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Radiation Protection and Safety Aspects

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Editorial

The harmful effects of radiation were recognized soon after the discovery of X-rays and radioactivity. The mass destruction caused by the atom bombs in Hiroshima and Nagasaki in Japan in 1945 made the common man aware and fearful of the deleterious effects of acute exposure to radiation. The high level of public awareness of radiation exposure had the positive effect, that is the nuclear industry had to strive hard to evolve very high safety standards to get acceptability.

At present, there are a large number of activities involving radiation and radioisotopes. These include mining and milling of uranium ores, fabrication of nuclear fuels, operation of nuclear reactors, reprocessing of spent fuel and safe disposal of nuclear wastes. Radiation and radioisotopes are also used in medicine, industry, agriculture and several other fields. In all the above operations exposure to radiation is inevitable. Nuclear industry has been giving the highest priority for safety of occupational workers and general public, under normal operation as well as in accidental situations. This fact is seen from the large quantum of published work in this field; and safety related literature brought out by various agencies such as the International Atomic Energy Agency (IAEA), International Commission on Radiological Protection (ICRP) and national regulatory bodies. It is mandatory to have well trained radiation safety officers in all installations where radioactive material or radiation is used.

The present issue of the IANCAS Bulletin is devoted to "Radiation Protection and Safety Aspects". This issue is guest edited by Dr. Pushparaja, Radiation Safety Systems Division, BARC. Dr. Pushparaja has done an excellent job in identifying the authors, relevant topics and editing the articles. I am thankful to Dr. Pushparaja and the authors who contributed to this issue. I am particularly glad in bringing out this issue as it reminds me of my brief stint in radiation protection in the early part of my career.

I look forward to readers views on IANCAS Bulletins and the contents therein.

M.R.A. Pillai

CONTENTS

Units and Quantities Used in Radiation Protection

Dr. A.N. Nandakumar holds a Ph.D. degree in Physics from the University of Mumbai. He has been working in the Radiological Physics and Advisory Division, Bhabha Atomic Research Centre, Mumbai since 1968. His fields of interest are radiation shielding, transport of radioactive materials and nuclear emergency preparedness. He has computed the shielding requirements of radiation installations and containers housing gamma sources of activities ranging from a few hundred GBq to several PBq (i.e. a few Curies to megaCurie). He has computed the dose received by workers and public during the various phases of transport under incident-free and also accident conditions of transport of a variety of radioactive materials in the Indian scenario. He has worked on the derivation of an empirical relation for calculating the dose received by persons handling packages containing radioactive material. He has many papers to his credit which have been published in National and International Journals. He has represented India in international conferences and IAEA Committees.

Units

In any branch of physical sciences, the 'units' are vital for quantifying any concept. In radiation physics, one studies the different types of radiation and their transport through matter. One is interested in calculating the intensity of radiation and the quantity of radiation energy that is absorbed in the medium in which radiation traverses and in determining the dose deposited in the medium by different types of radiation. Radiation protection is a derivative of radiation physics. We shall briefly discuss some of the most commonly encountered units in this discipline of physics.

Activity

In radiation physics, we deal with different radiation sources. In the case of radioactive materials, the quantity representing the emission rate of radiation is called the activity. The activity of a radionuclide (sometimes also referred to as the source strength) represents its disintegration rate. The unit of activity is Becquerel (Bq).

1 Bq = 1 disintegration per second

The dimension of Bq is number per second or s^{-1}.

The units, kiloBecquerel (kBq), megaBecquerel (MBq), gigaBecquerel (GBq) and petaBecquerel (PBq) are derived quantities.

In using this quantity, one should have a prior knowledge of the decay scheme of the nuclide whose activity is being calculated / measured. Consider the example of ^{137}Cs and ^{60}Co. From their decay schemes (Fig. 1 and Fig.2) it is clear that the emission rate of photon of ^{137}Cs would be numerically equal to the disintegration rate of the nuclide because each disintegration is accompanied by the emission of only one photon i.e. a 0.662 MeV gamma ray. However, in the case of ^{60}Co, the numerical value of the photon emission rate would be twice that of the disintegration rate.

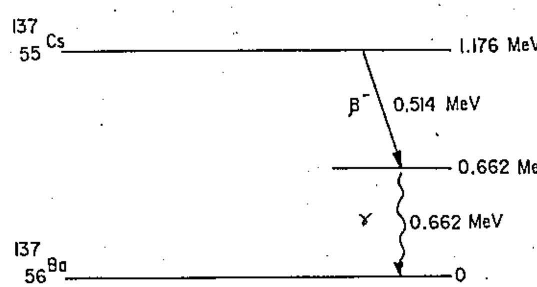

Fig. 1 Simplified decay scheme of ^{137}Cs

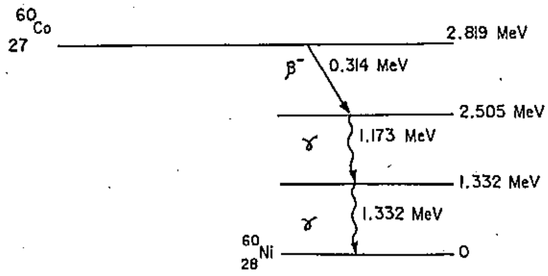

Fig. 2 Simplified decay scheme of ^{60}Co

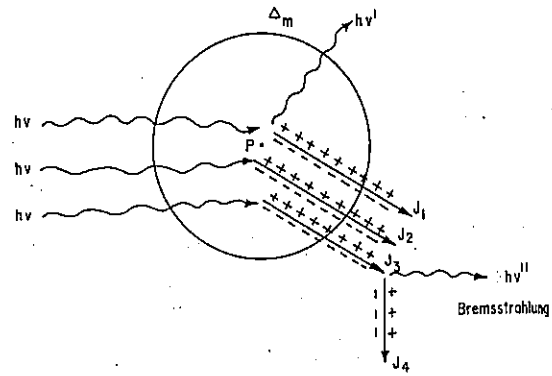

Fig. 3 Exposure at point P

Fluence

Nuclear disintegrations are accompanied by emission of radiation. Such radiation may be α particles, β particles or γ photons. The rate of emission of radiation is a function of the activity of the given radionuclide. The fluence at a point, P, in space is the number of particles, dN, incident on a sphere of cross-sectional area da (of a great circle of the sphere), surrounding the point, P. That is,

Fluence = dN/da

The dimension of fluence is number per square metre or m^{-2}.

This quantity is useful in determining the intensity of radiation at different points in a medium through which radiation passes.

Exposure

The most important property of radiations we are discussing is their ability to ionize medium they traverse. Hence, they are described as ionizing radiations. The presence of radiation at a given point in space can be detected by recognizing the ions produced by the radiation at that point. The X or γ radiation field at a point is described as exposure. Exposure is a measure of the ionization produced by electrons *originating* from a small mass of air, Δm, even if the ionization is not produced in the mass (Fig.3).

$$\text{Exposure} = \langle e \, \Sigma J_n \rangle / \Delta m$$

The average value, $\langle e \, \Sigma J_n \rangle$, denotes the statistical nature of interaction of radiation with matter. The unit of exposure is Roentgen.

Roentgen is defined as the exposure associated with the corpuscular emission of one esu of electric charge of either sign produced by X or γ radiation in air of mass, 0.001293 g.

In simpler terms, imagine a volume of one ml containing air at a temperature of $0^{o}C$ and pressure of 760 mm Hg. Suppose X or γ radiation is incident in the volume causing ionization. Ion pairs would be produced in the volume. The magnitude of the total quantity of positive ions produced would be equal to that of negative ions. If this quantity is 1 esu, then the amount of exposure is one Roentgen.

The dimension of Roentgen is Coulomb per kg of air or C kg^{-1}.

The most commonly used radiation survey meters read the exposure rate at the desired location in Roentgen per hour (R/h) or mR/h. The ionization chamber type survey meters measure the rate of ionization produced in the chamber volume. Accurate measurement demands that the electrons leaving the sensitive volume of the chamber are exactly compensated for by those entering the sensitive volume from outside. This situation is described as electronic equilibrium.

The limitation of Roentgen is that it applies only to X and γ radiations and that too of energy not exceeding 3 MeV. The energy limitation is caused

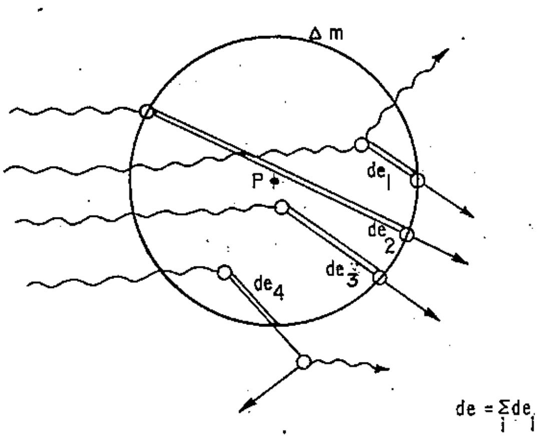

Fig. 4 Absorbed dose at point P

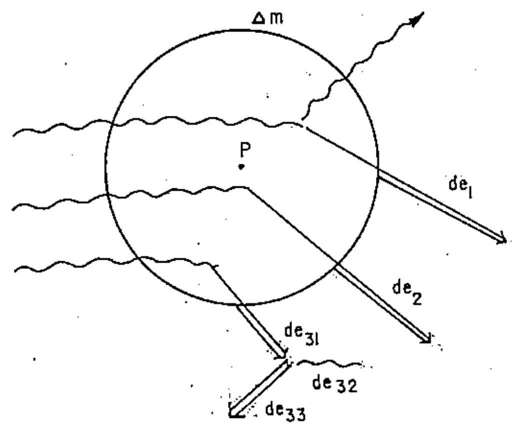

Fig. 5 KERMA at point P

by practical difficulties in designing ionization chambers capable of ensuring electronic equilibrium at higher energies.

Absorbed Dose

One R of X or γ radiation deposits 8.77×10^{-3} Joules per kg of air while the same exposure deposits 9.88×10^{-3} Joules per kg of tissue. In order to gauge or predict the effect of ionizing radiations absorbed in matter, it is necessary to determine the energy absorbed per unit mass of the medium. This quantity is called the absorbed dose. It applies to all types of radiations, viz., α, β, γ, X, neutrons, etc.

Absorbed dose, D, is given by

$$D = \langle de \rangle / \Delta m$$

Where <de> is the mean energy imparted by the ionizing radiation to matter of mass Δm. (Fig. 4). The unit of absorbed dose is Gray (Gy). The dimensions of Gy are Joules per kg or J/kg. Clearly, only the energy deposited in m is taken here. The advantage with this unit is that it applies to all types of radiation unlike Roentgen.

Kerma

The unit of exposure, R, may be replaced by Kerma which is an acronym for Kinetic Energy Released in Material. Kerma is given by

$$K = \Sigma de_i / \Delta m$$

Where Σde_i is the sum of the *initial* kinetic energies of all the *charged* ionizing particles released in a material of mass Δm (Fig. 5).

The unit of kerma is Gray (Gy), the dimensions being, Joules per kg, or J/kg.

The essential difference between absorbed dose and Kerma is that, the absorbed dose is the energy *deposited in* the mass and Kerma is the energy *released from* the mass.

Equivalent Dose

The effect of different types of radiation in a given medium depends upon the type of radiation, and in the case of neutrons, the energy of the radiation. That is the biological damage in tissue caused by one Gy of α radiation is not the same as that caused by one Gy of X, γ or neutrons. Thus an equivalence factor, usually referred to as the radiation weighting factor is associated with each type of radiation so that one might be in a position to state that $W_\alpha D_\alpha$ Gy of α dose is equivalent to $W_\gamma D_\gamma$ Gy of γ dose.

This brings in the concept of equivalent dose. The unit of equivalent dose is Sievert (Sv). The dimension of Sv is Joules per kg or J/kg.

The W_R values for different types of radiation are given in Table 1.

These values can be used thus:

Table 1. Values of radiation weighting factors, W_R

Type of radiation	W_R
γ	1
β	1
e^-	1
X	1
α	20
Heavy ions	20
Neutrons	5 to 20*

*The value depends upon the energy of neutrons.

Table 2. Tissue Weighting Factors

Tissue or organ	Tissue weighting factor, W_T
Gonads	0.20
Bone marrow (red)	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Oesophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surface	0.01
Remainder*	0.05

* These tissues/organs include adrenals, brain, small intestine, upper large intestine, kidney, muscle, pancreas, spleen, thymus and uterus. In those exceptional cases in which one of the remainder organs receive an equivalent dose in excess of the highest dose in any of the twelve organs with the W_T values specified above, a W_T of 0.025 should be applied to that tissue or organ and 0.025 to the rest of the remainder organs.
(The organs included in the table are those which are likely to be selectively irradiated. Some of them are known to be susceptible to cancer induction. The W_T values have been developed from a reference population of equal numbers of both sexes and a wide range of ages. In the definition of effective dose the W_T values apply to the whole population and to either sex.)

