ACR-1000® Design Features Minimizing Collective Occupational Radiation Exposures and Public Dose to ALARA

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Abstract

The ACR-1000®[1]nuclear power plant (NPP) is the next evolution of the proven CANDU®[2] reactor design. The ACR-1000 NPP is designed to meet the International Commission on Radiological Protection (ICRP) fundamental objectives of radiation protection and comply with Canadian Nuclear Safety Commission (CNSC) regulatory requirements for radiation protection for plant operating staff and the public, including CNSC guidelines for keeping radiation exposures As Low As Reasonably Achievable[3] (ALARA), taking into account social and economic factors.

During the ACR-1000 NPP design phase, a design improvement program was developed to minimize the collective occupational dose and public dose based on the ALARA approach, taking into consideration of technology advancement, the specific design features, operation and maintenance issues, and lessons learned from the operation of CANDU NPPs. The evolutionary ACR-1000 design features significant radiation exposure control improvements based on the latest technology and industry best practices with respect to the limitation of potential doses and the minimization of radionuclide production at source for all phases of the NPP life cycle (i.e. design, commissioning, operation, maintenance and decommissioning).

This paper summarizes the design aspects of the ACR-1000 NPP that contribute to minimize occupational radiation exposures and the public dose to ALARA during normal operations. As a result, it reduces collective occupational dose below the target defined by the Institute of Nuclear Power Operations (0.6 person-Sv/a) and the public dose well below the regulatory limits.

1. Introduction

The International Commission on Radiological Protection (ICRP) identified that the three fundamental objectives of radiation protection are: justification of radiation exposure, minimization of potential doses, and ensuring individual dose limits are ALARA [1]. The International Atomic Energy Agency (IAEA) [2] and the ICRP have created a recommended radiation protection framework such that the design of a nuclear power plant meets these objectives. This framework and the radiation protection recommendations of the ICRP set out in ICRP-60 [3] have been adopted by the CNSC within the Nuclear Safety and Control Act [4]

The ACR-1000 NPP is designed to meet these three fundamental objectives and comply with CNSC regulatory requirements [5] for radiation protection for plant operating staff and the public, and Sections 8.13 and 6.4 of CNSC's RD-337 [6]. The latest technology and industry best practices have been evaluated and considered in the ACR-1000 design with respect to the limitation of potential doses and the minimization of radionuclide production at source.

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[1] ACR-1000® (Advanced CANDU Reactor®) is a registered trademark of Atomic Energy of Canada Limited (AECL).

[2] CANDU is a registered trademark of the Atomic Energy of Canada Limited.

[3] Whenever the phrase "As Low As Reasonably Achievable" or "ALARA" is used, "social and economic factors taken into account" is implied.

This paper presents the ALARA process for the ACR-1000 design with respect to occupational dose and public dose and provides a summary of the design features and radiation protection practices implemented to ensure that the total worker exposure and public doses are below the design targets of 0.6 person-Sv/a and 10 μSv/a, respectively, over the operating life of a single unit NPP. The ALARA process target 0.6 person-Sv/a as the total worker exposure is based on the standards set by the Institute of Nuclear Power Operation (INPO) [7] and recommended by the CNSC.

2. CNSC Regulations

Under the umbrella of the Nuclear Safety and Control Act [4], the CNSC establishes rules and regulations to be followed for radiation protection in Canada. The ACR-1000 design complies with Canadian regulatory requirements and guidelines for keeping radiation exposures ALARA, taking into account social and economic as defined in Regulatory Guide G-129 [8] and Regulatory Document RD-337 [6].

2.1 Guideline G-129

The purpose of the ALARA assessment is to show that the designers complied with the ALARA principle during the design process and to describe what steps were taken to ensure that doses to plant operating staff and to the public are ALARA.

The following means of achieving ALARA exposures are listed in CNSC G-129 [8]:

i) Management control over work practices;

ii) Personnel qualification and training;

iii) Control of occupational and public exposure to radiation; and

iv) Planning for unusual situations.

The improvements in the design and practices described in this paper deal primarily with items iii) and iv). Items i) and ii) is normally handled by the licensee with guidance from the design authority.

