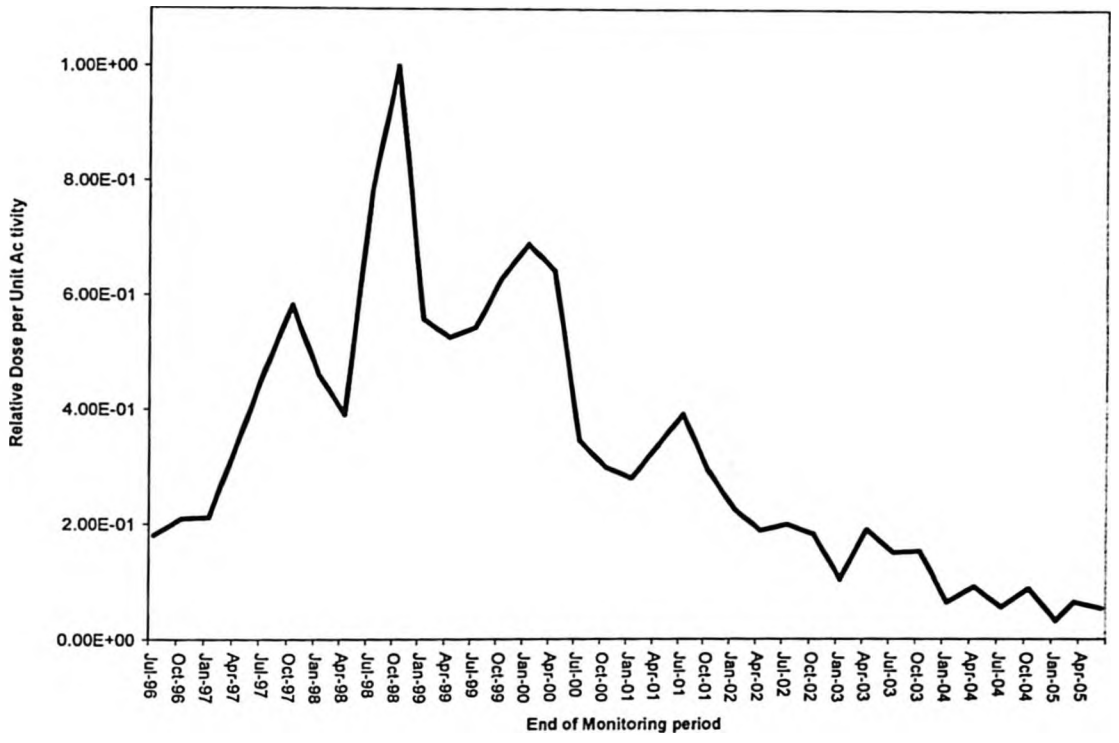
The benefit of issuing significantly exposed personnel with electronic alarming dosimeters was immediately evident. Provided with immediate, real-time feedback on dose accumulation, personnel quickly and independently identified sources of exposures and adjusted work practices (to the degree they were able) to mitigate them. This echoes the experience of the author in similar dose reduction exercises at other facilities. In the absence of any other action, issuing significantly exposed personnel with electronic personnel dosimeters can be expected to result in a significant reduction in dose in most instances.

A longer term dose reduction phase followed these early gains. The longer term reductions resulted mainly from controlling the amount of ^{126}I mobilized from the irradiation chamber of the production rigs. The higher than expected iodine mobility in the production apparatus was found to result from the inadvertent introduction of trace quantities of moisture. Because no ^{126}I recovery had been anticipated, the McMaster system included only minimal shielding and the dose consequences of even trace quantities outside the production chamber were significant. It should be noted that the motivation for the longer term phase of dose reduction activities may well have diminished had there not been a clear set of targets and exposure tracking against which to track performance.

It was determined that the most meaningful figure against which to track the effectiveness of dose reduction strategies was the collective dose per unit production. This was particularly true during the early phase of commercial production when both workforce size and production rates were changing rapidly. Figure 3-1 (an updated

version of a figure from the attached report) shows the collective dose per unit production. The short and longer term phases of dose reduction can be observed.

Figure 3-1: Relative Quarterly Collective Dose per Unit Activity of I-125 Produced



Summary

The situation described in the attached report clearly demonstrates the importance of establishing dose projections and budgets prior to initiating a new large scale project. The need for enhanced radiological surveillance through the commissioning phase of any new activity is also shown.

The potential sensitivity of doses in an ^{125}I production process to the level of radioactive impurities in the system is very high. Careful monitoring of the product impurities should be considered an important component of the facility radiation safety program.

Attached Report

The McMaster University Health Physics Department report HP-MNR-03-04, ALARA Review of I-125 Production (2003) is attached to this chapter.

The report was prepared solely by the author, using the references cited in the report as guidance. This report was prepared for submission to McMaster oversight committees for MNR and to the federal regulator. The report describes the work of a team of MNR and Health Physics personnel. The work was conducted under the general leadership of the author as the radiation safety manager for the facility. The planning and data analysis of the dose monitoring and dose accounting was performed by the author. The technical solutions described in the report were designed and implemented by a team of specialists and not primarily by the author.

Minor editorial changes have been made to the report for inclusion in this thesis, and proprietary information has been removed.

References

[1] International Commission on Radiological Protection, The 1990 Recommendations of the International Commission on Radiological Protection, ICRP 60, Annals of the ICRP Vol. 21 No. 1-3 (1991).

[2] McMaster Nuclear Reactor, ALARA Considerations for I-125 Production, IALARA7, Revision 7, (1993).

McMaster University **Health Physics**

Health Physics Report

HP-MNR-03-04 ALARA Review of I-125 Production

2003 December 19, Revision 0[1]

Prepared by: D. M. Tucker
Senior Health Physicist

DOCUMENT SUMMARY

This document describes the activities and results of the ALARA Review of I-125 production related doses carried out, primarily, during the period 1999 January to 2001 July. Dose trends and current performance are discussed.

Date of ROCC Approval:	2003 December 19
Date of HPAC Approval	2003 December 16

[1] With slight editorial revisions and removal of proprietary information – 2005 November.

Health Physics Report

HP-MNR-03-04 ALARA Review of I-125 Production

1. Reason for Report

This report has been prepared to document the activities and outcomes associate with an ALARA review of doses resulting from I-125 production. The review occurred, primarily, during the period 1999 January to 2001 July.

A commitment to provide such a report to the CNSC (then AECB) was made in 1999 January [1].

2. Background

I-125 production capabilities at MNR were developed from prototype to full scale commercial production throughout the 1990's. By mid 1999, staffing and training of a dedicated I-125 production group was well underway.

There are three significant radionuclides which have the potential to contribute to external exposures. They are I-125, Xe-125 and I-126. The production processes and associated equipment have been described in detail elsewhere, including previous licensing submissions.

- Xe-125 is routinely produced in TBq (multi-Ci) quantities by activation of Xe-124 as the first step of the production process. Although it has energetic photons (188 and 243 keV) it is kept under water shielding and no significant exposures associated with Xe-125 are expected in routine operation.

- I-125 is also produced in TBq (multi-Ci) quantities and is handled in close proximity to the production staff at several stages of the process. However, because it emits only low-energy photons (35.5 keV) it is easily shielded. Generally the thin stainless steel walls of the gas systems provide more than adequate shielding.

- I-126 is produced as an impurity in the production process. It is routinely produced in the irradiation chamber by activation of I-125 formed there. However, it is expected to remain fixed to the walls of the irradiation chamber under adequate water shielding. Because of its higher energy photons (388.6 and 666.3 keV) it is not easily shielded. Because it moves in the process exactly as I-125 does, its presence in the system outside of the irradiation chamber introduces several opportunities for significant contributions to external exposures.

Several safety analyses of the production processes and studies of anticipated and actual doses were completed, primarily in support of licence applications submitted to the (then) AECB. Predicted doses were low; primarily due to the relative ease with which I-125 photons are shielded and the configuration of the process such that the decay phase is carried on under water shielding. The predicted doses when large scale production was initiated are shown in Table 1. [2]

Initial production doses were higher than the predictions by approximately a factor of ten. In addition, there was a gradual increase in collective dose attributable to production that was evident through 1996 and 1997.

In 1999 January, dose results from the dosimetry quarter ending 1998 October 14 were received. The results revealed a significant increasing trend in the doses to the production staff. The average and

collective dose of the group, and the collective dose per unit production had all increased sharply. Figures 1, 2 and 3 and Table 1 all show the trend that was observed.

3. Actions Taken

An investigation and review of whole body exposures associated with I-125 production was undertaken immediately. The following steps were taken or initiated immediately:

- Dose history for I-125 production personnel was documented, communicated and reviewed.
- Personnel were issued with electronic personal dosimeters and were trained in their use.
- A dose accounting process based on recording doses for each task using electronic personal dosimeters was initiated.
- Additional routine and task based radiation surveys of the production areas and tasks were initiated.
- Additional dose awareness and survey technique training was provided to production personnel.
- The badge change frequency for I-125 personnel was changed to monthly (from quarterly).

Table 1: Predicted [2] and Actual I-125 Production Related Doses

		Predicted		Actual			
		Calculated	Estimated from trials	1996	1997	1998 Q3	2002
A	Iodine Personnel Dose per 10 Ci Batch	6.2 µSv	8.2 µSv				
B	Iodine Personnel Dose per Year	0.62 mSv	0.82 mSv	4.9 mSv	15.5 mSv	NA	16.8 mSv
C	Iodine Production Dose per Unit Activity	0.62 µSv	0.82 µSv	12 µSv	20 µSv	43 µSv	7 µSv

Notes: The "calculated" and "estimated from trials" values in Row A and B are from the original report.[2]
The calculated and "estimated from trials" values in Row C are inferred from the report. [2]
The calculated doses are based on calculation and do not include contribution from the elevated background in MNR [2]
The "estimated from trials" doses are based on extrapolation of the results of two test irradiations. [2]
The actual doses are based on TLD results for production personnel in the dosimetry periods indicated.
"Iodine Production Dose" is the dose to workers attributable to production activities.

4. Findings and Results

Generally, the findings were identified and the results were achieved in two phases. In the initial phase, sources of relatively high radiation fields were identified and mitigated. These were identified fairly quickly, mainly in the first two to three months of the assessment, mainly through surveys and the use of electronic personal dosimeters. This reversed the upward trend in doses but did not return the exposures to their previous low levels. A longer term dose accounting process and assessments of exposure sources followed the initial phase. This phase focused primarily on minimizing the quantity of I-126 present in the system outside of the irradiation chamber which was leading to generally increased radiation fields in all of the work area. The two phases are described further below and a summary of findings and corrective actions is provided in Appendix A. The two phases were not completely separate. For example, as the dose accounting identified dose intensive tasks, further surveys were conducted to identify radiation sources for mitigation.

Surveying and Dose Awareness

Additional surveys and dose awareness, in part attributable to the routine use of electronic personal dosimeters, quickly identified some significant contributors to doses. In each case, the item was associated with a bellows tube in the vacuum system or similar thin walled component (such as thermo-couple gauges). The thin walls provided inadequate shielding for the TBq (multi-Ci) quantities of I-125 contained in the systems.

One example was the shielding of a bellows tube in the Fish Tank – an apparatus used for drying decay chambers after recovery. This apparatus was well designed for containment but contained two significant weak spots with respect to shielding – a thermocouple gauge and a bellows tube. As production activities increased, a larger than anticipated quantity of I-125 accumulated in the system and radiation fields became high. Some shielding for the bellows tube was provided but it was moveable and difficult to position due to the need to access a valve located behind the shielding. Initially, this problem was identified in reviewing the daily electronic personal dosimeter results – an exposure of 0.6 mSv (60 mrem) occurred during one operation during the initial days of the assessment. Additional shielding was provided, personnel were briefed on the hazard, and a practice of routinely surveying the equipment was initiated. An administrative control was established and posted whereby Health Physics was to be alerted of radiation fields exceeding 0.1 mSv/h (10 mrem/h). These steps effectively eliminated the Fish Tank as a significant source of routine exposure. For the remainder of 1999, this work accounted for only 6% of dose to personnel, based on electronic personal dosimeter results – dropping to 4% by 2001.

Dose Accounting and I-126 reduction

Longer term reduction in doses was achieved primarily through minimization of the quantity of I-126 present in the system outside of the irradiation chamber. The production apparatus was not designed to accommodate significant quantities of I-126 – there is virtually no shielding for the higher energy photons in most of the system. The apparent strong influence of I-126 production on production related doses is indicated in Figure 4.

As mentioned previously, an intensive dose accounting program was initiated to identify the major sources of exposure so that attention could be appropriately focused. Personnel were issued with electronic personal dosimeters and trained in their use. They were required to complete a task list and record the dose reading before and after on the sheets shown in Appendix B. In addition, the use of the electronic personal dosimeters allowed for more timely exposure management to ensure University and Regulatory limits were not exceeded.

It should be noted that part of the dose reduction is likely to have resulted directly from issuing the electronic personal dosimeters and requiring personnel to keep detailed journals of dose. Having been provided with instant feedback, personnel made adjustments in their own routines that eliminated unnecessary exposures.

The dose accounting led to a clearer picture of doses attributable to each "job" (task) in the production process. Figures 5 and 6 show the breakdown of dose by task in 1999 and 2001 respectively. In the initial period of review, in 1999, it became clear that most of the ongoing exposures were associated with Interchange. Several sources of radiation were identified and eliminated or shielded early in the process; however, radiation fields around the Gas Handling Station and the docked Production Rigs remained generally high due to the presence of I-126.

Continued efforts through 1999 and 2000 led to ongoing reductions in the amount of I-126 present in the system. Several attempts were made to rectify the problem with varying results. It was generally understood that the I-126 was being allowed to move within the system due to the introduction of trace quantities of moisture to the irradiation chambers. Production processes were designed with this problem in mind. Eliminating these trace quantities of moisture proved to be fairly difficult. Different designs of moisture traps were deployed but these proved generally ineffective. Elimination of moisture loading in pump oil of the fish tank system by routine gas ballasting assisted in removal of moisture from the system. Changes in the procedures for decontamination of open fittings eliminated one significant source of moisture introduction. Some Gas Handling Station (GHS) activities were discontinued and the practice of actively pumping on the GHS traps between interchanges was reinstated. By 2001 May, a two-stage end of irradiation cryopump procedure was implemented for all of the production Rigs. Several procedure refinements were made to reduce moisture introduction potential and staff were trained and re-trained on the importance of eliminating moisture and techniques for doing so. By January of 2002 the problem was largely solved.

The general decrease in dose by job over the review period is shown in Figure 7 and the results in terms of overall dose per day are summarized in Figure 8.

Impact of Training

One additional consideration in explaining the initial spike in doses and the subsequent long term reduction is the number of production personnel that were in training beginning in 1999. The group had more than doubled in size with four new staff members. This had the effect of "doubling up" on some doses as tasks were observed closely by a trainee while performed by a qualified member of the staff. In addition, the production process requires highly skilled personnel. The initial staff had been involved since the prototype stage. They had developed a thorough understanding of the processes and procedures and were highly successful in carrying them out quickly with minimal contamination releases and minimal introduction of moisture to the systems. These skills took some time to transfer to the newer personnel.

5. Current Status

I-125 is now in full scale commercial production with a dedicated production staff. As shown in Figures 1 through 4, average doses and collective doses per unit activity are all currently as low as they were in the prototype stages of production in 1996.

The annual collective dose (and production) is shown in Figure 9. The collective dose attributable to I-125 production in 2002 (the last year for which complete results are available) was 16.8 person-mSv with a dose per unit activity production of 7 person-μSv. The maximum

dose to an individual was 3.9 mSv. These doses are considered to be as low as reasonably achievable, social and economic factors being taken into consideration (ALARA).

Doses are closely monitored through the use of electronic personal dosimeters and thermoluminescent dosimeters. Dose updates are provided to facility management and production personnel on a quarterly basis.

6. Conclusions

[1] An upward trend in doses associated with I-125 production was identified and responded to appropriately by facility personnel and management.

[2] Dose reduction activities have continued to the point where doses are considered to be ALARA.

[3] Continuing close monitoring of doses associated with I-125 production continues to ensure that any un-optimized doses that do occur are identified and mitigated.

7. List of Figures

[1] Iodine Production and Collective Dose by Quarter

[2] Iodine Production Group – Collective Dose Unit Activity Producted by Quarter

[3] Iodine Production Group – Average Dose by Quarter

[4] Iodine Production Doses and I-126 Production

[5] Dose Distribution by Job -1999

[6] Dose Distribution by Job 2001

[7] Annual Dose by Job 1999 to 2001

[8] Collective Dose by Day

[9] Annual Collective Dose and Production

8. List of APPENDICES

A Summary of identified sources of exposure and corrective actions

B Dose accounting sheet

9. References

[1] Dave Tucker to J. Kavanagh (CNSC), ALARA Review of I-125 Doses, 1999 January 21.

[2] IALARA7, ALARA Considerations For I-125 Production Revision 7, McMaster Nuclear Reactor, 1993 August.

10. Figures

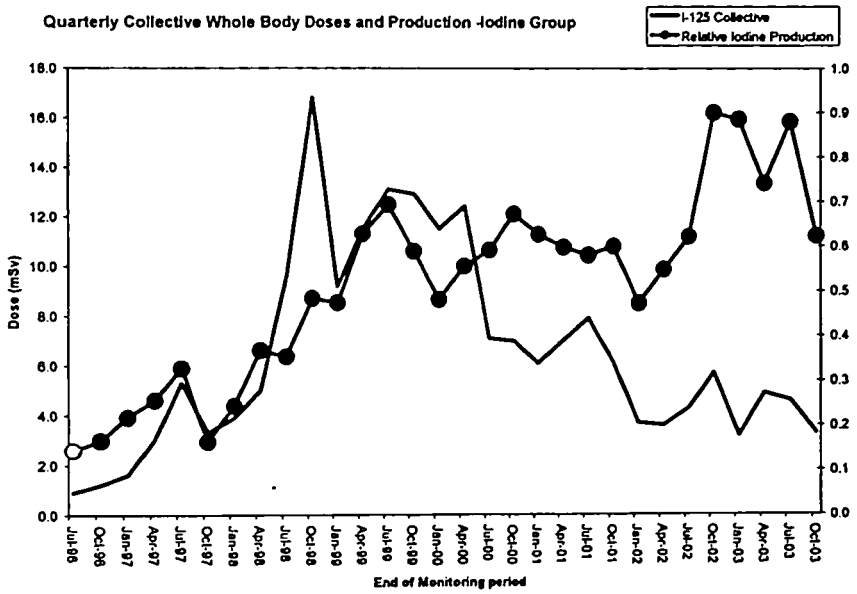


Figure 1: Iodine Production and Collective Dose by Quarter

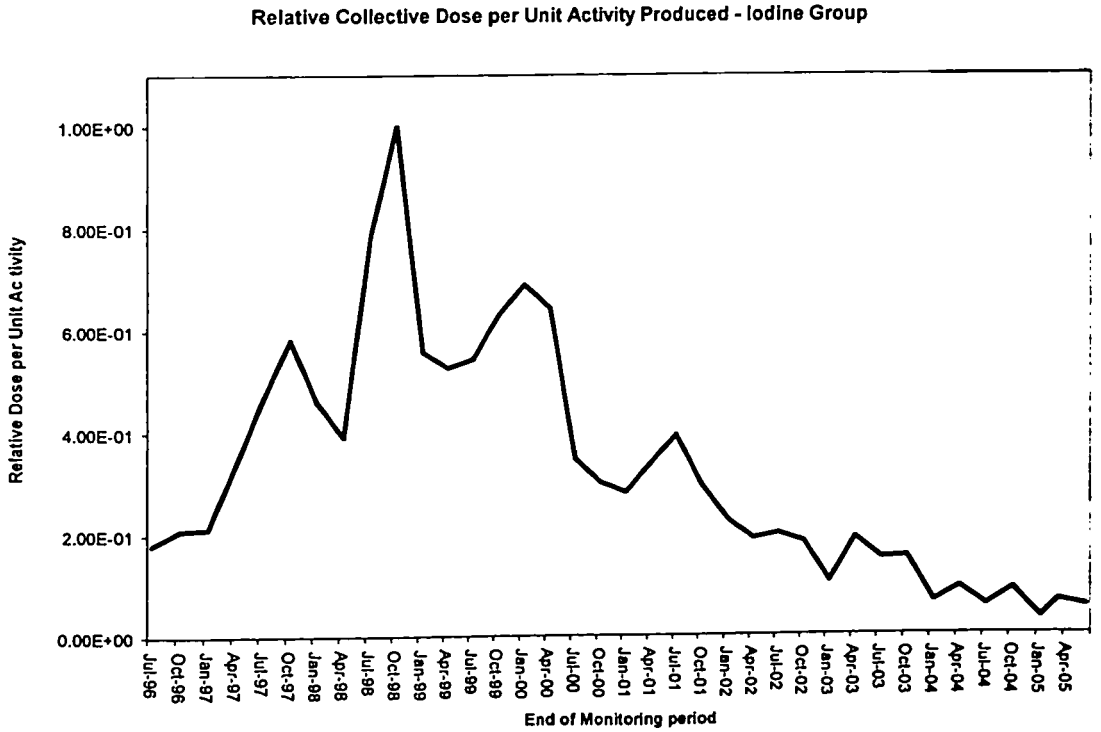


Figure 2: Iodine Production Team – Collective Dose per Unit Activity Produced by Quarter