If the absorbed dose resulting from α radiation is one Gy, the equivalent dose is determined as 20 Sv.

If the absorbed dose resulting from γ radiation is 20 Gy, the equivalent dose is determined as 20 Sv.

Thus in terms of biological damage one Gy of α dose is equivalent to 20 Gy of γ dose.

The advantage with this unit is that one Sv of α radiation dose is equivalent to one Sv of β radiation dose which is equivalent to one Sv of γ radiation dose, etc. That is once the equivalent dose is expressed in units of Sv, there is no need to specify the type of radiation.

This unit is generally used for the purpose of radiation protection, e.g. for stipulating dose limits and computation of accumulated dose. It should be used only when the dose received is at a low dose-rate, as for example, the chronic exposure received by persons occupationally exposed to radiation. Hence, if the exposure is acute then Gy would be the appropriate unit to be employed. This is because the values of W_R are not appropriate for acute exposure.

Effective Dose

Radiation dose may occur either as uniform exposure (usually described as whole body dose) or non-uniform exposure. When non-uniform exposure occurs, only certain individual tissues are likely to receive the dose. In order to translate the damage suffered by the affected tissue into the detriment which could be suffered by the individual, the concept of effective dose has been introduced. This dose is computed by multiplying the tissue dose by a weighting factor appropriate to the tissue, W_T. These factors have been derived on the basis of considerations such as relative importance of the tissue to the overall well-being of an individual and the treatability of the tissue. The values of the tissue weighting factors are given in Table 2. It must be noted that the sum of all W_T is unity, as should be.

If a given tissue, T, receives a dose of H_T Sv, then the effective dose is $W_T H_T$ Sv. If various tissues receive different doses then the resulting effective dose is given by:

$$E = \Sigma W_T H_T \text{ Sv}$$

Committed Effective Dose

Following intake of a radionuclide, the dose-rate to the tissue and also the effective dose-rate will decay according to the effective half-life of the radionuclide. The effective half-life is the sum of the reciprocals of the radioactive half-life of the radionuclide and the biological half-life of the element. Therefore, the total dose received due to the intake has to be computed thus:

$$E_c = \int E(t) \exp(-\lambda_{eff}\, t)\, dt$$

This has to be a definite integral in order to signify the total dose received during the specified period. If the integral is computed from 0 to 50 years, the resultant quantity is the committed effective dose. This value corresponds to the cumulative dose received by the person who incurred the intake, over a period of 50 years following the intake.

Dose Commitment

Dose commitment is a quantity which determines the total detriment resulting from an event or a practice. If an event or a practice results in dose received by a population, then let N(E) dE be the number of persons who received an effective dose between E and E+dE. The dose commitment is given by :

$$S = \int N(E)\, E\, dE$$

This quantity is useful in optimizing the dose receivable in certain practices and also for deciding whether a practice can be justified in terms of the net benefit arising there from. For computing the net benefit one compares the detriment resulting from the dose commitment with benefit accruing while incurring the dose.

Conclusion

The various units described above have specific applications. The interested reader would like to see the commonly used text books which are listed in suggested references.

Suggested References

1. G.F. Knoll, Radiation Detection and Measurement, John Wiley & Sons, New York, 1989.
2. W.J. Price, Nuclear Radiation Detection, McGraw-Hill, New York, 1964.
3. J.E. Turner, Atoms, Radiation, and Radiation Protection, Pergamon Press, New York, 1986.

Radiation Protection Standards

Dr. A.M. Bhagwat, after his post graduation in chemistry, joined Radiation Hazards Control (RHC) Section of Bhabha Atomic Research Centre (BARC), Mumbai in 1961 as Scientific Officer. He obtained his Ph.D. degree in physics in 1985 from the Gujarat University. He retired as Head, Radiation Safety Systems Division. He was responsible for safety of all the radiation facilities of BARC at Trombay, Calcutta and Baroda which include research reactors, fuel reprocessing plants, accelerator facilities and waste management plants. He has served in several safety committees. His fields of specialisation include operational radiation protection, application of solid state nuclear track detectors (SSNTD) and decontamination & radiation resistance properties of Indian paints and flooring tiles. He has over 100 publications to his credit including several publications in national and international journals of repute. He is elected Fellow of Maharashtra Academy of Sciences in 1996.

Introduction

Ionizing radiations and radioactive materials have always been features of our environment. But owing to lack of their impact on our senses, we became aware of them only towards the end of the last century. Since then, many important uses of radiations have been found and many technological processes have been developed which create them, either deliberately or as unwanted by-products.

The need to protect people from harmful effects due to over exposure to radiation, both early effects such as skin burns and late effects such as cancer and hereditary effects, was recognised quite soon after the discovery of radiations. Since there are thresholds for early deterministic effects, it is possible to avoid them by restricting the doses to individuals. On the other hand late (stochastic) effects cannot be completely avoided because no threshold can be invoked for them.

The primary aim of radiological protection is to provide an *appropriate standard of protection* to man without limiting the beneficial activities giving rise to radiation exposure. This aim cannot be achieved on the basis of scientific concepts alone. Value judgments are necessary about the relative importance of different kinds of risk and balancing of risk (costs and disadvantages) and benefits. In this way, the field of radiological protection is no different from other areas where control of hazards is a necessary concern.

Radiation protection concerns the protection of workers, members of the public and patients undergoing diagnosis and therapy, against the harmful effects of ionizing radiations. In order to cope up with expanding radiation and nuclear practices, and in view of the particular character (probabilistic) of the radiation risks, radiation protection has developed in the last few decades, into an elaborate system of concepts, principles and techniques for the prevention and control of radiological risks. It would therefore be worthwhile to know the nature of protection standards which govern this complex field.

Historical Aspects

The discovery of X-rays by W.C. Roentgen on November 8, 1895 placed, for the first time, a variable intensity ionizing radiation source in human hands. The harm these rays can cause on human beings was identified even before the announcement of its discovery on January 5, 1896 (Wiener Presse, 1896). This development marked the beginning of the field of radiation protection, the immediate aim being "to contain its hazards to radiologists while exploiting its possible medical benefits". The discovery of radioactivity and radium in immediate succession added extra dimension to the control of radiation hazards.

It may be recalled here that "British Roentgen Society" was formed within 18 months of announcement of the discovery of X-rays which amply highlighted the potential of the discovery in terms of medical benefits and health hazards. The measures to control radiation hazards were fairly simple, i.e. (i) use of protective shields (ii) restriction on number of working hours (iii) inspection of x-ray equipment & (iv) special medical examination, including blood test, for workers of x-ray and radium departments.

The next 30 years saw the growth of radiation protection at individual and national levels and efforts to quantify them resulted in coinage of terms like "Tolerance Dose" for external radiation exposure and "Maximum Permissible Body Burden (MPBB)" for internal contamination. The efforts acquired an international image in 1925 during the first International Congress of Radiology at London. In 1928, the International X-ray and Radium Protection Committee (IXRPC) was formed which changed its name to International Commission on Radiological Protection (ICRP) in 1946.

ICRP, in the subsequent years and decades, continued to make fresh and more elaborate recommendations based on new scientific information available in literature. This led to the introduction of new terms and principles like "Maximum Permissible Dose (MPD)", Acceptable Dose, Maximum Permissible Concentration (MPC-in air or water), Derived Air/Water Concentration, "As Low As Possible" etc., with special meaning attached to them. It may be noted that since 1934, the annual dose limits consistently went down but the single exposure values have remained high to provide freedom in operations on day-to-day basis.

The technological advances towards the end of the second world war created enormous possibilities for further radiation exposures, i.e. through reactor piles for research and power production, high voltage x-ray sources such as betatrons and linear accelerators, high activity radioactive sources for radiotherapy and industrial uses of radioisotope techniques. The possibility now existed for radiation exposure of whole population as a result of discharge of nuclear wastes into the environment, reactor accidents, peaceful atomic tests and use of atom bombs in war. Over the years, the ICRP have tried to gear themselves up to face these situations and to provide scientifically sound bases on which radiation protection policies could be evolved.

A detailed discussion of the evolution of protection standards over the years is outside the scope of this article. Conscious effort has therefore been made to confine the article to latest recommendations of ICRP as enunciated in 1990 and its implications.

Present Conceptual Framework

To clarify the 1990 ICRP recommendations, it is convenient to think of the processes causing human exposures as a network of events and situations. Each part of the network starts from a source. Radiation or radioactive materials pass through environmental pathways which may be simple in the workplace (e.g. inhalation of airborne activity), or very complex in the natural environment (e.g. ground accumulation, transfer to animals, then to foodstuffs) with some pathways being common to several sources. Eventually, individuals are exposed to one or many sources of radiation. It follows that assessments of the effectiveness of protection can be related to the source giving rise to the individual doses **(source-related)** or related to the individual dose received by a person from all relevant sources **(individual-related).**

The Radiation Protection Standards apply to both 'practices' and 'interventions' which are defined as follows :

Practices are activities that add radiation exposure to that which people normally receive due to background radiation, or that increase the likelihood of incurring exposure. These include the use of radiation or radioactive substances for medical, industrial, agricultural, educational, training and research purposes and, of course, the generation of energy by nuclear power. Also included are facilities containing radioactive substances or devices such as irradiation installations, mines and mills processing radioactive ores and radioactive waste management facilities.

Interventions are activities that seek to reduce the existing radiation exposure, or the likelihood of incurring exposure. These apply to both chronic exposure situations such as radon in buildings, and emergency situations such as those created by contamination in the aftermath of an accident. Dose limits are not applicable to interventions. For any justified intervention, the objective is achieved by keeping the individual doses lower than the threshold doses for deterministic effects and keeping all doses as low as reasonably achievable under the given circumstances.

Protection under the standards is based on the principles enunciated by ICRP, which can be summed up, for practices, as follows:

(i) *Justification of the practice*: No practice involving exposure to radiation should be adopted unless it produces a benefit that outweighs the harm it causes or could cause.

(ii) *Optimization of protection*: Radiation doses and risks should be kept as low as reasonably achievable while constraints should be applied to dose or risk to prevent an unfair distribution of exposure or risk.

(iii) *Limitation of individual risk*: Exposure of individuals should not exceed specified dose limits above which the dose or risk would be deemed unacceptable.

All the above three principles apply to the protection of workers and the public. These principles mean that when they are implemented for practices, it is necessary to consider not only normal operation but also the potential for exposures from accidents. Once the practice is justified and radiological protection considerations are only one aspect of decision making over the introduction of new practice, the doses and risks have to be optimised within the dose or risk limits specified for individuals. However, optimisation is a source-related process and limits apply to the individual to ensure protection from all sources.

ICRP has therefore introduced the concept of a constraint to dose or risk. A **constraint** is an individual-related criterion, but applied to a single source in order to ensure that doses or risk limits are not exceeded. A dose constraint would therefore be set at a fraction of the dose limit as a boundary on the optimisation of that source. ICRP considers that a constraint should be set on the basis of general knowledge about the performance of the source or by a generic optimisation. For potential exposures, risk constraints should be established in the same way. A constraint is therefore seen as a regulatory requirement, rather than as a design target or an operational investigation level.

Dose Limits

General

As could be seen, the dose limits or the limitation of individual risk is placed at the end of the three principles governing radiological protection for practices. Thus if the procedures of justification of practices and optimisation of protection can be carried out effectively, there will only be a few cases where limits on individual doses will have to be applied. However, justification & optimisation are subjective procedures. Dose limits are, therefore, necessary and provide clearly defined boundaries and prevent excessive individual detriment which might result from a combination of practices to which they apply. It may however be necessary to bear in mind that the definition and choice of dose limits do involve social judgements and that they cannot be based on health considerations alone.

Classes of Exposure

For the purpose of applying dose limits, the ICRP recognises three classes of exposure.

(i) *Occupational exposure* incurred at work (place) and principally as a result of work (situations) that can reasonably be regarded as being the responsibility of the management. From amongst the natural sources of exposure, those due to ^{40}K in the body, cosmic rays at the ground level, and radionuclides in earth's crust are all outside any reasonable scope of control. Only radon in work places and work with materials containing natural radionuclides can be regarded as the responsibility of the management.

(ii) *Medical exposures* defined as those received by (a) individuals undergoing diagnosis or treatment, (b) volunteers taking part in a programme of biomedical research, and (c) non-occupationally exposed individuals who knowingly and willingly incur exposure by helping, supporting or comforting patients undergoing radiation treatment or diagnosis.