The ACR-1000 NPP design conforms to the ALARA principle and the CNSC regulations by ensuring that, for a given system which has been identified as contributing to radioactivity production or release mechanisms, the design option was selected to minimize the dose to public while taking into account a wide range of factors including technological maturity, availability and reliability, operational safety, radiation protection, and social and economic factors.

2.2 Regulatory Document RD-337

The CNSC has issued RD-337 "Design of New Nuclear Power Plants" [6], in which the need to keep the doses ALARA is outlined in Section 4.1.1 of the document:

The radiation protection objective is to provide that during normal operation, or during anticipated operational occurrences, radiation exposures within the NPP or due to any planned release of radioactive material from the NPP are kept below prescribed limits and as low as reasonably achievable (ALARA).

3. Application of the ALARA Principle to the ACR-1000 Design

3.1 Overall ALARA Design Process

In the design phase of the ACR-1000, AECL followed a series of established procedures to ensure that the design proceeded through an orderly process of review and quality checks. Plant Performance Specification documents, Safety Design Guides, Design Guides, and Design Requirements were prepared for the use of designers during each design phase.

The long established and strong culture of continuous improvement reflected in the CANDU family is achieved through the use of Operating Experience (OPEX) and the Feedback Monitoring System (FMS), which captures operational feedback and ensures that issues are addressed not just in the plant that the issue directly relates to, but also to all future designs such as the ACR-1000. Feedback within the design process, formalized in the Project Feedback Disposition Tracking System, which is linked to the FMS, ensures that all issues and ALARA opportunities are addressed in an objective and consistent manner.

Application of the ALARA principle is imbedded in the design process by application of the ALARA assessment methodology described in Section 3.2. To meet the ALARA requirements, each stage of CANDU design evolution is a further iteration in the selection of the "most reasonably practicable" design options for the ACR-1000 NPP. For each system and component, design alternatives and changes are evaluated and incorporated based on previous operating experience and evolving safety requirements.

To reduce radiation exposure to ALARA, designers reviewed the operating experience from previous CANDU plants, for the corresponding system or major pieces of equipment. Operating experiences 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 are well understood by designers with support from in-house physicists familiar with the operation of CANDU reactors and OPEX.

Generally, radiation exposure of plant operating staff can be minimized by limiting the number of plant operating staff that must enter containment, the frequency with which they must enter, and the time spent there. This requires designers to give 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:

1) Minimize the amount of equipment that require maintenance in areas subject to high radiation hazard;
2) Reduce number of equipment (e.g., by using larger capacity pumps or heat exchangers);
3) Simplify equipment for a system to reduce maintenance durations;
4) Relocate equipment to a lower radiation field area (e.g., long stem valves, motorized valves);
5) Improve chemical control and purification to control corrosion;
6) Provide more reliable equipment to ensure a longer interval between maintenance;
7) Minimize the use of materials which may become activated under neutron irradiation in the reactor, including products of corrosion or wear (e.g., cobalt, antimony);
8) Arrange for quick removal of equipment for shop maintenance;
9) 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);
10) Control leakage from process and auxiliary systems that contains activity (including heavy water and light water) (e.g., leakage collection, fume hoods);
11) Design ventilation to minimize airborne radioactive contamination hazards;
12) Provide good monitoring coverage for radiation hazards (e.g., airborne radioactive contamination monitoring in all areas);
13) Provide facilities for removal of certain hazardous radionuclides (e.g., airborne radioactive contamination); and
14) Provide shielding.

Radiation exposure of the public is minimized by controlling radionuclide release from the plant in gaseous or liquid effluents, particularly the longer lived radionuclides. As a primary means of radiation protection of members of the public, radioactive wastes from the plant can be reduced by controlling the production of wastes at the source. Thus many of the options noted above aimed at reducing exposure of plant personnel can further be extended to protection of the public.

Once source terms have been minimized, particular effort for protection of the public is on:

- Minimizing and controlling gaseous emissions from the plant through improved ventilation system and filtration design; and
- Minimizing and controlling liquid effluent streams (i.e., by increasing the effectiveness of the liquid waste management system).

