AECL EACL

Safety Design Guide

RADIATION PROTECTION

ACR

108-03650-SDG-007

Revision 3

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2004/03/15
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Mars 2004

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Release and Revision History / Liste des documents et des révisions

0939B Rev. 13

Document Details / Détails sur le document

Title Titre	Total no. of pages N^bre total de pages
Radiation Protection	46

CONTROLLED - CONTRÔLÉ

Release and Revision History / Liste des documents et des révisions

Release Document No./N°	Date	Revision Révision No./N°	Date	Purpose of Release; Details of Rev./Amendement Objet du document; détails des rév. ou des modif.	Prepared by Rédigé par	Reviewed by Examiné par	Approved by Approuvé par
1	00-06-28	D1	00-06-28	Convert CANDU 9 SDG to NG CANDU format	A.H. Stretch		
2	00-09-01	D2	00-09-01	Review and Comment	A.L. Wight	D.H. Wren	
3		0	00-10-10	Issued as 'approved for use".	A.L. Wight	D.H. Wren	J.M. Hopwood
4		1D1	02-06-03	Issued for "Review and Comment".	C. Xu	A. H. Stretch	M. Bonechi
5		1D2	2002/09/03	Issued for "Review and Comment". 1) Reflect the current regulatory framework (deleted Reference 4 "CNSC Consultative Document C-118"). 2) Added table "Organ or Tissue Weighting Factors". 3) Table 3-2 & 3-3 have been put in Appendix. 4) Deleted Figure 3-1. 5) Revised the ALARA process. 6)Rearranged Section 3.3.2 to Section 3.3.5.	S. D. Nam	W. Rabbani V. Snell P. Allsop A. Stretch C. Boss/B. Rouben K. Hau E. Choy J. Millard M. Elgohary A. Josefowicz R. Ghai/O. Hines P. Chan H. Johal	
6		1	2002/10/15	Issued as 'approved for use".	S.D. Nam	W. Rabbani	M. Bonechi

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. Proj. conn.	Project Projet	SI	Section	Serial Série	Sheet Feuille No. N°	Of De	Unit No.(s) Tranche n°
	108	03650	SDG	007	1	4	

AECL EACL

Release and Revision History / Liste des documents et des révisions

0939B Rev. 13

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Release Document No./N°	Date	Revision Révision No./N°	Date	Purpose of Release; Details of Rev./Amendement Objet du document; détails des rév. ou des modif.	Prepared by Rédigé par	Reviewed by Examiné par	Approved by Approuvé par
7		2	2002/12/24	Issued as 'Approved for Use". Document was revised as per resolution of comments from the Technical Review. Additions were made to wording of various sections which were mostly of a clarification nature.	S.D. Nam	V. Snell P. Allsop A. Stretch C. Boss/B. Rouben K. Hau E. Choy J. Millard M. Elgohary A. Josefowicz R. Ghai/O. Hines P. Chan H. Johal J. Tong J. Waddington V. Langman N. Barkman P. Lee H. Johal	M. Bonechi

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. Proj. conn.	Project Projet	SI	Section	Serial Série	Sheet Feuille No. N°	Of De	Unit No.(s) Tranche n°
	108	03650	SDG	007	2	4	

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Release and Revision History / Liste des documents et des révisions

0939B Rev. 13

Document Details / Détails sur le document

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Radiation Protection	43

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Release and Revision History / Liste des documents et des révisions

Release Document No./N°	Date	Revision Révision No./N°	Date	Purpose of Release; Details of Rev./Amendement Objet du document; détails des rév. ou des modif.	Prepared by Rédigé par	Reviewed by Examiné par	Approved by Approuvé par
8		3	2004-03-15	Issued as 'Approved for Use". The following sections are revised as per ACR answers to CNSC review and comments: Sections 1, 3.1, 3.1.1, 3.1.2, 3.4.1, 3.4.2, 3.4.5, 4.2, 4.5, 4.6 and Appendix C. Changes are identified by underlining and major text additions and deletions are noted in this revision history sheet. Revisions consist of the following: • Section 1: removed last sentence in the fourth paragraph • Section 3.1: replaced CNSC Regulatory Guide G-129 by CNSC Radiation Protection Regulations • Section 3.1.1: revised radiation exposure requirements to address the effective dose limits and deleted last sentence • Section 3.1.2: revised regulations to address the action levels of Radiation Protection Regulations and removed last two paragraphs • Section 3.4: revised to clarify the average annual plant occupational exposure over the plant lifetime • Section 3.4.1: revised to clarify the maximum permissible dose for access control • Section 3.4.2 d): added new text for the requirement for personal protection equipment and revised to clarify the contamination control	S.D. Nam	H. Johal	R. Jaitly / M. Bonechi

DCS/RMS Input / Données SCD ou SGD

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Release and Revision History / Liste des documents et des révisions

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Release Document No./N°	Date	Revision Révision No./N°	Date	Purpose of Release; Details of Rev./Amendement / Objet du document; détails des rév. ou des modif.	Prepared by / Rédigé par	Reviewed by / Examiné par	Approved by / Approuvé par
				• Section 3.4.5: added requirements to determine radiation conditions in containment after an accident • Section 4.3.1 c): added requirements for control of airborne radioactive contamination in inaccessible areas • Section 4.3.2 revised to reflect current design concept about the zoning system • Section 4.4 a) 3): added shielding requirements • Section 4.5: item a) re-arranged requirements and removed items e) and f) • Added references [6], [7] and [8] • Appendix B: revised title of Table B-1 and removed Table B-2 • Appendix C: updated the application of ALARA objective in design and removed third paragraph in Section C.1 • Acronym table made more specific to ACR • Sections 1, 3, 3.2.1, 3.3, 3.3.1, 4.1, 4.3.4, 4.7 and Table D-1 contain minor changes to clarify the requirements. Replaced tritium by airborne radioactive contamination.			

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. / Proj. conn.	Project / Projet	SI	Section	Serial / Série	Sheet / Feuille No. / N°	Of / De	Unit No.(s) / Tranche n°
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1. PURPOSE

This Safety Design Guide describes the radiation protection philosophy, objectives and requirements to be used throughout the design of the plant. The guide identifies the regulatory limits which must be satisfied during operation of the plant, and outlines the principal features of a viable radiation protection design. Finally the guide identifies specific requirements that the systems and structures must satisfy.

The guide identifies the requirement to prepare a Radiation Exposure Control Program, which is a systematic review of the radiation exposures accumulated in the operation, inspection, and maintenance of each system or component.

The Radiation Exposure Control Program will incorporate the As Low As Reasonably Achievable (ALARA[1]) philosophy for radiation doses to plant staff and the public.

Designers should note that this guide only addresses radiation safety. The guide is not intended to address questions of conventional industrial health and safety, environmental protection or radiological protection of non-human biota.

Note that some requirements of this document are not directed just at designers but also at plant operators. The designers are required to ensure that the requirements directed at plant operators are clearly identified in design documents.

[1] Whenever the phrase "As Low As Reasonably Achievable" or "ALARA" is used, "social and economic factors taken into account" is implied, even when not stated.