(iii) *Public exposure* which comprise all other exposures.

Dose limits apply to occupational and public exposures.

Dose Limits In Occupational Exposure

Dose limits are needed as a part of control of occupational exposure and include exposure from (i) normal operations, (ii) minor mishaps and misjudgements in operations and (iii) maintenance and decommissioning under circumstances not necessarily envisaged by the designers. This is an extension of the commissions previous concepts of dose limits and makes the recommendations more stringent.

Basis for Dose Limits

The basis of choosing a limit on risks to which an individual may be subjected has always been difficult to specify. In 1977 recommendations, the Commission attempted to use a comparison with rates of accidental death in industries not associated with radiation. This comparison was not satisfactory for many reasons:

(i) Standards of industrial safety are neither constant nor uniform worldwide

(ii) Mortality data relate to average over the whole industries while dose limits apply to individuals

(iii) Quantitative comparisons were limited to mortality data though inclusion of nonfatal conditions on both sides of comparisons would have been more appropriate (making dose limits less restrictive) and

(iv) There are few grounds for believing that society expects the same standard of safety across a wide range of industries.

In the past, the Commission has used the attributable probability of death and severe hereditary conditions as the basis for judging the consequences of an exposure (risk). The Commission has now adopted a more comprehensive approach and considered other factors in the definition of detriment making it a multiattribute approach. They include the length of life lost due to an attributable death, the incidence of nonfatal conditions and serious as well as less severe genetic disorders in future generations. Thus, in principle, a single index of detriment has now been defined though it is still extremely difficult to judge its implications or tolerability.

Degree of Tolerability of Dose Limits

The Commission has, therefore, found it useful to use **three words** to indicate the degree of tolerability of an exposure (or risk) while arriving at the definition of dose limit. They are: **Unacceptable, Tolerable and Acceptable.**

The dose limit is set such that continued exposure at a dose just above the limit would be unacceptable on any reasonable basis. Continued exposure just below the dose limit might be tolerated but would not be welcome, so that acceptable doses are those somewhat below the limit. In order to decide where the boundary between unacceptable and tolerable is to be set, ICRP has taken into account a range of quantifiable factors of health detriment.

Acceptable Risk Figure and Attributes of Detriment

For occupational exposure, ICRP calculated the consequences of working from age 18 to 65 years, i.e. 47 years, at annual doses of 10, 20, 30 and 50 mSv. Fig. 1 shows the fatality probability due to exposure as a function of age, which because of the use of a multiplicative model, tends to follow probability of death from cancer for a general population. The peak risk rate arises in the late-70 years age group for all exposure groups. An annual risk figure of 10^{-3} per year is 'often the most' that is found in conventional industries and this figure would be exceeded at an age in the mid-50 years age group for someone receiving 50 mSv per year and in the mid-60 years age group for someone receiving 20 mSv per year. ICRP has also decided to allow for nonfatal and hereditary conditions in its considerations, and for the period of life lost or impaired. For nonfatal cancers, the weighted number is 20% of the number of fatalities. The corresponding figure for hereditary defects adds a further ~20% to the fatality rate (or 27% for a population of all ages). The result of using these figures is shown in Table 1 where, for comparison, the corresponding figures using the 1977 recommendations of ICRP are also included.

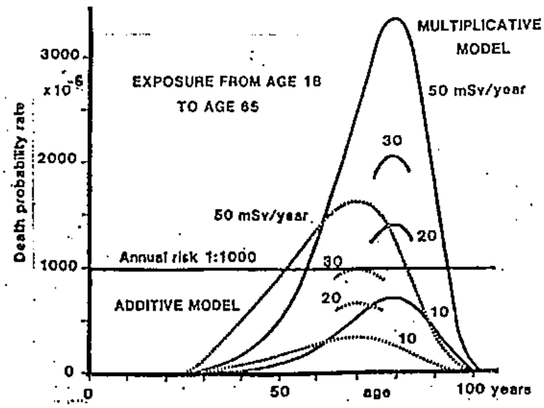

Fig. 1 Annual risk of death from radiation induced cancer for 10, 20, 30 and 50 mSv per year, received from age 18 to 65 years.

On considering these points and the data given in Table 1, ICRP decided to set a dose limit of 20 mSv per year to be averaged over a period of 5 years with no more than 50 mSv in a single year. At this rate of exposure, which ICRP considers verging on the unacceptable, the lifetime risk of induced cancer (fatal plus weighted nonfatal) plus hereditary defects is ~5%, slightly above the corresponding figure for 1977 data which, however, did not include nonfatal effects. The fatal cancer rate is similar, as is the loss of life expectancy for these exposures. Similarly for internal exposure control, Annual Limits of Intakes (ALIs) will be calculated on the basis of effective dose of 20 mSv per year but intakes will be allowed to be averaged over 5 years.

Because of the nature of definition of the term 'Dose Limit', the Commission does not now recommend the use of lifetime dose limit.

Organ Doses

The restrictions on effective dose are sufficient to avoid deterministic effects in almost all body tissues and organs except the lens of the eye (which makes no contribution to the effective dose) and the skin (which may well be subject to localised exposures). For lens of the eye, the Commission continues to recommend an annual equivalent dose limit of 150 mSv, as the threshold dose (received annually) for visual impairment (cataract) is greater than this dose. For external exposure to penetrating radiation over any substantial part of the whole body, including eyes, the effective dose limit will be more restrictive.

For the skin, the situation is more complicated. For stochastic effects, the equivalent dose can be averaged over the whole area of the skin. The stochastic effects are expected to arise in the basal layer at a nominal depth of 7 mg/cm^2 (range 2-10 mg/cm^2). Some deterministic effects also arise at the same depth, others arise in the deeper layers of the dermis (30-50 mg/cm^2). The limitation on the effective dose provides sufficient protection for the skin against stochastic effects. An additional limit is

Table 1. Attributes of detriment due to exposure of the working population

Annual effective dose (mSv) Approximate life time dose (Sv)	10 0.5	20 1.0	30 1.4	50 2.4	50 2.4 (1977 data)
Probability of attributable death (%)	1.8	3.6	5.3	8.6	2.9
Weighted contribution from nonfatal cancer (%)	0.36	0.7	1.1	1.7	--
Weighted contribution from hereditary effects (%)	0.36	0.7	1.1	1.7	1.2
Total (Aggregated detriment)(%)	2.5	5.0	7.5	12.0	4.1
Time lost due to an attributable death (y)	13	13	13	13	10-15
Mean loss of life expectancy at age 18 years (y)	0.2	0.5	0.7	1.1	0.3-0.5

Table 2. Recommended Dose Limits and Constraints

Application	Dose limit	
	For occupational workers	For public
Effective Dose	20 mSv/year averaged over a defined period of 5 years (max. of 50 mSv in any single year)	1 mSv/year or (1mSv/y averaged over 5 years in special circumstances)
Annual Equivalent Dose in		
Lens of the eye	150 mSv	15 mSv
Skin	500 mSv*	50 mSv
Hands & feet	500 mSv	--
Supplementary Eqv.Dose Limit for Pregnant Women (from the time of declaration of pregnancy)	2 mSv for the surface of the abdomen and 0.05 ALI for intake of radionuclides for the remaining period of pregnancy	--
Dose constraints for comforters and visitors of patients		5 mSv for adult 1 mSv for child

* Averaged over an area of no more than 1 cm^2, regardless of the area exposed.

however needed for localised exposures in order to prevent deterministic effects. The recommended annual limit is 500 mSv averaged over 1 cm^2, regardless of the area exposed, at a nominal depth of 7 mg/cm^2. This limit, applied to the skin of the face, will also provide protection for the lens of the eye against localised exposures to radiation of low penetrating power such as beta particles. The same limit, i.e. 500 mSv/year is applied to all the tissues in the hands and feet.

Occupational Exposure of Women

The basis for control of occupational exposure of women who are not pregnant is the same as that for men. However, if a woman is, or may be, pregnant, additional controls have to be considered to protect the unborn child. The ICRP policy is that methods of protection at work should provide a standard of protection for any conceptus broadly comparable with that provided for members of the public. ICRP believes that this will be achieved if the women is working under the normal system of protection before the pregnancy is declared. Once pregnancy has been declared, the conceptus should be protected by applying a supplementary dose limit to the surface of the woman's abdomen of 2 mSv for the remainder of the pregnancy and by restricting intakes of radionuclides to one-twentieth of an ALI.

It may be noted here that all the occupational dose limits stipulated above apply to persons of age 18 and above and are summarised in Table 2.

Occupational Exposure of Apprentices and Students

Apprentices undergoing training prior to employment and students (in the course of their studies) may get exposed to radiation as a part of their routine. For persons of age group 16 to 18 years falling in this category, the occupational exposure shall be governed by the following limits:

(i) An effective dose of 6 mSv in a year

(ii) An equivalent dose of 50 mSv in a year to the lens of the eye and

(iii) An equivalent dose of 150 mSv in a year to the body extremities or skin.

It may be noted that the above doses are 30% of the regular occupational dose limits for adult workers. No occupational exposure is permitted below the age of 16 years.

Special Circumstances

The recommendations on dose limits have a special relaxation clause (provision) to accommodate occupational exposures that are above the dose limits prescribed by the Commission. According to this provision, the dose averaging period may be extended to 10 years provided that

(i) The practice has been justified

(ii) The protection in the practice has been optimised and

(iii) It can be predicted that reasonable efforts in due course will bring the occupational exposures within the limits.

Under these conditions, the circumstances of work involving radiation exposure shall still be reviewed when the dose accumulated by any worker, under the extended averaging period, reaches 100 mSv. Averaging period shall not exceed 10 years under any circumstances and the dose in any single year shall not exceed 50 mSv.

Dose Limits in Medical Exposure

For Patients

Medical exposures are usually intended to provide a direct benefit to the individual undergoing diagnosis or treatment. Detriment, if any, would also accrue to the same individual. Therefore if,

(i) The practice has been justified (which is almost invariably so) and

(ii) The protection has been optimised (less attention has been paid to this area).

The dose to the patient will be **"As Low As is Compatible (ALAC)"** with medical purpose.

Any further application of limits may therefore be to the disadvantage of the patient. Dose limits are therefore not recommended for medical exposures. Consideration has therefore been given to the use of dose constraint/ investigation levels/ guidance levels (selected by appropriate professional or regulatory agency) which show what is achievable by a good practice in common diagnostic procedures. Similarly, the patients' doses do not form part of compliance with dose limits as applied to occupational or public exposures. However, attention is paid to all exposures, resulting from medical exposures, to judge whether protection has been optimised.

For Comforters and Visitors of Patients

Although no dose limits apply to this category of persons, their doses shall be individually constrained such that they will not exceed 5 mSv during the period of a patient's diagnostic examination or treatment. While no children are permitted to support a patient during his/her diagnosis/treatment, they may visit patients who have been given a dose of radioactive material (pharmaceutical). The dose of such children shall be constrained to less than 1 mSv. Further, it is considered highly unlikely that the same person will volunteer to support several patients undergoing radiation treatment.

Dose Limits for Public Exposure

The control of public exposure in all normal situations is exercised by the application of controls at the source and the concept of dose commitment has been found useful for this purpose. If a limit is set to effective dose commitment to a critical group from each year of practice that continues at a constant annual level, the average annual individual effective dose will never exceed that limit.

The Commission defines the scope of its dose limits for public exposure by confining it to doses incurred as a result of practices. The practices cover intended emission of radionuclides from installations, including the emission of naturally occurring radionuclides from installation like mines and waste disposal sites. The resulting doses are therefore subject to dose limits.

For members of the public, the same approach (as evolved for occupational workers) has been used to consider different results of exposure over a lifetime received at 1, 2, 3 or 5 mSv/y. On the basis of risk level between 10^{-5} and 10^{-4}/y and the variation in natural background radiation (excluding radon), ICRP has reconfirmed that the dose limit for members of the public should be 1 mSv/y or, in special circumstances, 1 mSv/y averaged over 5 years (see Table 2). Similarly the equivalent dose limit for the lens of the eye and skin shall be 15 mSv and 50 mSv in a year respectively (see Table 2).

Table 3. Exposure Limits on Intake of Radon/ Thoron Progeny for Occupational Workers

Quantity	Radon Progeny		Thoron Progeny	
	Ann. Average over 5 y	Max. in 1 y	Ann. Average over 5 y	Max. in 1 y
(i) Potential α-energy intake (mJ)	17	42	51	127
(ii) Potential α-energy exposure (WLM)	4	10	12	30
(iii) Potential α-energy exposure conc. ($\mu J/m^3$)	7	17.5	21.2	53
Equilibrium equivalent concentration (EEC) (Bq/m^3)	1235	3080	275	690

WLM - Working Level Months

Exposure Limits on Intake of Radon/Thoron Progeny

In the special case of inhalation of shortlived daughters of ^{222}Rn and ^{220}Rn, intake limits may be expressed in terms of inhaled potential alpha energy or time-integrated exposure in practical units of Working Level Months (WLM) or in concentration units. The limits are as shown in the Table 3.