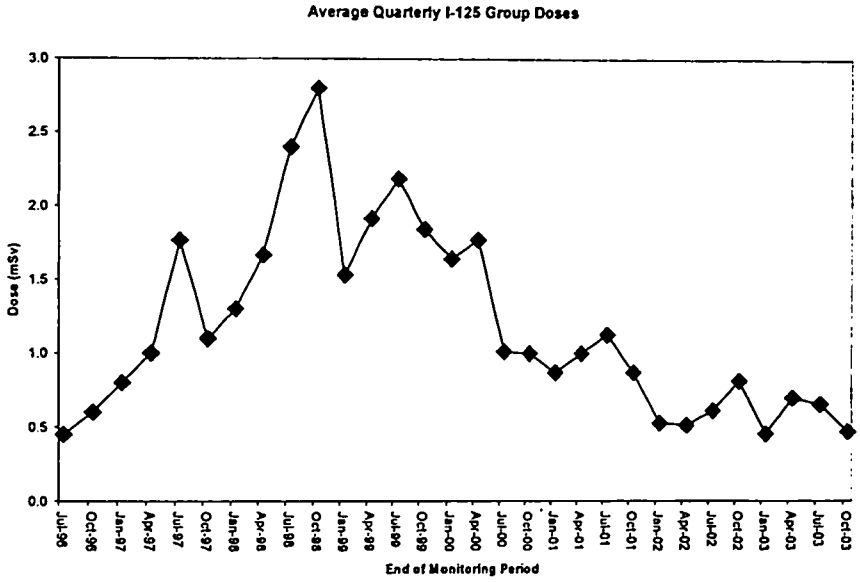


Figure 3: Iodine Production Team – Average Dose by Quarter

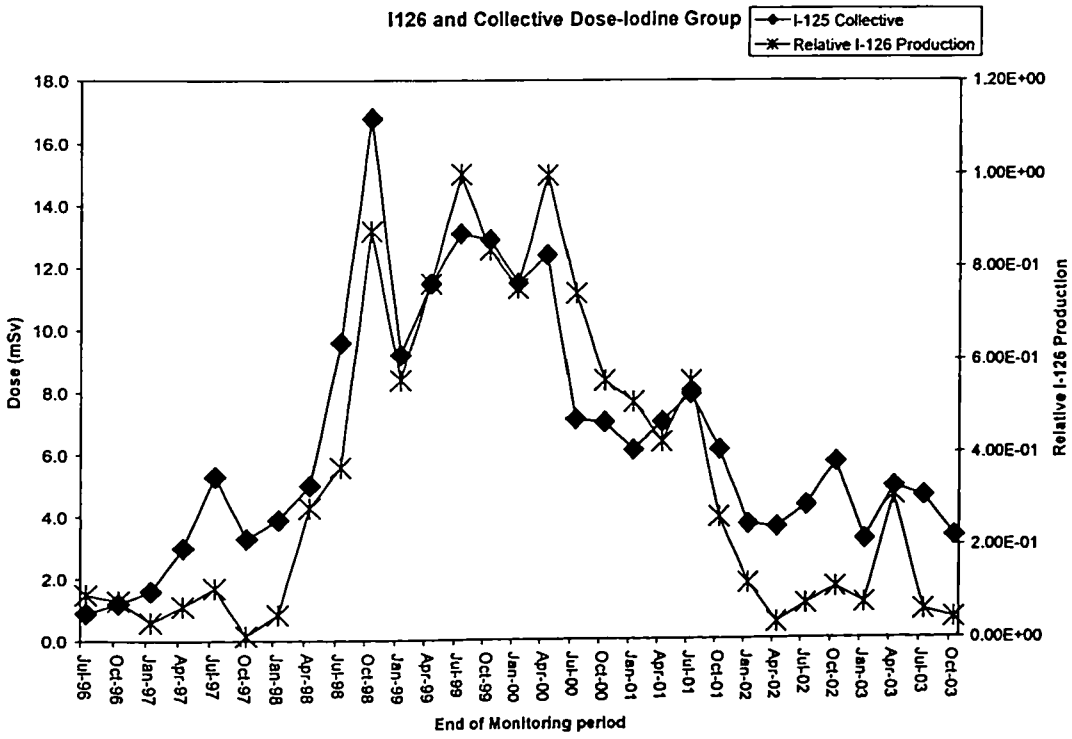


Figure 4: Iodine Production Doses and I-126 Production

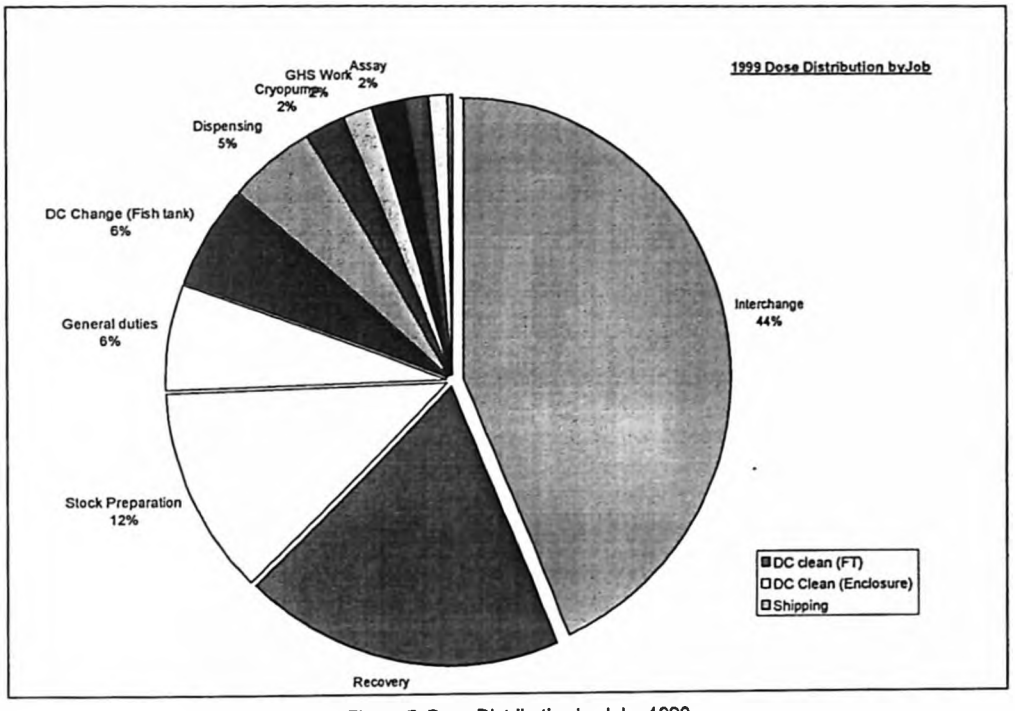


Figure 5: Dose Distribution by Job - 1999

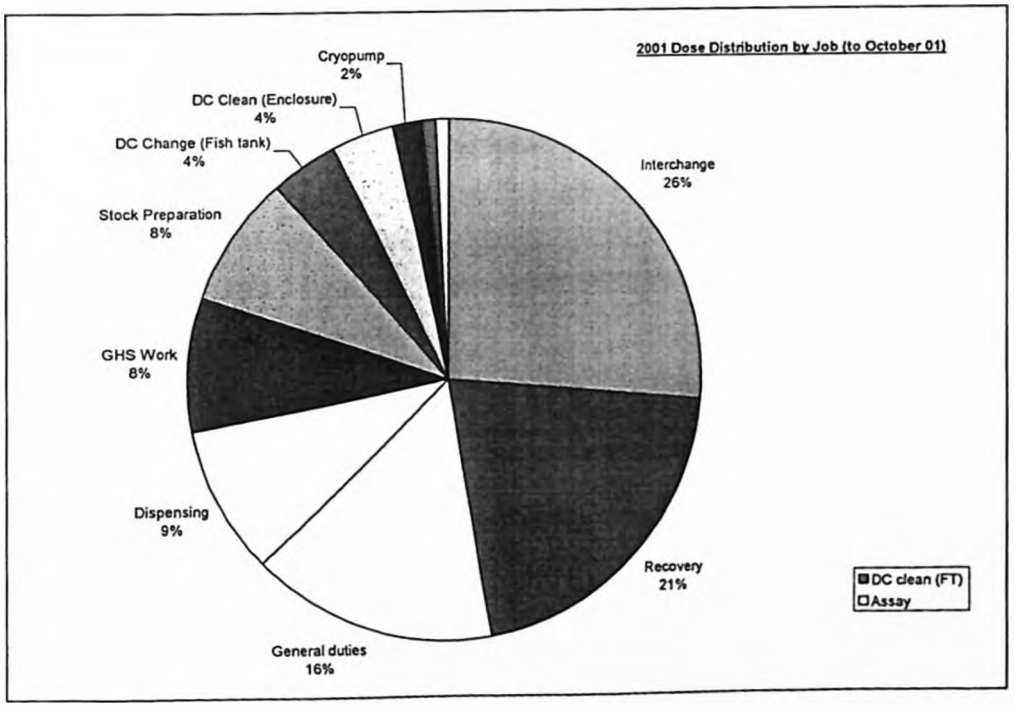
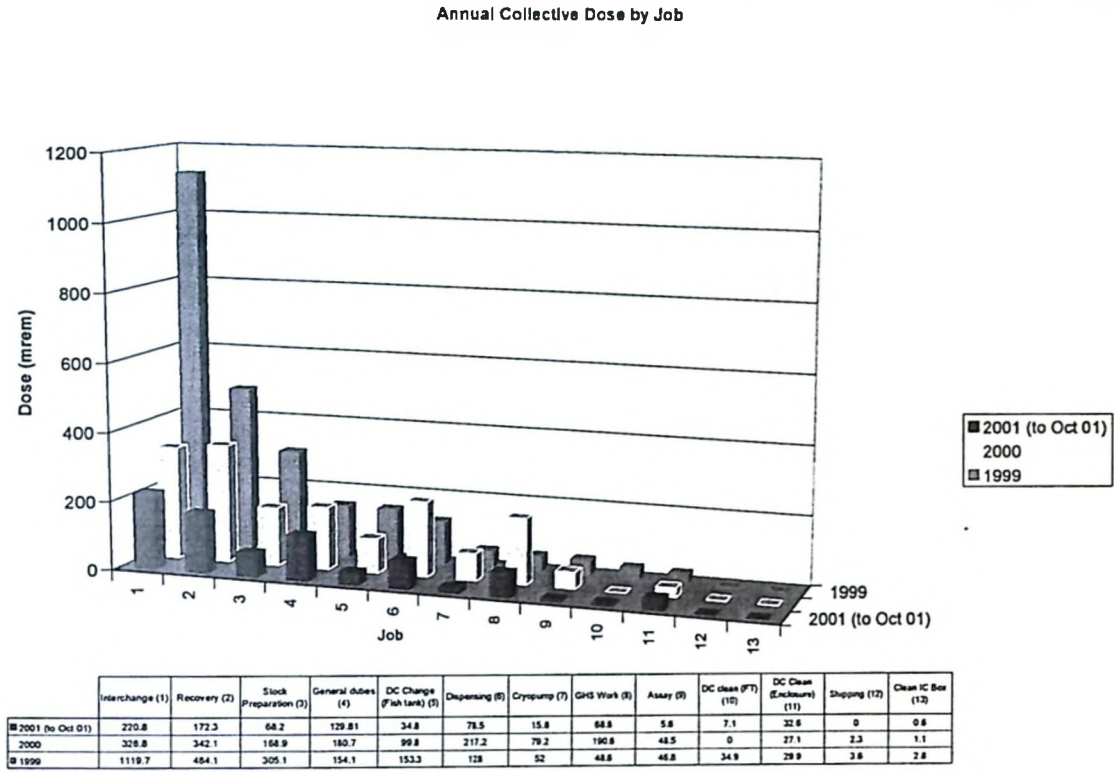


Figure 6: Dose Distribution by Job 2001



	Interchange (1)	Recovery (2)	Stock Preparation (3)	General duties (4)	DC Change (Fish tank) (5)	Dispensing (6)	Cryopump (7)	GHS Work (8)	Assay (9)	DC clean (FT) (10)	DC Clean (Enclosure) (11)	Shipping (12)	Clean IC Box (13)
2001 (to Oct 01)	220.8	172.3	68.2	129.81	34.8	78.5	15.8	68.8	5.8	7.1	32.6	0	0.6
2000	326.8	342.1	168.9	180.7	99.8	217.2	79.2	190.6	48.5	0	27.1	2.3	1.1
1999	1119.7	484.1	305.1	154.1	153.3	128	52	48.8	46.8	34.9	29.9	3.6	2.8

Figure 7: Annual Dose by Job

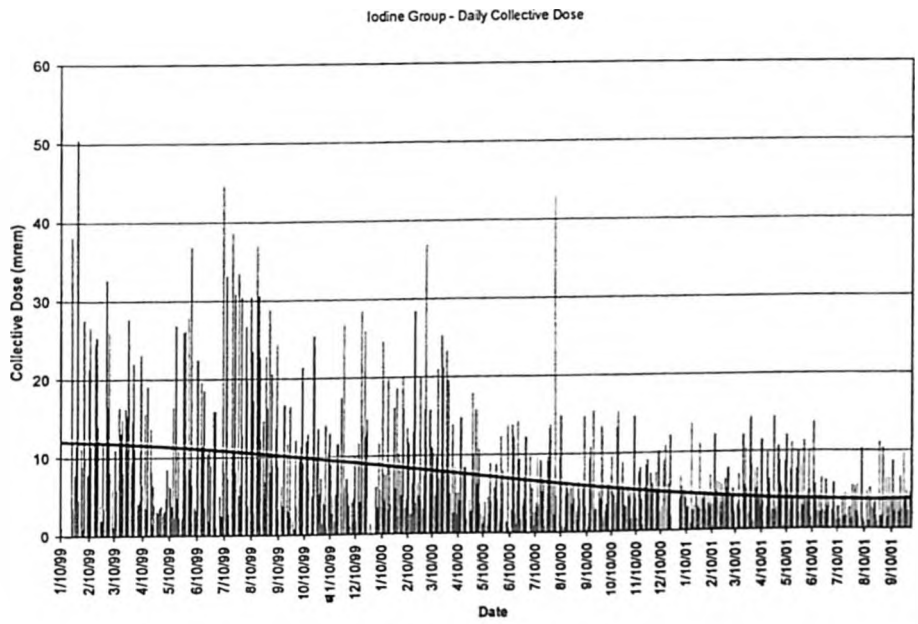
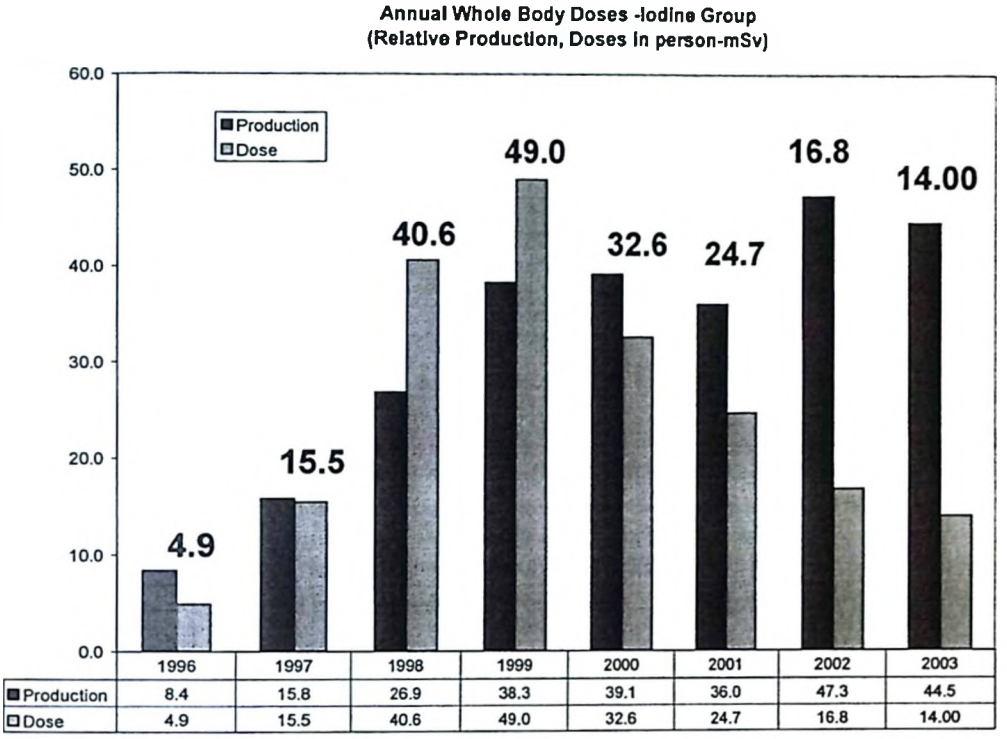


Figure 8: Collective Dose by Day



	1996	1997	1998	1999	2000	2001	2002	2003
Production	8.4	15.8	26.9	38.3	39.1	36.0	47.3	44.5
Dose	4.9	15.5	40.6	49.0	32.6	24.7	16.8	14.00

Figure 9: Annual Collective Dose and Production

Appendix A – Summary of Identified Sources of Exposure and Corrective Actions

Finding	Notes	Corrections
Inadequate shielding of bellows tube and thermocouple gauges in the "Fish Tank"	The fish tank is used to remove moisture from decay chambers following product recovery. Design focused on contamination control primarily – shielding was provided but Ci level quantities were not anticipated at this stage of the production process. One exposure of 0.63 mSv was identified after the review was initiated. Corrections were put in place immediately afterwards. This was likely to have been a significant contributor to the increase in doses.	• Additional fixed and moveable shielding for components was installed. • Pre and post-use surveys were initiated. An administrative control was initiated whereby Health Physics had to be notified of any radiation fields exceeding 0.1 mSv/h (10 mrem/h).
Inadequate shielding of bellows tube on Rig 5	The design of one of the production rigs still in use differed from that of the others with the inclusion of a bellows tube near the Decay Chamber. The bellows tube was unshielded and led to high transient radiation fields (several mSv/h) at the operator's position during gas movement.	• Shielding was added to the Rig.
Inadequate shielding of bellows tube on Gas Handling Station	Non-transient radiation fields up to 1 mSv/h (100 mrem/h) at 30 cm and 0.2 mSv/h (20 mrem/h) at the operator's position were encountered.	• The length of the bellows tube was minimized. • Shielding was added.
High radiation fields associated with bellows tube on transfer line	Non-transient radiation fields of 4 mSv/h (400 mrem/h) at 30 cm were detected on the bellows tube at the end of the transfer line connecting the gas handling station to the decay chambers in the rigs.	• The bellows tube was replaced and a practice of monitoring the accumulation of activity on this component was initiated. • Shielding was added to the containment vessel on the Gas Handling Station where this component is stored between uses.

Finding	Notes	Corrections
General problem of inadequate shielding attributed to bellows tubes and thin walled components	Design of new and improved components for the production process continues.	• Use of bellows tubes and thin walled components minimized • Shielding is incorporated into the design as needed when use of components cannot be avoided. • Commissioning surveys of new equipment performed to verify design • Use of electronic personal dosimeters assists in ensuring that unanticipated transient fields will be detected in the future.
General problem of higher than expected doses from I-126 production	I-126 impurities in the product were higher than anticipated leading to a general increases in exposures attributable to the work.	• Production processes were reviewed and improved over several months to minimize the amount of moisture to the systems. • Training provided to production personnel on techniques for, and importance of, minimizing introduction of moisture.

Appendix B: Dose Accounting Sheet

Name:			Job Performed												Operator			
Date	Start Dose	End Dose	Interchange	GHS Work	DC Change (Fish tank)	Recovery	Stock Preparation	DC clean (FT)	Cryopump	Dispensing	DC Clean (Enclosure)	Clean IC Box	Assay	General duties	Primary	Secondary	Observer	Notes

Chapter 4: Derived Release Limits for ^{125}I

Introduction

In Chapter 1, the radiation safety program components relevant to public radiation control were discussed. Establishing limits for releases to the environment was described as a crucial component of the control of public doses. This chapter describes in detail the derivation of one such set of limits – the limits on releases of ^{125}I contamination in air effluents. The release limits for ^{41}Ar are also derived in the report but will not be discussed here.

Discussion

In Canada, the derivation of release limits for radioactive effluents from nuclear facilities is governed by a national standard: CAN/CSA-N288.1-M87 Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities [1]. The standard ("N288.1") specifies the methodologies and approach for determining a facility's routine release limits.

Limits which are derived are submitted to the federal regulator and, if approved, are incorporated into the facility's license.

The general approach to calculating derived release limits is to determine the steady state concentration in an effluent pathway that will lead to a critical group of exposed persons receiving an exposure at the applicable dose limit. The critical group for a given radionuclide and source is "a fairly homogenous group of people whose location, age, habits, diet, etc, cause them to receive dose equivalents higher than the average received by typical people in all other groups in the exposed population[1]." N288.1 lists two criteria that should be satisfied in selection of the critical group: that they be representative of those expected to receive the highest dose and that they be relatively homogenous with respect to those factors that effect the dose received. It is recognized in the standard that when the number of people exposed is relatively small, one may have to perform calculations based on an exceptional individual. In the case of MNR, release limits have been derived considering two potential critical groups: workers in neighbouring buildings (a critical group) and an infant permanently located at the point of maximum ground level concentration (an exceptional individual).

The selection of these groups differs from the approach taken at most nuclear facilities. In part, this is because there is no physical boundary surrounding MNR as there is at other nuclear facilities. In that case, it is more feasible to look beyond the facility boundary and identify a group of persons likely to receive limiting exposure. At MNR, the space around the facility is open to the public. No effort is made to control occupancy and so,

the most conservative case of an infant at the point of maximum ground level concentration is evaluated. The presence of an air intake at the neighbouring nuclear research building, immediately adjacent to MNR in the direction of the prevailing wind, leads one to consider the dose impact of releases on persons working in that building. Because of the proximity of persons to the facility, and because of the scarcity of agricultural land in the vicinity of the campus, direct exposure to the plume is the limiting exposure path for MNR. Other, more indirect, pathways (e.g. contamination of crops resulting in intakes directly or through contamination of the milk supply) can be dismissed.

After selecting the critical group to be considered, the transfer parameter is calculated based on the exposure pathway. The transfer parameter relates the released rate (e.g. in Bq s^{-1}) concentration to the exposure rate for the critical group being evaluated. In the case of exposure to iodine, the transfer parameter depends upon the diffusion coefficient calculated according to a modified Gaussian plume dispersion model. The exposure rate for the critical group under consideration for unit release rate is then the product of the diffusion coefficient (s m^{-3}), the dose coefficient (Sv Bq^{-1}) and the breathing rate (m^3 a^{-1}). The limiting release rate is easily determined by the ratio of the exposure rate for unit release rate and the limiting dose rate (Sv a^{-1}). Values calculated for the two critical groups considered at MNR are provided in Table 4-1.

Table 4-1: Derived Released Limits (Bq a^{-1}) for Critical Groups

	Boundary	Nearby Buildings	Selected
^{125}I	2.2 E13	1.1 E13	1.1 E13

For ease of comparison with monitored parameters, the derived release limit is expressed as a rate limit corresponding to the weekly frequency of measurements.

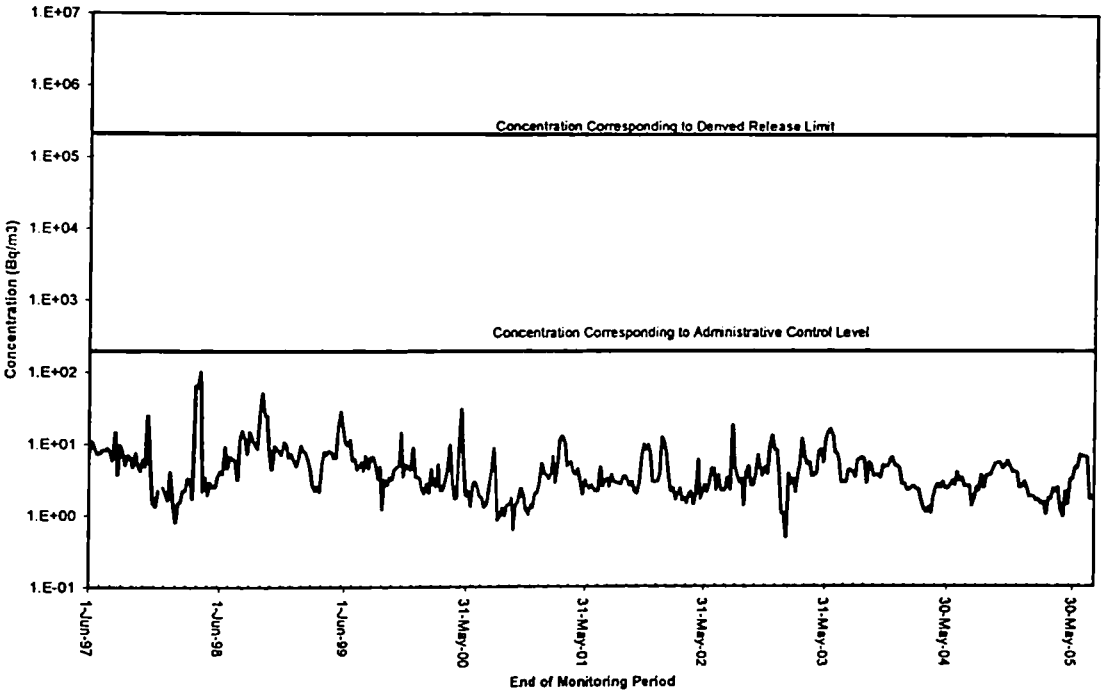
Table 4-2: Derived Release Limits for Reference in Facility Licence

	Rate Limit		Annual Limit
	(Bq/m^3)	(Bq/week)	(Bq/a)
^{125}I	2.1E5	2.1 E11	1.1 E13

Liquid effluents are not considered because MNR does not routinely release radioactive liquid effluents.