2. COMPLIANCE

Compliance with the Safety Design Guides is mandatory. A listing of the Safety Design Guides is included in Appendix A. Deviations from the requirements identified in the guides may be allowed, after an internal safety review. All such deviations shall be approved and documented by completion of a Safety Design Guide Supplement, AECL form 0729-00.

3. RADIATION PROTECTION PHILOSOPHY

This section of the Safety Design Guide describes the radiation protection concepts and design philosophy used to develop the requirements. It is to be considered as information for the interpretation and application of safety requirements listed in Section 4.

The objective of the radiation protection philosophy is to ensure that:

1. Regulatory requirements for the radiation exposure of the public and plant staff are satisfied,
2. Radiation exposures during normal plant operation are As Low As Reasonably Achievable (ALARA), social and economic factors being taken into account, and
3. Total station staff exposures average 1 person-Sv/a over the operating life of a single unit plant. An equal exposure would apply to a second or twinned unit.

3.1 Regulatory Dose Limits for Normal Operations

A Canadian licensee of a nuclear facility must comply with the Canadian Nuclear Safety Commission (CNSC) Regulations (Reference [1]). These regulations are based on the recommendations of the International Commission on Radiological Protection (ICRP) as outlined in ICRP 60 (Reference [2]).

The CNSC "Radiation Protection Regulations" (Reference [3]), requires that "every licensee shall implement a radiation protection program and shall, as part of that program, (a) keep the amount of exposure to radon progeny and the effective dose and the equivalent dose received by and committed to persons as low as is reasonably achievable, social and economic factors being taken into account". The ALARA process used is outlined in Section 3.3.

3.1.1 Plant Operating Staff

The CNSC effective dose limits for plant operating staff, Nuclear Energy Workers[2], (NEW) are an average of 20 mSv effective dose per year over a five year period (100 mSv in five consecutive years), with a further provision that the dose not exceed 50 mSv effective dose in any year (Reference [1]). However, an effective dose limit of 20 mSv per year will be used for design purposes; the dose limits adopted for ACR™[*] are listed in Tables 3-1 and B-1.

3.1.2 Members of the Public

In accordance with the CNSC Radiation Protection Regulations, the effective dose limit for a member of the public is set at 1 mSv per year (see Tables 3-1, under "A person who is not a Nuclear Energy Worker" and B-1, under "any other person"). The annual dose limits for members of the public apply to the doses received from all sources of radiation other than natural background and medical exposures.

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[2] Formerly called "Atomic Radiation Workers (ARW)".

[*] ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

Action levels for public exposures should be established for the plant, specified in terms of effluent emission rates, or dose rates, and expressed in a form that compliance can be demonstrated in a practical manner. These action levels are not limits, but values at which the licensee must determine the cause and take appropriate measures to reduce the emissions.

Table 3-1
Whole Body Radiation Dose Limits for Normal Operation

Item	Person	Period	Effective Dose (mSv)
1.	Nuclear Energy Worker, including a pregnant Nuclear Energy Worker	One year dosimetry period	20
2.	Pregnant Nuclear Energy Worker	Balance of pregnancy	1 *
3.	A person who is not a Nuclear Energy Worker	One calendar year	1

Note: * The CNSC radiation dose limit is 4 mSv. The dose limit, 1 mSv, is set as per the international consensus. This requirement should be satisfied by administrative measures rather than by plant design.

3.2 Regulatory Dose Limits Following an Accident

3.2.1 Plant Operating Staff

The Radiation Protection Regulations (Reference [1]) allow plant staff to receive a whole body radiation dose of 500 mSv (and 5000 mSv to the skin) during the control of an "emergency" and the consequent immediate and urgent remedial work. Once the emergency is under control, the doses received as a result of further remedial work shall be limited by the regular occupational dose limits.

For design purposes[3], the accumulated dose limit for a NEW over a 90-day period following an accident shall be less than 50 mSv in an area of continuous occupancy. In this context continuous occupancy is taken to be 24-hour occupancy over the first day and then 8-hour periods per day per operator for 89 days following the accident. After 90 days, the regular occupational dose limits will apply.

For events that result in accident conditions, a post accident habitability study will be prepared during the plant design, which will demonstrate that the above dose limit is satisfied. It will also identify those areas to which access is required to satisfy the safety functions, and in which the external radiation field exceeds 2 mSv/h.

[3] For example, in determining the habitability of a control room.

3.2.2 Members of the Public

The dose limits for members of the public following an accident are specified in CNSC Consultative Document C-6 (Reference [4]), reproduced in Table 3-2. Compliance with these limits will be demonstrated in the safety analyses.

Table 3-2
Dose and Release Limits for Postulated Accidents

Requirement	Event Class				
	1	2	3	4	5
Effective dose (mSv)	0.5	5	30	100	250
Lens of the eye (mSv)	5	50	300	1000	1500
Skin (mSv averaged over 1 cm^2)	20	200	1200	4000	5000
30 day emissions of liquid effluent are within the derived annual emission limits for normal operation	T	T	N	N	N

Notes:

Event Class - Classification of the Event Class are specified in CNSC C-6 (Reference [4]).

T – the limit shall be met by the worst failure sequence in the event class

N – not required

3.3 Limiting Occupational Radiation Exposure

Throughout the design process, designers must be aware that a principal objective is to minimize radiation exposure. This is the basis of the As Low As Reasonably Achievable (ALARA) philosophy as given in the Radiation Protection Regulations and outlined in the CNSC Regulatory Guide G-129 [3].

3.3.1 Radiation Protection Objectives

For normal operation and anticipated operational occurrences, the design must meet two specific radiation protection objectives for limiting radiation exposures:

1. Radiation exposures must be kept within prescribed regulatory limits, and
2. Radiation exposures must be kept as low as reasonably achievable (ALARA), social and economic factors being taken into account.

Compliance with the first objective is demonstrated by comparing the calculated equivalent dose with the dose limit or the prescribed limit specified by the Regulator. (See Section 3.1)

The second objective (the optimization objective or ALARA principle) involves balancing detriments (for example, costs) against benefits, dose reduction. Further information on the ALARA principle is given in References [6], [7] and [8].

3.4 Principles for the Control of Radiation Exposures

The current operating experience from existing CANDU 6 plants shows that the average annual plant occupational exposure is about 1.0~1.5 person-Sv. About 75 percent of the total is accumulated from external radiation and 25 percent of the total is accumulated from airborne radioactive contaminations. Almost 80 percent of the external radiation is accumulated during the annual outage. The airborne radioactive contaminations exposures accumulate almost equally during operation and the annual outage.

For an ACR plant, the design objective is to achieve a reduction in occupational exposure relative to the level for previous CANDU plants. For design purposes, an average of 1.0 person-Sv/a over the plant lifetime with the understanding that some years (e.g., during a maintenance shutdown) may result in collective doses higher than this value and in some years the value will be lower. Ways of controlling exposures in the plant design are identified in the following sections.

3.4.1 Access Control

Access control restricts the movement of personnel immediately outside the nuclear plant, preventing unauthorized access by members of the public and thereby ensuring their safety.