In the ALARA process, the designer must decide whether the benefit of efforts to reduce doses is commensurate with the incurred costs. Some radiation protection problems may be resolved using a cost-benefit analysis or other quantitative technique. Many other problems do not lend themselves to a quantitative analysis as these techniques have their 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 doses are received, and broader social factors.

The judgement of reasonableness is inherent in the ALARA process. The following are suggested for the designer to help judge if an action is reasonable:

- 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.
- Good Practice: This involves comparing the current design with other similar facilities to determine the most effective solution to the problem.
- Feasibility: This involves assessing if it is practical to make the change.

In order to substantiate the judgement of how reasonable a proposal is, the designer must document the basis of the judgement using an ALARA assessment, which includes:

1) Design options which have been considered for the reduction of dose;
2) Where possible, a review of radiation protection experience relating to the proposed options from operating plants and an explanation of where this information was obtained (e.g., dose records, inspection and instrument maintenance records, effluent releases, incident reports, etc.); and
3) Rationale for either adopting or rejecting each proposed design option for the reduction of dose.

3.2. Application of ALARA Principle

To minimize the commitment of resources which are likely to have limited return in improvement of safety, further ALARA review is not normally required if the ACR-1000 design dose targets are met:

1) Dose to individual members of the public is unlikely to exceed 10 μSv/a.
2) The collective dose (both occupational and public) is unlikely to exceed 0.6 person-Sv/a.

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, if an expenditure in excess of a certain value is required to reduce the dose to plant personnel or members of the public, the action is not economically justifiable.

There may be situations (e.g., the collective dose may be less than 0.6 person-Sv/a, a limited number of persons may still be receiving a significant fraction of the individual dose limit) in which it would be appropriate to carry out an ALARA review even if doses are less than the design dose targets given above.

The ACR Safety Design Guide for Radiation Protection outlines the procedure that is used to determine if an ALARA assessment is required for a system.

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

1) Identify systems and practices that result in radiation exposure to workers and members of public.

2) Estimate the total doses resulting from identified systems and practices.
3) If the doses are above the effective dose limits given in Table 1 and equivalent dose limits [5], radiation protection must be improved.
4) If the doses are below the ACR-1000 design dose targets given above, no further ALARA assessment is required. Doses should be reduced below the design target if this can be done at a cost that is justifiable [8, 9]. Designers are encouraged to reduce doses below the design targets where this can be done without significant expenditure.
5) If the doses are above the ACR-1000 design dose targets given above, an ALARA assessment is required, as described below.
6) The process and its results must be documented in the overall assessment document for specific systems.

Other factors, including organizational values, regulatory requirements, non-radiological detriments, and public expectations were also taken into consideration. The design process ensures that all steps are taken to reduce dose where no significant expenditure is required.

To apply the ALARA principle to ACR-1000 NPP systems, the following procedure is used in determining which design option is the ALARA option. The procedure is as follows:

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

This process is illustrated in Figure 1. An example of the ALARA implementation process for minimizing worker and public doses is shown in Figure 2.

4. ACR-1000 Design Features to Minimize Occupational Radiation Exposure and Public Dose

The ACR-1000 design incorporates design features and improvements (see Tables 2 and 3) based on information from operating CANDU plants to reduce occupational radiation exposures and public dose to ALARA.

4.1 Occupational Radiation Exposure

The radiation hazard can be either an external or internal radiation hazard. The external hazard arises from a radiation source outside the body, emitting penetrating radiation, which deposits in body tissues. The radiation may be in the form of neutron radiation, gamma rays, or beta particles (electrons). The internal hazard arises from the ingestion, inhalation, or absorption of radioactive matter into the body. In an ACR-1000 plant, the internal hazard is from tritium and alpha contamination. Tritium is the radioactive isotope of hydrogen, which has a radiological half-life of 12.3 years and emits a low energy beta particle. Tritium is mainly produced by neutron activation of deuterium in heavy water of the moderator system. Tritium is the principal source of internal radiation exposure to worker.

As shown in Table 2, suitable provisions are provided in the design and layout of the ACR-1000 plant to minimize exposure and contamination hazards from all sources. These features include design features to minimize exposures, layout and contamination control, radiological zoning and access control, shielding, radiation monitoring, ventilation and filtering systems. In Table 2, ACR-1000 design features and improvements to minimize external and internal radiation hazards are provided for heat transport and auxiliary systems, moderator and auxiliary systems, fuel handling system and vapour recovery system.