In practice, it has not been challenging to maintain facility emissions far below the derived release limits. Figure 4-1 shows the weekly average ^{125}I exhaust concentrations at MNR and the corresponding release limit. Typical values are more than five orders of magnitude lower than the limit. An administrative control level (described in Chapter 1 as an "action level") has been established to aid in ensuring that the releases are maintained as low as reasonably achievable and that any deviations from good practice or loss of control will be identified. This limit has been set at 2E2 Bq m^{-3}.

Figure 4-1: ^{125}I Concentration in the MNR Exhaust



Summary

Derived release limits are highly facility specific, so it is difficult to anticipate how similar the value will be for other facilities. The approach outlined in the attached report is likely to be applicable for a wide range of facilities, however. In particular, it is likely to be a relevant approach for facilities located in campuses or industrial parks with no exclusion zone surrounding establishing a formal boundary for members of the public.

It is likely that a facility with suitable ventilation filtration will have little difficulty operating well below the derived release limit. A lower action level should be set to

assist in focussing efforts on maintaining releases ALARA and to flag unusual releases which may indicate a loss of control or deviation from good practices.

Attached Report

The McMaster University Health Physics Department report HP-MNR-03-05, Derived Release Limits for the McMaster Nuclear Reactor (2005) is attached to this chapter.

The report was prepared solely by the author, using the references cited in the report as guidance. This report was prepared for submission to McMaster oversight committees for MNR and to the federal regulator. The report describes the derivation of the release limits for MNR.

Minor editorial changes have been made to the report for inclusion in this thesis, and proprietary information has been removed.

References

[1] Canadian Standards Association, Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities, CAN/CSA-N288.1-M87 (1987).

McMaster University **Health Physics**

Health Physics Report

HP-MNR-03-05 Derived Release Limits for the McMaster Nuclear Reactor

Revision R1.2 – 2005 February 16[1]

Prepared by: D. M. Tucker
Senior Health Physicist

DOCUMENT SUMMARY

This document provides the Derived Release Limits for airborne effluents from the McMaster Nuclear Reactor. The basis and methods for the calculations are provided.

Date of ROCC Approval:	2004 August 24
Date of HPAC Approval	2004 August 24

[1] With minor editorial changes and removal of proprietary information – 2005 November.

Revision History

Revision	Date	Author	Comments
0	19DEC03	D.M. Tucker	Initial issue. Approved by HPAC 2003 December 16. Approved by ROCC 2003 December 19. Not accepted by CNSC due to dose potential for occupants of nearby buildings exceeding the public dose limits with sustained releases at the DRL. See CNSC letter of 2004 May 26 (reference 26-1-1-4-0).
1.0	9JUN04	D.M. Tucker	Revised to incorporate occupants of most restrictive nearby building (NRB) as the basis for the I-125 DRL.
1.1	6JUL04	D.M. Tucker	Revised to include rationale for selection and rejection of pathways as requested following CNSC specialist preliminary review (2004 June 29 email from Rao Avandhanula)
1.2	16FEB05	D.M. Tucker	Revised to include description of current routine release rates with respect to the DRLs as requested in the 2005 February 15 letter from Lisa Lang (CNSC). The description was excerpted from a version of the document previously approved by the ROCC and HPAC

Health Physics Report

HP-MNR-03-05 derived Release Limits for the McMaster Nuclear Reactor

1 Reason for Report

This document provides the Derived Release Limits for the McMaster Nuclear Reactor (MNR). The basis and methods for the calculations are provided. This document replaces HP-5-94-0-Rev.2, "Derived Emission Limits for the McMaster Nuclear Reactor" previously submitted to the CNSC.

This is a licence basis document. It will be referenced by the CNSC licence for MNR.

2 Background

The McMaster Nuclear Reactor is located on the campus of McMaster University in Hamilton Ontario. The facility location and site are described in the McMaster Nuclear Reactor Safety Analysis Report [1].

Operation of the McMaster Nuclear Reactor leads to ongoing airborne releases of ^{41}Ar. In addition, isotope production activities result in ongoing emissions of ^{125}I. These are the only two radionuclides known to be routinely released to the environment.

For consistency with previous submissions, two critical groups are evaluated. The DRL is calculated on the basis of boundary dose for a hypothetical infant permanently located at the point of maximum ground level concentration. The potential doses to individuals working in buildings at closer distances are also evaluated.

3 Meteorological Data

Meteorological data relevant to the MNR site, collected at the Royal Botanical Gardens, has been provided by Environment Canada. The data are described in the McMaster Nuclear Reactor Safety Analysis Report [1]. A summary of the relevant data for calculation of the DRLs is presented in Appendix A.

4 Facility Data

A stack height of 17m has been used in the calculations. The nominal exhaust rate for MNR is 3500 cfm or 1.65 $m^3 s^{-1}$.[1]

5 Derived Release Limit Calculation Based on Boundary

5.1 Critical Group and Pathway Selection

MNR does not have a site boundary in the way larger reactors do. Therefore the critical group selected is a hypothetical 1 year old infant present at the point of maximum ground level concentration in the most exposed sector throughout the year. This treatment is conservative. The applicable annual dose limit is 1 mSv.

For I-125, the only exposure pathway considered is inhalation of activity from the plume. While external exposure from immersion also occurs, the dose conversion factor for this exposure pathway [2] is more than three orders of magnitude less than that for inhalation. Another exposure pathway commonly considered but not of significance here is the consumption of milk by infants. The University is surrounded by residential and industrial land [1]. It can be shown, following the methodology of the CSA [2] that an infant would have to consume entirely milk produced less than 10 km from MNR for this pathway to become limiting.

For Ar-41, only external exposure from immersion is considered, following convention and the guidance of the CSA [2] that "air immersion is the most important exposure pathway to man for the noble gases". This is the only pathway for which dose coefficients are published for Ar-41 [3].

5.2 Calculation of Transfer Parameters

Transfer parameters at ground level were calculated for each sector in the wind rose following the guidance of the relevant CSA standard [2] equation 5.3 and Appendix D. A sample of the spreadsheets used in the calculations is presented in Appendix B. Figure B1 shows the transfer parameters for each sector as a function of distance. Also shown in Figure B1 is the Transfer Parameter calculated using the default weather patterns proposed by CSA [2].

The maximum ground level transfer parameter occurs in Sector 1, due east of MNR at approximately 90 m. The value of the transfer parameter is approximately 1.5E-5 s m^{-3}.

5.3 Radiological and Physiological Data

Radiological data relevant to the calculation are presented in Table 1.

Table 1: Radiological and Physiological Data

Parameter	Value	Source	Comment
I-125 Vapour inhalation Dose Conversion Factor for 1 year old infant	5.2 E-8 Sv Bq^{-1}	ICRP-72 [3]	Use of vapour inhalation dose coefficients recommended by standard
Ar-41 Immersion Dose Coefficient	8.0 E-9 Sv d^{-1} Bq^{-1} m^3	ICRP-72 [3] CSA [2]	The ICRP immersion dose coefficient for adults was multiplied by 1.5 following the guidance of CSA
Inhalation rate for 1 year old infant	5.1 m^3 d^{-1}	ICRP-89[4]	

5.4 Derived Release Limits - Boundary

For ^{125}I, the DRL is obtained by dividing the annual dose limit (1 mSv) by the product of the dose conversion factor, the transfer parameter and inhalation rate, with appropriate unit conversion.

For ^{41}Ar, the DRL is obtained by dividing the annual dose limit (1 mSv) by the product of the transfer parameter and the immersion dose coefficient, with appropriate unit conversion.

The DRLs calculated on the basis of Boundary Dose are presented in Table 2.

Table 2: Derived Released Limits for Boundary Dose

	DRL (Bq a^{-1})
^{125}I	2.2 E13
^{41}Ar	7.2 E14

6 Derived Release Limit Calculation Based on Nearby Buildings

6.1 Critical Group and Pathway Selection

The nearest building to MNR is the Nuclear Research Building (NRB). An air intake for a portion of the building is located on the West side of NRB, approximately 30 m to the East of MNR. The centre of the air intake is at an elevation of 11.3m. This location means that the air intake is also in the direct line of the prevailing wind. Therefore, persons occupationally exposed in NRB are taken as the critical group for exposure in nearby buildings.

Although most of the people working routinely in NRB are designated Nuclear Energy Workers, such designation is not currently an administrative requirement. Therefore, the public dose limit of 1 mSv a^{-1} has been applied to this group.

Occupational occupancy of 2000 hours per year is assumed.

As explained in Section 5.1, the only exposure pathways considered are inhalation for I-125 and external exposure from immersion for Ar-41.

6.2 Calculation of Transfer Parameters

The transfer parameter was calculated at a receptor elevation (z) of 11.3 m as a function of distance for Sector 1 following the guidance of the relevant CSA standard [2] equation 5.2. This represents the physical configuration of the buildings with slight conservatism by neglecting dilution of the air with air ingress and intake from sources further from MNR. The spreadsheet used in the calculations is presented in Appendix C. Figure C1 shows the transfer parameter for Sector 1 as a function of distance at the elevation of the NRB intake.

The transfer parameter at 30 m is approximately 8.2E-5 s m^{-3}.

6.3 Radiological and Physiological Data

Radiological data relevant to the calculation are presented in Table 3.

Table 3: Radiological and Physiological Data

Parameter	Value	Source	Comment
I-125 Vapour inhalation Dose Conversion Factor for adult.	1.4 E-8 Sv Bq^{-1}	ICRP-72 [3]	Use of vapour inhalation dose coefficients recommended by standard [3].
Ar-41 Immersion Dose Coefficient	5.3 E-9 Sv d^{-1} Bq^{-1} m^3	ICRP-72 [3]	
Inhalation rate for adults	9.6 m^3 d^{-1} (8 hour occupational)	ICRP-89[4]	The value for sedentary male workers has been selected.

6.4 Derived Release Limits – Nearby Buildings

For ^{125}I, the DRL is obtained by dividing the annual dose limit (1 mSv) by the product of the dose conversion factor, the transfer parameter and inhalation rate, with appropriate unit conversion.

For ^{41}Ar, the DRL is obtained by dividing the annual dose limit (1 mSv) by the product of the transfer parameter and the immersion dose coefficient, with appropriate unit conversion.

The DRLs calculated on the basis of occupational occupancy of nearby buildings are presented in Table 4.

Table 4: Derived Released Limits for Nearby Buildings

	DRL (Bq a^{-1})
^{125}I	1.1 E13
^{41}Ar	8.7 E14

7 Selection of DRLs

The DRLs calculated for each of the two critical groups considered are summarized in Table 5. The selected DRL is the lower of the two values for each radionuclide.

Table 5: Derived Released Limits (Bq a^{-1})for Critical Groups

	Boundary	Nearby Buildings	Selected
^{125}I	2.2 E13	1.1 E13	1.1 E13
^{41}Ar	7.2 E14	8.7 E14	7.2 E14

8 Conclusions

Derived Release Limits have been calculated based on occupationally exposed occupants of nearby buildings and on a hypothetical infant at the point of maximum ground-level concentration (the "boundary dose"). The lower value for each radionuclide has been selected.

In practice, samples are normally collected on a weekly basis and it is industry practice to establish a limit based on emission rate. Thus, the limits proposed for reference with respect to the facility licence are as shown in Table 6.

Table 6: Derived Release Limits for Reference in Facility Licence

	Rate Limit (Bq/week)	Annual Limit (Bq/a)
^{125}I	2.1 E11	1.1 E13
^{41}Ar	1.4 E13	7.2 E14

It should be noted that the typical rates of emission of ^{125}I and ^{41}Ar are approximately 0.002% and 0.1% respectively of the proposed DRLs. In addition, results are evaluated on a weekly basis and any releases exceeding the (much lower) administrative control levels in the MNR radiation safety program are formally investigated and reported to the CNSC. There is no potential for the public dose limit to be exceeded as a result of routine emissions.

9 List of APPENDICES

A Meteorological Data for MNR.

B Calculation of Transfer Parameters at Ground Level - Sample Spreadsheets

C Calculation of Transfer Parameter for Nearby Buildings – Spreadsheet

10 References

[1] Safety Analysis Report for the McMaster Nuclear Reactor, February 2002.

[2] CAN/CSA-N288.1-M87 Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities, Canadian Standards Association, 1987.

[3] ICRP-72, Age Dependant Doses to Members of the Public from Intake of Radionuclides, Annals of the ICRP Volume 26 No.1, 1996

[4] ICRP-89, Basic Anatomical and Physiological Data for Use in Radiological Protection: Reference Values, Annals of the ICRP Volume 32, No. 3-4, 2002.

Sector	Direction	Stability	Persistance	Mean Wind Speed (m/s)
1 E	A		0.008048	2
	B		0.018390	2.7
	C		0.085185	4.3
	D		0.008954	6.5
	E		0.010960	3.4
	F		0.001200	2.2
2 ENE	A		0.006121	1.8
	B		0.013799	2.6
	C		0.084845	4.2
	D		0.007187	7
	E		0.007504	3.4
	F		0.000724	2.2
3 NE	A		0.004846	1.9
	B		0.012440	2.6
	C		0.078832	4.5
	D		0.009378	6.9
	E		0.007291	3.4
	F		0.001000	2.3
4 NNE	A		0.004200	1.7
	B		0.010440	2.6
	C		0.058370	4.2
	D		0.007781	6.1
	E		0.008186	3.3
	F		0.008186	2.2
5 N	A		0.000678	1.9
	B		0.005555	2.5
	C		0.011271	3.9
	D		0.029432	4.9
	E		0.008456	2.9
	F		0.006632	2
6 NNW	A		0.000452	1.6
	B		0.002930	2.4
	C		0.002847	3.1
	D		0.008403	4.1
	E		0.001546	2.6
	F		0.002191	1.8
7 NW	A		0.000452	1.8
	B		0.002162	2.2
	C		0.002014	2.8
	D		0.004298	3.4
	E		0.000866	2.5
	F		0.001571	1.8
8 WNW	A		0.000498	1.4
	B		0.002559	2.2
	C		0.002564	2.7
	D		0.004676	3.2
	E		0.000748	2.6
	F		0.001765	1.8

Sector	Direction	Stability	Persistance	Mean Wind Speed (m/s)
9 W	A		0.000656	1.6
	B		0.006169	2.4
	C		0.007471	3.2
	D		0.020226	4.4
	E		0.003279	2.9
	F		0.004983	1.9
10 WSW	A		0.000543	1.7
	B		0.005521	2.8
	C		0.013256	4.2
	D		0.007257	6.3
	E		0.000337	3.2
	F		0.003620	2.1
11 SW	A		0.000452	2
	B		0.004638	2.7
	C		0.010883	4.2
	D		0.011146	6.1
	E		0.005624	3.3
	F		0.004382	2.2
12 SSW	A		0.000204	1.7
	B		0.003110	2.4
	C		0.005237	3.7
	D		0.025410	5.2
	E		0.003808	3.5
	F		0.003587	2.2
13 S	A		0.000666	1.7
	B		0.003137	2.3
	C		0.005402	3.4
	D		0.015344	4.8
	E		0.003143	3.5
	F		0.003936	2.1
14 SSE	A		0.000271	1.7
	B		0.002371	2.4
	C		0.004488	3.8
	D		0.015653	5.2
	E		0.003245	3.7
	F		0.003054	2
15 SE	A		0.000384	1.5
	B		0.002454	2.5
	C		0.005436	4
	D		0.028704	5.8
	E		0.004536	3.7
	F		0.003601	2.1
16 ESE	A		0.000905	1.9
	B		0.003904	2.7
	C		0.009274	4.3
	D		0.043760	6.2
	E		0.004944	3.5
	F		0.006128	2.2

Appendix B: Calculation of Transfer Parameters at Ground Level - Sample Spreadsheets

Table D3 [2]

Coefficients	Stability A	B	C	D	E	F
a1	0.112	0.13	0.112	0.098	0.0838	0.0838
b1	1.03	0.95	0.92	0.889	0.835	0.783
a2	5.38E-04	6.52E-04	9.05E-04	1.36E-03	1.96E-03	1.36E-03
b2	0.815	0.75	0.718	0.688	0.684	0.672

Table D4 [2]

Roughness Length Zo (cm)	c1	d1	c2	d2
1	1.56	0.048	8.25E-04	0.45
4	2.02	0.0269	7.76E-04	0.37
10	271828	0	0	0
40	5.16	-0.098	18.6	-0.225
100	7.37	-0.0957	4.29E+03	-0.6
400	11.7	-0.128	4.59E+04	-0.76

See Table D2[2]