Inside the plant fence, procedural and engineered access controls must prevent access to any area where the external or internal radiation hazard would otherwise result in a Nuclear Energy Worker accumulating the maximum permissible dose (20 mSv) within one shift, that is, several hours or where a dose rate of 250 μSv/h may be present.

Throughout the station, procedural controls should be used to control the access by station staff to areas with a radiation hazard.

3.4.2 Contamination Control

The principles used to minimize the potential hazard from radioactive contamination are:

a) identify and limit the sources or potential sources of radioactive contamination,

b) provide engineered systems, such as zoning, ventilation, driers, monitors, collection facilities, and decontamination facilities to control contamination,

c) administrative procedures, such as monitoring and controlling transport of material between zones, and

d) personal protection equipment such as full plastics with air supply.

An important objective of contamination control is to restrict or limit airborne concentrations of radioactive material in the accessible areas of the reactor building to a committed dose rate of 5 μSv/h, which, if worked in for 2000 hours, would commit a person to 10 mSv of effective whole body dose. Since the shielding for the accessible areas of the plant is designed to meet the dose rate of 5 µSv/h, there will be an external source of 10 mSv of effective whole body dose. Thus an individual working in an environment containing airborne radioactive contamination and external radiation could accumulate 20 mSv over a year. Higher concentrations may exist in normally inaccessible areas, such as the moderator equipment enclosures, D_2O storage tanks, and active waste treatment and other isolated areas. Access to these areas will only be permitted if appropriate protection equipment (plastics) is used.

At shutdown, maintenance activities on the components in the heavy water systems may allow the escape of airborne radioactive contamination to the atmosphere directly or from spills. Doses from such airborne radioactive contamination sources must be controlled by the station operator by appropriate work practices which might include increased ventilation and use of protection equipment at shutdown. Other potential sources of contamination are identified in Table D-1.

3.4.3 Radiation Shielding

The principles used to minimize the hazard from external radiation shall be:

a) to identify the sources;

b) where possible, to limit the source or potential source of external radiation (in some situations it is impossible to reduce a radiation source strength);

c) to provide shielding to limit the external radiation fields; and

d) to provide monitors to alert station staff to an external radiation field.

3.4.4 Waste Management

The plant shall be designed to manage the radioactive solid, liquid, or gaseous waste produced during the operating life of the station. The radioactive waste must be managed in such a way that the radiation exposure of station staff and members of the public meets dose limit and regulatory limits, and is ALARA. Any shipment of radioactive waste from the plant will be subject to the Transport Regulations of the host country (e.g., Reference [5]).

3.4.5 Radiation Monitoring

The radiation monitoring system shall be designed to prevent the inadvertent exposure of plant staff or members of the public under normal operation. In addition, suitable design provisions must be made to enable the plant operators to determine radiation conditions in containment after a LOCA.

4. RADIATION PROTECTION DESIGN REQUIREMENTS

4.1 Station Budget

a) The designer shall review the sources of radiation and radioactive materials for each system, to identify the potential radiation hazards associated with the relevant equipment or system. The major sources for the more important systems are listed, in Table D-1.

b) The designers shall assess the significance of the radiation sources with radiation physics personnel and agree on what engineered features can be used to reduce the exposure. Note that radiation exposures must meet design targets and regulatory limits, and be ALARA.

c) The design target for annual exposures for the plant staff shall be <u>an average of</u> 1.0 person-Sv <u>throughout</u> plant life. Because the achievement of low exposures in an operating station is strongly dependent on the performance of the owners, management, and staff, the design objective is that this target is capable of being met <u>in conjunction</u> with good radiation protection practices.

d) The Radiation Exposure Control Program shall demonstrate that the design target can be met for the anticipated performance of the plant, treated as an average over several years to satisfy outage targets. Exposures accumulated during major refurbishing of the plant (for example, large scale fuel channel replacement) shall be part of the target. Exposures accumulated during a major refurbishment can be treated as an individual review item rather than included in the reviews of individual systems or components.

e) The Radiation Exposure Control Program shall contain an ALARA evaluation of potential options to reduce exposures to the public and to plant staff (see Appendix C).

4.2 Access Control

a) A security fence around the plant shall prevent unauthorized access by any person. This ensures the safety of members of the public, as well as protecting the facility from intrusion.

b) Inside the security fence, engineered access controls shall prevent access to any area where the external or internal radiation hazard would allow a Nuclear Energy Worker (NEW) to accumulate the <u>administrative dose limit of 10 mSv</u> in a five hour period. That is, access controls shall prevent access to an area:

   1) with an external radiation hazard of 2 mSv/h or greater,

   2) with an internal (<u>airborne radioactive contamination</u>) hazard of 2 mSv/h or greater, or

   3) with external and internal hazards whose total effect would deliver the <u>administrative dose limit of 10 mSv</u> in five hours, e.g., 1.5 mSv/h (external) and 0.5 mSv/hr (internal).

c) Access controls shall minimize the possibility of a change in radiation hazard once access is made. Overrides to engineered access controls shall be avoided. However, should the external radiation hazard <u>increase</u> during access such that the accumulated dose in 5 hour may exceed 10 mSv, then monitors should alarm and prompt immediate evacuation.

d) Entry/exit barriers shall be designed to permit egress from an access controlled area without tools or external help (other than designed into the mechanism). The operation of emergency egress features shall be indicated in the control room.

e) Elsewhere in the station, procedural controls and locked doors shall control the access by station staff to areas where a potential radiation hazard exists that could cause dose limits to be exceeded.

4.3 Contamination Control

4.3.1 Control of Airborne Radioactive Contaminations

a) The contamination control features shall limit dose due to airborne radioactive contamination in the accessible areas of the reactor building and heavy water management areas outside the reactor building to a design target of 5 μSv/h.

b) In other areas of the plant which are accessible during normal operation excluding heavy water management areas outside the reactor building, the dose rates due to airborne radioactive contamination shall be less than 1 μSv/h.

c) Limit the dose rate in inaccessible areas such as the moderator equipment rooms to 500 μSv/h. However use of protective equipment will reduce the internal dose rate to 10 μSv/h or less.

d) Every effort shall be made to reduce airborne radioactive contamination concentrations at shutdown. When such systems must be open for maintenance or other reason, provisions are made to control airborne radioactive contamination exposure.

e) The Design Requirements for the Reactor Building Ventilation System (ASI 73120) shall require identification of confinement barriers or other equipment credited in achieving the airborne radioactive contamination concentration design limits.

4.3.2 Zoning System

a) A zoning system shall be used to classify all areas of the station according to the levels of potential or actual contamination.

b) The zoning system shall direct the movements of personnel or equipment and thereby control the spread of contamination.

c) The designers shall consider no more than four classifications, as follows:

    1) ZONE 0 contains no radioactive equipment and is normally free from contamination. However, radioactive material may move through the area provided it is suitably contained. Eating is permitted in designated areas. To be consistent with the zoning classification, the external radiation field shall be less than 0.5 μSv/h. The internal hazard from airborne radioactive contaminations in air shall also be less than 0.5 μSv/h. Typically, this Zone includes parts of the yard area and buildings which would be expected to be free of contamination (e.g., pump house).