Conclusions

In generating the above standards for radiation protection, the Commission has aimed at using risk estimates that are not likely to underestimate the consequences of exposure. In the radiation protection community, there is a largely held view that the degree of scientific knowledge available today does constitute an acceptable basis for a conservative system of protection.

It may be noted that while retaining the scientific approach in the formulation of limits, their social and economic implications have not been ignored. The Commission has also tried to make the basis of all such judgements as clear as possible. This is a tribute to openness in their own deliberations.

References for Further Reading

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3. Radiation Safety, A Booklet of IAEA, Division of Public Information, April 1996, (96-00725 IAEA/PI/A47E).
4. Safety series no. 115, International Basic Safety Standards for Protection against Ionising Radiation and for Safety of Radiation Sources, IAEA, Vienna, Austria, Feb. 1996.
5. Radiation Protection: Today and Tomorrow (OECD, 1994).

Glossary

Annual limit on intake (ALI): The intake by inhalation, ingestion or through the skin, of a given radionuclide in a year by the reference man which would result in a committed dose equal to the relevant dose limit. ALI is expressed in units of activity.

Dose limit: The value of the effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded.

Effective dose: The quantity E, defined as a summation of the tissue equivalent doses, each multiplied by the appropriate tissue weighting factor:

$$E = \sum_{T} W_T \cdot H_T$$

where H_T is the equivalent dose in tissue T and W_T is the tissue weighting factor for tissue T. From the definition of equivalent dose, it follows that:

$$E = \sum_T W_T \,.\, \sum_R W_R \,.\, D_{T,R}$$

where W_R is the radiation weighting factor for radiation R and $D_{T,R}$ is the average absorbed dose in the organ or tissue, T. The unit of effective dose is $J.kg^{-1}$, termed the sievert (Sv).

Equivalent dose: The quantity $H_{T,R}$, defined as:

$$H_{T,R} = D_{T,R}.W_R$$

where $D_{T,R}$ is the absorbed dose delivered by radiation type R averaged over a tissue or organ T and W_R is the radiation weighting factor for radiation type R.

When the radiation field is composed of different radiation types with different values of W_R, the equivalent dose is:

$$H_T = \sum_R W_R.D_{T,R}$$

The unit of equivalent dose is $J.kg^{-1}$, termed the sievert (Sv).

Exposure: The act or condition of being subject to irradiation. Exposure can be either external exposure (irradiation by sources outside the body) or internal exposure (irradiation by sources inside the body). Exposure can be classified as either normal exposure or potential exposure; either occupational, medical or public exposure; and, in intervention situations, either emergency exposure or chronic exposure.

Safety culture: The assembly (or summation) of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance.

Working level: A unit for potential alpha energy concentration (i.e. the sum of the total energy per unit volume of air carried by alpha particles emitted during the complete decay of each atom and its progeny in that volume of air) resulting from the presence of radon progeny or thoron progeny equal to emission of 1.3×10^5 MeV of alpha energy per litre of air. In SI units the WL corresponds to 2.1×10^{-5} J/m^3.

Working Level Month (WLM): A unit of exposure to radon progeny or thoron progeny.

1 WLM = 170 WL.h

One working level month is equivalent to 3.54 $mJ.h.m^{-3}$.

Change: Alteration in the character or makeup of a system. A change may or may not be harmful.

Damage: It represents some degree of deleterious change. For example, damage to cells but it may not necessarily be deleterious to the exposed individual.

Harm: It is used to denote clinically observable deleterious effects that are expressed in individuals (somatic effects) or their descendants (hereditary effects).

Detriment: It is a complex concept combining probability, severity and time of expression of harm and may be briefly stated as mathematical expectation of harm taking into account the above factors. It is thus a measure of total harm. Detriment here refers to health detriment.

Radiation Protection – In Perspective

Dr. U.C. Mishra joined the Department of Atomic Energy (DAE) in 1958 after completing an one year training in Nuclear Science and Engineering. He was later trained at the Argone National Laboratory, USA during 1962-63. He has more than 40 years of research and development experience in various areas of environmental sciences. He is a research guide of Mumbai and Gujarat Universities. Dr. Mishra is a Pitamber Pant Fellow of Ministry of Environment and Forests. He is also a Fellow of the National Academy of Sciences and the Maharashtra Academy of Sciences. He has served in several national/ international organisations as advisor/consultant. He is been in the Editorial Board of several journals. He has over 300 publications in national and international journals and chapters in books. Dr. U.C. Mishra retired as Director, Health, Safety and Environment Group of BARC and the group, comprised of more than 500 scientists was responsible for providing radiation protection services for all the units of DAE covering the entire nuclear fuel cycle, and all radiation users in the country.

Introduction

The most visible of the benefits derived from ionizing radiation are the 437 nuclear power plants (status as of March 1998) now in operation worldwide, accounting for about 17% of the total electricity production the world over. Application of radiation and radioisotopes is also widespread in diagnostic and therapeutic medicine, in industry and agriculture. India has ambitious programmes in these applications to support industrial and technological growth and basic health care programme. At present, the Indian nuclear power contribution is small as compared to the ten other countries (Table 1). However, the Department of Atomic Energy has plans to increase the power output substantially by the year 2020.

Table 1. Ten largest consumers of Nuclear Power

Country	No. of plants	Total MW(e)
USA	109	99,784
France	56	58,493
Japan	59	38,875
Germany	21	22,657
Russian Federation	29	19,843
Canada	22	15,755
Ukraine	15	12,679
United Kingdom	12	11,720
Sweden	12	10,002
Republic of Korea	10	8,170
Total	335	297,978
World	437	340,347

For supporting the above objectives, a full-fledged and wide-ranging facilities are established in the country. The facilities consist of uranium mines and mills, fuel fabrication plants, nuclear reactors, fuel reprocessing plants to recover plutonium and radioactive waste management facilities. The cycle of operations together is called nuclear fuel cycle. At present, natural uranium is used as the nuclear fuel. In addition to the above nuclear fuel cycle operations, facilities are built to process and produce reactor-produced isotopes for use in industry and medicine. Accelerators are the latest addition to the line of irradiators erected by the Department. The future Indian nuclear power progamme envisages utilization of the vast resources of thorium present in monazite sand.

As we all know, exposure to ionizing radiation at higher levels results in biological effects which can be harmful to the body. In order to control radiation exposure of occupational workers and to protect the public and environment from the exposures due to the discharges from nuclear

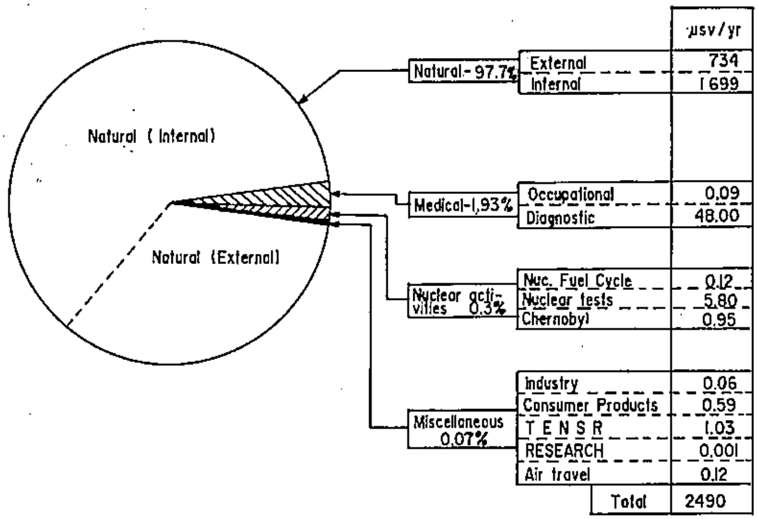

Fig. 1 Per caput radiation dose to Indian population from all sources

facilities, a comprehensive radiation protection progamme is organised and implemented by Health, Safety & Environment Group of Bhabha Atomic Research Centre (BARC). Environmental impact of the man-made radiation sources is continuously assessed so that contribution from these sources to radiation dose to the public remains negligible as compared to the natural background radiation dose. The per caput population dose arising from the natural and the man-made radiation sources is depicted in Fig. 1. About 98% of the dose is from exposure to natural sources. A separate operational Radiation Hazards Control Unit, is established in all the nuclear facilities to provide the necessary radiation/protection surveillance. A separate Environmental Survey Laboratory is located in six Centres in India for assessing the environmental impact from the operation of these facilities. A strong Research and Development Programme in all the related topics such as radiation monitoring instrumentation, protection standards, operational limits, etc. backs up these radiation protection activities. In addition to this, radiation protection expertise is provided for implementation of new departmental projects and faculty support is provided for development of trained manpower at different levels, for various activities. At international level, regional training courses are conducted in collaboration with bodies such as IAEA for the benefit of radiation protection professionals.

Radiation from different facilities

Depending upon the nature of operation and the facility, the radionuclides handled or generated are of different types, in terms of emission and chemical or physical state. Various sources of radiation in typical nuclear reactor facility is shown in Fig. 2. Radiological toxicity depends mainly on the metabolic behaviour of the radioactive material, type of emission and its energy. Table 2 shows the type of radionuclides, which are generally, encountered in some typical nuclear fuel cycle operation.

Biological Effects

Nuclear radiations cause ionization of atoms/molecules of biological interest, which in turn can cause undesirable chemical changes. Background radiations, which are natural and hence unavoidable also, causes similar biological effects. Similar biological changes occur from exposure to many other conventional pollutants present in the environment of occupational workers and the public. However, the living systems have evolved a very efficient repair mechanism that usually limits the damage caused to the body cells.

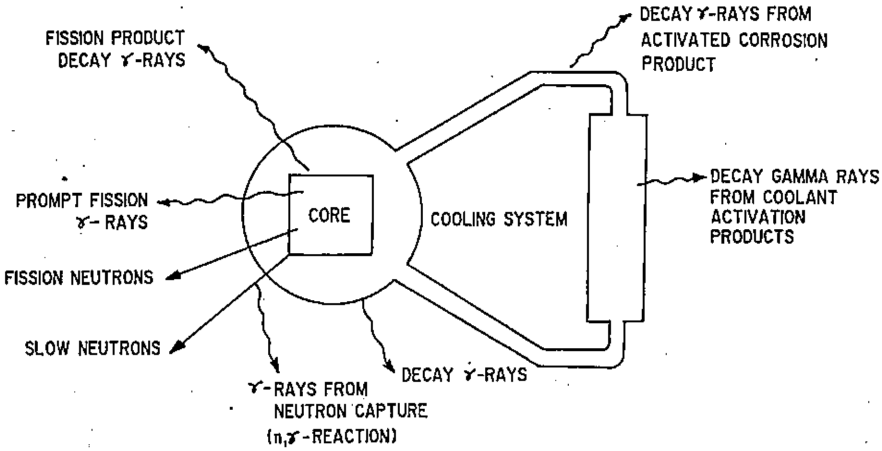

Fig. 2 Sources of radiation in a nuclear reactor system.

Table 2. Typical radionuclides in nuclear fuel cycle operations

Operation type	Radionuclides of interest	Type of hazard
Mining and Milling	Low specific activity natural radionuclides e.g. ^{238}U, ^{232}Th, ^{228}Th, ^{226}Ra, ^{228}Ra, ^{222}Rn, ^{220}Rn and their short-lived progeny	Inhalation of radioactive dust, and gaseous ^{222}Rn & ^{220}Rn and their daughters. External exposure from the concentrate.
Nuclear Fuel Fabrication	Natural uranium and its beta emitting daughters (e.g. ^{234}Th, ^{234}Pa) and ^{230}Th	Inhalation of dust containing natural radionuclides.
Reactor Operations	Neutron activation products (e.g. ^{3}H and ^{41}Ar)	Inhalation of tritium and ^{41}Ar gas.
Nuclear Fuel Reprocessing	Fission Products (e.g. ^{131}I, $^{137+134}Cs$, ^{104}Ru, $^{89+90}Sr$, etc.) ^{85}Kr, depleted U, and Pu isotopes	External exposure, inhalation of Pu, criticality hazard from Pu.
Nuclear Waste Management	Fission Products, (e.g. ^{137}Cs, ^{90}Sr), Actinide wastes of high specific activity	External exposure, potential public exposure to discharges by ingestion route.