For the ACR-1000, the estimated annual occupational collective dose is below the design target of 0.6 person-Sv/a over the 60 years operating life of a single unit ACR-1000 plant. The estimated breakdown of doses for reactor operation and outage is approximately 20% from reactor operations and 80% from outage. The external and internal doses (tritium) contribute 97% and 3% of the collective dose estimate (0.6 person-Sv/a), respectively. The low contribution of internal dose (tritium) is attributable to traces level of tritium in the heat transport system.

4.2 Public Dose

The upper-bound annual radiation dose received by individual members of the public from radioactive gaseous and liquid effluent emissions are well below the design target (10 μSv/a) at the boundary of the exclusion zone of 500 m for two-unit ACR-1000 NPP.

ACR-1000 design features and improvements to reduce airborne tritium, carbon-14 and noble gas emissions and public doses are summarized in Table 3. Additional design features minimizing radioactive airborne and waterborne emissions are provided in [9].

5. Conclusions

The evolutionary ACR-1000 design incorporates significant radiation exposure control improvements based on ALARA assessment using the latest technology and industry best practices with respect to the limitation of potential worker and public doses and the minimization of radionuclide production at source for all phases of the ACR-1000 NPP life cycle.

The ACR-1000 design meets the international radiation protection practices and CNSC regulatory requirements for radiation protection for plant operating staff and the public, including CNSC guidelines for keeping radiation exposures ALARA, taking into account social and economic factors. The estimated annual occupational dose and public dose are ALARA and are below design targets 0.6 person-Sv/a and 10 μSv/a, respectively.

6. References

[1] Annals of the ICRP, Publication 55, "Optimization and Decision-Making in Radiological Protection", 1989.

[2] IAEA Safety Series No. 101, "Operational Radiation Protection: A Guide to Optimization", 1990.

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

[4] CNSC Act, "The Nuclear Safety and Control Act", May 2000.

[5] Canada Gazette Part II, Vol. 134, No. 13, "Nuclear Safety and Control Act - Radiation Protection Regulations", May 2000.

[6] CNSC Regulatory Document RD-337, "Design of New Nuclear Power Plants", November, 2008.

[7] INPO Convention on Nuclear Safety Report, "The role of the Institute of Nuclear Power Operations in supporting the United States Commercial Nuclear Electric Utility Industry's , Focus on Nuclear Safety", NUREG-1650, Revision 2, September 2007.

[8] CNSC Regulatory Guide G-129, "Keeping Radiation Exposures and Doses "As Low as Reasonably Achievable (ALARA)", Revision 1, October 2004.

[9] Sachar, M, Julien, S. and K. Hau, "ACR-1000 Environmental Performance Design Improvements", 30th Annual Conference of Canadian Nuclear Society, CNS Bulletin, Vol. 30, No. 3, September 2009.

Table 1: Effective Dose Limits for Normal Operation [5]

Item	Person	Period	Effective Dose (mSv)
1.	Nuclear Energy Worker, including a pregnant Nuclear Energy Worker	a) One year dosimetry period	50
		b) Five-year dosimetry period	100*
2.	Pregnant Nuclear Energy Worker	Balance of pregnancy	4**
3.	A person who is not a Nuclear Energy Worker	One calendar year	1

Note: * The CNSC radiation exposure limits for plant staff designated as nuclear energy workers (NEW) are 50 mSv in a one-year dosimetry period and 100 mSv in a five-year dosimetry period.

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**The CNSC radiation dose limit is 4 mSv. The target dose limit, 1 mSv, is used and satisfied by administrative measures rather than by plant design.