2) ZONE 1 contains no radioactive equipment and is free from contamination. Eating is permitted in designated areas. No form of radioactivity is allowed to enter this zone. Typically, this includes the administration and engineering offices.
3) ZONE 2 contains no radioactive equipment and is normally free from contamination. However, the movement of radioactive material from Zone 3 can temporarily create contamination, which will be cleaned up on completion of the activity. Typically, this includes parts of the service building, reactor auxiliary building and turbine building.
4) ZONE 3 contains radioactive material and equipment and the presence of contamination should be expected. Any sources of contamination are localized and under control. Typically, this includes reactor building and parts of reactor auxiliary building.

The designers may choose to simplify this zoning system.

d) Movements between these zones shall be restricted by physical barriers (e.g., walls and doors) and procedural controls.
e) Movements of personnel from Zone 3 to Zone 2 or from Zone 2 to Zone 1 shall be monitored.
f) A full body contamination monitor shall be provided at the interfaces between Zone 2 and Zone 1.
g) Provision of suitable power outlets and space allocations for hand/foot and frisker monitors and whole body monitors shall be made at all interzonal boundaries.
h) Sufficient space and Class II power outlets shall be provided for the monitors.

4.3.3 Ventilation System

a) A ventilation system shall be designed to control the concentration of airborne contamination, and maintain a flow pattern from areas of lower contamination to areas of higher contamination.
b) This ventilation system shall be complemented by a system of containment and confinement which will localize any areas with high levels of contamination. The layout of equipment shall consider the need to create temporary confinement areas around individual components.
c) As far as practical, the design of ventilation system shall consider all modes of plant operation, such as maintenance and emergency situations.

4.3.4 Driers

a) A system of driers shall be provided to reduce airborne radioactive contamination concentrations in the air in appropriate confinement areas. The designers shall consider the need to reduce airborne radioactive contamination concentrations at shutdown as well as at power.

4.3.5 Liquid Waste and Decontamination

a) A system of radioactive drains and liquid waste treatment shall be provided to collect and manage liquid contamination.

b) A decontamination centre to reduce the levels of particulate contamination on components or equipment shall be provided.

4.4 Shielding

a) The radiation fields shall be designed to achieve operating and shutdown fields comparable with or lower than those of CANDU 6. To that end the designers shall provide shields to reduce dose rates to:

1) less than 0.5 μSv/h in areas normally occupied by non-Nuclear Energy Workers;
2) less than 5 μSv/h in areas of high occupancy by Nuclear Energy Workers;
3) less than 25 μSv/h inside the reactor building and other areas of low occupancy by Nuclear Energy Workers;
4) less than 250 μSv/h in controlled access areas, accessed by Nuclear Energy Workers on an infrequent basis, e.g., at shutdown (this is a target, which may be adjusted based on the expected occupancy of the area).

b) External radiation exposures from neutrons shall be minimized through use of appropriate shielding.

c) The post-LOCA shielding requirements shall be established. (See Section 4.8)

4.5 Waste Management

a) The designers must ensure that, exposure of the public from all radionuclides in the effluent streams is less than 1 mSv/a. To ensure this limit is met, the designer will need to calculate Derived Emission Limits (DELs) for the gaseous and liquid effluent streams.

b) Releases of liquid and gaseous waste shall be controlled and monitored.

c) Releases of liquid and gaseous waste shall be quantified. That is, for any release of radioactivity the radionuclides must be identified and the amount of activity released measured.

d) The waste management systems shall have a dose target of 50 μSv per year averaged over the plant lifetime at the exclusion area boundary. Should the dose exceed this target, the systems shall be subject to ALARA optimization (see Appendix C).

4.6 Radiation Monitoring

a) The monitoring system shall:

1) detect the presence of, and locate failed fuel in the core;
2) measure the external radiation fields around the plant, and survey personnel, work locations and equipment for any sources of contamination (special attention should be given to monitoring for airborne radioactive contamination at work locations with consideration given to alarms in the control room);
3) provide a system of personnel dosimetry;

4) monitor the release of airborne and liquid effluents from the station and alarm upon the release of activity concentration above some preset level;
5) provide offsite monitoring of radioactive material in environmental media;
6) provide a low level radiation laboratory for the analysis of environmental and biological samples and the monitoring of individuals; and
7) provide a calibration facility to calibrate all radiation monitors used in the plant.

b) In the event of an accident the radiation monitoring system shall provide:
   1) data needed to evaluate the radiological conditions around the plant and assess the need for off-site action;
   2) data on the ambient and radiological conditions inside the plant to aid safe evacuation and to minimize exposure of the operating staff in the performance of necessary accident related work activities.

4.7 Radiation Exposure Control Program

a) The designers shall demonstrate compliance with the overall station budget by estimating the exposures associated with each system or component in the Radiation Exposure Control Program.

b) The Radiation Exposure Control Program shall estimate the exposures involved in the operation, maintenance, inspection, repair or replacement of any equipment in the nuclear island. The program shall include the explanation or justification of the exposures involved.

c) Designers shall include an ALARA review for exposures of the public and plant staff, to comply with the requirements of G-129 (Reference [3]), to the extent applicable to design. See Appendix C to determine when an ALARA analysis must be performed.

d) Any system, equipment, component or activity that involves releases resulting in public exposure or significant exposure of plant staff shall be evaluated under the ALARA process (see Appendix C), unless an ALARA analysis is shown not to be required in Step c.

e) For consideration of exposure of plant staff, the inputs to the program are:
   1) a breakdown of work activities on a system,
   2) the number of operators involved in each work activity (including operators involved in a support function),
   3) the time taken for each activity, and
   4) the external radiation field and internal airborne hazards.

f) Operating experience from previous CANDU plants shall be used throughout the review. Where CANDU operating experience is unavailable, other experience or engineering judgement can be used.

4.8 Post Accident Habitability

a) The post accident habitability study shall:

1) Identify those areas of the station where access is required after an accident,
2) Calculate the effective doses up to 90 days, and
3) Identify any design changes needed to comply with the targets for the various types of accidents. The fission product source strengths and the distribution of the fission products within the plant (source terms) should take account of the time dependence throughout the 90-day period. The source terms should be taken from the appropriate safety analyses.

5. DOCUMENTATION

a) The findings of the Radiation Exposure Control Program shall be reported in a formal project document.

b) The requirements for:

   1) shielding,

   2) solid waste management system(s),

   3) liquid waste management system(s), and

   4) gaseous waste management system(s)

   shall be established in the corresponding Design Requirements documents. The specific features shall be identified in the corresponding Design Description documents.

c) The need for the consideration of radiation protection in the design of systems or equipment forming part of the nuclear island shall be identified in the Design Requirements for that system. The Design Requirements shall state that the system or equipment will be reviewed as part of the Radiation Exposure Control Program. The principal features of the radiation protection provided shall be identified in the Design Description for that system.

d) Where the structure acts as a radiation protection shield or confinement barrier the requirements shall be stated in the Design Requirements document. The specific design features shall be identified in the Design Description for that structure.

e) The ALARA analyses of each system subject to ALARA shall be documented. Where a system is exempted from ALARA analysis the rationale shall be documented.

f) Deviations from the requirements of this Safety Design Guide shall be recorded and approved as outlined in Section 2.