The biological effects are of two types: stochastic and deterministic. The deterministic effects (e.g. cataract of eyes) appear after exposure to relatively high radiation doses. The severity of the effect is proportional to the radiation dose.

The stochastic effect is of interest in radiation protection. Genetic effect and cancer are the two important stochastic effects, which occur due to the mutagenic effect of ionizing radiation. The effect is assumed to have no threshold and the probability of the effect occurring is proportional to the dose. Carcinogenic chemical agents such as benzo(a) pyrene which are present in automobile exhaust; and the tobacco smoke also cause mutation, and is known to cause lung cancer and other cardiovascular disorders. Cancers caused by exposure to radiation and other conventional pollutants are not different and are indistinguishable.

Radiation Protection Philosophy

Philosophy of radiation protection differs from other systems of protection against conventional agents which are potentially harmful to health. The system of radiation protection is quite comprehensive and covers all types of exposures: normal and potential. Protection of occupational workers, their descendents, individual members of the public, and population as a whole is the objective of radiation protection. In addition, environmental safety has been given a high priority while controlling discharges from various facilities.

The basic principles of radiation protection i.e. justification, optimization and dose limitation, are established by an independent, international body of experts called International Commission on Radiological Protection (ICRP). In its recent recommendations, the ICRP has emphasised that ionizing radiation needs to be treated with care rather than fear and its risks should be seen in perspective with other risks. The principles and recommendations of the ICRP generally form the basis of national regulatory bodies and the recommendations have been incorporated by the International Atomic Energy Agency (IAEA) into the its, International Basic Safety Standards for Radiation Protection, (IAEA, Safety Series 115, published in 1996) in collaboration with other organizations such as: World Health Organization (WHO), International Labour Organization (ILO), the Nuclear Energy Agency of the Organisation for Economic Co-operation and Development (OECD/NEP), and the Pan American health Organization (PAHO). India was one of the participating countries which contributed to the development of the IAEA International Basic Safety Standards. The details, in parts, are given elsewhere in this issue. One of the important principles, i.e. optimization of protection calls for making all reasonable efforts to reduce radiation exposure (As Low As Reasonably Achievable – ALARA), and cost the of protection should not be disproportional to the potential reduction in the health detriment.

Radiation Protection of Workers

Routes of exposure

There are four routes of exposures possible. They are: inhalation of airborne radionuclides, ingestion of radioactive materials via food and drinks, injection of radioactive materials directly into the blood through cuts or wounds and lastly absorption of radioactivity via skin (e.g. tritiated water vapour).

Type of exposures

Radiation exposures of persons are broadly categorized into two types, namely normal exposures and potential exposures. The potential exposures may arise as a consequence of incident/accident. The exposures are further classified into three types, namely, occupational exposures that occur during the course of work, public exposures that are incurred by members of public from man-made radiation sources and medical exposures (exposures of the patients in their own diagnosis or treatment). Medical exposures also include exposures of person knowingly exposed while voluntarily helping in the support and comfort of patients, and by volunteers involved in biomedical research programme.

Exposure Control Methods and Procedures

There are three approaches to control external exposures. They are reduction in the exposure time, keeping sufficient distance from the source for reducing radiation levels, and use of shielding between the source and the radiation worker. Internal exposures are controlled by adopting design based good containment for radioactive material, elaborate ventilation and air-cleaning system and an effective radiation safety surveillance programme.

A notable feature of the nuclear power programme is that detailed safety analysis and explicit safety clearances are prerequisites for all its activities. Right from the stage of the design of the reactor, through site selection, construction, commissioning and operation, there exist expert safety groups or committees acting serially before the practice is cleared finally by the Atomic Energy Regulatory Board. The safety requirements are very stringent and have to be complied with. At the beginning itself elaborate accident analysis is carried out using a concept of Design Basis Accidents (DBA), to make sure that even in the case of most severe accident such as the Chernobyl type, with as low a probability as 10^{-6}/year (i.e. one in million years), the potential exposure to the public is low and to the acceptable dose limit.

Radiation Protection Monitoring

Radiation monitoring is an integral part of Safety Surveillance of nuclear facilities. A well-qualified and trained group of health physicists which is independent of the plant management, is located at these facilities for this purpose. These groups serve as eyes and ears for the Atomic Energy Regulatory Board. The types of monitoring include area monitoring, air monitoring, contamination monitoring and individual monitoring of radiation workers. A complete record of all the data related to the radiation exposure of the personnel and radioactivity discharges from the facility is maintained. The radiation dose to man from these releases is calculated taking into account all the possible exposure pathways in the environment (Fig. 3). A strong research & development support is also provided to review and upgrade existing radiation monitoring systems, and to develop new safety systems to cater to the development of new activities by the department.

Environmental Protection

Environmental monitoring is a well-planned comprehensive programme which takes into account all the possible pathways such as air, water, diet, etc. through which radioactivity can cause exposure to man. The environmental impact of nuclear power and other related activities is assessed to make an estimate of the possible enhancement of public exposure to radiation. Table 3 gives the data obtained by such a study at our nuclear power stations.

The approach to protect the environment starts right from: siting of the nuclear facility, design based on defence-in-depth principle for total containment of radioactivity during normal as well as abnormal situations, control of effluents, environmental monitoring and a good safety culture among the workers. Some of these aspects are discussed

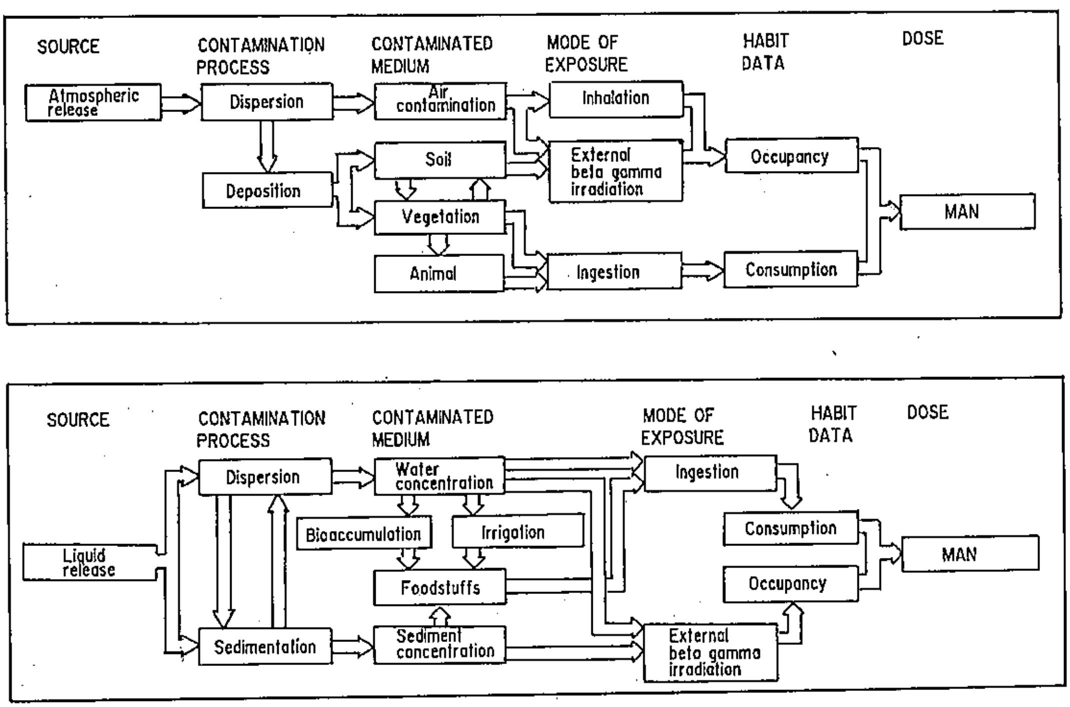

Fig. 3 Possible pathways of human exposure in the environment

Table 3. Environmental radiation dose at Nuclear Power Plant sites (in units of mSv/y)

	TAPS 1,2	RAPS 1,2	MAPS 1,2	NAPS 1,2
Natural background radiation dose	1.96	2.12	2.40	3.22
Dose limit to public from Man-made sources	1.00	1.00	1.00	1.00
Technical specification (as per the design value)	0.35	0.18	0.27	0.45
Actual dose to public at site boundary (1.6 km) (average for 1989-93)	0.048	0.055	0.044	0.0009

elsewhere in this issue. Fig. 4. Shows the multiple barriers to contain the release of radioactivity. Independence of the safety related systems is another way of ensuring safety during any abnormal/accidental situations. Meteorological parameters and seismic parameters for the site are also taken into account in the choice of a site for a nuclear power station. As a further precaution, a nuclear power plant has an exclusion zone of about 1.6 km around it where permanent residence by the public is not allowed. A sterilized zone of 5 km where growth and developments are restricted or controlled surrounds this exclusion zone. Emergency plan is prepared for area covering 16 km radius from a nuclear power plant. The comprehensive plan covers various aspects of emergency preparedness at the plant. In addition to this, a green belt of planted trees is created to enhance the aesthetics of the site as well as to help in dust control and reduce the impact of radioactive release in an unlikely event of an accident. All the gaseous emissions are treated and are made to pass through high efficiency particulate air (HEPA) filters. Inert radioactive gases such as ^{41}Ar and ^{85}Kr are diluted with large quantities of plant ventilation air and discharged through a tall stack.

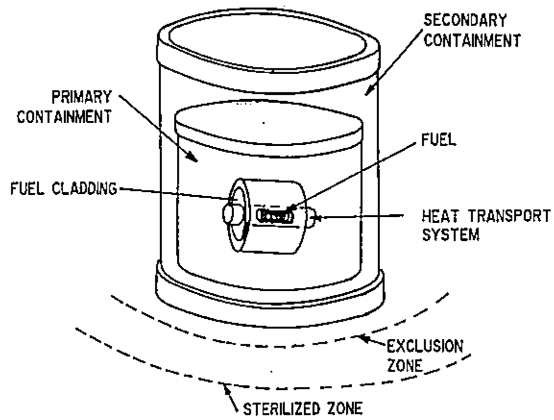

Fig. 4 Barriers to contain the release of radioactivity

Regulatory Control

The purpose of Regulatory Control is to ensure effective implementation of radiation protection programme in all the facilities where radiation/radioactive sources are handled. In India, Atomic Energy Regulatory Board is the national regulatory body, which has the authority (through legislation), to enforce radiation protection standards. India is one of the few countries to have implemented the recent ICRP recommendations which brought down the annual dose limit from 0.05 Sv to an annual average of 0.02 Sv. Further details on regulatory aspects are given in a separate article in this issue.

Comparison of Risks

Risk is the term used to express probability of harm or undesirable health effect occurring due to exposure to pollutants, including ionising radiation. Work areas are often polluted by toxic chemical pollutants such as oxides of nitrogen (NO_x), sulfur dioxide, and carbon dioxide. Exposure to such chemicals at enhanced levels causes morbidity or even mortality. Long term health effects of exposure to such chemical pollutants particularly at lower level i.e. below threshold limit values, are not well documented or quantified. However, in case of ionsing radiation, health effects are well documented

Table 4. Specific activities of natural uranium, thorium, radium in Indian coal and fly ash

Material	Isotopic (Bq/g)	Coal (Bq/g)	Fly ash (Bq/g)	Committed effective dose (Sv/Bq)
Uranium	2.5×10^{4}	7.7×10^{-2}	1.9×10^{-1}	3.33×10^{-5}
Thorium	8.1×10^{3}	2.4×10^{-2}	7.8×10^{-2}	2.22×10^{-6}
Radium-226	3.7×10^{10}	2.4×10^{-2}	7.8×10^{-2}	2.22×10^{-6}

Table 5. Environmental comparison of a 1000 MW(e) coal-fired Thermal Power Plant with a 1000 MW(e) Nuclear Power Plant (Ref: BARC Newsletter 158, 1997)

	Nuclear	Coal
Fuel	125 t/y	2.5×10^{6} t/y
Transport	15 trucks/y	2500 trains/y
Waste	Most of it is reusable (U + Pu) ≈ 15 m^3 of vitrified high level waste	5×10^{5} t/y ash 5×10^{6} t/y CO_2 2×10^{5} t/y SO_2 3×10^{4} t/y NO_x 10 t of lead, mercury and arsenic

and quantified which enables realistic assessment of risk at different exposure level.

The amount of radiation 'dose' may be defined as energy of the radiation absorbed by unit mass of body tissue, and is expressed in Sievert (=1J/kg) in International System of units. The risk equivalent of radiation dose (RERD) for chemical pollutants in air at their permissible levels work out much higher than the risk from radiation exposure at the dose limits.