Table 2: ACR-1000 Design Features and Improvements[4] to Minimize Worker Dose

Hazard	Systems	ACR-1000 Design Features and Improvements
External Radiation Exposure (neutron radiation, gamma rays, or beta particles (electrons))	Common Equipment and Component and Layout	• Large equipment from various process systems, e.g., delay tanks, heat exchangers, steam generators, and the pressurizer, is shielded by ordinary concrete shielding from the accessible areas. • Wherever practicable, radioactive pipes are run through inaccessible areas during operation and they are shielded behind a wall or inside trenches, so that the radiation dose rate received from these pipes remains below acceptable levels in accessible areas. • Shielding of pumps is accomplished by separating the pump motor from the pump bowl with an internal wall. The Heat Transport System (HTS) pump bowl also provides some shielding for the pump motor. Some pumps are not shielded (e.g., moderator system pumps) because they are located in inaccessible areas during reactor operation. In addition, the shielding walls of these rooms provide adequate shielding to reduce dose rates in accessible areas in their close proximity. • Valves are located in valve galleries or behind shielded walls with holes for valve manipulation and shielding against nearby equipment. • New fuel loading is located in the Reactor Auxiliary Building to avoid radiation sources from reactor operations. • Plant has two major radiological zones, i.e., Radiological Control Area (RCA) and non-RCA. This reduces the time taken to enter and exit the facility, particularly during reactor shutdown, and makes it easier to control contamination. Fewer monitoring stations at zone boundaries are needed, leading to reduced manpower needs at the fewer monitoring stations and reduced collective doses. • Improved plant layout and improved access controls result in avoidance of high-radiation areas and provide increased radiation protection for operations and maintenance staff. • Where possible, equipment requiring more frequent access and maintenance is located in low-dose and non-RCA areas. • A maintenance-based design provides space allocation and reduction in temporary scaffolds and hoists, and includes provisions for built-in electrical, water, and air supplies for on-power and normal shutdown maintenance.
External Radiation Exposure	Heat Transport System	• Stainless steel for the HTS feeders reduces flow-assisted corrosion and therefore reduces the quantity of mobile material available for activation and reduces the requirement for feeder-thinning inspections. • Provision to supply nitrogen gas to the HTS to provide inert cover gas during drained state to reduce corrosion due to oxidation.

[4] Design improvements are based on information from operating CANDU plants to reduce occupational radiation exposures to ALARA.

Table 2: ACR-1000 Design Features and Improvements[5] to Minimize Worker Dose (cont'd)

Hazard	Systems	ACR-1000 Design Features and Improvements
External Radiation Exposure	HT Purification System	• Use of sub-micron filters to remove particulates from the HTS coolant. • Use of a high flow purification system to provide a purification half-life of one hour or less to reduce activity transport and disposition outside the reactor core. • Purification in-service during most shutdown configurations to maintain chemistry control, and to minimize start-up chemistry transients (e.g., crud bursts).
External Radiation Exposure	Fuel Handling and Storage System	• Use of a snout blowdown system to collect water from the fuelling machine snout after refuelling, thus reducing the quantity of HTS water released into the reactor vault. This reduces airborne contamination in these areas and worker dose from resulting cleanup. • Spent fuel transfer is under water. This ensures less off-gassing of radioiodines and noble gases during handling of depleted fuel.
Internal Radiation Hazard (tritium)	Common Equipment and Component	• The use of low-enriched uranium (LEU) and a smaller lattice pitch reduces thermal neutron flux, and the presence of smaller amount of D_2O inside the calandria, lowers tritium production and therefore reduces the internal dose hazard from tritium escaping from the moderator system. • HTS coolant is light water in the ACR-1000 plant, the tritium-in-air hazard from the HTS leakage is insignificant. Lithium, which is added to the coolant for chemistry control, is depleted in Li-6 (0.1%) to reduce tritium production from Li-6 (n,α) ^{3}H reaction in the HTS. Thus, the internal radiation hazard is associated with leakage from the heavy water - filled moderator system components that are located in confinement areas.
Internal Radiation Hazard (tritium)	Vapour Recovery Systems	The RB vapour recovery system minimizes tritium-in-air concentration in the moderator system areas to reduce worker dose: • Increased moderator dryer capacity (three rotary desiccant wheel dryer units) serving the Reactor Building (RB) and improved atmospheric control in the RB with higher purge flow from dried moderator areas. • Improved layout of moderator enclosure and moderator auxiliary systems by moving all moderator and moderator auxiliary equipment into dedicated dried rooms in the RB, making the management of their atmospheres more efficient and preventing diffusion of heavy water vapour to other regions of the RB. • Addition of a single rotary desiccant wheel dryer unit to the Maintenance Building D_2O Management Area minimizes tritium-in-air concentration in this area to reduce worker dose.