6. REFERENCES

[1] Canadian Nuclear Safety Commission, “Radiation Protection Regulations”, 2000-05-31.

[2] Annals of the ICRP, “1990 Recommendations of the International Commission on Radiological Protection”, ICRP Publication 60, Pergamon Press, 1991.

[3] CNSC Regulatory Guide G-129, “Guidelines on How to Meet the Requirement to Keep All Exposures As Low As Reasonably Achievable”, 1997-09-26.

[4] CNSC Consultative Document C-6, Rev. 1, “Requirements for the Safety Analysis of CANDU Nuclear Power Plants”, issued for comment 1999-09.

[5] Canadian Nuclear Safety Commission, “Packaging and Transport of Nuclear Substances Regulations”, 2000-05-31.

[6] <u>Annals of ICRP, “Optimization and Decision-Making in Radiological Protection”, ICRP Publication 55, Pergamon Press, 1988.</u>

[7] <u>Safety Series No. 50-SG-D9, “Design Aspects of Radiation Protection for Nuclear Power Plants”, IAEA Safety Guides, IAEA, Vienna, 1985.</u>

[8] <u>Safety Series No. 101, “Operational Radiation Protection A Guide to Optimization”, IAEA Safety Guides, IAEA, Vienna, 1990.</u>

Appendix A

List of Safety Design Guides

Identification	Title
108-03650-SDG-001	Safety Related Systems
108-03650-SDG-002	Seismic Requirements
108-03650-SDG-003	Environmental Qualification
108-03650-SDG-004	Separation of Systems and Components
108-03650-SDG-005	Fire Protection
108-03650-SDG-006	Containment
108-03650-SDG-007	Radiation Protection

Appendix B

Non-Stochastic Radiation Dose Limits

Table B-1
Non-Stochastic Radiation Dose Limits for Normal Operation

Item	Organ or Tissue	Person	Period	Equivalent Dose (mSv)
1.	Lens of an eye	(a) Nuclear energy worker	One year dosimetry period	150
		(b) Any other person	One calendar year	15
2.	Skin (averaged over 1 cm^2 that receives highest dose)	(a) Nuclear energy worker	One year dosimetry period	500
		(b) Any other person	One calendar year	50
3.	Hands and feet	(a) Nuclear energy worker	One year dosimetry period	500
		(b) Any other person	One calendar year	50

Appendix C

Application of ALARA in Design

The Radiation Exposure Control Program will ensure that the radiation dose limits for plant staff and the public can be met, and will ensure that the ALARA objective is satisfied by:

1. identifying and evaluating potential means of reducing dose for planned operating and maintenance activities for each system, and anticipated operational occurrences; and
2. documenting the rationale for accepting or rejecting the means identified in 2.

The ALARA principle requires the minimization of radiation exposure consistent with overall project requirements. To minimize radiation exposure, designers must review the operating experience from previous CANDU plants, for the corresponding system or major piece of equipment. That operating experience should include details of equipment reliability, frequency of maintenance, inspection or repair, as well as the station exposures associated with the equipment. The radiation sources leading to those exposures should be understood. Table D-1 lists some of the principal radiation sources associated with the major systems.

Designers should concentrate on those features of previous CANDU plants which make the largest contributions to exposures of the public and plant staff. Particular efforts should be made to reduce these exposures. Designers should also give special attention to features that are different from earlier CANDU designs. For example, the use of Slightly Enriched Uranium (SEU) fuel with a more demanding duty cycle and higher burnup, or the failed fuel location strategy.

Generally, radiation exposure of plant staff can be minimized by limiting the number of staff that must enter containment, the frequency with which they must enter, and the time spent in containment. This involves careful attention to layout, access, and shielding, taking into account past experience and station data, and exercising good engineering judgement where appropriate. Suggested options for reducing radiation exposure, ranked in order of their expected effectiveness, are:

a) minimize amount of equipment in areas subject to a radiation hazard,

b) reduce number of equipment (e.g., by using larger capacity pumps or heat exchangers),

c) simplify equipment for a system to reduce maintenance durations,

d) relocate equipment to a lower radiation field area (e.g., long stem valves, motorized valves),

e) improve chemical control and purification,

f) provide more reliable equipment to ensure a longer interval between maintenance,

g) minimize the use of materials which may become activated due to exposure to the neutron flux of the reactor, including products of corrosion or wear (e.g., cobalt, antimony),

h) arrange for quick removal of equipment for shop maintenance (e.g., plate type rather than tube-in shell heat exchangers),

i) arrange for shorter time required for in-situ maintenance or inspections (e.g., doorway into steam generator chimney to access lower manway, provide more space between equipment, minimize use of scaffolding),

j) control leakage from systems handling water containing activity (including heavy water and light water) (e.g., leakage collection, fume hoods),

k) design ventilation to minimize airborne radioactive contamination hazards,

l) provide good monitoring coverage for radiation hazards (e.g., airborne radioactive contamination monitoring in all areas),

m) provide facilities for removal of certain hazardous radionuclides (e.g., airborne radioactive contamination), and

n) provide shielding.

Radiation exposure of the public can be reduced by the reduction of radionuclides released from the plant as gaseous or liquid effluents, particularly the longer lived radionuclides.

Generally radioactive wastes from the plant can be reduced by controlling the production of wastes at source. Thus the efforts to reduce the use of materials such as cobalt and antimony that become active will also reduce the amount of activity handled in the liquid waste management system. However, other options to reduce public exposures include:

o) strict control of air flows between areas with airborne contamination including atmospheric segregation of area with possible airborne hazard,

p) design of equipment that eliminates any traps for contaminated particles that will be removed in a decontamination facility, and

q) design of equipment that eliminates or reduces operator actions on equipment, that is automated equipment.

C.1 Determining if an ALARA Assessment is Required

The CNSC accepts that there may be dose levels for which it would not be financially justifiable to undertake an ALARA assessment. To minimize the commitment of resources which are likely to have a poor return in improvement of safety, further ALARA analysis is not normally required if the following criteria are met (Reference [3]):

1. individual occupational doses are unlikely to exceed 1 mSv per year,
2. dose to individual members of the public is unlikely to exceed 50 micro-sievert per year, and
3. the annual collective dose (both occupational and public) is unlikely to exceed 1 person-Sv.

In some situations, a decision is required on whether it is economically justifiable to take action to reduce dose levels. According to the CNSC Regulatory Guide G-129 (Reference [3]), if an expenditure in excess of $100,000 to reduce collective dose by 1 person-Sv is required, the action is not economically justifiable.

There may be situations (e.g., the collective dose may be less than 1 person-Sv, a limited number of persons may still be receiving a significant fraction of the individual dose limit) in which it would be appropriate for the designer to carry out an ALARA review even if doses are less than the criteria given above. Designers are encouraged to reduce doses where this can be done without significant expenditure.