	Calculation of g(x) for each stability						Calculation of F(zo,x) for Zo=100		Sigmaz-I					
x	A	B	C	D	E	F	Guts	Ln(guts)	A	B	C	D	E	F
10	1.28E+00	1.15E+00	9.27E-01	7.54E-01	4.74E-01	3.85E-01	5.917937	1.78E+00	2.28E+00	2.05E+00	1.65E+00	1.34E+00	8.42E-01	6.84E-01
20	2.06E+00	2.22E+00	1.75E+00	1.39E+00	8.70E-01	6.59E-01	5.54070	1.71E+00	4.56E+00	3.81E+00	2.69E+00	2.38E+00	1.50E+00	1.13E+00
30	4.09E+00	3.26E+00	2.53E+00	1.99E+00	1.25E+00	9.03E-01	5.331942	1.67E+00	6.84E+00	5.46E+00	4.24E+00	3.33E+00	2.10E+00	1.51E+00
40	5.53E+00	4.26E+00	3.29E+00	2.56E+00	1.61E+00	1.13E+00	5.188859	1.65E+00	9.10E+00	7.05E+00	5.42E+00	4.21E+00	2.66E+00	1.88E+00
50	6.09E+00	5.28E+00	4.03E+00	3.11E+00	1.96E+00	1.34E+00	5.080814	1.63E+00	1.14E+01	8.58E+00	6.56E+00	5.08E+00	3.19E+00	2.18E+00
60	8.46E+00	6.27E+00	4.78E+00	3.65E+00	2.30E+00	1.54E+00	4.994336	1.61E+00	1.36E+01	1.01E+01	7.66E+00	5.87E+00	3.70E+00	2.48E+00
70	9.95E+00	7.24E+00	5.48E+00	4.18E+00	2.63E+00	1.74E+00	4.922492	1.59E+00	1.59E+01	1.15E+01	8.73E+00	6.66E+00	4.20E+00	2.77E+00
80	1.14E+01	8.2 E+00	6.18E+00	4.69E+00	2.96E+00	1.92E+00	4.861104	1.58E+00	1.81E+01	1.30E+01	9.77E+00	7.42E+00	4.68E+00	3.04E+00
90	1.29E+01	9.17E+00	6.86E+00	5.20E+00	3.28E+00	2.10E+00	4.80784	1.57E+00	2.03E+01	1.44E+01	1.08E+01	8.18E+00	5.15E+00	3.30E+00
100	1.44E+01	1.0 E+01	7.56E+00	5.70E+00	3.59E+00	2.29E+00	4.76088	1.56E+00	2.25E+01	1.58E+01	1.18E+01	8.89E+00	5.60E+00	3.58E+00
110	1.58E+01	1.1 E+01	8.24E+00	6.19E+00	3.89E+00	2.45E+00	4.718478	1.55E+00	2.47E+01	1.72E+01	1.28E+01	9.60E+00	6.05E+00	3.80E+00
120	1.74E+01	1.20E+01	8.91E+00	6.67E+00	4.20E+00	2.32E+00	4.680328	1.54E+00	2.69E+01	1.85E+01	1.38E+01	1.03E+01	6.49E+00	4.04E+00
130	1.90E+01	1.29E+01	9.58E+00	7.16E+00	4.50E+00	2.78E+00	4.645540	1.64E+00	2.91E+01	1.99E+01	1.47E+01	1.10E+01	6.92E+00	4.29E+00
140	2.05E+01	1.38E+01	1.02E+01	7.62E+00	4.80E+00	2.95E+00	4.613622	1.63E+00	3.13E+01	2.12E+01	1.57E+01	1.16E+01	7.34E+00	4.50E+00
150	2.20E+01	1.48E+01	1.09E+01	8.09E+00	5.09E+00	3.10E+00	4.584102	1.52E+00	3.35E+01	2.25E+01	1.66E+01	1.23E+01	7.75E+00	4.70E+00
160	2.36E+01	1.57E+01	1.15E+01	8.55E+00	5.38E+00	3.26E+00	4.56675	1.52E+00	3.57E+01	2.38E+01	1.75E+01	1.30E+01	8.18E+00	4.94E+00
170	2.50E+01	1.68E+01	1.22E+01	9.00E+00	5.67E+00	3.41E+00	4.531207	1.51E+00	3.78E+01	2.51E+01	1.84E+01	1.36E+01	8.56E+00	5.16E+00
180	2.65E+01	1.75E+01	1.28E+01	9.46E+00	5.95E+00	3.56E+00	4.507283	1.51E+00	4.00E+01	2.63E+01	1.93E+01	1.42E+01	8.96E+00	5.38E+00
190	2.81E+01	1.84E+01	1.35E+01	9.91E+00	6.23E+00	3.71E+00	4.484795	1.50E+00	4.21E+01	2.76E+01	2.02E+01	1.49E+01	9.35E+00	5.57E+00
200	2.96E+01	1.93E+01	1.41E+01	1.04E+01	6.50E+00	3.86E+00	4.463587	1.50E+00	4.43E+01	2.88E+01	2.11E+01	1.56E+01	9.73E+00	5.77E+00
210	3.11E+01	2.02E+01	1.47E+01	1.08E+01	6.78E+00	4.30E+00	4.443529	1.49E+00	4.64E+01	3.01E+01	2.19E+01	1.61E+01	1.01E+01	5.97E+00
220	3.26E+01	2.11E+01	1.53E+01	1.12E+01	7.05E+00	4.14E+00	4.424509	1.49E+00	4.85E+01	3.13E+01	2.28E+01	1.67E+01	1.05E+01	6.16E+00
230	3.42E+01	2.19E+01	1.60E+01	1.17E+01	7.32E+00	4.28E+00	4.406429	1.48E+00	5.07E+01	3.25E+01	2.37E+01	1.73E+01	1.09E+01	6.35E+00
240	3.57E+01	2.28E+01	1.66E+01	1.21E+01	7.59E+00	4.42E+00	4.389205	1.48E+00	5.28E+01	3.37E+01	2.45E+01	1.79E+01	1.12E+01	6.54E+00
250	3.72E+01	2.37E+01	1.72E+01	1.25E+01	7.85E+00	4.56E+00	4.372784	1.48E+00	5.49E+01	3.50E+01	2.54E+01	1.85E+01	1.16E+01	6.73E+00
260	3.87E+01	2.46E+01	1.78E+01	1.29E+01	8.12E+00	4.70E+00	4.357049	1.47E+00	5.70E+01	3.61E+01	2.62E+01	1.90E+01	1.19E+01	6.91E+00
270	4.02E+01	2.54E+01	1.84E+01	1.34E+01	8.38E+00	4.83E+00	4.341981	1.47E+00	5.91E+01	3.73E+01	2.70E+01	1.96E+01	1.23E+01	7.09E+00
280	4.18E+01	2.63E+01	1.90E+01	1.39E+01	8.64E+00	4.96E+00	4.327532	1.46E+00	6.12E+01	3.85E+01	2.78E+01	2.02E+01	1.27E+01	7.27E+00
290	4.33E+01	2.72E+01	1.96E+01	1.42E+01	8.90E+00	5.09E+00	4.313848	1.46E+00	6.33E+01	3.97E+01	2.86E+01	2.08E+01	1.30E+01	7.45E+00
300	4.48E+01	2.80E+01	2.02E+01	1.46E+01	9.15E+00	5.22E+00	4.300209	1.46E+00	6.53E+01	4.09E+01	2.95E+01	2.15E+01	1.33E+01	7.62E+00
310	4.63E+01	2.89E+01	2.08E+01	1.50E+01	9.40E+00	5.35E+00	4.28742	1.46E+00	6.74E+01	4.20E+01	3.03E+01	2.19E+01	1.37E+01	7.79E+00
320	4.78E+01	2.97E+01	2.14E+01	1.54E+01	9.66E+00	5.48E+00	4.275008	1.45E+00	6.95E+01	4.32E+01	3.11E+01	2.24E+01	1.40E+01	7.96E+00
330	4.93E+01	3.06E+01	2.20E+01	1.58E+01	9.91E+00	5.61E+00	4.263022	1.45E+00	7.16E+01	4.43E+01	3.18E+01	2.30E+01	1.44E+01	8.13E+00
340	5.09E+01	3.14E+01	2.25E+01	1.62E+01	1.02E+01	5.73E+00	4.251437	1.45E+00	7.36E+01	4.54E+01	3.26E+01	2.35E+01	1.47E+01	8.29E+00
350	5.24E+01	3.22E+01	2.31E+01	1.66E+01	1.04E+01	5.86E+00	4.240228	1.44E+00	7.57E+01	4.66E+01	3.34E+01	2.40E+01	1.50E+01	8.46E+00
360	5.39E+01	3.31E+01	2.37E+01	1.70E+01	1.06E+01	5.98E+00	4.229372	1.44E+00	7.77E+01	4.77E+01	3.42E+01	2.46E+01	1.54E+01	8.62E+00
370	5.54E+01	3.39E+01	2.43E+01	1.74E+01	1.09E+01	6.10E+00	4.219849	1.44E+00	7.97E+01	4.88E+01	3.50E+01	2.51E+01	1.57E+01	8.78E+00
380	5.69E+01	3.48E+01	2.49E+01	1.78E+01	1.11E+01	6.22E+00	4.208841	1.44E+00	8.18E+01	4.99E+01	3.57E+01	2.56E+01	1.60E+01	8.94E+00
390	5.84E+01	3.56E+01	2.54E+01	1.82E+01	1.14E+01	6.34E+00	4.198731	1.43E+00	8.38E+01	5.11E+01	3.65E+01	2.61E+01	1.63E+01	9.10E+00
400	5.99E+01	3.64E+01	2.60E+01	1.86E+01	1.16E+01	6.46E+00	4.189101	1.43E+00	8.58E+01	5.22E+01	3.72E+01	2.67E+01	1.66E+01	9.26E+00
410	6.14E+01	3.72E+01	2.66E+01	1.90E+01	1.19E+01	6.58E+00	4.179739	1.43E+00	8.79E+01	5.33E+01	3.80E+01	2.72E+01	1.70E+01	9.41E+00
420	6.29E+01	3.8 E+01	2.71E+01	1.94E+01	1.21E+01	6.70E+00	4.170629	1.43E+00	8.99E+01	5.44E+01	3.88E+01	2.77E+01	1.73E+01	9.56E+00
430	6.44E+01	3.89E+01	2.77E+01	1.98E+01	1.23E+01	6.81E+00	4.161761	1.43E+00	9.19E+01	5.54E+01	3.95E+01	2.82E+01	1.76E+01	9.72E+00
440	6.59E+01	3.97E+01	2.83E+01	2.01E+01	1.26E+01	6.93E+00	4.153121	1.42E+00	9.39E+01	5.65E+01	4.02E+01	2.87E+01	1.79E+01	9.87E+00
450	6.74E+01	4.05E+01	2.88E+01	2.05E+01	1.28E+01	7.04E+00	4.1447	1.42E+00	9.59E+01	5.76E+01	4.10E+01	2.92E+01	1.82E+01	1.00E+01
460	6.89E+01	4.13E+01	2.94E+01	2.09E+01	1.30E+01	7.16E+00	4.136487	1.42E+00	9.79E+01	5.87E+01	4.17E+01	2.97E+01	1.85E+01	1.02E+01
470	7.04E+01	4.22E+01	2.99E+01	2.13E+01	1.33E+01	7.27E+00	4.128473	1.42E+00	9.99E+01	5.98E+01	4.24E+01	3.02E+01	1.88E+01	1.03E+01
480	7.19E+01	4.30E+01	3.05E+01	2.17E+01	1.35E+01	7.38E+00	4.120648	1.42E+00	1.02E+02	6.08E+01	4.32E+01	3.07E+01	1.91E+01	1.05E+01
490	7.34E+01	4.38E+01	3.10E+01	2.20E+01	1.37E+01	7.50E+00	4.113005	1.41E+00	1.04E+02	6.19E+01	4.39E+01	3.12E+01	1.94E+01	1.08E+01
500	7.49E+01	4.46E+01	3.16E+01	2.24E+01	1.39E+01	7.31E+00	4.105538	1.41E+00	1.06E+02	6.29E+01	4.46E+01	3.16E+01	1.97E+01	1.07E+01

Stack Height			
h=	17 m	z	0

Calculation of F(h,z,sima-z-i) from eq 5.2

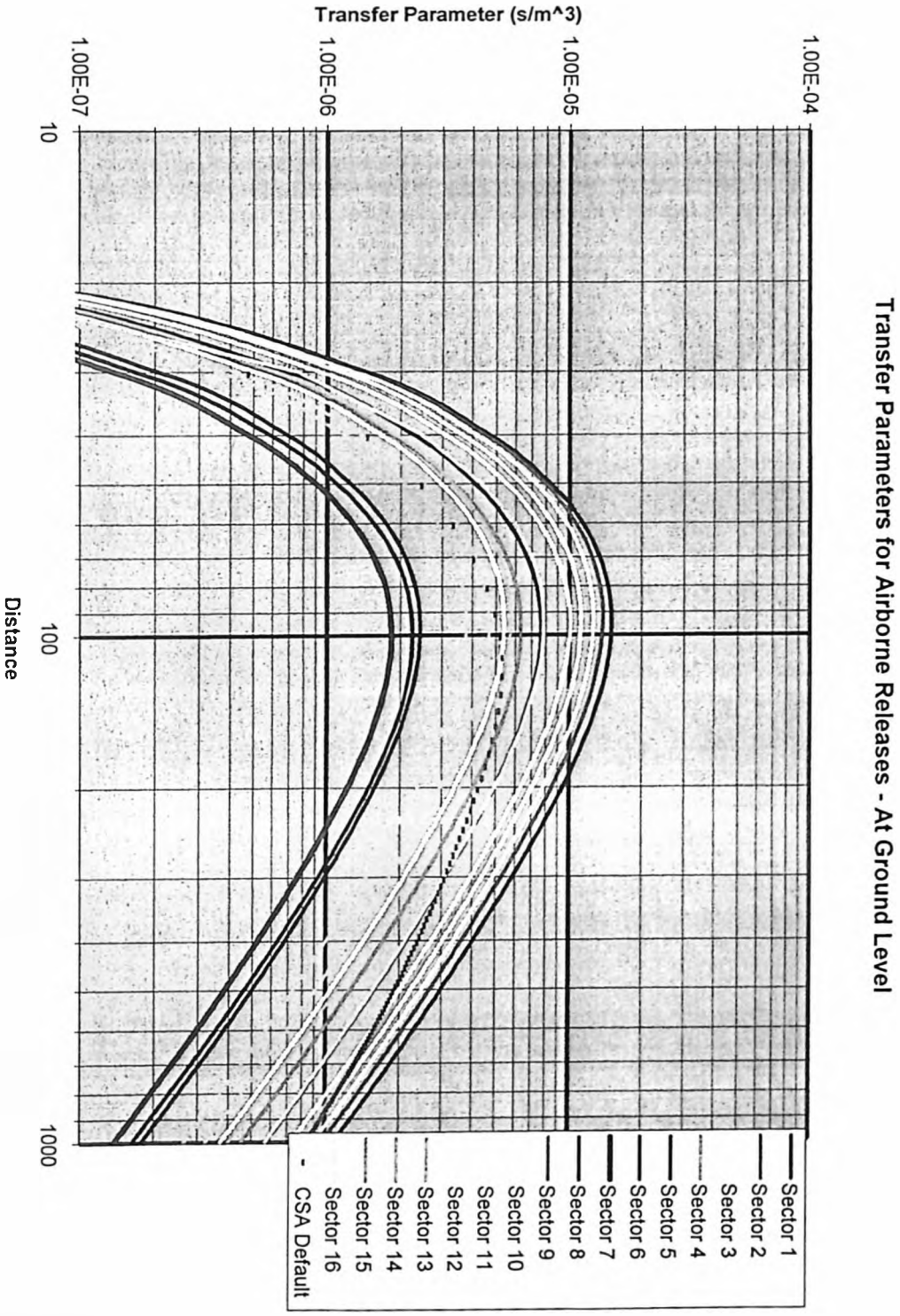
A	B	C	D	E	F
1.83E-12	2.54E-15	1.82E-23	2.42E-35	6.94E-89	1.32E-134
1.93E-03	9.44E-05	2.01E-07	1.71E-11	2.51E-28	1.15E-49
9.10E-02	1.57E-02	6.46E-04	4.27E-08	1.09E-14	6.63E-28
3.50E-01	1.09E-01	1.47E-02	5.84E-04	2.62E-09	1.27E-18
6.53E-01	2.81E-01	8.95E-02	7.08E-03	1.38E-06	1.17E-13
9.17E-01	4.82E-01	1.70E-01	3.02E-02	5.33E-05	1.23E-10
1.13E+00	6.78E-01	3.00E-01	7.86E-02	5.51E-04	1.25E-08
1.29E+00	8.49E-01	4.41E-01	1.45E-01	2.72E-03	3.24E-07
1.41E+00	9.98E-01	5.79E-01	2.28E-01	8.55E-03	3.55E-06
1.50E+00	1.12E+00	7.08E-01	3.21E-01	2.00E-02	2.21E-05
1.58E+00	1.22E+00	8.26E-01	4.17E-01	3.86E-02	9.23E-05
1.64E+00	1.31E+00	9.32E-01	5.11E-01	6.45E-02	2.91E-04
1.69E+00	1.39E+00	1.03E+00	6.03E-01	9.75E-02	7.41E-04
1.73E+00	1.45E+00	1.11E+00	6.90E-01	1.37E-01	1.61E-03
1.76E+00	1.50E+00	1.18E+00	7.71E-01	1.80E-01	3.10E-03
1.79E+00	1.55E+00	1.25E+00	8.46E-01	2.28E-01	5.40E-03
1.81E+00	1.59E+00	1.31E+00	9.16E-01	2.78E-01	8.71E-03
1.83E+00	1.62E+00	1.36E+00	9.81E-01	3.30E-01	1.32E-02
1.84E+00	1.65E+00	1.40E+00	1.04E+00	3.82E-01	1.89E-02
1.86E+00	1.68E+00	1.44E+00	1.09E+00	4.35E-01	2.61E-02
1.87E+00	1.70E+00	1.48E+00	1.14E+00	4.87E-01	3.46E-02
1.88E+00	1.73E+00	1.51E+00	1.19E+00	5.38E-01	4.45E-02
1.89E+00	1.74E+00	1.55E+00	1.23E+00	5.87E-01	5.57E-02
1.90E+00	1.76E+00	1.57E+00	1.27E+00	6.35E-01	6.83E-02
1.91E+00	1.78E+00	1.60E+00	1.31E+00	6.82E-01	8.21E-02
1.91E+00	1.79E+00	1.62E+00	1.34E+00	7.27E-01	9.70E-02
1.92E+00	1.80E+00	1.64E+00	1.37E+00	7.70E-01	1.13E-01
1.92E+00	1.81E+00	1.66E+00	1.40E+00	8.11E-01	1.30E-01
1.93E+00	1.82E+00	1.68E+00	1.43E+00	8.51E-01	1.48E-01
1.93E+00	1.83E+00	1.60E+00	1.46E+00	8.80E-01	1.66E-01
1.94E+00	1.84E+00	1.71E+00	1.48E+00	9.26E-01	1.85E-01
1.94E+00	1.85E+00	1.72E+00	1.50E+00	9.60E-01	2.05E-01
1.94E+00	1.86E+00	1.73E+00	1.52E+00	9.93E-01	2.25E-01
1.95E+00	1.86E+00	1.75E+00	1.54E+00	1.02E+00	2.45E-01
1.95E+00	1.87E+00	1.76E+00	1.56E+00	1.05E+00	2.65E-01
1.95E+00	1.88E+00	1.77E+00	1.57E+00	1.08E+00	2.80E-01
1.96E+00	1.88E+00	1.78E+00	1.59E+00	1.11E+00	3.07E-01
1.96E+00	1.89E+00	1.79E+00	1.60E+00	1.14E+00	3.28E-01
1.96E+00	1.89E+00	1.79E+00	1.62E+00	1.16E+00	3.49E-01
1.96E+00	1.90E+00	1.80E+00	1.63E+00	1.19E+00	3.70E-01
1.96E+00	1.90E+00	1.81E+00	1.64E+00	1.21E+00	3.91E-01
1.96E+00	1.90E+00	1.82E+00	1.66E+00	1.23E+00	4.12E-01
1.97E+00	1.91E+00	1.82E+00	1.67E+00	1.25E+00	4.33E-01
1.97E+00	1.91E+00	1.83E+00	1.68E+00	1.27E+00	4.53E-01
1.97E+00	1.91E+00	1.84E+00	1.69E+00	1.29E+00	4.74E-01
1.97E+00	1.92E+00	1.84E+00	1.70E+00	1.31E+00	4.94E-01
1.97E+00	1.92E+00	1.85E+00	1.71E+00	1.33E+00	5.14E-01
1.97E+00	1.92E+00	1.85E+00	1.72E+00	1.35E+00	5.33E-01
1.97E+00	1.93E+00	1.86E+00	1.72E+00	1.36E+00	5.53E-01
1.97E+00	1.93E+00	1.86E+00	1.73E+00	1.38E+00	5.72E-01

Calculation of Fo' for Sector 1 from equation 5.3

Sector	Direction	Stability	Persistence	Speed (m/s)
1	E	A	0.029364	2
		B	0.099578	2.7
		C	0.388375	4.3
		D	0.356609	6.5
		E	0.076471	3.4
		F	0.054970	2.2
			0.133000	
		(2pi)^(3/2)/n=	0.084252	

Termsin sum

x	A	B	C	D	E	F	F01
10	1.05E-14	4.56E-17	3.37E-25	9.90E-37	1.85E-90	4.33E-136	1.42366E-15
20	5.22E-06	9.14E-07	5.76E-09	3.94E-13	3.77E-30	2.54E-51	4.82142E-08
30	1.95E-04	1.00E-04	1.30E-05	7.05E-08	1.17E-16	1.10E-29	1.42060E-05
40	5.64E-04	5.70E-04	2.44E-04	7.61E-06	2.21E-11	1.71E-20	4.68379E-05
50	9.44E-04	1.21E-03	9.57E-04	7.65E-05	9.72E-09	1.34E-15	8.34055E-05
60	9.09E-04	1.77E-03	2.31E-03	2.02E-04	3.23E-07	1.24E-12	1.10099E-05
70	1.04E-03	2.16E-03	3.11E-03	6.32E-04	2.95E-06	1.13E-10	1.34031E-05
80	1.04E-03	2.41E-03	4.37E-03	1.07E-03	1.31E-05	2.66E-09	1.45423E-05
90	1.02E-03	2.55E-03	4.34E-03	1.54E-03	3.74E-05	2.69E-08	1.49934E-05
100	9.81E-04	2.62E-03	5.42E-03	1.98E-03	8.05E-05	1.55E-07	1.49743E-05
110	9.38E-04	2.63E-03	5.34E-03	2.38E-03	1.43E-04	6.06E-07	1.46669E-05
120	8.93E-04	2.61E-03	5.12E-03	2.73E-03	2.24E-04	1.80E-06	1.41627E-05
130	8.50E-04	2.58E-03	5.30E-03	3.01E-03	3.17E-04	4.33E-06	1.3573E-05
140	8.09E-04	2.52E-03	5.40E-03	3.25E-03	4.18E-04	8.95E-06	1.29407E-05
150	7.71E-04	2.47E-03	5.44E-03	3.44E-03	5.24E-04	1.64E-05	1.22988E-05
160	7.35E-04	2.40E-03	5.44E-03	3.58E-03	6.29E-04	2.73E-05	1.1668E-05
170	7.02E-04	2.34E-03	5.41E-03	3.69E-03	7.31E-04	4.22E-05	1.10604E-05
180	6.71E-04	2.27E-03	5.35E-03	3.78E-03	8.29E-04	6.14E-05	1.04826E-05
190	6.43E-04	2.21E-03	5.28E-03	3.84E-03	9.20E-04	8.50E-05	9.93792E-06
200	6.16E-04	2.15E-03	5.19E-03	3.88E-03	1.01E-03	1.13E-04	9.42723E-06
210	5.92E-04	2.09E-03	5.10E-03	3.90E-03	1.08E-03	1.45E-04	8.95017E-06
220	5.69E-04	2.03E-03	5.00E-03	3.91E-03	1.15E-03	1.80E-04	8.50648E-06
230	5.48E-04	1.98E-03	5.30E-03	3.91E-03	1.22E-03	2.19E-04	8.09144E-06
240	5.28E-04	1.93E-03	5.79E-03	3.91E-03	1.27E-03	2.61E-04	7.70609E-06
250	5.10E-04	1.88E-03	[illegible]	3.89E-03	1.32E-03	3.05E-04	7.34742E-06
260	4.93E-04	1.83E-03	5.50E-03	3.87E-03	1.37E-03	3.51E-04	7.01341E-06
270	4.77E-04	1.78E-03	5.49E-03	3.84E-03	1.41E-03	3.98E-04	6.70212E-06
280	4.62E-04	1.74E-03	5.39E-03	3.81E-03	1.44E-03	4.46E-04	6.41171E-06
290	4.48E-04	1.70E-03	5.29E-03	3.78E-03	1.47E-03	4.95E-04	6.14049E-06
300	4.34E-04	1.66E-03	5.19E-03	3.75E-03	1.50E-03	5.44E-04	5.88687E-06
310	4.22E-04	1.62E-03	5.10E-03	3.71E-03	1.52E-03	5.93E-04	5.64942E-06
320	4.10E-04	1.59E-03	5.01E-03	3.67E-03	1.54E-03	6.42E-04	5.42001E-06
330	3.99E-04	1.56E-03	4.92E-03	3.63E-03	1.55E-03	6.90E-04	5.21785E-06
340	3.88E-04	1.51E-03	4.33E-03	3.59E-03	1.57E-03	7.38E-04	5.02143E-06
350	3.70E-04	1.40E-03	4.75E-03	3.55E-03	1.50E-03	7.04E-04	4.00067E-06
360	3.69E-04	1.45E-03	4.57E-03	3.52E-03	1.59E-03	8.30E-04	4.66236E-06
370	3.60E-04	1.42E-03	4.59E-03	3.48E-03	1.59E-03	8.74E-04	4.49799E-06
380	3.51E-04	1.39E-03	4.51E-03	3.44E-03	1.60E-03	9.17E-04	4.34271E-06
390	3.43E-04	1.37E-03	4.44E-03	3.40E-03	1.60E-03	9.59E-04	4.19685E-06
400	3.35E-04	1.34E-03	4.37E-03	3.36E-03	1.60E-03	1.00E-03	4.05679E-06
410	3.28E-04	1.32E-03	4.30E-03	3.32E-03	1.60E-03	1.04E-03	3.92498E-06
420	3.21E-04	1.29E-03	4.23E-03	3.28E-03	1.60E-03	1.08E-03	3.7999E-06
430	3.14E-04	1.27E-03	4.17E-03	3.25E-03	1.60E-03	1.11E-03	3.68108E-06
440	3.08E-04	1.25E-03	4.11E-03	3.21E-03	1.60E-03	1.15E-03	3.5681E-06
450	3.01E-04	1.23E-03	4.04E-03	3.17E-03	1.60E-03	1.18E-03	3.46058E-06
460	2.95E-04	1.21E-03	3.99E-03	3.14E-03	1.59E-03	1.21E-03	3.35815E-06
470	2.90E-04	1.19E-03	3.93E-03	3.10E-03	1.59E-03	1.25E-03	3.26048E-06
480	2.84E-04	1.17E-03	3.87E-03	3.07E-03	1.58E-03	1.27E-03	3.16728E-06
490	2.79E-04	1.15E-03	3.82E-03	3.03E-03	1.58E-03	1.30E-03	3.07826E-06
500	2.74E-04	1.13E-03	3.77E-03	3.00E-03	1.57E-03	1.33E-03	2.99318E-06



Transfer Parameters for Airborne Releases - At Ground Level

Figure B1

Appendix C: Calculation of Transfer Parameters at NRB Intake Height - Sample Spreadsheets

Table D3 [2]

Coefficients	Stability A	B	C	D	E	F
a1	0.112	0.13	0.112	0.090	0.0309	0.0630
b1	1.06	0.95	0.92	0.889	0.895	0.783
a2	5.38E-04	8.52E-04	9.06E-04	1.35E-03	1.96E-03	1.38E-03
b2	0.315	0.75	0.718	0.688	0.684	0.672

Table D4 [2]

Roughness Length Zo (km)	c1	d1	c2	d2
1	1.56	0.048	6.25E-04	0.45
4	2.02	0.0369	7.78E-04	0.37
10	2.71828	0	0	0
40	5.16	-0.098	18.6	-0.225
100	7.37	-0.0957	4.26E-03	-0.6
400	11.7	-0.128	4.66E-04	-0.76