Nature is the largest source of radiation exposure. We are exposed to this natural background radiation, which is unavoidable. Detailed surveys have shown that the normal natural background radiation in different parts of India varies from place to place depending upon the altitude from the sea level and on the radioactivity content of the local soil. The variation is in the range 1.3 mSv to 2.5 mSv/y. In monazite areas in southern coast of India and mineralised belt of Bihar, the annual dose may be as high as 10 mSv.

The coal used in Thermal Power Plants (TPPs) contain small amounts of radioactivity, the amount depends upon the the origin of the coal. The radioactivity present in the coal gets concentrated in the coal-ash (Table 4) and forms another source of radioactivity in addition to the sources in nuclear industry.

Table 5 gives an environmental impact comparison of a 1000 MW(e) coal-fired Thermal Power Plant with a 1000 MW(e) Nuclear Power Plant. It has been estimated that the population dose from the inhalation pathway due exposure to radioactivity in the coal-ash is higher as compared to the population dose from exposure to atmospheric releases from Nuclear Power Plants. Similarly, the dose from exposure to man-made radioactive substance is negligible as compared to the natural background radiation dose, indicating that these have no observable impact whatsoever on the health of population. Further, the burning of fossil fuel in thermal power plants generate chemical pollutants causing acid rains and the carbon dioxide adds to the greenhouse gas load in the environment. It can warm up the earth's atmosphere leading to changes in our environment. These pollutants are not generated in nuclear power plants. The small amounts of radioactive waste encountered in the nuclear plants can be managed through intensive efforts and technical solutions to the nuclear waste management so that it is completely isolated from the public environment.

Future Outlook

Since we live in a naturally radioactive environment, small additional amounts of man-made nuclear radiation need not deter us from enjoying the inumerable advantages of radiation applications. The per caput radiation dose to Indian population from natural as well as from all the man-made radiation sources is presented in Fig. 1. The contribution from the nuclear activities is only 0.37% of the average annual exposure of 2.49 mSv. The effects of radiation are well studied and radioactivity measurement and monitoring techniques are well developed. The radiation protection standard is very stringent and is based on the experimental data. Systematic work on radioactive emissions and maintaining exposures as low as reasonably achievable (ALARA) has brought significant success in limiting already low radiological impacts of nuclear fuel cycle operations and other application of radiations and radioisotopes in medicine, industry and agriculture. It is now necessary to look at the radiation risk in proper perspective along with other conventional risk to which a worker is always exposed.

Environmental Radiation Exposure: Natural and Man-made

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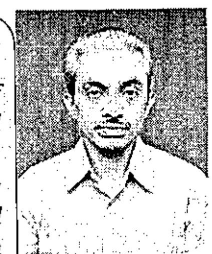

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Introduction

Exposure to ionizing radiation originates from two major sources: natural (background) radiation and man-made radiation. The natural radiation present on earth distinctly falls into two categories such as virgin natural sources (VNS) of radiation and modified natural sources (MNS) of radiation. The virgin sources of radiation are of cosmogenic or primordial (terrestrial) origin. The virgin natural sources of radiation have existed on the earth long before life emerged as they were present in space before the formation of earth itself. The modified natural sources of radiation consists of mining activities, usage of fossil fuel, fertilizers, natural material for building and air/space travel. The sources of the man-made radiation include nuclear fuel cycle operation, nuclear weapon tests, medical and industrial application of radionuclides. The extent of exposure to natural and man-made radiation depends on the occupation, type of dwelling, location of habitation, life style and level of medical care they receive. Natural radiation is of particular importance because this source is the largest contributor to the collective dose of the world's population. The relatively constant exposure rate to the humans is the distinctive characteristics of this radiation. In the midst of a radiation environment, however low it may be, it is not possible to avoid radiation exposure from natural sources. All what is needed and possible is to be conscious to this fact with a constant endeavour to control the radiation from man-made sources to levels as low as reasonably achievable.

Virgin Natural Sources

Cosmic Radiation

Galactic cosmic rays are created outside the solar system. They are produced and accelerated as a consequence of stellar flares, supernova explosions, pulsar acceleration or the explosions of galactic nuclei. Cosmic rays consist of 85% protons, 14% alpha particles, and about 1% nuclei of atomic number between Z=4 and Z=26. These particles are highly penetrating and have a mean energy of 10^{10} eV and maximum energies up to 10^{19} eV. The higher energy primary particles interact with atmospheric nuclei and undergo nuclear reactions (spallation) producing neutrons, protons, pions, and kaons as well as a variety of reaction products (cosmogenic nuclides). The high energy spallation particles thus formed react to form secondary particles. Primary cosmic rays are substantially affected by the earth's magnetic field which deflects the lower-energy radiation back into space. Hence, the flux of cosmic rays is greater at the north and south poles than that at the equator. Attenuation in the atmosphere decreases the flux of cosmic rays at the earth's surface. As a result, the cosmic ray exposure becomes double [1] at every 1500 m above the earth's surface. The cosmic ray dose rate at various altitudes are given in Table 1.

Table 1. Cosmic ray dose rate at various altitudes

Elevation above sea level (m)	Equivalent dose rate (μSv/y)
0-150	260-270
150-305	270-280
305-610	280-310
610-1220	310-390
1220-1828	390-520
1828-2438	520-740
2438-3408	740-1070
> 3408	1070

Cosmic rays contribute mainly to the external exposure. The attenuation by buildings, causes 20% reduction in cosmic radiation indoors. In general, the individual annual effective dose from cosmic rays ranges between 260 μSv/y to 2000 μSv/y. The average effective dose from cosmic radiation in India is about 355 μSv/y and the world average works out to be 380 μSv/y [2]. Since the dose rate from cosmic radiation increases with altitude, in high altitude places like Gulmarg in India, the annual effective dose to residents is about 830 μSv [3].

Cosmogenic Radionuclides

A number of radionuclides are produced by cosmic radiation through (n,α), (n,p) and (n,γ) reactions. Most of them contribute little to the natural radiation exposure and a few of those contributing significantly to the natural radiation include ^{3}H, ^{7}Be, ^{14}C and ^{22}Na. Tritium is produced in the atmosphere by interaction of cosmic ray neutrons with nitrogen. It is then converted to tritiated water and participates in the normal water cycle. The cosmogenic ^{7}Be is produced by the interaction of cosmic ray neutrons with lithium (n-α reaction) in the atmosphere. The main pathway to humans for ^{7}Be is through the ingestion of leafy vegetables. The cosmogenic nuclide ^{14}C is produced in the upper atmosphere by interaction of slow cosmic neutrons with nitrogen. This nuclide is metabolized as CO_2 and appears in the photosynthetic cycle in plants. The production rates and global inventories of cosmogenic radionuclides [4] are presented in Table 2. The cosmogenic radionuclides contribute to the internal exposure mainly through inhalation.

Table 2. Production rates and inventories of cosmogenic radionuclides

Nuclide	Half-life (y)	Production rate (atoms/ cm²/s)	Global inventory (MCi)
^{3}H	1.23×10^{1}	2.50×10^{-1}	3.40×10^{1}
^{7}Be	1.45×10^{-1}	8.10×10^{-2}	1.10×10^{1}
^{10}Be	2.70×10^{6}	4.50×10^{-2}	6.00
^{14}C	5.73×10^{3}	1.67	2.30×10^{2}
^{26}Al	7.40×10^{5}	1.40×10^{-4}	2.00×10^{-2}
^{22}Na	2.60	8.70×10^{-5}	1.00×10^{-2}
^{32}P	3.92×10^{-2}	8.10×10^{-4}	1.00×10^{-1}
^{37}S	2.38×10^{-1}	1.40×10^{-3}	2.00×10^{-1}
^{36}Cl	3.10×10^{5}	1.10×10^{-3}	1.50×10^{-1}

1 MCi = 37 PBq

The concentration of a few cosmogenic radionuclides in different environmental matrices along with their annual intake and the resulting average effective dose to man [1, 5] are presented in Table 3.

Table 3 indicates the importance of ^{14}C and ^{7}Be in terms of effective dose received by man annually. The values for ^{3}H and ^{14}C represent pre-nuclear explosion and they can be marginally higher in local regions wherein technologically induced activities prevail. The global average annual

Table 3. Annual intake of cosmogenic radionuclides and the resulting effective dose

Nuclide	Concentration (Bq/m^3)	Intake (Bq/y)	Effective dose (μSv/y)
^{3}H	Surface water : 400 Ocean water : 100	5.0×10^{2}	1.0×10^{-2}
^{7}Be	Air : 0.03 Rain water : 700	1.0×10^{2}	3.0
^{14}C	Specific activity : 227 (Bq/kgC)	2.0×10^{4}	1.2×10^{1}
^{22}Na	Air : 3.0×10^{-7}	5.0×10^{1}	2.0×10^{-1}

effective dose contribution from cosmogenic radionuclides through internal exposure amounts to be about 15 μSv.

Terrestrial Sources

Primordial Radionuclides

The radionuclides existed on the earth's crust since its formation are referred to as primordial radionuclides. The primordial radionuclides having half-lives <10^8 years might have disappeared from the earth's crust due to radioactive decay and those existing now are very long-lived having half lives >10^{10} years. Some of the important primordial radionuclides are ^{40}K ($T_{1/2}$=1.26×10^9 y), ^{50}V (6.0×10^{15} y), ^{87}Rb (4.8×10^{10} y), ^{123}Te (1.2×10^{13} y), ^{232}Th (1.4×10^{10} y), ^{235}U (7.04×10^8 y) and ^{238}U (4.47×10^9 y). Apart from these singly occurring radionuclides, the primordial nuclides include 3 distinct chains of radionuclides viz: uranium series (Parent : ^{238}U), thorium series (Parent : ^{232}Th) and actinium series (Parent : ^{235}U). The isotopic composition of natural uranium is 99.28% of ^{238}U, 0.71% of ^{235}U and 0.0058% of ^{234}U (by weight) and this nuclide is found in various quantities in most rocks and soils. The uranium isotopes being alpha emitters do not contribute to the gamma background radiation. Their concentrations in environment is too low to contribute significantly to the internal alpha dose. However, food and human tissues contain uranium derived from soil and fertilizers. The concentration of some important primordial radionuclides in different environmental matrices [4,5,6] are given in Table 4.

The levels of terrestrial radiation differ from place to place around the world, as the concentrations of these primordial radionuclides in the earth's crust vary considerably. Internal exposure especially via inhalation is the predominant pathway for radiation exposure by primordial radionuclides. Potassium is an essential element for human body. The radioactive ^{40}K is absorbed in the body along with its stable counterpart. The body content of potassium in an Indian adult works out to be 3.06 g/kg of body weight. the ^{40}K present in this will contribute to an annual effective dose of 189 μSv/y for an average adult in India [7]. The global average effective dose to a person due to intake of ^{40}K and ^{87}Rb is about 180 μSv/y and 6 μSv/y respectively [8].

Primordial radionuclides such as U, Th and actinium produce series of daughter radionuclides by radioactive decay. The whole spectrum of primordial radionuclides together with their daughter products form the major source of terrestrial radiation. Thus, the isotope ^{226}Ra which originates from uranium series, and its daughter products (Radon, RaA, RaB etc.) assume special importance as they contribute to the internal exposure to man through inhalation pathway. ^{226}Ra, an alpha emitter, is present in varying amounts in all rocks, soils and water. This nuclide ($T_{1/2}$=1622 y) decays to ^{222}Rn (radon) which is a noble gas ($T_{1/2}$=3.8 days). The daughter products of radon are alpha-beta emitting radionuclides. In thorium series, thoron (^{220}Rn) is produced by decay of ^{224}Ra and decays to ^{216}Po by alpha emission and subsequently to stable ^{208}Pb. Radium is chemically similar to calcium and is absorbed from the soil by plant and passes through the food chain to humans. The average content of ^{226}Ra in total diet ranges between 192 Bq/kg to 270 Bq/kg [9]. The average content of ^{226}Ra in human skeleton is about 85 mBq

Table 4. Concentration of primordial radionuclides in different environmental matrices

Matrix	^{238}U	^{226}Ra	^{40}K	^{87}Rb
Igneous rock (Bq/g)	0.04	0.048	1.2	-
Phosphate rock (Bq/g)	1.60	1.5	0.4	-
Limestone (mBq/g)	16.0	5.0 - 20.0	30.0 - 150.0	-
Soil (mBq/g)	37.0	16.0	100.0	-
Air (μBq/m^3)	1.2	1.5	22.0	-
Surface water (mBq/L)	0.18 - 62.9	0.4 - 111.0	3.7 - 244.2^a	0.9
Ocean surface water (mBq/L)	44.4	1.3 - 3.1	11.0^a	100.0
Ocean bottom water (mBq/L)	40.0	3.0 - 5.6	11.0^a	-
Human (Bq)	1.3 - 1.6	1.0 - 1.5	6300.0	455.0
Daily intake by human (mBq)	13.0	190.0 - 270.0	100.0 - 140.0^b	7000.0
Annual effective dose (μSv)	1.2	7.0	180.0	6.0

^{a}Bq/L and ^{b}Bq

and that delivers an annual effective dose of 7 μSv. Latest investigations on outdoor external radiation levels in 23 countries reveal that about 95% of people receive an average annual dose of 400 μSv from the terrestrial radiation sources [8].