The steps in determining if an ALARA assessment is required are as follows:

a) Identify systems and practices that results in radiation exposure to people.

b) Estimate the total doses resulting from identified systems and practices.

c) If the doses are above the Dose Limit given in Section 3.1, radiation protection must be improved.

d) If the doses are below the criteria given in points 1, 2 and 3 above, no further ALARA assessment is required.

e) If the doses are above the criteria given above, an ALARA assessment is required, as described in Section C.2 below.

The process and its results must be documented, as required in Section 5.

C.2 ALARA Assessment Process

When an ALARA assessment is to be undertaken, it is necessary to determine what options are available for further reduction of radiological dose. The options are then evaluated using quantitative and/or qualitative techniques to decide which, if any, measures are to be implemented.

In determining if a dose is ALARA, it is necessary to determine whether further risk reduction would require a grossly disproportionate expenditure to reduce the dose. In performing the cost benefit analysis, $100,000 per person Sievert averted is used as a rough measure for indicating the appropriate level of expenditure. Other factors, such as organizational values, regulatory requirements, non-radiological detriments and public expectations are also taken into account. These may over-ride considerations of cost.

The steps in the process are as follows:

1. Define the situation which requires consideration of a reduction in dose (see Section C.1).
2. Identify the options for dose reduction and factors to be assessed.
3. Evaluate the options using quantitative techniques or judgements (see Section C.3).
4. Evaluate other non-radiological and non-quantifiable factors.
5. Decide which, if any, options to implement.

The evaluation process, the options considered and the rationale for accepting or rejecting options is to be documented, as required in Section 5.

This process is illustrated in Figure C-1.

C.3 Judgement of Reasonableness

In the ALARA process, it must be decided whether the benefit of efforts to reduce doses is worthwhile. Some problems may be resolved using cost benefit analysis or other quantitative techniques. Many other situations, however, do not lend themselves to quantitative analysis. It must be recognized that such techniques have limitations, and it may not be possible to quantify all the factors involved, such as the balance between collective and individual doses, the rate at which dose is received, and broader social factors.

The judgement of reasonableness is inherent in the ALARA process. The following are suggested to help judge if an action is reasonable (Reference [3], Section F):

1. Common sense: this reflects experience, knowledge and the exercise of professional judgement. For example, a very low cost, yet practical, change that reduces dose should probably be made even if doses are already low.
2. Good practice: this involves comparing with other similar facilities.
3. Feasibility: this involves assessing if the change can be made in a practical fashion.

In order to substantiate the judgement of how reasonable a proposal is, the designer must document the basis of the judgement, as required in Section 5. Where relevant to a particular situation, this could include:

a) design options considered for the reduction of dose;

b) review of the radiation protection experience relating to the proposed options from operating plants, where this information can be obtained (e.g., dose records, effluent releases, incident reports, etc.); and

c) rationale for either adopting or rejecting the proposed design options for the reduction of dose.

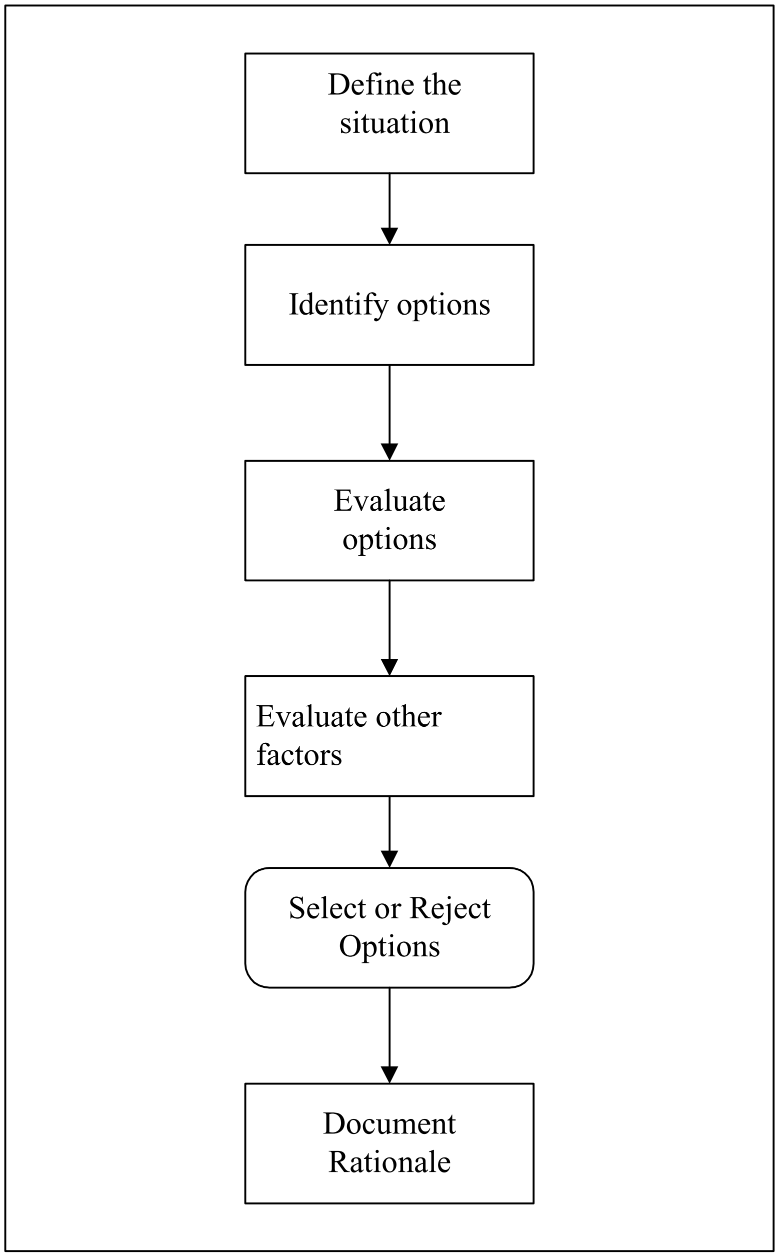

Figure C-1 ALARA Assessment Process

Appendix D

Review of Radiation Hazards Associated with Systems and Assemblies

Table D-1 shows the principal radiation hazards from the different process systems and components. The purpose is to alert the designer to the principal radiation hazards associated with that system or equipment. The Table D-1 shows the hazards with the reactor at power and shutdown and groups the hazards as external, that is, the radiation dose originates from outside the body and internal, that is, the radiation hazard exists from activity that is taken into the body.

Table D-1
Principal Radiation Hazards

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
24250	Spent Fuel Bay	Fission product decay gammas from irradiated fuel.		Fission product decay gammas from irradiated fuel.		
30000	Reactor, Reactor Systems & Auxiliaries	See below for details.	See below for details.	See below for details.	See below for details.	
30100	Nuclear Safeguards System and Equipment	Fission product decay gammas from irradiated fuel.		Fission product decay gammas from irradiated fuel.		