See Table C.2[2]

x	Calculation of g(x) for each stability A	B	C	D	E	F	Calculation of F(zo,x) for Zo=100 Units	Lengths	Sigma z-l A	B	C	D	E	F
10	1.28E+00	1.15E+00	9.27E-01	7.54E-01	4.74E-01	3.85E-01	5.917937	1.78E-00	2.28E+00	2.05E+00	1.65E+00	1.34E+00	8.42E-01	6.84E-01
20	2.66E+00	2.22E+00	1.72E+00	1.39E+00	0.73E-01	6.59E-01	5.54076	1.71E-00	4.56E+00	3.01E+00	2.99E+00	2.30E+00	1.50E+00	1.10E+00
30	4.09E+00	3.26E+00	2.53E+00	1.99E+00	1.25E+00	9.03E-01	5.331942	1.67E-00	6.84E+00	5.46E+00	4.24E+00	3.33E+00	2.10E+00	1.51E+00
40	5.53E+00	4.28E+00	3.25E+00	2.58E+00	1.61E+00	1.13E+00	5.188809	1.65E-00	9.10E+00	7.05E+00	5.42E+00	4.27E+00	2.68E+00	1.83E+00
60	6.99E+00	5.28E+00	4.03E+00	3.11E+00	1.96E+00	1.34E+00	5.080814	1.63E-00	1.14E+01	8.55E+00	6.56E+00	5.06E+00	3.19E+00	2.13E+00
60	8.46E+00	6.27E+00	4.78E+00	3.65E+00	2.30E+00	1.54E+00	4.994336	1.61E-00	1.36E+01	1.01E+0'	7.66E+00	5.87E+00	3.70E+00	2.43E+00
70	9.95E+00	7.24E+00	5.46E+00	4.18E+00	2.61E+00	1.74E+00	4.922492	1.59E-00	1.59E+01	1.15E+0'	8.73E+00	6.66E+00	4.20E+00	2.77E+00
80	1.14E+01	8.21E+00	6.15E+00	4.69E+00	2.9EE+00	1.92E+00	4.861194	1.58E-00	1.81E+01	1.30E+0'	9.77E+00	7.42E+00	4.68E+00	3.04E+00
90	1.29E+01	9.17E+00	6.8EE+00	5.20E+00	3.2EE+00	2.10E+00	4.80784	1.57E-00	2.03E+01	1.44E+0'	1.08E+01	8.16E+00	5.15E+00	3.33E+00
100	1.43E+01	1.01E+01	7.5EE+00	5.70E+00	3.5EE+00	2.28E+00	4.76068	1.56E-00	2.25E+01	1.5EE+0'	1.18E+01	8.89E+00	5.60E+00	3.63E+00
110	1.59E+01	1.11E+01	8.24E+00	6.19E+00	3.9CE+00	2.45E+00	4.718475	1.55E-00	2.47E+01	1.72E+0'	1.28E+01	9.60E+00	6.05E+00	3.9EE+00
120	1.74E+01	1.20E+01	8.91E+00	6.67E+00	4.2CE+00	2.62E+00	4.680326	1.54E-00	2.69E+01	1.8EE+0'	1.38E+01	1.03E+01	6.49E+00	4.04E+00
130	1.90E+01	1.29E+01	9.5EE+00	7.15E+00	4.5CE+00	2.78E+00	4.645549	1.54E-00	2.91E+01	1.9EE+0'	1.47E+01	1.10E+01	6.92E+00	4.23E+00
140	2.05E+01	1.39E+01	1.0EE+01	7.62E+00	4.8CE+00	2.95E+00	4.613622	1.53E-00	3.13E+01	2.1EE+0'	1.57E+01	1.16E+01	7.34E+00	4.5CE+00
150	2.20E+01	1.48E+01	1.0EE+01	8.09E+00	5.0CE+00	3.10E+00	4.584132	1.52E-00	3.35E+01	2.2EE+0'	1.66E+01	1.22E+01	7.75E+00	4.73E+00
160	2.35E+01	1.57E+01	1.1EE+01	8.55E+00	5.3EE+00	3.26E+00	4.55975	1.52E-00	3.57E+01	2.3EE+0'	1.75E+01	1.30E+01	8.16E+00	4.9EE+00
170	2.50E+01	1.66E+01	1.2EE+01	9.00E+00	5.6TE+00	3.41E+00	4.531207	1.51E-00	3.78E+01	2.5EE+0'	1.84E+01	1.36E+01	8.56E+00	5.13E+00
180	2.66E+01	1.75E+01	1.2EE+01	9.46E+00	5.9EE+00	3.56E+00	4.507283	1.51E-00	4.00E+01	2.6EE+0'	1.93E+01	1.42E+01	8.96E+00	5.3EE+00
190	2.81E+01	1.84E+01	1.3EE+01	9.91E+00	6.2EE+00	3.71E+00	4.484795	1.50E-00	4.21E+01	2.7EE+0'	2.02E+01	1.49E+01	9.35E+00	5.5TE+00
200	2.96E+01	1.93E+01	1.4EE+01	1.04E+01	6.5CE+00	3.88E+00	4.463587	1.50E-00	4.43E+01	2.8EE+0'	2.11E+01	1.55E+01	9.73E+00	5.7TE+00
210	3.11E+01	2.02E+01	1.4TE+01	1.09E+01	6.7EE+00	4.00E+00	4.443529	1.49E-00	4.64E+01	3.0EE+0'	2.19E+01	1.6'E+01	1.01E+01	5.9TE+00
220	3.26E+01	2.11E+01	1.5EE+01	1.12E+01	7.0EE+00	4.14E+00	4.424609	1.49E-00	4.85E+01	3.1EE+0'	2.28E+01	1.67E+01	1.05E+01	6.1EE+00
230	3.42E+01	2.19E+01	1.6CE+01	1.17E+01	7.3ZE+00	4.28E+00	4.406429	1.48E-00	5.07E+01	3.2EE+0'	2.37E+01	1.73E+01	1.09E+01	6.3EE+00
240	3.57E+01	2.28E+01	1.6EE+01	1.21E+01	7.5EE+00	4.42E+00	4.389205	1.48E-00	5.28E+01	3.3TE+0'	2.45E+01	1.79E+01	1.12E+01	6.5TE+00
250	3.72E+01	2.37E+01	1.7EE+01	1.25E+01	7.8EE+00	4.56E+00	4.372784	1.48E-00	5.49E+01	3.5CE+0'	2.54E+01	1.85E+01	1.16E+01	6.7EE+00
260	3.87E+01	2.46E+01	1.7EE+01	1.29E+01	8.1EE+00	4.70E+00	4.357047	1.47E-00	5.70E+01	3.6EE+0'	2.62E+01	1.90E+01	1.19E+01	6.9TE+00
270	4.02E+01	2.54E+01	1.8EE+01	1.34E+01	8.3EE+00	4.83E+00	4.34198	1.47E-00	5.91E+01	3.7EE+0'	2.70E+01	1.96E+01	1.23E+01	7.0EE+00
280	4.18C+01	2.63C+01	1.9CC+01	1.30C+01	8.64C+00	4.96C+00	4.327532	1.46C-00	6.12C+01	3.8EC+0'	2.79C+01	2.02C+01	1.27C+01	7.27C+00
290	4.33E+01	2.71E+01	1.9EE+01	1.42E+01	8.9CE+00	5.09E+00	4.313648	1.46E-00	6.33E+01	3.9TE+0'	2.88E+01	2.08E+01	1.30E+01	7.45E+00
300	4.48E+01	2.80E+01	2.0EE+01	1.46E+01	9.1EE+00	5.22E+00	4.300289	1.46E-00	6.53E+01	4.0EE+0'	2.95E+01	2.13E+01	1.33E+01	7.6ZE+00
310	4.63E+01	2.89E+01	2.0EE+01	1.50E+01	9.4CE+00	5.35E+00	4.28742	1.46E-00	6.74E+01	4.2CE+0'	3.03E+01	2.19E+01	1.37E+01	7.7TE+00
320	4.78E+01	2.97E+01	2.1EE+01	1.54E+01	9.6EE+00	5.48E+00	4.275008	1.45E-00	6.95E+01	4.3ZE+0'	3.11E+01	2.24E+01	1.40E+01	7.9EE+00
330	4.93E+01	3.06E+01	2.2CE+01	1.59E+01	9.9TE+00	5.61E+00	4.263022	1.45E-00	7.15E+01	4.4EE+0'	3.19E+01	2.30E+01	1.44E+01	8.1EE+00
340	5.09E+01	3.14E+01	2.2EE+01	1.62E+01	1.0ZE+01	5.73E+00	4.251437	1.45E-00	7.36E+01	4.5EE+0'	3.28E+01	2.35E+01	1.47E+01	8.2EE+00
350	5.24E+01	3.22E+01	2.3TE+01	1.66E+01	1.0EE+01	5.86E+00	4.240228	1.44E-00	7.57E+01	4.6CE+0'	3.34E+01	2.40E+01	1.50E+01	8.4EE+00
360	5.39E+01	3.31E+01	2.3TE+01	1.70E+01	1.0EE+01	5.98E+00	4.229372	1.44E-00	7.77E+01	4.7TE+0'	3.42E+01	2.46E+01	1.54E+01	8.6ZE+00
370	5.54E+01	3.39E+01	2.4EE+01	1.74E+01	1.0EE+01	6.10E+00	4.218849	1.44E-00	7.97E+01	4.8EE+0'	3.50E+01	2.5'E+01	1.57E+01	8.7TE+00
380	5.69E+01	3.48E+01	2.4EE+01	1.78E+01	1.1TE+01	6.22E+00	4.208841	1.44E-00	8.18E+01	4.9EE+0'	3.57E+01	2.56E+01	1.60E+01	8.9EE+00
390	5.84E+01	3.56E+01	2.5EE+01	1.82E+01	1.1EE+01	6.34E+00	4.199731	1.43E-00	8.38E+01	5.1TE+0'	3.65E+01	2.6'E+01	1.63E+01	9.1CE+00
400	5.99E+01	3.64E+01	2.6EE+01	1.86E+01	1.1EE+01	6.46E+00	4.189101	1.43E-00	8.58E+01	5.2EE+0'	3.72E+01	2.67E+01	1.66E+01	9.2EE+00
410	6.14E+01	3.72E+01	2.6EE+01	1.90E+01	1.1EE+01	6.58E+00	4.179739	1.43E-00	8.79E+01	5.3EE+0'	3.80E+01	2.72E+01	1.70E+01	9.4TE+00
420	6.29E+01	3.81E+01	2.7TE+01	1.94E+01	1.2TE+01	6.70E+00	4.170629	1.43E-00	8.99E+01	5.4EE+0'	3.88E+01	2.77E+01	1.73E+01	9.5EE+00
430	6.44E+01	3.89E+01	2.7TE+01	1.98E+01	1.2EE+01	6.81E+00	4.161761	1.43E-00	9.19E+01	5.5TE+0'	3.95E+01	2.82E+01	1.76E+01	9.7ZE+00
440	6.59E+01	3.97E+01	2.8EE+01	2.01E+01	1.2EE+01	6.93E+00	4.153121	1.42E-00	9.39E+01	5.6EE+0'	4.02E+01	2.87E+01	1.79E+01	9.8TE+00
450	6.74E+01	4.05E+01	2.8EE+01	2.05E+01	1.2EE+01	7.04E+00	4.1447	1.42E-00	9.59E+01	5.7EE+0'	4.10E+01	2.92E+01	1.82E+01	1.0CE+01
460	6.89E+01	4.13E+01	2.9TE+01	2.09E+01	1.3CE+01	7.16E+00	4.136497	1.42E-00	9.79E+01	5.8TE+0'	4.17E+01	2.97E+01	1.85E+01	1.02E+01
470	7.04E+01	4.21E+01	2.9EE+01	2.13E+01	1.3EE+01	7.27E+00	4.128473	1.42E-00	9.00E+01	6.0EE+0'	4.24E+01	3.02E+01	1.88E+01	1.03E+01
480	7.19E+01	4.30E+01	3.0EE+01	2.17E+01	1.3EE+01	7.38E+00	4.120648	1.42E-00	1.02E+02	6.0EE+0'	4.32E+01	3.07E+01	1.91E+01	1.05E+01
490	7.34E+01	4.38E+01	3.1CE+01	2.20E+01	1.37E+01	7.50E+00	4.113005	1.41E-00	1.04E+02	6.1EE+0'	4.39E+01	3.12E+01	1.94E+01	1.0SE+01
500	7.49E+01	4.46E+01	3.1EE+01	2.24E+01	1.3EE+01	7.61E+00	4.105536	1.41E-00	1.06E+02	6.2EE+0'	4.46E+01	3.16E+01	1.97E+01	1.07E+01

Stack Height			
h=	17 m	z	11.3

Calculation of F(h,z,sima-z-i) from eq 5.2

A	B	C	D	E	F
4.37E-02	2.11E-02	2.54E-03	1.19E-04	1.14E-10	8.23E-16
4.58E-01	3.26E-01	1.63E-01	5.70E-02	7.30E-04	2.91E-06
7.07E-01	5.80E-01	4.05E-01	2.30E-01	2.49E-02	8.14E-04
8.30E-01	7.21E-01	5.75E-01	4.01E-01	1.00E-01	9.00E-03
9.27E-01	8.06E-01	6.86E-01	5.30E-01	2.03E-01	3.26E-02
1.03E+00	8.72E-01	7.59E-01	6.24E-01	3.06E-01	7.12E-02
1.14E+00	9.35E-01	8.13E-01	6.93E-01	3.98E-01	1.20E-01
1.26E+00	1.00E+00	8.59E-01	7.45E-01	4.76E-01	1.72E-01
1.34E+00	1.07E+00	9.02E-01	7.86E-01	5.42E-01	2.26E-01
1.42E+00	1.14E+00	9.46E-01	8.20E-01	5.96E-01	2.77E-01
1.49E+00	1.20E+00	9.92E-01	8.51E-01	6.42E-01	3.26E-01
1.55E+00	1.26E+00	1.04E+00	8.81E-01	6.80E-01	3.70E-01
1.60E+00	1.32E+00	1.08E+00	9.10E-01	7.12E-01	4.11E-01
1.65E+00	1.37E+00	1.13E+00	9.40E-01	7.40E-01	4.49E-01
1.69E+00	1.42E+00	1.18E+00	9.70E-01	7.64E-01	4.83E-01
1.72E+00	1.46E+00	1.22E+00	1.00E+00	7.86E-01	5.14E-01
1.74E+00	1.50E+00	1.26E+00	1.03E+00	8.05E-01	5.43E-01
1.77E+00	1.54E+00	1.30E+00	1.06E+00	8.23E-01	5.69E-01
1.79E+00	1.57E+00	1.34E+00	1.09E+00	8.40E-01	5.92E-01
1.81E+00	1.60E+00	1.37E+00	1.12E+00	8.57E-01	6.14E-01
1.82E+00	1.62E+00	1.40E+00	1.15E+00	8.73E-01	6.34E-01
1.84E+00	1.65E+00	1.43E+00	1.18E+00	8.89E-01	6.52E-01
1.85E+00	1.67E+00	1.46E+00	1.21E+00	9.05E-01	6.69E-01
1.86E+00	1.69E+00	1.49E+00	1.24E+00	9.21E-01	6.84E-01
1.87E+00	1.71E+00	1.51E+00	1.26E+00	9.37E-01	6.99E-01
1.88E+00	1.72E+00	1.53E+00	1.29E+00	9.53E-01	7.12E-01
1.89E+00	1.74E+00	1.56E+00	1.31E+00	9.69E-01	7.24E-01
1.89E+00	1.75E+00	1.58E+00	1.34E+00	9.86E-01	7.36E-01
1.90E+00	1.77E+00	1.59E+00	1.36E+00	1.00E+00	7.47E-01
1.91E+00	1.78E+00	1.61E+00	1.38E+00	1.02E+00	7.57E-01
1.91E+00	1.79E+00	1.63E+00	1.40E+00	1.04E+00	7.67E-01
1.92E+00	1.80E+00	1.64E+00	1.42E+00	1.05E+00	7.76E-01
1.92E+00	1.81E+00	1.66E+00	1.44E+00	1.07E+00	7.84E-01
1.93E+00	1.82E+00	1.67E+00	1.46E+00	1.08E+00	7.93E-01
1.93E+00	1.82E+00	1.68E+00	1.47E+00	1.10E+00	8.01E-01
1.93E+00	1.83E+00	1.70E+00	1.49E+00	1.12E+00	8.08E-01
1.94E+00	1.84E+00	1.71E+00	1.50E+00	1.13E+00	8.16E-01
1.94E+00	1.85E+00	1.72E+00	1.52E+00	1.15E+00	8.23E-01
1.94E+00	1.85E+00	1.73E+00	1.53E+00	1.16E+00	8.30E-01
1.94E+00	1.86E+00	1.74E+00	1.55E+00	1.18E+00	8.37E-01
1.95E+00	1.86E+00	1.75E+00	1.56E+00	1.19E+00	8.43E-01
1.95E+00	1.87E+00	1.76E+00	1.57E+00	1.21E+00	8.50E-01
1.95E+00	1.87E+00	1.76E+00	1.58E+00	1.22E+00	8.56E-01
1.95E+00	1.88E+00	1.77E+00	1.60E+00	1.24E+00	8.63E-01
1.96E+00	1.88E+00	1.78E+00	1.61E+00	1.25E+00	8.69E-01
1.96E+00	1.89E+00	1.79E+00	1.62E+00	1.26E+00	8.75E-01
1.96E+00	1.89E+00	1.79E+00	1.63E+00	1.28E+00	8.81E-01
1.96E+00	1.89E+00	1.80E+00	1.64E+00	1.29E+00	8.88E-01
1.96E+00	1.90E+00	1.80E+00	1.65E+00	1.30E+00	8.94E-01
1.96E+00	1.90E+00	1.81E+00	1.65E+00	1.31E+00	9.00E-01

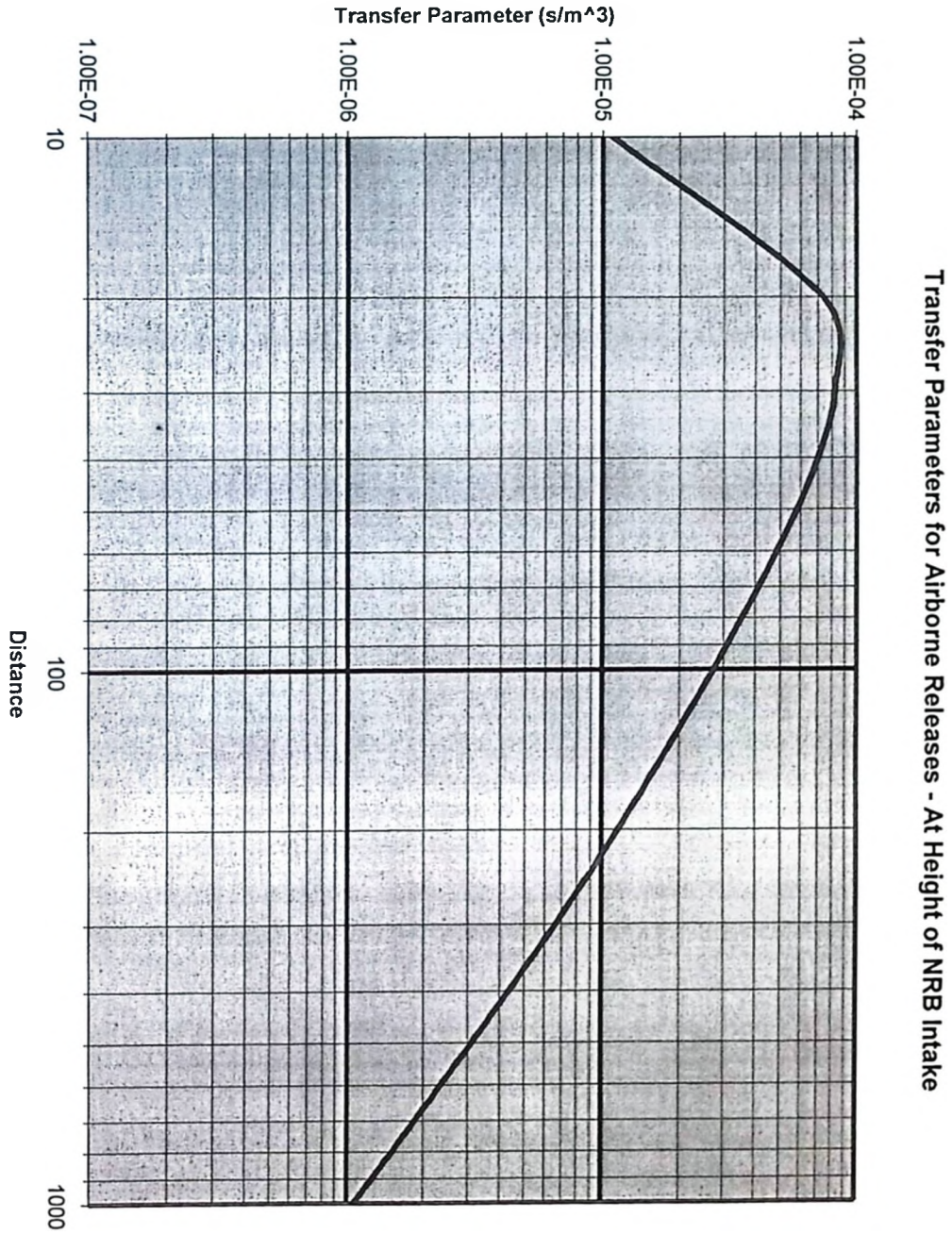
Calculation of Pc1 for Sector 1 from equation 5.2

Sector	Direction	Stability	Persistence	Speed (m/s)
1	E	A	0.022984	2
		B	0.095578	2.7
		C	0.388375	4.3
		D	0.356609	6.5
		E	0.076471	3.4
		F	0.054970	2.2
			0.100000	
		(2pi)^(3/2)/h=	0.984252	