In India, regions of Maharashtra and South Gujarat are covered by the Deccan lava basalt with very low radioactivity. The Gangetic alluvial regions covering parts of Uttar Pradesh, Bihar and West Bengal have somewhat higher radioactivity. The granite region of Andhra Pradesh exhibits high levels of the primordial radioactivity. The mean external dose from natural sources in high background areas like the coastal strips of Kerala and Tamilnadu, where ^{232}Th content is high in the monazite sand, is estimated as 4000 μSv/y and in some locations along these coastal strips, the annual external dose exceeds even 32500 μSv [10, 11]. A countrywide survey of outdoor natural gamma radiation levels using thermoluminescent dosimeters reveals that the average external dose is about 734 μSv/y [12]. The cosmic and terrestrial components of this estimates work out to be 355 μSv/y and 379 μSv/y respectively. Out of the terrestrial component, 48.7% is contributed by ^{40}K and the remainder by uranium and thorium series [13].

The major contribution of dose from natural radiation arises due to inhalation of radon and its daughters. The contribution from thoron and its daughters originating from thorium series though small, is also significant. ^{222}Rn gas emanates from soil and building material that contain ^{226}Ra. The emanation rate depends on soil porosity, concentration of ^{226}Ra, temperature and pressure. Typical worldwide outdoor and indoor levels of radon are about 7 Bq/m^3 and 45 Bq/m^3, respectively [1]. A national survey in Indian houses indicates that the indoor radon concentration varies between 2.2 to 56 Bq/m^3 with geometric mean of 15.1 Bq/m^3 [14]. The outdoor concentrations of ^{222}Rn varies with day and season. It is highest at midnight and minimum at noon. The translation of exposure to radon and its daughter products in the form of dose is slightly complex. The conversion of radon concentration to working level (1WL=1.3x10^5 MeV alpha energy per litre or 1WL=3700 Bq/m^3) is valid only if the gas is in equilibrium with its daughter products. Further, the proportion of the radon daughters attached to ambient aerosols is not easily known. All the attached fraction deposits in the bronchial tree and

Table 5. Total radiation exposure from virgin natural sources

Sources	India		Worldwide	
	Annual effective dose (μSv)	% contribution	Annual effective dose (μSv)	% contribution
External				
Cosmic radiation	355.0	13.03	370.0	16.14
Terrestrial	379.0	13.91	460.0	19.53
Internal				
Cosmogenic nuclides (inhalation)	15.0	0.55	10.0	0.42
Radon and thoron (inhalation)	1660.0	60.94	1275.0	54.14
Terrestrial (Ingestion)	315.0	11.56	230.0	9.77
Total	2724.0	100.00	2355.0	100.00

has a greater biological impact. Thoron decay series is similar to ^{222}Rn decay series and it is produced wherever ^{232}Th is found. The typical outdoor concentration of ^{220}Rn is estimated to be about 0.2 $Bq.m^{-3}$ [1]. Indian surveys estimate the average annual effective dose from indoor radon-thoron and their progeny through inhalation as 1595 μSv. The same from outdoor radon-thoron and their progeny works out to be 65 μSv. The world average annual effective dose from indoor and outdoor radon-thoron and their progeny is estimated as 1275 μSv [2]. This estimate is based on the assumption that a person spends two third of the day indoors and the remaining period outdoors.

The final annual effective dose from virgin natural sources to members of the Indian population is estimated as 2724 μSv. The contribution of each natural component to the total dose is given in Table 5. A comparison with the worldwide natural radiation dose is also presented in the same table.

Modified Natural Sources

Fossil Fuel contains small amounts of cosmogenic radionuclides and large amounts of primordial radionuclides especially ^{226}Ra and ^{40}K. Among the fossil fuels, coal contains the highest radioactivity and coal burning produces large amount of particulate emissions. The flyash contains more radioactivity than coal itself. The major pathway for radiation exposure from coal-fired power plants is through inhalation. The radioactivity concentration in coal and flyash [1] are presented in Table 6.

Table 6. Radioactivity concentration in coal and fly ash samples

Radionuclide	Coal sample (Bq/kg)	Flyash (Bq/kg)
^{40}K	50 - 100	250 - 700
^{238}U series	16 - 27	200 - 900
^{232}Th series	8 - 27	50 - 150

The annual average release rates of radioactivity from different thermal power plants in India using coal as fuel [15] are given in Table 7.

The annual per caput dose to Indian population from thermal power plants in India is estimated as 0.20 μSv [16]. Other technologically modified sources of radiation include mining operation of heavy metals (such as lead, zinc, copper, manganese and gold), mining of phosphate rock and its use as fertilizer, use of natural gas, air travel and use of consumer products. The annual effective dose from technologically modified sources in India [16] is given in Table 8.

Table 7. Annual average release of radioactivity from thermal power plants in India

Installed capacity (MWe)	Radioactivity released (GBq/y)		
	^{226}Ra	^{228}Th	^{40}K
200 - 400	0.16 - 0.84	0.22 - 1.70	0.37 - 11.2
400 - 600	0.48 - 3.70	1.22 - 2.01	3.04 - 19.5
600 - 800	0.45 - 1.90	0.70 - 4.00	2.24 - 5.9

Table 8. Per caput dose to Indian population from modified natural sources of radiation

Source	Radionuclide	Effective dose (mSv/y)
Mining of heavy metals	Radon and daughters	0.69
Coal fired power plants	^{226}Ra and ^{228}Th	0.20
Mining of phosphate rock and its use as fertilizer	^{238}U decay series, ^{232}Th, ^{40}K	0.012
Natural gas	Radon and daughters	0.13
Production of gas mantles	^{232}Th	4.54×10^{-2}
Luminous dial productions	^{3}H	3.80×10^{-2}
Air travel	Cosmic radiation	0.12
Total		1.24

Man-made Sources

Man-made sources of radiation exposure include fallout due to nuclear weapon tests, operation of nuclear research reactors and nuclear power plants as well as medical and industrial applications of radionuclides.

Weapon Tests

About 450 atmospheric and 1000 underground nuclear explosions had occurred between 1945 and 1982. More than 300 underground explosions had occurred later. Radionuclides released in atmospheric explosions may reach human either directly through inhalation or indirectly from deposits on earth's surface. The radionuclides deposited on the surface cause exposure to man either externally or internally through food chain. Most of the atmospheric explosions lead to worldwide contamination of long-lived fission products by stratosphere fallout. Some of the radionuclides of importance released through fallout are ^{3}H, ^{14}C, ^{131}I, ^{55}Fe, ^{89}Sr, ^{90}Sr, ^{106}Ru, ^{136}Cs, ^{137}Cs, ^{140}Ba, ^{144}Ce and transuranics including plutonium isotopes. The estimates of radioactivity released into the atmosphere from nuclear weapon tests conducted upto 1980 and the resultant effective dose commitment to the world population [17] is presented in Table 9. The radionuclide that contributes the highest percentage (70%) of the total dose commitment is ^{14}C. Other radionuclides such as ^{137}Cs accounts for 13%, ^{90}Sr for 3 % and ^{95}Zr for 2% of the dose commitment. This commitment will be delivered over thousands of years to the world population. Since most of the atmospheric nuclear weapon tests have occurred in the northern hemisphere, the dose commitment is slightly higher in this hemisphere than in the southern hemisphere.

The other radionuclides in Table 9 include plutonium isotopes and their contribution to the total dose commitment accounts for 1% only. However, ^{95}Nb in this group alone accounts for another 1% to the total effective dose commitment. The annual per capita absorbed dose from these fallout radionuclides is estimated approximately as 45 μSv

Table 9. Estimates of radioactivity released into the atmosphere and the resultant effective dose commitment to the world population from various weapon tests

Nuclide	Estimated release (EBq)	Effective dose commitment (μSv)	% contribution
^{3}H	240	47	1.3
^{14}C	0.22	2580	70
^{54}Mn	5.2	57	1.6
^{90}Sr	0.604	110	3.0
^{95}Zr	143	87	2.4
^{106}Ru	11.8	69	1.9
^{131}I	651	51	1.4
^{137}Cs	0.91	470	12.8
Others	1464	216	5.6
Total	2517	3687	100

Table 10. Measured concentrations of fallout radionuclides in India

Matrix	Concentration of fallout radionuclides		
	^{90}Sr	^{137}Cs	$^{239,240}Pu$
Air	191.0 - 140.0 pBq/L	10.0 - 30.0 nBq/L	2.1 - 2.3 pBq/L
Drinking water	1.1 mB1/L	< 2.0 mBq/L	39.2 µBq/L
Diet	40.2 - 44.8 mBq	7.0 - 20.0 mBq	0.17 - 0.18 mBq

based on 70 years lifetime commitment to a person. For fallout radionuclides, the contribution from ingestion route is about 4 times higher than that of external exposure while the external exposure itself is about 5 times greater than the doses that arise from inhalation of these radionuclides. The fallout of these radionuclides is observed in India and the concentration of a few fallout radionuclides in different environmental matrices measured in India are given in Table 10.

Nuclear Power Production Operations

The sources of radiation exposure from nuclear power production should include nuclear fuel cycle and this consists of mining and milling of uranium, fuel fabrication, nuclear reactor operation, fuel reprocessing and radioactive waste storage and management. At each stage of the nuclear fuel cycle, radionuclides are released to the environment in minor quantities resulting in radiation exposure to man. The important radionuclides released to the environment from uranium milling along with their release rates [17] are given in Table 11.

Table 11 shows that radon in airborne stream and ^{238}U in liquid stream contribute to the major fraction of radionuclides released during this operation. The major radionuclides being released to the environment due to reactor operations are noble gases, ^{3}H, ^{14}C, ^{131}I and particulates containing ^{90}Sr and ^{137}Cs. The normalized worldwide release of radionuclides from nuclear reactors during 1985 to 1989 [17] are presented in Table 12.

Table 11. Normalized release rates of radionuclides from uranium milling operation

Release rate (GBq/GWa)			
Nuclide	Airborne release	Nuclide	Liquid release
^{222}Rn	3.00×10^{3}	^{210}Pb	0.01
^{226}Ra	0.02	^{226}Ra	0.02
^{230}Th	0.02	^{230}Th	0.01
^{238}U	0.40	^{238}U	0.30
^{210}Pb	0.02		

Table 12. Normalized release rates of radionuclides from nuclear reactors

Nuclide	Release from PWR (TBq/GWa)	Release from HWR (TBq/GWa)
Noble gases	8.1×10^{1}	1.9×10^{2}
^{3}H (Gas)	2.8	4.8×10^{2}
^{14}C	1.2×10^{-1}	4.8
^{131}I	9.0×10^{-4}	2.0×10^{-4}
Particulate	2.0×10^{-3}	2.0×10^{-4}
^{3}H (Liquid)	2.5×10^{1}	3.7×10^{2}
^{90}Sr and ^{137}Cs (Liquid)	4.5×10^{-2}	3.0×10^{-2}

Tritium releases from heavy water reactors are high since they use heavy water as moderator and primary heat transport system. The normalized worldwide releases of radionuclides from fuel reprocessing plants during 1985 to 1989 [17] are presented in Table 13.

It can be seen that in the airborne stream, the release of ^{85}Kr is the highest followed by ^{3}H. In the liquid stream, ^{3}H and ^{106}Ru have higher release rates.

The radiation dose to members of the public from the nuclear fuel cycle could arise from external

Table 13. Normalized release rates of radionuclides from fuel reprocessing plants

Nuclide	Airborne release (TBq/GWa)	Nuclide	Liquid release (TBq/GWa)
^{85}Kr	1.23×10^{4}	^{3}H	6.43×10^{2}
^{3}H	4.10×10^{1}	^{106}Ru	3.90×10^{1}
^{14}C	2.00	^{137}Cs	1.30×10^{1}
^{129}I	6.00×10^{-3}	^{90}Sr	1.10×10^{1}
^{137}Cs	2.00×10^{-3}	^{14}C	5.40×10^{-1}
^{131}I	7.00×10^{-4}	^{129}I	4.00×10^{-2}

exposure due to atmospheric discharges of fission product noble gases and ^{41}Ar; and from internal exposure due to inhalation and ingestion of radionuclides such as ^{3}H, ^{131}I, ^{90}Sr, ^{137}Cs, radon-thoron and their progeny. The overall impact of nuclear power generation on total population exposure is very small. This is evident from the annual per caput dose delivered to a member of the public (Table 14) due to releases (1995-1997) from Indian nuclear power plants [18, 19].