[5] Design improvements are based on information from operating CANDU plants to reduce occupational radiation exposures to ALARA.

Table 3: ACR-1000 Design Features and Improvements[6] to Minimize Airborne Emissions and Public Dose

Radionuclides	ACR-1000 Design Features and Improvements
Tritium	• Design features minimizing internal radiation exposure also reduce airborne tritium emissions (See Table 2, Internal Radiation Exposure)
Carbon-14	• Lower carbon-14 production rate in the moderator and HTS due to the use of LEU and a reduced lattice pitch. • Installation of sub-micron filters downstream of the moderator ion exchange (IX) columns to capture resin fines. This will prevent resin fines from reaching the reactor core where the fines form carbonate and bicarbonate ions from carbon-12. Consequently, saturation of the IX columns with carbon-12, which competes for IX sites with carbon-14, will be avoided and the risk of increased emissions of carbon-14 during chemistry excursions will be reduced. • Circulating Moderator Cover Gas through vertical reactivity mechanism thimbles to prevent stagnation and build up of gases including carbon-14 and thereby reducing the potential consequences of a mechanism leak.
Noble gases (Argon-41 and radioisotopes of xenon and krypton)	• Improved design of the Annulus Gas System (AGS) compressor, provisions to purge air from the AGS piping after component replacement/maintenance to minimize argon-40 in air ingress and reduce argon-41 production. Addition of the AGS purge delay tanks subsystem of the OGMS to remove argon-41 in the AGS purge stream before release to the environment. • An OGMS is included in the reference design that collects xenon and krypton gases from several specified off-gas streams per unit and delays them for decay. • Less off-gassing of xenon and krypton gases from defective fuel during fuel handling since the entire fuel handling process will be in water with a nitrogen cover gas.

[6] Design improvements are based on information from operating CANDU plants to reduce airborne emissions and public dose to ALARA.