Terrain sum

x	A	B	C	D	E	F	PC1
10	2.82E-04	3.60E-04	1.39E-04	4.66E-06	3.03E-12	3.01E-17	1.08864E-05
20	1.47E-03	3.16E-03	4.33E-03	1.31E-03	1.09E-05	6.45E-08	7.35561E-05
30	1.52E-03	3.52E-03	8.50E-03	3.00E-03	2.07E-04	1.35E-05	8.17200E-05
40	1.34E-03	3.78E-03	9.59E-03	5.22E-03	8.49E-04	1.21E-04	7.05522E-05
50	1.20E-03	3.47E-03	9.44E-03	5.75E-03	1.43E-03	3.73E-04	5.86252E-05
60	1.11E-03	3.19E-03	8.95E-03	5.83E-03	1.86E-03	7.17E-04	4.87883E-05
70	1.06E-03	2.99E-03	8.41E-03	5.71E-03	2.13E-03	1.08E-03	4.12778E-05
80	1.01E-03	2.84E-03	7.94E-03	5.51E-03	2.29E-03	1.42E-03	3.54832E-05
90	9.69E-04	2.74E-03	7.56E-03	5.28E-03	2.37E-03	1.71E-03	3.09795E-05
100	9.27E-04	2.66E-03	7.24E-03	5.06E-03	2.39E-03	1.95E-03	2.73075E-05
110	8.87E-04	2.59E-03	7.01E-03	4.87E-03	2.39E-03	2.14E-03	2.44057E-05
120	8.47E-04	2.52E-03	6.82E-03	4.69E-03	2.36E-03	2.29E-03	2.19833E-05
130	8.09E-04	2.46E-03	6.68E-03	4.55E-03	2.32E-03	2.40E-03	1.99496E-05
140	7.73E-04	2.39E-03	6.53E-03	4.42E-03	2.27E-03	2.49E-03	1.82183E-05
150	7.39E-04	2.33E-03	6.40E-03	4.32E-03	2.22E-03	2.55E-03	1.67265E-05
160	7.07E-04	2.27E-03	6.29E-03	4.23E-03	2.17E-03	2.60E-03	1.54277E-05
170	6.77E-04	2.21E-03	6.16E-03	4.16E-03	2.12E-03	2.63E-03	1.42872E-05
180	6.49E-04	2.15E-03	6.06E-03	4.09E-03	2.07E-03	2.65E-03	1.32781E-05
190	6.24E-04	2.10E-03	5.97E-03	4.03E-03	2.02E-03	2.66E-03	1.23795E-05
200	5.99E-04	2.04E-03	5.87E-03	3.98E-03	1.98E-03	2.66E-03	1.15748E-05
210	5.77E-04	1.99E-03	5.77E-03	3.93E-03	1.94E-03	2.65E-03	1.08507E-05
220	5.56E-04	1.94E-03	5.67E-03	3.88E-03	1.91E-03	2.64E-03	1.01961E-05
230	5.36E-04	1.89E-03	5.57E-03	3.84E-03	1.87E-03	2.63E-03	9.60216E-06
240	5.18E-04	1.85E-03	5.48E-03	3.79E-03	1.84E-03	2.61E-03	9.0612E-06
250	5.00E-04	1.80E-03	5.38E-03	3.75E-03	1.82E-03	2.59E-03	8.56692E-06
260	4.84E-04	1.76E-03	5.29E-03	3.71E-03	1.79E-03	2.57E-03	8.11392E-06
270	4.69E-04	1.72E-03	5.20E-03	3.67E-03	1.77E-03	2.55E-03	7.6976E-06
280	4.55E-04	1.68E-03	5.11E-03	3.63E-03	1.75E-03	2.53E-03	7.314E-06
290	4.41E-04	1.64E-03	5.03E-03	3.59E-03	1.73E-03	2.51E-03	6.95997E-06
300	4.28E-04	1.60E-03	4.94E-03	3.55E-03	1.72E-03	2.48E-03	6.63172E-06
310	4.16E-04	1.57E-03	4.86E-03	3.51E-03	1.70E-03	2.46E-03	6.32746E-06
320	4.05E-04	1.54E-03	4.78E-03	3.47E-03	1.69E-03	2.43E-03	6.04466E-06
330	3.94E-04	1.50E-03	4.70E-03	3.43E-03	1.67E-03	2.41E-03	5.7813E-06
340	3.84E-04	1.47E-03	4.63E-03	3.40E-03	1.66E-03	2.39E-03	5.53562E-06
350	3.74E-04	1.44E-03	4.55E-03	3.36E-03	1.65E-03	2.36E-03	5.30604E-06
360	3.65E-04	1.42E-03	4.48E-03	3.32E-03	1.64E-03	2.34E-03	5.09117E-06
370	3.57E-04	1.39E-03	4.41E-03	3.29E-03	1.62E-03	2.32E-03	4.88976E-06
380	3.48E-04	1.36E-03	4.34E-03	3.25E-03	1.61E-03	2.30E-03	4.7007E-06
390	3.40E-04	1.34E-03	4.28E-03	3.22E-03	1.60E-03	2.28E-03	4.52299E-06
400	3.33E-04	1.31E-03	4.21E-03	3.18E-03	1.59E-03	2.26E-03	4.35572E-06
410	3.25E-04	1.29E-03	4.15E-03	3.15E-03	1.58E-03	2.24E-03	4.19808E-06
420	3.19E-04	1.27E-03	4.09E-03	3.12E-03	1.57E-03	2.22E-03	4.04934E-06
430	3.12E-04	1.25E-03	4.03E-03	3.08E-03	1.56E-03	2.20E-03	3.90884E-06
440	3.06E-04	1.22E-03	3.97E-03	3.05E-03	1.55E-03	2.18E-03	3.77596E-06
450	2.99E-04	1.20E-03	3.92E-03	3.02E-03	1.55E-03	2.17E-03	3.65017E-06
460	2.94E-04	1.18E-03	3.87E-03	2.99E-03	1.54E-03	2.15E-03	3.53095E-06
470	2.88E-04	1.17E-03	3.81E-03	2.96E-03	1.53E-03	2.14E-03	3.41785E-06
480	2.83E-04	1.15E-03	3.76E-03	2.93E-03	1.52E-03	2.12E-03	3.31046E-06
490	2.77E-04	1.13E-03	3.71E-03	2.90E-03	1.51E-03	2.11E-03	3.20839E-06
500	2.72E-04	1.11E-03	3.66E-03	2.87E-03	1.50E-03	2.09E-03	3.11129E-06

Figure C1


Transfer Parameter (s/m^3)
1.00E-07
1.00E-06
1.00E-05
1.00E-04
10
100
1000
Distance
Transfer Parameters for Airborne Releases - At Height of NRB Intake

Chapter 5: Preliminary Assessment of Emergency Plan

Introduction

In Chapter 1, the importance of evaluating the impact of proposed ^{125}I operations on the facility's emergency plan is emphasized. This chapter describes a scoping calculation that was performed in order to assess the potential of ^{125}I operations to trigger the top level MNR emergency procedure (the "Type D" emergency procedure).

Discussion

In Canada, the methods for assessment of potential consequences from nuclear facility emergencies are prescribed in a national standard[1]. The standard lays out the methods for calculating doses from exposure to the plume following accidental releases based on a modified Gaussian plume model. Dose consequences are assessed in a manner similar to that described in Chapter 4. Dilution factors for points of interest were determined. In this case, a dilution factor for each weather stability class is determined. Because this is a bounding calculation, the dilution factor was assessed along the plume centre-line where the highest doses will occur. This can be contrasted to the approach for long-term release calculations discussed in Chapter 4 where average weather conditions are used and the dilution factor is averaged over an entire compass sector. Dose conversion factors for ^{125}I are calculated from the product of breathing rate and the dose coefficient. Doses are then calculated as the product of the dilution factor, the activity released and the dose conversion factor. For ^{125}Xe, the exposure pathway of concern is immersion. The dose can be calculated from the product of the dilution factor, the activity released and the appropriate dose conversion factor.

This general method was applied to four hypothetical release situations. Consequences of releases of 2 TBQ of ^{125}I were evaluated for both release through the stack and for release at ground level. The activity chosen corresponds to the licensed limit for the inventory of ^{125}I in one production rig. This is the maximum amount likely to be in process and available for release at any given time. Consequences of the release of 150 TBq of ^{125}Xe were evaluated in the same fashion. The activity in this case was chosen because it is the ^{125}Xe activity that will eventually decay to produce the limiting inventory of ^{125}I in a rig. As the ^{125}I is produced over several production cycles, this represents a conservative upper bound on the ^{125}Xe activity likely to be present at the facility at any time. Both stack and ground height releases were assessed because of the possibility of a release occurring at some point when the containment system of the facility is compromised. ^{126}I was listed in Chapter 1 as one of the radionuclides of radiological significance in ^{125}I production. While it contributes significantly to chronic exposures, that activity of ^{126}I produced is too small to have an impact on doses from accidental releases.

It should be noted that no attempt has been made in this analysis to create a realistic estimate of the releases that may occur following an accident at the facility. The effects of filtration, plate out, solubility in pool water and other factors have been disregarded. The purpose of this calculation was to bound the potential dose consequences and to determine whether or not additional analysis was required.

The criteria against which the doses are evaluated arise from the MNR emergency plan[2]. This plan requires activation of the Type D emergency procedure when a radiation hazard is introduced outside of MNR which could lead to an effective dose of 1 mSv or a gamma radiation field greater than 0.25 mSv/h.

It is shown in the attached report that ^{125}I operations cannot be excluded as a potential trigger for the Type D emergency procedure. The centre-line doses for 2 TBq releases of ^{125}I exceed the 1 mSv effective dose criteria for several hundred metres out from the reactor in both the stack and ground-level release scenarios. Similarly, the peak dose rates from rapidly occurring ^{125}Xe releases can exceed the 0.25 mSv/h criterion.

Further analysis is underway to determine realistic dose estimates for credible accidents.

Summary

A bounding assessment of the consequences of accidental release of radionuclides from the ^{125}I has been performed in a manner consistent with the national standard for nuclear facility accident analysis. ^{125}I operations cannot be excluded as a trigger for the MNR Type D emergency procedure on the basis of this assessment. Additional analysis is required to determine realistic dose consequences from credible accidents at the facility.

Attached Report

The McMaster University Health Physics Department report HP-MNR-03-01, Preliminary Assessment of I-125 Operations as a Trigger for a Type D Emergency (2003) is attached to this chapter.

The report was prepared solely by the author, using the references cited in the report as guidance. This report was prepared primarily for consideration by the committee responsible for MNR emergency planning. The report describes a bounding assessment of the dose consequences for ^{125}I and ^{125}Xe releases from the facility

Minor editorial changes have been made to the report for inclusion in this thesis, and proprietary information has been removed.

References

[1] Canadian Standards Association, Guidelines for Calculating Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors, CAN/CSA-N288.2-M91 (1991).

[2] McMaster Nuclear Reactor, The MNR Emergency Preparedness Plan, EP7000 (1994).

McMaster University **Health Physics**

Health Physics Report

HP-MNR 03-01 Preliminary Assessment of I-125 Operations as Trigger for a Type D Emergency

2003 April 25

Prepared by: D. M. Tucker
Senior Health Physicist

Health Physics Report

HP-MNR-03-01 Preliminary Assessment of I-125 Operations as a Trigger for a Type D Emergency

Reason for Report

This report has been prepared to document an assessment performed to determine the potential for releases arising from I-125 operations to trigger a Type D Emergency. This internal report was prepared for the Emergency Preparedness Review Group (a sub-committee of the Reactor Operational Control Committee) and for reference by MNR (McMaster Nuclear Reactor) and Health Physics personnel.

The results presented in this report do not represent an estimate of doses that would result in the event of an accidental release during iodine production operations. No consideration has been given to mitigating factors such as filtration and plate out. The calculations were performed to bound the upper limit of potential consequences without regard to probability.

Background

The MNR Type D Emergency procedures are currently being revised to account for several significant changes in the regulatory and operating environments, updated source terms derived in the revision of the Safety Analysis Report and changes in the internal and external emergency organizations. As part of this revision, the various incidents and upsets capable of triggering a Type D Emergency have been reviewed. This report documents the preliminary assessment of the I-125 Operations.

A Type D emergency is deemed to exist in the event of an incident that might create a hazard of radiation exposure to persons outside the Reactor Building. The phrase "a hazard of radiation exposure" is interpreted as an exposure that could lead to an effective dose in excess of 1 mSv (100 mrem) or the introduction of a gamma radiation field greater than 0.25 mSv/hr (25 mrem/hr) at 25 meters from the Reactor Building [1].

Analysis

Estimates of effective doses from accidental releases arising from I-125 Operations were assessed in a manner consistent with the applicable CSA Standard[2]. Releases of I-125 were assessed with respect to the possibility of causing effective doses in excess of 1 mSv. Releases of Xe-125, the precursor to I-125 in the production process and the only other radiologically significant radionuclide associated with production operations, were assessed with respect to the potential for the introduction of radiation fields greater than 0.25 mSv/h (25 mrem/h) at 25 m from the reactor building.

Releases of I-125:

For I-125 releases, two scenarios were considered. The first is a release of 50 Ci (1.85E12 Bq) from the stack; the second is a release of the same magnitude at ground level. The magnitude of the release was selected to correspond to the current maximum inventory permitted per production rig. In both cases, an instantaneous release was assumed, which provides the highest doses. No credit is taken for hold-up or filtration.

A stack height of 18 m was used in the calculation. Building wake effects were conservatively neglected. A nominal wind speed of 2 ms^{-1} was used for the calculations. The application of this low

wind speed to some of the weather classes is an additional conservatism – the dilution factor scales inversely proportional to the mean wind speed. Equation 5.2 from Reference [2] was evaluated for y=o (plume centerline) at various values of distance from the reactor. The calculated diffusion coefficients are shown in Table A1 of Attachment A.

Dose Conversion Factors (DCFs) were determined from the product of the breathing rate from ICRP Publication 66[3] and the dose per unit intake from ICRP Publication 72[4]. Dose Conversion factors were derived for adults and for 3-month-old infants, both assumed to be exercising lightly. The precise chemical form which may be involved in a hypothetical release is not known, so elemental iodine has been conservatively assumed. The relevant data and conversion factors are shown in Table 1.

Table 1: Dose Conversion Factors for I-125

Parameter		Value
Dose Coefficient – Elemental I-125	Adults	1.4E-8 (Sv/Bq)
Dose Coefficient – Methyl I-125	Adults	1.1E-8 (Sv/Bq)
Dose Coefficient – Elemental I-125	Infant (3 month)	4.7E-8 (Sv/Bq)
Dose Coefficient – Methyl I-125	Infant (3 month)	3.7E-8 (Sv/Bq)
Breathing Rate – Light Exercise	Adults	1.5 (m^3/h)
Breathing Rate – Light Exercise	Infant (3 month)	0.19 (m^3/h)
Dose Conversion Factor	Adult	5.8E-12 (Sv m^3/Bq s)
Dose Conversion Factor	Infant (3 month)	2.5E-12 (Sv m^3/Bq s)

The committed effective dose from I-125 exposure was calculated according to equation 9.1 of Reference [2] as the product of the diffusion coefficient, the activity released and the dose conversion factor. The calculated values are listed in Table A2 of Attachment A.

The results are shown in Figure 1.

For ground level releases of I-125, the calculation of diffusion coefficient and effective doses was performed in the manner described above for a value of H=0. The calculated values are shown in Table A3 and A4.

The results are shown in Figure 2.

Releases of Xe-125

An upper bound on the Xe-125 activity that may be released at one time is taken as that activity required to decay to 50 Ci (2 TBq) of I-125. Neglecting losses, this activity is given by the inverse of the

ratio of the decay constants of the radionuclides. The upper bound of Xe-125 considered is thus 4 kCi (150 TBq).

Calculations of Diffusion Coefficients were performed as for I-125.

The committed effective dose from Xe-125 exposure was calculated according to equation 9.1 of Reference [2] as the product of the diffusion coefficient, the activity released and the dose conversion factor. The values for stack and ground-level releases are listed in Table A5 and A6 of Attachment A.

A Dose Conversion Factor for adults is published in ICRP Publication 72 [4].

Table 2: Dose Conversion Factor for Xe-125

Parameter		Value
Dose Conversion Factor	Adult	9.3 E-10 (Sv m^3/Bq d)
		3.9 E-11 (Sv m^3/Bq h)

The results for stack and ground-level releases are shown in Figure 3 and Figure 4 respectively.

From the dose conversion factor, it can be determined that a dose rate of 0.25 mSv/h will result from a semi-infinite cloud of Xe-125 with a concentration of 6.4E6 Bq/m^3. The reactor is exhausted at a rate of approximately 100 m^3/min. In the bounding case, the entire inventory is released over approximately 10 minutes, so that the highest conceivable outlet concentration is 1.5 E11 Bq/m^3. Therefore, areas with diffusion coefficients of greater than 4.2 E-5 s m^{-3} need be considered as potentially having radiation fields (transiently) exceeding 0.25 mSv/h. As can be seen in Tables A5 and A6 of Attachment A, this is the case for several hundred meters in the bounding analysis of both stack and ground-level releases. Releases over a longer time period, of say 100 minutes, would not result in radiation fields exceeding 0.25 mSv/h due to the lower initial concentration and greater dilution with wind meander.

Conclusions

1. I-125 Operations cannot be excluded, on the basis of this analysis, as possible triggers for a Type D Emergency. The upper bound for individual dose in stack and ground-level releases of I-125 exceeds 1 mSv. The upper bound for radiation fields caused by a release of Xe-125 exceeds 0.25 mSv/h (albeit for a short duration).

2. Further investigation should be performed to determine and analyze the worst credible accident (with a frequency >10^{-6}) associated with I-125 production operations.

Attachments

A Calculated Values (spreadsheets)

References

[1] The MNR Emergency Preparedness Plan, March 1994, McMaster Nuclear Reactor, Hamilton, Ontario.

[2] CAN/CSA-N288.2-M91, Guidelines for Calculating Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactor, Canadian Standards Association, Toronto, Canada, 1991.

[3] ICRP Publication 66, Human Respiratory Tract Model for Radiological Protection, Annals of the ICRP Vol 24 No. 1-3, 1994 Pergamon Press, Oxford, United Kingdom.

[4] ICRP Publication 72., Age-Dependant Doses to Members of the Public from Intake of Radionuclides: Part 5 – Compilation of Ingestion and Inhalation Dose Coefficients, Annals of the ICRP Vol 26 No.1, 1996. Pergamon Press, Oxford, United Kingdom.

Figures

Figure 1

Effective Dose on Plume Centreline - 2TBq Release - Stack Release

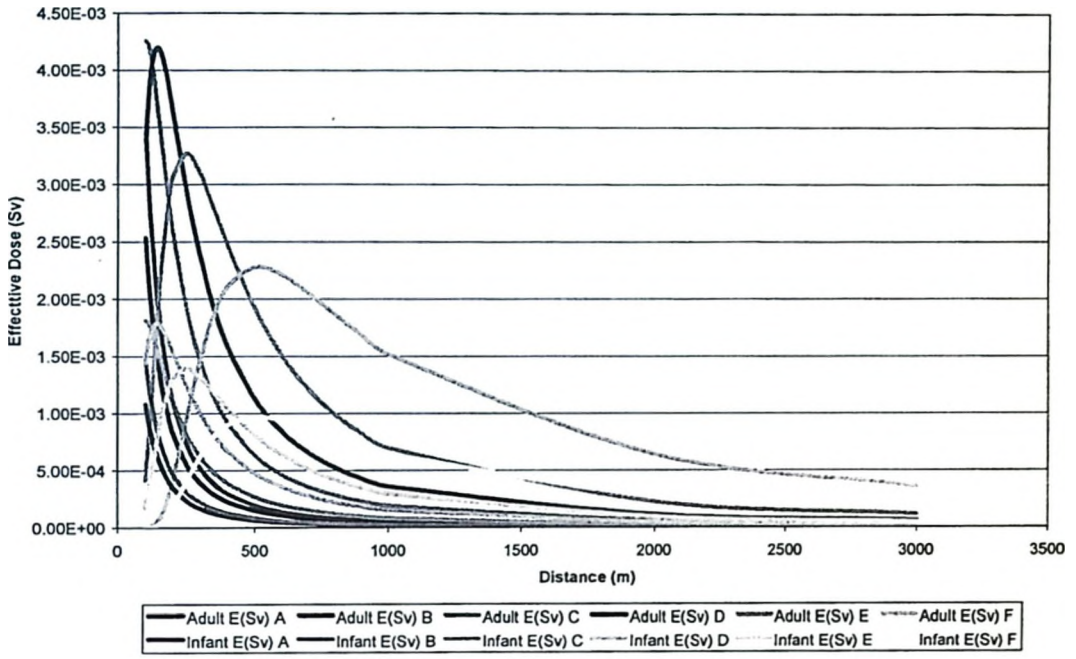


Figure 2

Effective Dose on Plume Centreline - 2TBq Release - Ground Release

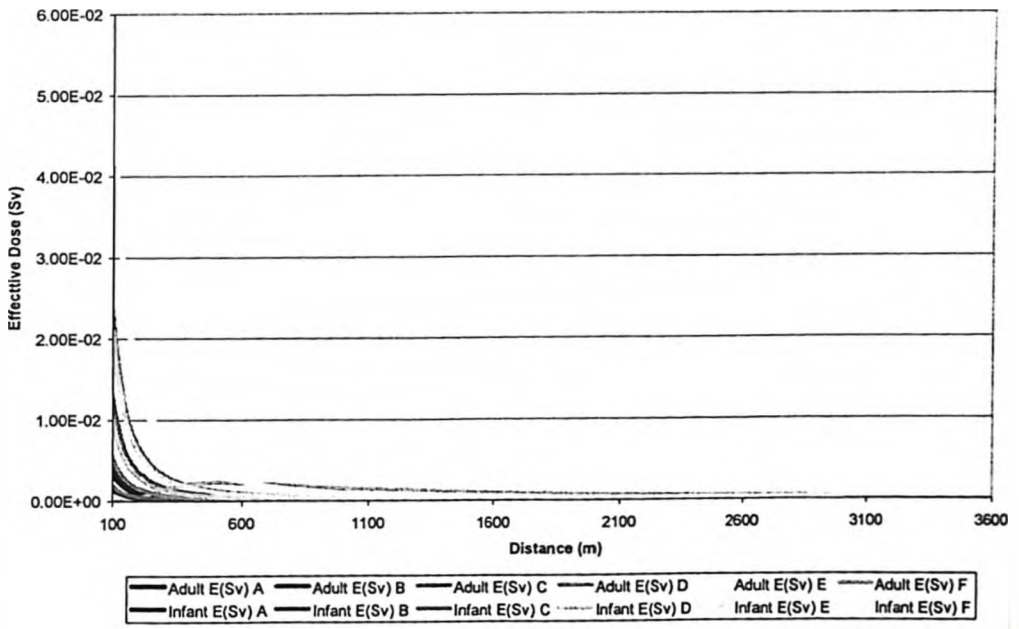


Figure 3

Effective Dose on Plume Centreline - Xe-125 150TBq Release - Stack Release

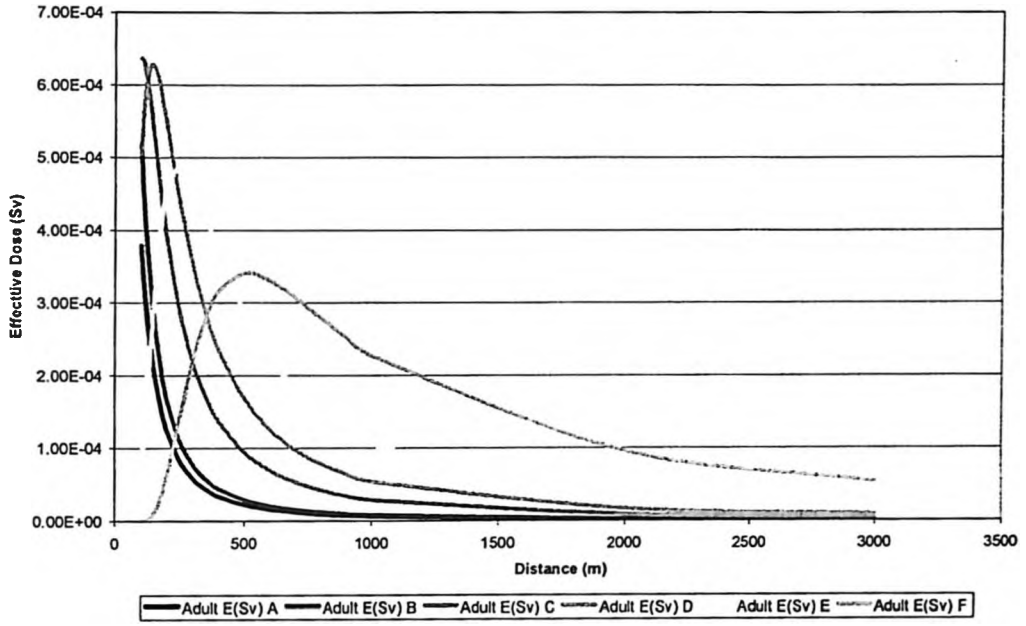
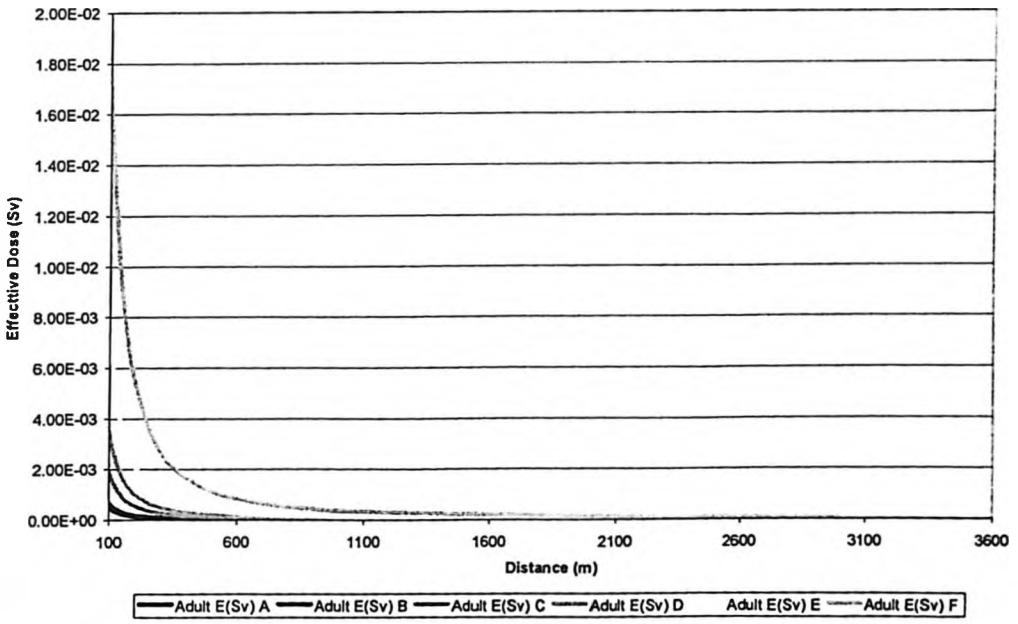