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
30700	Special and Consumable Materials	^{41}Ar decay gammas from argon impurity in carbon dioxide. Corrosion and fission product activities collected on ion exchange resins and filter elements.	^{14}C activity in carbon dioxide.	^{41}Ar decay gammas from argon impurity in carbon dioxide. Corrosion and Fission product activities collected on ion exchange resins and filter elements.	^{14}C activity in carbon dioxide.	
30800	Process System Components	^{16}N decay gammas and photoneutrons in heat transport system; ^{16}N, ^{19}O and ^{17}F decay gammas and photoneutrons in moderator systems (see below).	Airborne radioactive contaminations from heat transport[4] and moderator systems.	Fission products and corrosion products deposited on swept surfaces of heat transport system and corrosion products deposited on moderator system surfaces.	Airborne radioactive contaminations from heat transport and moderator systems.	Cobalt content of steel components in heat transport and moderator system must be restricted. Stellite materials must be avoided.
30900	Misc. Components for Nuclear Process Systems	Decay gammas from activation products in system (see below breakdown for important systems).	Airborne radioactive contaminations from moderator.	Deposited corrosion product activities.	Airborne radioactive contaminations from moderator.	
31000	Reactor	See below for details.	See below for details.	See below for details.	See below for details.	

4 The airborne radioactive contaminations hazard associated with the moderator will be orders of magnitude greater than that associated with the HTS, owing to the use of light water in the HTS. In this table, the airborne radioactive contaminations hazard associated with the HTS will only be identified separately where the moderator tritium is not a factor.

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
31100	Fuel Channel Assemblies	Neutrons from fuel, prompt gammas from fuel and calandria.		Fission product decay gammas; ^{95}Zr, ^{95}Nb and ^{60}Co decay gammas from pressure tube. ^{60}Co decay gammas from end fittings activation. ^{60}Co decay gammas from deposited corrosion product activity.	^{14}C activity from carbon dioxide in gas annulus deposited on pressure tube and calandria tube, etc.	All fuel channel hardware including pressure tubes must have a specified limit on cobalt content. ^{94}Nb formed in irradiation of niobium creates a long-term decommissioning hazard.
31200	Calandria Shield Tank Assembly	Prompt gammas from neutron capture in calandria shell and tubesheets. Neutrons from fuel.	Airborne radioactive contaminations from moderator.	^{60}Co decay gammas from activation of calandria shell and tubesheets. Fission product decay gammas from fuel.	Airborne radioactive contaminations from moderator.	Calandria main shell, annular plate sub-shell and tubesheets must have a specified limit on cobalt content. Shield tank must be filled with water while irradiated fuel is present in the reactor.

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
31700	Reactivity Control Units	Neutrons from fuel, prompt gammas from capture in calandria shell or lower end of thimbles.	Airborne radioactive contaminations from moderator.	^{60}Co decay gammas from activation of lower end of thimbles. ^{95}Zr and ^{95}Nb gammas from in-core section of guide tubes and LISS nozzles. ^{60}Co decay gammas from out of core section of guide tubes.	Airborne radioactive contaminations from moderator.	Shutoff rod elements should be relatively inactive. Lead impurities in ion chamber housing must be strictly controlled to ensure proper operation of ion chambers at shutdown.
31710	Reactivity Mechanisms Deck	Neutrons from fuel, prompt gammas from neutron capture in calandria shell. Streaming through the vertical reactivity mechanism penetrations.	Airborne radioactive contaminations from moderator.	^{60}Co decay gammas from activation of lower end of thimbles and from parked reactivity devices.	Airborne radioactive contaminations from moderator.	Reactivity mechanisms must be parked in core at shutdown particularly for access around shield tank. Shield tank must be filled with water while irradiated fuel exists in the reactor.

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
31800	Reactor Shielding	Neutrons from fuel, gammas from fuel and activation of core internals and reactor shielding		Sub-critical neutrons from fuel, decay gammas from fuel and activation of core internals and reactor shielding		Activation of shielding materials is hazard for removal and decommissioning
31900	Installation and Maintenance Equipment	Not applicable.	Not applicable.	Fission products decay gammas. ^{60}Co decay gammas from corrosion, products, ^{95}Zr, ^{95}Nb and ^{60}Co decay gammas from components.	Airborne radioactive contaminations from moderator. High loose surface contamination.	Reactor maintenance equipment has to protect operators from fields from fuel in the core, from installed reactor equipment and from reactor components being removed. It should not pick up surface contamination and it should be easy to decontaminate.
32100	Moderator Systems	^{16}N, ^{19}O and ^{17}F decay gammas from activity induced in heavy water. Photoneutron source from photodisintegration of deuterium driven by gammas from ^{16}N.	Airborne radioactive contaminations from moderator.	^{60}Co decay gammas from transport of activated corrosion products of stainless steel.	Airborne radioactive contaminations from moderator.	Low cobalt content steels should be used in system.

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
32200	(Moderator) Purification System	^{19}O and ^{17}F decay gammas.	Airborne radioactive contaminations from moderator.	^{60}Co decay gammas from corrosion products.	Airborne radioactive contaminations from moderator.	^{16}N activity will normally decay in main circuits and be insignificant in auxiliary circuits.
32300	(Moderator) Cover Gas System	^{41}Ar decay gammas from activation of argon impurity in helium.	Airborne radioactive contaminations from moderator.	^{41}Ar decay gammas.	Airborne radioactive contaminations from moderator.	
32500	(Moderator) Heavy Water Collection Systems	Corrosion product activity in heavy water.	Airborne radioactive contaminations from moderator.	Corrosion product activity.	Airborne radioactive contaminations from moderator.	
32600	(Moderator) Heavy Water Sampling Systems	^{16}N, ^{19}O and ^{16}F decay gammas from activity induced in heavy water.	Airborne radioactive contaminations from moderator.	Corrosion product activity.	Airborne radioactive contaminations from moderator.	

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
32700	(Moderator) Liquid Poison Systems	Not applicable.	Airborne radioactive contaminations from moderator.	Not applicable	Airborne radioactive contaminations from moderator.	Activation products in the heavy water should be small since the system is fed from the D_2O supply system.
33100	Heat Transport Circuit	^{16}N decay gammas: Photoneutron source from photo-disintegration of deuterium.	Airborne radioactive contaminations from lithium in HTS water[5].	Corrosion and fission product activity.	High loose surface contamination. Radioactive iodine. Airborne radioactive contaminations from lithium in HTS water.	Fuel defects must be removed from the core promptly. The chemistry of the heat transport system must be carefully specified. The steel including all material used in the heat transport system must have limits on cobalt and antimony contents.
33310	Heat Transport Pressure and Inventory Control System	^{16}N, ^{19}O and ^{18}F decay gammas. Fission products from defect fuel.	Airborne radioactive contaminations from lithium in HTS water.	Corrosion and Fission product activity in heavy water collected as off-gas and deposited on wetted surfaces.	Airborne radioactive contaminations from lithium in HTS water.	

5 The HTS airborne radioactive contaminations content will be very low compared to the moderator due to the use of light water in the HTS. In addition depleted lithium (low Li-6 content) will be used in both the chemicals and IX resins used for chemical control of pH in the coolant.