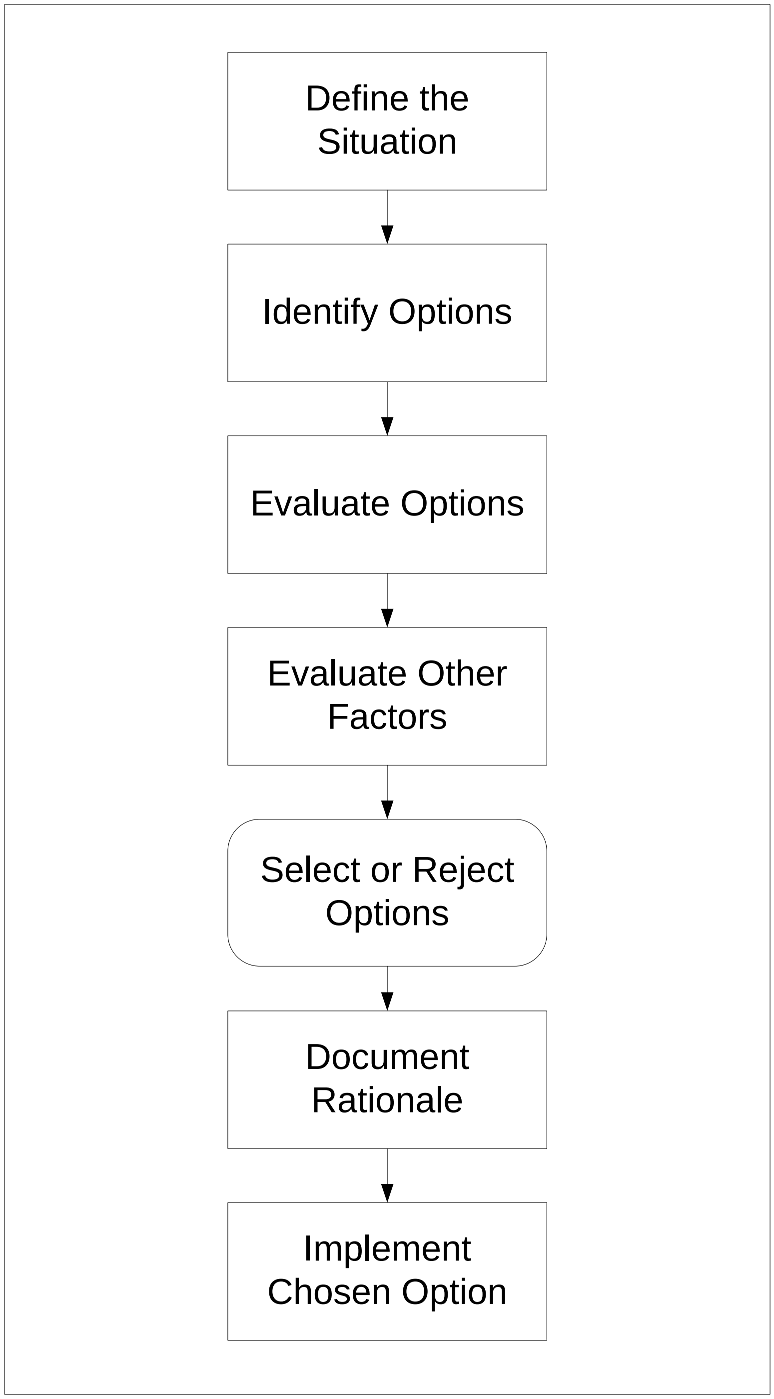

Figure 1: Simplified ALARA Assessment Process

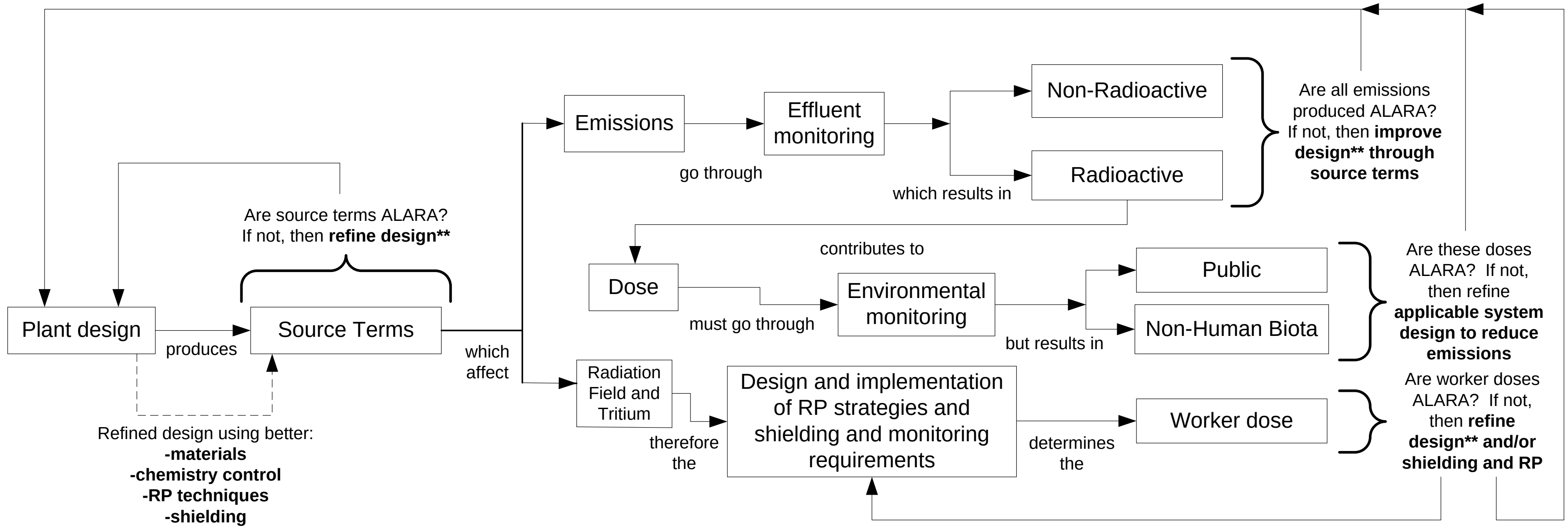

Figure 2: ALARA Implementation Process for Worker and Public Doses