Figure 4

Effective Dose on Plume Centreline -Xe-125 150 TBq Release - Ground Release



Attachment A – Calculations

Calculations for I-125 Stack Release

Table A1 – Diffusion Coefficients

	g(x)						F(Zo,x)	Sigma Z						Sigma Y					
x	A	B	C	D	E	F		A	B	C	D	E	F	A	B	C	D	E	F
100	1.44E+01	1.47E+01	7.70E+00	5.85E+00	3.73E+00	2.34E+00	1.560	2.25E+01	2.29E+01	1.20E+01	9.12E+00	5.82E+00	3.65E+00	2.19E+01	1.59E+01	1.09E+01	7.96E+00	5.97E+00	3.98E+00
110	1.59E+01	1.63E+01	8.40E+00	6.36E+00	4.06E+00	2.52E+00	1.551	2.47E+01	2.52E+01	1.30E+01	9.87E+00	6.30E+00	3.91E+00	2.41E+01	1.75E+01	1.20E+01	8.75E+00	6.56E+00	4.38E+00
120	1.74E+01	1.78E+01	9.10E+00	6.87E+00	4.39E+00	2.70E+00	1.543	2.69E+01	2.75E+01	1.40E+01	1.06E+01	6.77E+00	4.17E+00	2.62E+01	1.91E+01	1.31E+01	9.54E+00	7.16E+00	4.77E+00
130	1.90E+01	1.94E+01	9.79E+00	7.37E+00	4.71E+00	2.87E+00	1.535	2.91E+01	2.98E+01	1.50E+01	1.13E+01	7.24E+00	4.41E+00	2.84E+01	2.07E+01	1.42E+01	1.03E+01	7.75E+00	5.17E+00
140	2.05E+01	2.10E+01	1.05E+01	7.87E+00	5.03E+00	3.04E+00	1.529	3.13E+01	3.20E+01	1.60E+01	1.20E+01	7.70E+00	4.65E+00	3.06E+01	2.22E+01	1.53E+01	1.11E+01	8.34E+00	5.56E+00
150	2.20E+01	2.25E+01	1.12E+01	8.37E+00	5.35E+00	3.21E+00	1.522	3.35E+01	3.43E+01	1.70E+01	1.27E+01	8.15E+00	4.89E+00	3.28E+01	2.38E+01	1.64E+01	1.19E+01	8.93E+00	5.96E+00
160	2.35E+01	2.41E+01	1.18E+01	8.86E+00	5.67E+00	3.38E+00	1.516	3.57E+01	3.66E+01	1.79E+01	1.34E+01	8.60E+00	5.12E+00	3.49E+01	2.54E+01	1.75E+01	1.27E+01	9.52E+00	6.35E+00
170	2.50E+01	2.57E+01	1.25E+01	9.35E+00	5.98E+00	3.54E+00	1.510	3.78E+01	3.89E+01	1.89E+01	1.41E+01	9.04E+00	5.35E+00	3.71E+01	2.70E+01	1.85E+01	1.35E+01	1.01E+01	6.74E+00
180	2.65E+01	2.73E+01	1.32E+01	9.83E+00	6.30E+00	3.70E+00	1.505	4.00E+01	4.11E+01	1.98E+01	1.48E+01	9.48E+00	5.58E+00	3.92E+01	2.85E+01	1.96E+01	1.43E+01	1.07E+01	7.14E+00
190	2.81E+01	2.89E+01	1.38E+01	1.03E+01	6.61E+00	3.86E+00	1.500	4.21E+01	4.34E+01	2.08E+01	1.55E+01	9.91E+00	5.80E+00	4.14E+01	3.01E+01	2.07E+01	1.51E+01	1.13E+01	7.53E+00
200	2.96E+01	3.05E+01	1.45E+01	1.08E+01	6.91E+00	4.02E+00	1.495	4.43E+01	4.57E+01	2.17E+01	1.61E+01	1.03E+01	6.02E+00	4.36E+01	3.17E+01	2.18E+01	1.58E+01	1.19E+01	7.92E+00
250	3.72E+01	3.86E+01	1.78E+01	1.31E+01	8.43E+00	4.79E+00	1.475	5.49E+01	5.70E+01	2.62E+01	1.94E+01	1.24E+01	7.06E+00	5.43E+01	3.95E+01	2.72E+01	1.98E+01	1.48E+01	9.88E+00
300	4.48E+01	4.68E+01	2.10E+01	1.54E+01	9.91E+00	5.52E+00	1.458	6.53E+01	6.82E+01	3.06E+01	2.25E+01	1.45E+01	8.05E+00	6.50E+01	4.73E+01	3.25E+01	2.36E+01	1.77E+01	1.18E+01
350	5.24E+01	5.50E+01	2.41E+01	1.77E+01	1.14E+01	6.22E+00	1.444	7.57E+01	7.94E+01	3.49E+01	2.56E+01	1.64E+01	8.99E+00	7.57E+01	5.50E+01	3.78E+01	2.75E+01	2.06E+01	1.38E+01
400	5.99E+01	6.32E+01	2.73E+01	1.99E+01	1.28E+01	6.90E+00	1.432	8.58E+01	9.06E+01	3.90E+01	2.85E+01	1.83E+01	9.89E+00	8.63E+01	6.28E+01	4.31E+01	3.14E+01	2.35E+01	1.57E+01
500	7.49E+01	7.99E+01	3.34E+01	2.42E+01	1.56E+01	8.21E+00	1.412	1.06E+02	1.13E+02	4.71E+01	3.42E+01	2.20E+01	1.16E+01	1.07E+02	7.81E+01	5.37E+01	3.90E+01	2.93E+01	1.95E+01
600	8.98E+01	9.66E+01	3.93E+01	2.84E+01	1.83E+01	9.46E+00	1.395	1.25E+02	1.35E+02	5.49E+01	3.97E+01	2.56E+01	1.32E+01	1.28E+02	9.32E+01	6.41E+01	4.66E+01	3.50E+01	2.33E+01
700	1.04E+02	1.13E+02	4.52E+01	3.25E+01	2.10E+01	1.07E+01	1.382	1.44E+02	1.57E+02	6.25E+01	4.50E+01	2.90E+01	1.47E+01	1.49E+02	1.08E+02	7.44E+01	5.41E+01	4.06E+01	2.71E+01
800	1.19E+02	1.30E+02	5.10E+01	3.66E+01	2.36E+01	1.18E+01	1.373	1.63E+02	1.79E+02	6.98E+01	5.01E+01	3.24E+01	1.62E+01	1.69E+02	1.23E+02	8.47E+01	6.16E+01	4.62E+01	3.08E+01
900	1.33E+02	1.47E+02	5.66E+01	4.06E+01	2.62E+01	1.30E+01	1.360	1.81E+02	2.00E+02	7.70E+01	5.52E+01	3.56E+01	1.76E+01	1.90E+02	1.38E+02	9.48E+01	6.90E+01	5.17E+01	3.45E+01
1000	1.47E+02	1.64E+02	6.23E+01	4.45E+01	2.87E+01	1.41E+01	1.350	1.99E+02	2.22E+02	8.41E+01	6.01E+01	3.88E+01	1.90E+01	2.10E+02	1.53E+02	1.05E+02	7.63E+01	5.72E+01	3.81E+01
2000	2.80E+02	3.36E+02	1.15E+02	8.14E+01	5.27E+01	2.41E+01	1.292	3.61E+02	4.34E+02	1.49E+02	1.05E+02	6.81E+01	3.11E+01	4.02E+02	2.92E+02	2.01E+02	1.46E+02	1.10E+02	7.30E+01
3000	3.97E+02	5.07E+02	1.64E+02	1.16E+02	7.50E+01	3.29E+01	1.259	5.00E+02	6.39E+02	2.07E+02	1.46E+02	9.45E+01	4.14E+01	5.79E+02	4.21E+02	2.89E+02	2.10E+02	1.58E+02	1.05E+02

Table A2 – Committed Effective Doses

H(m) 18 u(m/s) 2

	X(x,0,0)/Q						Q	Adult E(Sv)						Infant E(Sv)					
x	A	B	C	D	E	F	(Bq)	A	B	C	D	E	F	A	B	C	D	E	F
100	2.35E-04	3.21E-04	3.94E-04	3.13E-04	3.84E-05	5.84E-08	1.85E+12	2.53E-03	3.46E-03	4.25E-03	3.38E-03	4.15E-04	6.31E-07	1.08E-03	1.47E-03	1.81E-03	1.44E-03	1.76E-04	2.68E-07
110	2.05E-04	2.80E-04	3.91E-04	3.49E-04	6.51E-05	2.36E-07	1.85E+12	2.22E-03	3.02E-03	4.22E-03	3.77E-03	7.02E-04	2.55E-06	9.42E-04	1.28E-03	1.80E-03	1.60E-03	2.99E-04	1.08E-06
120	1.80E-04	2.45E-04	3.80E-04	3.72E-04	9.61E-05	7.07E-07	1.85E+12	1.94E-03	2.64E-03	4.10E-03	4.02E-03	1.04E-03	7.63E-06	8.27E-04	1.12E-03	1.74E-03	1.71E-03	4.41E-04	3.24E-06
130	1.59E-04	2.16E-04	3.64E-04	3.85E-04	1.29E-04	1.70E-06	1.85E+12	1.72E-03	2.33E-03	3.93E-03	4.15E-03	1.39E-03	1.84E-05	7.30E-04	9.89E-04	1.67E-03	1.77E-03	5.92E-04	7.82E-06
140	1.41E-04	1.91E-04	3.46E-04	3.89E-04	1.61E-04	3.48E-06	1.85E+12	1.52E-03	2.06E-03	3.73E-03	4.20E-03	1.74E-03	3.76E-05	6.47E-04	8.75E-04	1.59E-03	1.78E-03	7.39E-04	1.60E-05
150	1.26E-04	1.70E-04	3.26E-04	3.87E-04	1.91E-04	6.27E-06	1.85E+12	1.36E-03	1.83E-03	3.52E-03	4.17E-03	2.06E-03	6.77E-05	5.77E-04	7.79E-04	1.50E-03	1.78E-03	8.76E-04	2.88E-05
160	1.13E-04	1.52E-04	3.07E-04	3.81E-04	2.17E-04	1.02E-05	1.85E+12	1.22E-03	1.64E-03	3.32E-03	4.11E-03	2.35E-03	1.10E-04	5.17E-04	6.97E-04	1.41E-03	1.75E-03	9.97E-04	4.70E-05
170	1.01E-04	1.36E-04	2.89E-04	3.71E-04	2.40E-04	1.55E-05	1.85E+12	1.09E-03	1.47E-03	3.12E-03	4.01E-03	2.59E-03	1.67E-04	4.65E-04	6.26E-04	1.32E-03	1.70E-03	1.10E-03	7.09E-05
180	9.17E-05	1.23E-04	2.71E-04	3.60E-04	2.59E-04	2.19E-05	1.85E+12	9.90E-04	1.33E-03	2.92E-03	3.88E-03	2.79E-03	2.36E-04	4.21E-04	5.66E-04	1.24E-03	1.65E-03	1.19E-03	1.00E-04
190	8.33E-05	1.12E-04	2.54E-04	3.47E-04	2.74E-04	2.95E-05	1.85E+12	8.99E-04	1.21E-03	2.74E-03	3.75E-03	2.95E-03	3.18E-04	3.82E-04	5.13E-04	1.17E-03	1.59E-03	1.26E-03	1.35E-04
200	7.60E-05	1.02E-04	2.39E-04	3.34E-04	2.85E-04	3.80E-05	1.85E+12	8.20E-04	1.10E-03	2.58E-03	3.61E-03	3.08E-03	4.10E-04	3.49E-04	4.67E-04	1.10E-03	1.53E-03	1.31E-03	1.74E-04
250	5.06E-05	6.73E-05	1.77E-04	2.70E-04	3.03E-04	8.86E-05	1.85E+12	5.46E-04	7.26E-04	1.91E-03	2.92E-03	3.27E-03	9.56E-04	2.32E-04	3.09E-04	8.10E-04	1.24E-03	1.39E-03	4.06E-04
300	3.61E-05	4.77E-05	1.35E-04	2.17E-04	2.86E-04	1.37E-04	1.85E+12	3.89E-04	5.15E-04	1.45E-03	2.34E-03	3.09E-03	1.48E-03	1.66E-04	2.19E-04	6.18E-04	9.97E-04	1.31E-03	6.29E-04
350	2.70E-05	3.55E-05	1.06E-04	1.77E-04	2.58E-04	1.73E-04	1.85E+12	2.92E-04	3.83E-04	1.14E-03	1.91E-03	2.78E-03	1.87E-03	1.24E-04	1.63E-04	4.85E-04	8.11E-04	1.18E-03	7.95E-04
400	2.10E-05	2.75E-05	8.50E-05	1.46E-04	2.28E-04	1.96E-04	1.85E+12	2.27E-04	2.96E-04	9.17E-04	1.57E-03	2.46E-03	2.11E-03	9.65E-05	1.26E-04	3.90E-04	6.69E-04	1.05E-03	8.98E-04
500	1.38E-05	1.79E-05	5.85E-05	1.04E-04	1.77E-04	2.11E-04	1.85E+12	1.49E-04	1.93E-04	6.32E-04	1.12E-03	1.91E-03	2.28E-03	6.34E-05	8.19E-05	2.69E-04	4.77E-04	8.12E-04	9.68E-04
600	9.81E-06	1.26E-05	4.29E-05	7.77E-05	1.39E-04	2.04E-04	1.85E+12	1.06E-04	1.35E-04	4.63E-04	8.38E-04	1.50E-03	2.21E-03	4.50E-05	5.76E-05	1.97E-04	3.56E-04	6.38E-04	9.38E-04
700	7.35E-06	9.32E-06	3.28E-05	6.03E-05	1.12E-04	1.89E-04	1.85E+12	7.93E-05	1.01E-04	3.54E-04	6.51E-04	1.20E-03	2.04E-03	3.37E-05	4.28E-05	1.51E-04	2.77E-04	5.12E-04	8.69E-04
800	5.73E-06	7.20E-06	2.60E-05	4.83E-05	9.13E-05	1.72E-04	1.85E+12	6.19E-05	7.77E-05	2.81E-04	5.22E-04	9.85E-04	1.86E-03	2.63E-05	3.30E-05	1.20E-04	2.22E-04	4.19E-04	7.90E-04
900	4.61E-06	5.74E-06	2.12E-05	3.97E-05	7.61E-05	1.56E-04	1.85E+12	4.97E-05	6.19E-05	2.29E-04	4.28E-04	8.21E-04	1.68E-03	2.12E-05	2.63E-05	9.73E-05	1.82E-04	3.49E-04	7.14E-04
1000	3.80E-06	4.69E-06	1.76E-05	3.32E-05	6.44E-05	1.40E-04	1.85E+12	4.10E-05	5.06E-05	1.90E-04	3.58E-04	6.95E-04	1.51E-03	1.74E-05	2.15E-05	8.10E-05	1.52E-04	2.96E-04	6.44E-04
2000	1.10E-06	1.26E-06	5.29E-06	1.02E-05	2.06E-05	5.93E-05	1.85E+12	1.18E-05	1.36E-05	5.71E-05	1.10E-04	2.22E-04	6.40E-04	5.03E-06	5.76E-06	2.43E-05	4.69E-05	9.45E-05	2.72E-04
3000	5.49E-07	5.92E-07	2.65E-06	5.15E-06	1.05E-05	3.33E-05	1.85E+12	5.93E-06	6.39E-06	2.86E-05	5.56E-05	1.13E-04	3.59E-04	2.52E-06	2.72E-06	1.22E-05	2.37E-05	4.81E-05	1.53E-04

Calculations for I-125 Ground Release

Table A3 – Diffusion Coefficients

	g(x)						F(Zo,x)	Sigma Z						Sigma Y					
x	A	B	C	D	E	F		A	B	C	D	E	F	A	B	C	D	E	F
100	1.44E+01	1.47E+01	7.70E+00	5.85E+00	3.73E+00	2.34E+00	1.5603	2.25E+01	2.29E+01	1.20E+01	9.12E+00	5.82E+00	3.65E+00	2.19E+01	1.59E+01	1.09E+01	7.96E+00	5.97E+00	3.98E+00
110	1.59E+01	1.63E+01	8.40E+00	6.36E+00	4.06E+00	2.52E+00	1.5514	2.47E+01	2.52E+01	1.30E+01	9.87E+00	6.30E+00	3.91E+00	2.41E+01	1.75E+01	1.20E+01	8.75E+00	6.56E+00	4.38E+00
120	1.74E+01	1.78E+01	9.10E+00	6.87E+00	4.39E+00	2.70E+00	1.5433	2.69E+01	2.75E+01	1.40E+01	1.06E+01	6.77E+00	4.17E+00	2.62E+01	1.91E+01	1.31E+01	9.54E+00	7.16E+00	4.77E+00
130	1.90E+01	1.94E+01	9.79E+00	7.37E+00	4.71E+00	2.87E+00	1.5359	2.91E+01	2.98E+01	1.50E+01	1.13E+01	7.24E+00	4.41E+00	2.84E+01	2.07E+01	1.42E+01	1.03E+01	7.75E+00	5.17E+00
140	2.05E+01	2.10E+01	1.05E+01	7.87E+00	5.03E+00	3.04E+00	1.5290	3.13E+01	3.20E+01	1.60E+01	1.20E+01	7.70E+00	4.65E+00	3.06E+01	2.22E+01	1.53E+01	1.11E+01	8.34E+00	5.56E+00
150	2.20E+01	2.25E+01	1.12E+01	8.37E+00	5.35E+00	3.21E+00	1.5226	3.35E+01	3.43E+01	1.70E+01	1.27E+01	8.15E+00	4.89E+00	3.28E+01	2.38E+01	1.64E+01	1.19E+01	8.93E+00	5.96E+00
160	2.35E+01	2.41E+01	1.18E+01	8.86E+00	5.67E+00	3.38E+00	1.5166	3.57E+01	3.66E+01	1.79E+01	1.34E+01	8.60E+00	5.12E+00	3.49E+01	2.54E+01	1.75E+01	1.27E+01	9.52E+00	6.35E+00
170	2.50E+01	2.57E+01	1.25E+01	9.35E+00	5.98E+00	3.54E+00	1.5109	3.78E+01	3.89E+01	1.89E+01	1.41E+01	9.04E+00	5.35E+00	3.71E+01	2.70E+01	1.85E+01	1.35E+01	1.01E+01	6.74E+00
180	2.65E+01	2.73E+01	1.32E+01	9.83E+00	6.30E+00	3.70E+00	1.5056	4.00E+01	4.11E+01	1.98E+01	1.48E+01	9.48E+00	5.58E+00	3.92E+01	2.85E+01	1.96E+01	1.43E+01	1.07E+01	7.14E+00
190	2.81E+01	2.89E+01	1.38E+01	1.03E+01	6.61E+00	3.86E+00	1.5006	4.21E+01	4.34E+01	2.08E+01	1.55E+01	9.91E+00	5.80E+00	4.14E+01	3.01E+01	2.07E+01	1.51E+01	1.13E+01	7.53E+00
200	2.96E+01	3.05E+01	1.45E+01	1.08E+01	6.91E+00	4.02E+00	1.4959	4.43E+01	4.57E+01	2.17E+01	1.61E+01	1.03E+01	6.02E+00	4.36E+01	3.17E+01	2.18E+01	1.58E+01	1.19E+01	7.92E+00
250	3.72E+01	3.86E+01	1.78E+01	1.31E+01	8.43E+00	4.79E+00	1.4753	5.49E+01	5.70E+01	2.62E+01	1.94E+01	1.24E+01	7.06E+00	5.43E+01	3.95E+01	2.72E+01	1.98E+01	1.48E+01	9.88E+00
300	4.48E+01	4.68E+01	2.10E+01	1.54E+01	9.91E+00	5.52E+00	1.4586	6.53E+01	6.82E+01	3.06E+01	2.25E+01	1.45E+01	8.05E+00	6.50E+01	4.73E+01	3.25E+01	2.36E+01	1.77E+01	1.18E+01
350	5.24E+01	5.50E+01	2.41E+01	1.77E+01	1.14E+01	6.22E+00	1.4446	7.57E+01	7.94E+01	3.49E+01	2.56E+01	1.64E+01	8.99E+00	7.57E+01	5.50E+01	3.78E+01	2.75E+01	2.06E+01	1.38E+01
400	5.99E+01	6.32E+01	2.73E+01	1.99E+01	1.28E+01	6.90E+00	1.4324	8.58E+01	9.06E+01	3.90E+01	2.85E+01	1.83E+01	9.89E+00	8.63E+01	6.28E+01	4.31E+01	3.14E+01	2.35E+01	1.57E+01
500	7.49E+01	7.99E+01	3.34E+01	2.42E+01	1.56E+01	8.21E+00	1.4123	1.06E+02	1.13E+02	4.71E+01	3.42E+01	2.20E+01	1.16E+01	1.07E+02	7.81E+01	5.37E+01	3.90E+01	2.93E+01	1.95E+01
600	8.98E+01	9.66E+01	3.93E+01	2.84E+01	1.83E+01	9.46E+00	1.3959	1.25E+02	1.35E+02	5.49E+01	3.97E+01	2.56E+01	1.32E+01	1.28E+02	9.32E+01	6.41E+01	4.66E+01	3.50E+01	2.33E+01
700	1.04E+02	1.13E+02	4.52E+01	3.25E+01	2.10E+01	1.07E+01	1.3822	1.44E+02	1.57E+02	6.25E+01	4.50E+01	2.90E+01	1.47E+01	1.49E+02	1.08E+02	7.44E+01	5.41E+01	4.06E+01	2.71E+01
800	1.19E+02	1.30E+02	5.10E+01	3.66E+01	2.36E+01	1.18E+01	1.3704	1.63E+02	1.79E+02	6.98E+01	5.01E+01	3.24E+01	1.62E+01	1.69E+02	1.23E+02	8.47E+01	6.16E+01	4.62E+01	3.08E+01
900	1.33E+02	1.47E+02	5.66E+01	4.06E+01	2.62E+01	1.30E+01	1.3601	1.81E+02	2.00E+02	7.70E+01	5.52E+01	3.56E+01	1.76E+01	1.90E+02	1.38E+02	9.48E+01	6.90E+01	5.17E+01	3.45E+01
1000	1.47E+02	1.64E+02	6.23E+01	4.45E+01	2.87E+01	1.41E+01	1.3509	1.99E+02	2.22E+02	8.41E+01	6.01E+01	3.88E+01	1.90E+01	2.10E+02	1.53E+02	1.05E+02	7.63E+01	5.72E+01	3.81E+01
2000	2.80E+02	3.36E+02	1.15E+02	8.14E+01	5.27E+01	2.41E+01	1.2920	3.61E+02	4.34E+02	1.49E+02	1.05E+02	6.81E+01	3.11E+01	4.02E+02	2.92E+02	2.01E+02	1.46E+02	1.10E+02	7.30E+01
3000	3.97E+02	5.07E+02	1.64E+02	1.16E+02	7.50E+01	3.29E+01	1.2592	5.00E+02	6.39E+02	2.07E+02	1.46E+02	9.45E+01	4.14E+01	5.79E+02	4.21E+02	2.89E+02	2.10E+02	1.58E+02	1.05E+02