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
33350	Heat Transport Purification System	^{16}N, ^{19}O and ^{18}F decay gammas. Corrosion and fission product activity collected on filters, etc.	Airborne radioactive contaminations from lithium in HTS water.	Corrosion and fission product activity.	Airborne radioactive contaminations from lithium in HTS water.	
33700	Heat Transport Water Sampling System	^{16}N, ^{19}O and ^{18}F decay gammas. Deposited corrosion and fission product activity.	Airborne radioactive contaminations from lithium in HTS water.	Corrosion and fission product activity deposited on piping surfaces.	Airborne radioactive contaminations from lithium in HTS water.	
33810	Heat Transport Water Collection System	Corrosion and fission product activity deposited on piping surfaces.	Airborne radioactive contaminations from lithium in HTS water.	Corrosion and fission product activity deposited on piping surfaces.	Airborne radioactive contaminations from lithium in HTS water.	
34100	Cooling Systems	^{16}N and ^{19}O plus corrosion product activities in shield cooling system. Fission products and corrosion products activities in irradiated fuel cooling and purification system.		^{16}N and ^{19}O plus corrosion product activities in shield cooling system. Fission products and corrosion products activities in irradiated fuel cooling and purification system.		
34300 34320	Emergency Core Cooling System Emergency Coolant Injection System	None	None	Fission product decay gammas following an accident.	None	None

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
34350	Long term Cooling Systems	Deposited corrosion and fission product activity.	Airborne radioactive contaminations from lithium in HTS water.	Corrosion and fission product activity deposited on piping surfaces.		
34410	(Spent Fuel Bay) Cooling and Purification System	Fission products from fuel defects.		Fission products from fuel defects.		The purification system will not control the gaseous fission products from defective fuels.
34510	Resin Transfer System	Corrosion product activities on resins	None	Corrosion product activities on resins	None	None
34700	Liquid Injection Shutdown (System)	None	Airborne radioactive contaminations from moderator heavy water.	Fission product decay gammas streaming down a drained penetration.	Airborne radioactive contaminations from moderator heavy water.	
34900	Miscellaneous Reactor Process Systems (Annulus Gas)	^{41}Ar decay gammas from argon present as a result of air ingress to system.		^{41}Ar decay gammas from argon impurity in carbon dioxide.	Airborne radioactive contaminations from moderator system. ^{14}C in gas annulus system.	^{14}C activity formed in gas annulus system deposited on pressure tube and calandria tube to create contamination hazard during fuel channel replacement.
35100	New Fuel Transfer and Storage	Beta particle and gamma dose rates from actinide activity.	Alpha particle activity of actinides.	Beta particle and gamma dose rates from actinide activity.	Alpha particle activity of actinides.	

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
35200	Fuel Changing	Neutrons from fuel, prompt gammas from fuel and tubesheets leaving the end shield. ^{16}N activity in heat transport system. Fission products decay gammas from irradiated fuel.	During maintenance work on a spare fuelling machine when the reactor is operating there will be a surface contamination hazard from the components in that spare machine.	Fission product decay gammas. Beta radiation	High loose surface contamination. Radioactive iodine	Cobalt content of material in the fuelling machine exposed to the heat transport water must be restricted.
35300	Spent Fuel Transfer and Storage	Fission product decay gammas. Beta radiation	Gaseous fission products from defect fuel. High loose surface contamination. Radioactive iodine	Fission product decay gammas. Beta radiation	Gaseous fission products from defect fuel. High loose surface contamination. Radioactive iodine	
35510	Fuel Channel Component Handling Tools	Not applicable.	Not applicable.	^{95}Zr and ^{95}Nb decay gammas from pressure tube; ^{60}Co decay gammas from end-fitting and shield plug.	<u>Airborne radioactive contaminations</u> from moderator system. High loose surface contamination.	

ASI	System or Assembly	Principal Radiation Hazards				Comments
		At Power		At Shutdown		
		External	Internal	External	Internal	
37000	Fuel	Neutrons from uranium fission, prompt gammas and delayed gammas from fission product and actinide decay.		Fission product decay gammas; ^{95}Zr, ^{95}Nb and ^{60}Co decay gammas from sheath.		Fuel sheath must have a specified limit on cobalt content. ^{94}Nb formed in irradiation of niobium creates a long-term decommissioning hazard.
38100	Heavy Water Supply System	Not applicable.	Not applicable.		Airborne radioactive contaminations from moderator heavy water.	
38310	Heavy Water Vapour Recovery System	Not applicable.	Airborne radioactive contaminations from moderator heavy water.		Airborne radioactive contaminations from moderator heavy water.	
67312	(Reactor Building) Ventilation System	Fission and corrosion products collected on exhaust filter.		Fission and corrosion products collected on exhaust filter.		Precautious must be taken during HEPA and charcoal filter replacement (inhalation hazard) operations.

Appendix E

Acronyms

AC	Alternating Current
ACR™*	Advanced CANDU Reactor™
AECL	Atomic Energy of Canada Limited
ALARA	As Low As Reasonably Achievable
ASDV	Atmospheric Steam Discharge Valves
ASI	AECL Subject Index
BOP	Balance Of Plant
CA	Control Absorber
CANDU®	Canadian Deuterium Uranium
CCP	Critical Channel Power
CCW	Condenser Cooling Water
CHF	Critical Heat Flux
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	The Canadian Standards Association
D_2O	Heavy Water
DBE	Design Basis Earthquake
DC	Direct Current
DCS	Distributed Control System
DEL	Derived Emission Limit
DG	Diesel Generator
EAB	Exclusion Area Boundary
ECCS	Emergency Core Cooling System
ECI	Emergency Core Injection
EDS	Electrical power Distribution System
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
ICRP	International Commission for Radiation Protection
ISO	International Organization for Standardization
LCDA	Limited Core Damage Accident
LOCA	Loss Of Coolant Accident
LTC	Long Term Cooling
LWR	Light Water Reactor

* ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Limited (AECL).

® CANDU is a registered trademark of Atomic Energy of Canada Limited.

MCR	Main Control Room
MOT	Main Output Transformer
MSIV	Main Steam Isolation Valves
MSSV	Main Steam Safety Valves
NEW	Nuclear Energy Worker
NSP	Nuclear Steam Plant
NSSS	Nuclear Steam Supply System
OM&A	Operation, Maintenance and Administration
PAM	Post Accident Management
PSA	Probabilistic Safety Assessment
PTR	Pressure Tube Reactor
PWR	Pressurized Water Reactor
RAB	Reactor Auxiliary Building
RB	Reactor Building
RCU	Reactivity Control Unit
RCW	Recirculated Cooling Water
RSW	Raw Service Water
RWS	Reserve Water System
SCA	Secondary Control Area
SDS 1	Shut Down System 1
SDS 2	Shut Down System 2
SEU	Slightly Enriched Uranium
SFC	Single Failure Criterion
SST	System Service Transformer
SU	Shutoff Unit
ULC	Underwriter's Laboratories Canada
UPS	Uninterrupted Power Supply
UST	Unit Service Transformer
ZCU	Zone Control Unit