Table A4-Committed Effective Doses

H(m) 0 u(m/s) 2

	X(x,0,0)/Q						Q	Adult E(Sv)						Infant E(Sv)					
x	A	B	C	D	E	F	(Bq)	A	B	C	D	E	F	A	B	C	D	E	F
100	3.23E-04	4.36E-04	1.21E-03	2.19E-03	4.58E-03	1.10E-02	1.85E+12	3.49E-03	4.71E-03	1.31E-02	2.37E-02	4.94E-02	1.18E-01	1.48E-03	2.00E-03	5.56E-03	1.01E-02	2.10E-02	5.03E-02
110	2.68E-04	3.61E-04	1.02E-03	1.84E-03	3.85E-03	9.30E-03	1.85E+12	2.89E-03	3.89E-03	1.10E-02	1.99E-02	4.15E-02	1.00E-01	1.23E-03	1.66E-03	4.66E-03	8.46E-03	1.77E-02	4.27E-02
120	2.25E-04	3.03E-04	8.64E-04	1.57E-03	3.28E-03	8.01E-03	1.85E+12	2.43E-03	3.27E-03	9.33E-03	1.70E-02	3.54E-02	8.65E-02	1.03E-03	1.39E-03	3.97E-03	7.22E-03	1.51E-02	3.68E-02
130	1.92E-04	2.59E-04	7.45E-04	1.36E-03	2.84E-03	6.98E-03	1.85E+12	2.08E-03	2.79E-03	8.04E-03	1.47E-02	3.06E-02	7.54E-02	8.83E-04	1.19E-03	3.42E-03	6.24E-03	1.30E-02	3.21E-02
140	1.66E-04	2.23E-04	6.50E-04	1.19E-03	2.48E-03	6.15E-03	1.85E+12	1.79E-03	2.41E-03	7.02E-03	1.28E-02	2.68E-02	6.64E-02	7.63E-04	1.03E-03	2.98E-03	5.46E-03	1.14E-02	2.82E-02
150	1.45E-04	1.95E-04	5.72E-04	1.05E-03	2.19E-03	5.47E-03	1.85E+12	1.57E-03	2.10E-03	6.18E-03	1.13E-02	2.36E-02	5.90E-02	6.66E-04	8.94E-04	2.63E-03	4.81E-03	1.00E-02	2.51E-02
160	1.28E-04	1.71E-04	5.08E-04	9.33E-04	1.94E-03	4.89E-03	1.85E+12	1.38E-03	1.85E-03	5.48E-03	1.01E-02	2.10E-02	5.28E-02	5.87E-04	7.86E-04	2.33E-03	4.28E-03	8.92E-03	2.25E-02
170	1.14E-04	1.52E-04	4.54E-04	8.36E-04	1.74E-03	4.41E-03	1.85E+12	1.23E-03	1.64E-03	4.90E-03	9.02E-03	1.88E-02	4.76E-02	5.21E-04	6.97E-04	2.09E-03	3.84E-03	7.99E-03	2.02E-02
180	1.02E-04	1.36E-04	4.09E-04	7.54E-04	1.57E-03	4.00E-03	1.85E+12	1.10E-03	1.46E-03	4.41E-03	8.13E-03	1.69E-02	4.32E-02	4.66E-04	6.22E-04	1.88E-03	3.46E-03	7.20E-03	1.84E-02
190	9.13E-05	1.22E-04	3.70E-04	6.83E-04	1.42E-03	3.65E-03	1.85E+12	9.85E-04	1.31E-03	3.99E-03	7.37E-03	1.53E-02	3.94E-02	4.19E-04	5.59E-04	1.70E-03	3.13E-03	6.53E-03	1.67E-02
200	8.26E-05	1.10E-04	3.37E-04	6.22E-04	1.30E-03	3.34E-03	1.85E+12	8.91E-04	1.19E-03	3.63E-03	6.72E-03	1.40E-02	3.61E-02	3.79E-04	5.05E-04	1.55E-03	2.86E-03	5.95E-03	1.53E-02
250	5.34E-05	7.08E-05	2.23E-04	4.16E-04	8.64E-04	2.28E-03	1.85E+12	5.76E-04	7.64E-04	2.41E-03	4.49E-03	9.32E-03	2.46E-02	2.45E-04	3.25E-04	1.03E-03	1.91E-03	3.97E-03	1.05E-02
300	3.75E-05	4.94E-05	1.60E-04	2.99E-04	6.21E-04	1.67E-03	1.85E+12	4.04E-04	5.33E-04	1.73E-03	3.23E-03	6.70E-03	1.81E-02	1.72E-04	2.27E-04	7.34E-04	1.37E-03	2.85E-03	7.68E-03
350	2.78E-05	3.64E-05	1.21E-04	2.26E-04	4.70E-04	1.29E-03	1.85E+12	3.00E-04	3.93E-04	1.30E-03	2.44E-03	5.07E-03	1.39E-02	1.28E-04	1.67E-04	5.54E-04	1.04E-03	2.16E-03	5.91E-03
400	2.15E-05	2.80E-05	9.45E-05	1.78E-04	3.69E-04	1.03E-03	1.85E+12	2.32E-04	3.02E-04	1.02E-03	1.92E-03	3.98E-03	1.11E-02	9.87E-05	1.29E-04	4.34E-04	8.17E-04	1.69E-03	4.71E-03
500	1.40E-05	1.81E-05	6.30E-05	1.19E-04	2.47E-04	7.04E-04	1.85E+12	1.51E-04	1.95E-04	6.79E-04	1.29E-03	2.67E-03	7.59E-03	6.43E-05	8.30E-05	2.89E-04	5.47E-04	1.13E-03	3.23E-03
600	9.91E-06	1.27E-05	4.52E-05	8.61E-05	1.78E-04	5.17E-04	1.85E+12	1.07E-04	1.37E-04	4.88E-04	9.29E-04	1.92E-03	5.58E-03	4.55E-05	5.81E-05	2.08E-04	3.95E-04	8.17E-04	2.37E-03
700	7.41E-06	9.38E-06	3.42E-05	6.54E-05	1.35E-04	3.99E-04	1.85E+12	7.99E-05	1.01E-04	3.69E-04	7.06E-04	1.46E-03	4.31E-03	3.40E-05	4.30E-05	1.57E-04	3.00E-04	6.21E-04	1.83E-03
800	5.77E-06	7.24E-06	2.69E-05	5.16E-05	1.07E-04	3.19E-04	1.85E+12	6.22E-05	7.81E-05	2.91E-04	5.56E-04	1.15E-03	3.44E-03	2.65E-05	3.32E-05	1.24E-04	2.37E-04	4.89E-04	1.46E-03
900	4.63E-06	5.76E-06	2.18E-05	4.18E-05	8.64E-05	2.62E-04	1.85E+12	5.00E-05	6.22E-05	2.35E-04	4.52E-04	9.33E-04	2.83E-03	2.13E-05	2.64E-05	1.00E-04	1.92E-04	3.97E-04	1.20E-03
1000	3.81E-06	4.70E-06	1.81E-05	3.47E-05	7.17E-05	2.20E-04	1.85E+12	4.11E-05	5.07E-05	1.95E-04	3.75E-04	7.74E-04	2.37E-03	1.75E-05	2.16E-05	8.28E-05	1.59E-04	3.29E-04	1.01E-03
2000	1.10E-06	1.26E-06	5.33E-06	1.04E-05	2.13E-05	7.02E-05	1.85E+12	1.18E-05	1.36E-05	5.75E-05	1.12E-04	2.30E-04	7.57E-04	5.03E-06	5.77E-06	2.45E-05	4.76E-05	9.79E-05	3.22E-04
3000	5.50E-07	5.92E-07	2.66E-06	5.19E-06	1.07E-05	3.66E-05	1.85E+12	5.93E-06	6.39E-06	2.87E-05	5.60E-05	1.15E-04	3.94E-04	2.52E-06	2.72E-06	1.22E-05	2.38E-05	4.90E-05	1.68E-04

Calculations for Xe-125 Stack Release

Table A5-Committed Effective Doses

H(m) 18 u(m/s) 2

	X(x,0,0)/Q						Q	Adult E(Sv)					
x	A	B	C	D	E	F	(Bq)	A	B	C	D	E	F
100	2.35E-04	3.21E-04	3.94E-04	3.13E-04	3.84E-05	5.84E-08	1.50E+14	3.79E-04	5.18E-04	6.37E-04	5.05E-04	6.21E-05	9.43E-08
110	2.05E-04	2.80E-04	3.91E-04	3.49E-04	6.51E-05	2.36E-07	1.50E+14	3.31E-04	4.52E-04	6.32E-04	5.64E-04	1.05E-04	3.81E-07
120	1.80E-04	2.45E-04	3.80E-04	3.72E-04	9.61E-05	7.07E-07	1.50E+14	2.91E-04	3.95E-04	6.14E-04	6.01E-04	1.55E-04	1.14E-06
130	1.59E-04	2.16E-04	3.64E-04	3.85E-04	1.29E-04	1.70E-06	1.50E+14	2.57E-04	3.48E-04	5.88E-04	6.21E-04	2.08E-04	2.75E-06
140	1.41E-04	1.91E-04	3.46E-04	3.89E-04	1.61E-04	3.48E-06	1.50E+14	2.28E-04	3.08E-04	5.58E-04	6.28E-04	2.60E-04	5.62E-06
150	1.26E-04	1.70E-04	3.26E-04	3.87E-04	1.91E-04	6.27E-06	1.50E+14	2.03E-04	2.74E-04	5.27E-04	6.25E-04	3.08E-04	1.01E-05
160	1.13E-04	1.52E-04	3.07E-04	3.81E-04	2.17E-04	1.02E-05	1.50E+14	1.82E-04	2.45E-04	4.96E-04	6.14E-04	3.51E-04	1.65E-05
170	1.01E-04	1.36E-04	2.89E-04	3.71E-04	2.40E-04	1.55E-05	1.50E+14	1.64E-04	2.20E-04	4.66E-04	5.99E-04	3.87E-04	2.49E-05
180	9.17E-05	1.23E-04	2.71E-04	3.60E-04	2.59E-04	2.19E-05	1.50E+14	1.48E-04	1.99E-04	4.38E-04	5.81E-04	4.18E-04	3.53E-05
190	8.33E-05	1.12E-04	2.54E-04	3.47E-04	2.74E-04	2.95E-05	1.50E+14	1.35E-04	1.81E-04	4.11E-04	5.61E-04	4.42E-04	4.76E-05
200	7.60E-05	1.02E-04	2.39E-04	3.34E-04	2.85E-04	3.80E-05	1.50E+14	1.23E-04	1.64E-04	3.86E-04	5.40E-04	4.60E-04	6.14E-05
250	5.06E-05	6.73E-05	1.77E-04	2.70E-04	3.03E-04	8.86E-05	1.50E+14	8.17E-05	1.09E-04	2.85E-04	4.36E-04	4.90E-04	1.43E-04
300	3.61E-05	4.77E-05	1.35E-04	2.17E-04	2.86E-04	1.37E-04	1.50E+14	5.83E-05	7.70E-05	2.17E-04	3.51E-04	4.62E-04	2.21E-04
350	2.70E-05	3.55E-05	1.06E-04	1.77E-04	2.58E-04	1.73E-04	1.50E+14	4.36E-05	5.73E-05	1.70E-04	2.85E-04	4.16E-04	2.80E-04
400	2.10E-05	2.75E-05	8.50E-05	1.46E-04	2.28E-04	1.96E-04	1.50E+14	3.40E-05	4.43E-05	1.37E-04	2.35E-04	3.68E-04	3.16E-04
500	1.38E-05	1.79E-05	5.85E-05	1.04E-04	1.77E-04	2.11E-04	1.50E+14	2.23E-05	2.88E-05	9.45E-05	1.68E-04	2.86E-04	3.41E-04
600	9.81E-06	1.26E-05	4.29E-05	7.77E-05	1.39E-04	2.04E-04	1.50E+14	1.58E-05	2.03E-05	6.92E-05	1.25E-04	2.24E-04	3.30E-04
700	7.35E-06	9.32E-06	3.28E-05	6.03E-05	1.12E-04	1.89E-04	1.50E+14	1.19E-05	1.50E-05	5.30E-05	9.74E-05	1.80E-04	3.06E-04
800	5.73E-06	7.20E-06	2.60E-05	4.83E-05	9.13E-05	1.72E-04	1.50E+14	9.26E-06	1.16E-05	4.20E-05	7.81E-05	1.47E-04	2.78E-04
900	4.61E-06	5.74E-06	2.12E-05	3.97E-05	7.61E-05	1.56E-04	1.50E+14	7.44E-06	9.26E-06	3.42E-05	6.41E-05	1.23E-04	2.51E-04
1000	3.80E-06	4.69E-06	1.76E-05	3.32E-05	6.44E-05	1.40E-04	1.50E+14	6.13E-06	7.57E-06	2.85E-05	5.36E-05	1.04E-04	2.27E-04
2000	1.10E-06	1.26E-06	5.29E-06	1.02E-05	2.06E-05	5.93E-05	1.50E+14	1.77E-06	2.03E-06	8.54E-06	1.65E-05	3.33E-05	9.58E-05
3000	5.49E-07	5.92E-07	2.65E-06	5.15E-06	1.05E-05	3.33E-05	1.50E+14	8.87E-07	9.56E-07	4.28E-06	8.32E-06	1.69E-05	5.37E-05

Calculations for Xe-125 Ground Release

Table A6-Committed Effective Doses

H(m) 0 u(m/s) 2

	X(x,0,0)/Q						Q	Adult E(Sv)					
x	A	B	C	D	E	F	(Bq)	A	B	C	D	E	F
100	3.23E-04	4.36E-04	1.21E-03	2.19E-03	4.58E-03	1.10E-02	1.50E+14	5.21E-04	7.04E-04	1.95E-03	3.54E-03	7.40E-03	1.77E-02
110	2.68E-04	3.61E-04	1.02E-03	1.84E-03	3.85E-03	9.30E-03	1.50E+14	4.32E-04	5.83E-04	1.64E-03	2.98E-03	6.22E-03	1.50E-02
120	2.25E-04	3.03E-04	8.64E-04	1.57E-03	3.28E-03	8.01E-03	1.50E+14	3.64E-04	4.90E-04	1.40E-03	2.54E-03	5.30E-03	1.29E-02
130	1.92E-04	2.59E-04	7.45E-04	1.36E-03	2.84E-03	6.98E-03	1.50E+14	3.11E-04	4.18E-04	1.20E-03	2.20E-03	4.58E-03	1.13E-02
140	1.66E-04	2.23E-04	6.50E-04	1.19E-03	2.48E-03	6.15E-03	1.50E+14	2.69E-04	3.61E-04	1.05E-03	1.92E-03	4.00E-03	9.93E-03
150	1.45E-04	1.95E-04	5.72E-04	1.05E-03	2.19E-03	5.47E-03	1.50E+14	2.34E-04	3.14E-04	9.24E-04	1.69E-03	3.53E-03	8.83E-03
160	1.28E-04	1.71E-04	5.08E-04	9.33E-04	1.94E-03	4.89E-03	1.50E+14	2.07E-04	2.77E-04	8.20E-04	1.51E-03	3.14E-03	7.90E-03
170	1.14E-04	1.52E-04	4.54E-04	8.36E-04	1.74E-03	4.41E-03	1.50E+14	1.83E-04	2.45E-04	7.34E-04	1.35E-03	2.81E-03	7.12E-03
180	1.02E-04	1.36E-04	4.09E-04	7.54E-04	1.57E-03	4.00E-03	1.50E+14	1.64E-04	2.19E-04	6.60E-04	1.22E-03	2.53E-03	6.46E-03
190	9.13E-05	1.22E-04	3.70E-04	6.83E-04	1.42E-03	3.65E-03	1.50E+14	1.47E-04	1.97E-04	5.98E-04	1.10E-03	2.30E-03	5.89E-03
200	8.26E-05	1.10E-04	3.37E-04	6.22E-04	1.30E-03	3.34E-03	1.50E+14	1.33E-04	1.78E-04	5.44E-04	1.01E-03	2.09E-03	5.40E-03
250	5.34E-05	7.08E-05	2.23E-04	4.16E-04	8.64E-04	2.28E-03	1.50E+14	8.62E-05	1.14E-04	3.61E-04	6.71E-04	1.40E-03	3.69E-03
300	3.75E-05	4.94E-05	1.60E-04	2.99E-04	6.21E-04	1.67E-03	1.50E+14	6.05E-05	7.97E-05	2.58E-04	4.83E-04	1.00E-03	2.70E-03
350	2.78E-05	3.64E-05	1.21E-04	2.26E-04	4.70E-04	1.29E-03	1.50E+14	4.49E-05	5.88E-05	1.95E-04	3.66E-04	7.59E-04	2.08E-03
400	2.15E-05	2.80E-05	9.45E-05	1.78E-04	3.69E-04	1.03E-03	1.50E+14	3.47E-05	4.52E-05	1.53E-04	2.87E-04	5.96E-04	1.66E-03
500	1.40E-05	1.81E-05	6.30E-05	1.19E-04	2.47E-04	7.04E-04	1.50E+14	2.26E-05	2.92E-05	1.02E-04	1.93E-04	3.99E-04	1.14E-03
600	9.91E-06	1.27E-05	4.52E-05	8.61E-05	1.78E-04	5.17E-04	1.50E+14	1.60E-05	2.04E-05	7.30E-05	1.39E-04	2.88E-04	8.35E-04
700	7.41E-06	9.38E-06	3.42E-05	6.54E-05	1.35E-04	3.99E-04	1.50E+14	1.20E-05	1.51E-05	5.53E-05	1.06E-04	2.18E-04	6.44E-04
800	5.77E-06	7.24E-06	2.69E-05	5.16E-05	1.07E-04	3.19E-04	1.50E+14	9.31E-06	1.17E-05	4.35E-05	8.32E-05	1.72E-04	5.15E-04
900	4.63E-06	5.76E-06	2.18E-05	4.18E-05	8.64E-05	2.62E-04	1.50E+14	7.48E-06	9.30E-06	3.52E-05	6.76E-05	1.40E-04	4.23E-04
1000	3.81E-06	4.70E-06	1.81E-05	3.47E-05	7.17E-05	2.20E-04	1.50E+14	6.15E-06	7.59E-06	2.91E-05	5.61E-05	1.16E-04	3.55E-04
2000	1.10E-06	1.26E-06	5.33E-06	1.04E-05	2.13E-05	7.02E-05	1.50E+14	1.77E-06	2.03E-06	8.60E-06	1.67E-05	3.44E-05	1.13E-04
3000	5.50E-07	5.92E-07	2.66E-06	5.19E-06	1.07E-05	3.66E-05	1.50E+14	8.88E-07	9.56E-07	4.30E-06	8.39E-06	1.72E-05	5.90E-05

Chapter 6: Concluding Remarks

The elements of a radiation safety program for the production of ^{125}I have been reviewed and discussed from the perspective of the changes that are required when adding production activities to a facility with an established radiation safety program. This reflects the experience at the McMaster Nuclear Reactor over the past several years as prototype production activities shifted to full scale commercial ^{125}I production.

Above all else, the importance of careful pre-planning cannot be over emphasized. Safe addition of ^{125}I production activities required substantive changes to virtually every aspect of the facility's radiation safety program, from documentation to instrumentation to facilities to training. Making these changes at MNR has required significant additional radiation safety effort in recent years, and the process is not yet complete.

One crucial aspect to a safe and successful transition was the work performed by others at the early stages of prototype production to predict doses that would result from commercial scale production. As outlined in Chapter 3, prolonged and sustained effort was required to identify and mitigate all of the non-optimized exposures associated with the production activities. Establishing a formal ALARA program for this work was key to communicating the problem and obtaining buy-in and support from the management and staff involved. If the theoretically obtainable doses had not been previously established, focus and enthusiasm could easily have waned after the first early successes in reducing exposures. Implementation of the ALARA program reduced the collective dose to the group by nearly 90%, saving nearly 40 person-mSv per year.

Careful facility design is another priority for a safe transition. Significant investment in engineered controls, such as ventilation, charcoal filtration and containment systems, are vitally important to executing production tasks in a manner which is safe for both workers and members of the public. These controls play an important role in managing both ongoing occupational exposures and potential exposures which may result from a production related accident. It is a testament to the staff responsible for the design and maintenance of the MNR facilities, and to the decision making process for safety related investments, that after several years of commercial scale production it has never been necessary to assign a dose due to internal exposure for any worker.

Finally, the experiences gained at MNR, as outlined in this thesis, may be useful at other facilities considering the introduction of large scale production of virtually any radionuclide. Reviewing the changes that are required for ^{125}I production will provide a general checklist that can be used in considering the changes required for other radionuclides. The specific changes and solutions to be implemented will depend on the properties of the radionuclide, the characteristics of the facility, the nature of the production process and the state of the existing radiation safety program, but the topics to be evaluated are those which are discussed in this report.