Table 14 shows that the total per caput dose to members of Indian population is about 1.5×10^{-2} μSv/y which is significantly lower compared to the dose from natural sources of radiation. The annual per caput dose to the population in India from other stages of the nuclear fuel cycle is estimated as 2.6×10^{-2} μSv (50.0%) from mining and milling operation; 1.4×10^{-3} μSv (2.7%) from fuel fabrication; 3.3×10^{-3} μSv (6.3%) from fuel reprocessing; 1.4×10^{-3} μSv (2.7 %) from radioactive waste management practice and 4.8×10^{-3} μSv (9.3%) from other related activities [20]. The total per caput dose from the complete nuclear fuel cycle works out to be about 5.1×10^{-2} μSv/y of which 29.2% is due to nuclear power plants operation. It is estimated that the worldwide collective effective dose from nuclear power plant operation (4.0×10^{5} man Sv) is 2 orders of magnitude less than that from atmospheric weapon testing and this will generate a per capita effective dose of 0.1 μSv to world population. This estimate represents only 5.0×10^{-3}

Table 14. Annual per caput dose due to releases form Indian nuclear power plants

Airborne releases		Liquid release	
Reactor / Nuclide	Per caput dose (µSv/y)	Reactor / Nuclide	Per caput dose (µSv/y)
TAPS			
Noble gases Iodine, particulate	1.6×10^{-4} 1.3×10^{-5}	Gross beta	8.2×10^{-5}
RAPS			
^{41}Ar ^{3}H ^{14}C	4.5×10^{-3} 2.4×10^{-3} 5.6×10^{-4}	^{3}H Gross beta	7.9×10^{-5} 8.8×10^{-7}
MAPS			
^{41}Ar ^{3}H ^{14}C	1.5×10^{-4} 1.9×10^{-3} 3.1×10^{-4}	^{3}H Gross beta	3.7×10^{-7} 1.3×10^{-7}
NAPS			
Noble gases ^{3}H	4.0×10^{-5} 3.3×10^{-3}	^{3}H Gross beta	5.4×10^{-4} 5.0×10^{-6}
KAPS			
Noble gases ^{3}H	7.4×10^{-6} 7.4×10^{-4}	^{3}H Gross beta	6.0×10^{-4} 2.9×10^{-5}
Total	1.4×10^{-2}		1.3×10^{-3}

percent of the average exposure to natural sources of radiation. The global average percentage contribution of each stage of nuclear fuel cycle to the total annual effective dose of 0.1 µSv are presented in Table 15.

Thus, it is evident that the contribution from mining-milling and nuclear power plant releases are almost equal to the global average per caput dose from nuclear fuel cycle. In India, the contribution from the former is almost double of that from the later.

Other Man-made Sources

Other man-made sources include industrial application of radionuclides and medical exposure. The radiation dose due to the former is negligible (0.65 µSv/y) compared to medical exposure and this exposure is predominantly due to television viewing

Table 15. Global average contribution of each stage of nuclear fuel cycle to the annual per caput dose of 0.1 µSv

Stage	Percentage contribution
Mining	34.35
Milling	2.86
Mine and mill tailings	14.31
Fuel fabrication	0.07
Reactor releases-airborne	37.22
Reactor releases-liquid	1.15
Fuel reprocessing	7.16
Transportation	2.86

[21]. Among the sources of radiation, medical exposure stands next to natural radiation sources. The annual frequency of X-ray examination at present in the country is about 100 exposures per 1000 persons and the total dose from these 100 exposures of all types works out to be about 0.02 Sv [22]. This will result in the per caput dose of 21 μSv/y to the Indian population. Similarly the per caput annual dose from nuclear medicine diagnostic and treatment procedures is about 27 μSv/y [23]. Thus, the total per caput annual dose from medical exposure to the Indian population is estimated as 48 μSv/y. The worldwide average of medical exposure ranges between 400-1000 μSv/y.

Conclusions

The major sources of radiation exposure to the population are natural background and medical exposure. In India, the total annual effective dose from natural and man-made sources of radiation works out to be 2819 μSv. Out of this, virgin natural radiation sources contribute an annual effective dose of 2724 μSv (96.6%); modified natural sources 1.24 μSv (0.04%); atmospheric nuclear weapon tests 45.0 μSv (1.6%); nuclear fuel cycle 0.05 μSv (1.77×10^{-3} %); medical exposure 48.0 μSv (1.7%) and industrial applications 0.65 μSv (2.31×10^{-2} %). Thus, the contribution from man-made radiation sources to the annual effective dose is about 3.3% only. The worldwide average annual effective dose distribution also shows the same pattern. However the dose from medical exposure is about 10 to 50 times higher than that in India because of highly advanced nuclear medical treatment existing in many developed countries.

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Basic Criteria for Design and Operation of a Radiation Facility

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Introduction

Health effects of exposure to ionizing radiation is the important consideration in design and operation of a radiation facility. it is essential that the radiation workers, members of the public and the eco-system as a whole are protected from the possible harmful effects of exposure to ionizing radiation arising from the operation of the facility. The health risks associated with the radiation exposure can only be restricted and not eliminated completely because radiation and radioactive substances are natural features of our environment. The natural background radiation level prevailing in the country vary from place to place depending on the radioactivity content of the soil and the altitude of the place.

The radiation sources are handled with utmost care. The level of protection of and safety existing in the nuclear industry is not only comparable to other safe industries, but also serves as model for other large industrial undertakings. This article describes in brief the special features incorporated in the design and operation of radiation facility to ensure protection and safety of the workers, the public and the environment.

Radiation Effects

When ionizing radiation passes through living tissues, positive ions and negative ions are produced, which eventually recombine resulting in formation of highly reactive chemical species, called free radicals. The free radicals can induce chemical changes leading to biological effects which can be detrimental to health. Such effects depend on the type of radiation, its specific activity and the total amount of the energy deposited per unit mass of the tissue, called radiation dose. The effect on the cells may be mutation or cell death depending on the dose. If the mutated cell happens to be a normal cell in the body and if the cell survives the repair mechanism inherent in the body, the mutagenic disturbance can lead to loss of control of cell activity leading finally to malignant growth. If the mutated cell is a germ cell, the mutated genetic information may be passed on to the descendents as genetic effects.

It must be mentioned here that the mutations are not specific to ionizing radiation. Host of other agents found in our environment such as certain chemicals, asbestos, etc. are also capable of causing the mutation in the cells. It is also not that every mutated cell would lead to malignancy or genetic disorder. The chance or the probability of causing the

Table 1. A typical arrangement of radionuclides based on limiting Derived Air Concentration based on recent ICRP-68 data (1994)

Up to 5 Bq/m^3	^{231}Pa, ^{228}Th, ^{230}Th, ^{232}Th, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, ^{242}Cm, ^{238}U(M&S), ^{235}U, ^{235}U, ^{234}U, ^{228}Ra, ^{224}Ra(M), ^{147}Sm(M), ^{252}Cf(M), ^{210}Po(M)
6 – 10 Bq/m^3	^{241}Pu(M), ^{238}U(F), ^{235}U(F), ^{234}U(F), ^{210}Pb, ^{210}Po(F)
11 – 100 Bq/m^3	^{225}Ra(M), ^{241}Pu(S)
101 – 500 Bq/m^3	^{144}Ce, ^{131}I (Vapour), ^{106}Ru (M,S), ^{90}Sr, ^{60}Co (S), ^{152}Eu (M)
501 – 1000 Bq/m^3	^{134}Cs, ^{137}Cs, ^{131}I(F), ^{125}I(Vapour), ^{106}Ru(F), ^{89}Sr(S), ^{60}Co(M)
1000 – 6000 Bq/m^3	^{125}Sb, ^{95}Zr, ^{59}Fe, ^{32}P(M), ^{99}Tc(M), ^{95}Nb, ^{89}Sr(F), ^{237}U(S), ^{237}U(M), ^{234}Th(M,S), ^{203}Hg(M) Inorg, ^{192}Ir, ^{58}Co(M,S), ^{99}Mo(S), ^{90}Y(M,S), ^{65}Zn(S)
6001 and above Bq/m^3	^{239}Np, ^{125}I(F), ^{85}Sr, ^{55}Fe, ^{32}P(F), ^{14}C, ^{3}H, ^{35}S, ^{24}Na, ^{82}Br(F,M), ^{99}Mo(F), ^{99}Tc(F), ^{237}U(F), ^{234}Pa(M,S), ^{203}Hg(F, Org.), ^{203}Hg(F, Inorg.), ^{51}Cr, ^{99m}Tc

Note: F, M, S are the ICRP classification of inhaled compounds based on absorption by the body fluids (F-Fast; M-Moderate and S-Slowly absorbed from the respiratory tract)

above effects is statistical in nature and hence they are called "stochastic effects". In the other type of effect, where the affected cells are extensively damaged, the effects may appear within a few hours to a few weeks after exposure to ionizing radiation. The effect has a threshold and it manifests only if the radiation dose received is above the minimum level. Such effects are called "deterministic effects", and cataract of the eye is an example of such effect.

For the purpose of radiation protection, the stochastic effects are considered as cumulative without any threshold, and the probability of the effect is assumed to vary linearly with the dose.

Radiological Toxicity

Radiological toxicity of radionuclides depends upon the type and energy of the radiation, specific activity and metabolic behaviour, particularly the sites of deposition in the body after its intake. Alpha emitting radionuclides are radiologically more hazardous as compared to beta gamma emitters. Radionuclides of high specific activity and bone seeking radionuclides such as plutonium and strontium-90 are considered as highly toxic. Derived air concentration (DAC) for occupational workers are based on such considerations. Table 1 shows some of the radionuclides listed in the order of limiting DAC values. The DAC values in Bq/m^3 were derived by dividing annual limit on intake (ALI) by volume of air breathed in a working year i.e. 2,400 m^3. The ALI, in Bq, can be obtained by dividing annual dose limit by the committed effective dose coefficient given in ICRP-68. Radionuclides with low DAC values are considered more toxic and hence need safe handling facilities.

Modes of Radiation Exposure in Work Place

In work places, workers may be exposed to (i) external exposure from sources external to the body. Exposure from an industrial radiography source or exposure from a chest x-ray examination are the examples of external exposure, and (ii) internal exposure from radioactive materials, natural and man-made, incorporated within the body. Exposure from ^{40}K naturally present in the body, and exposure from occupational intakes by the workers constitute internal exposure. The intakes may be by inhalation of contaminated air, or ingestion of contaminated food or water. Intakes may also result through wound and absorption by skin.

Table 2. Typical Classification of areas

Area	Access Control	Typical examples and requirements
White	Unrestricted	Office rooms, where there is no radioactive source and there is no radioactive contamination.
Green	Access generally restricted to radiation workers. Change of clothing not necessary.	Heath physics room, control room, etc. Potential for radioactive contamination being adjacent to radioactive areas, Small sealed sources may be present.
Amber	Controlled areas, Access limited to radiation workers, Entry through barrier, change of clothing at the barrier.	Laboratory areas where radioactivity is handled. A certain level of surface contamination may be present. A good ventilation system is necessary. Personal dosimeter provided.
Red	Access is only through Special Work Permit (SWP).	Inside fume hoods, glove boxes and hot-cells, areas where open radioactive sources may be handled. External radiation exposure levels and or radioactive contamination levels could be high.

Design Criteria for Exposure Control

Design Considerations

The hazards peculiar to the handling of radioactive materials necessitate special features of design and construction, which are not required for conventional work areas. The general design of the plant and equipment should be such that normal operations should not involve any personnel exposures, and the radioactivity is safely contained during any foreseen accidents.

Segregation of Areas

Access to the areas in which exposure to radiation and radionuclides could occur is strictly controlled through proper design of the facility. Areas are segregated based on the potential for external exposure, and spread of radioactive contamination from active areas to other low-active and inactive areas. The working areas are classified as "white", "green", "amber" and "red" areas. A typical classification is described in Table 2. A barrier, either over-shoe cover type or change of foot-wear type, is provided at the entry point to the amber areas. Additional localised barrier may be erected as and when necessary to control spread of contamination. A separate change room, closer to the entry point to the amber areas, is provided for the change of clothing. A shower and decontamination facility is also made available for personnel contamination.

The amber areas are further classified into A,B and C type depending upon the radiological toxicity and the quantity of the radionuclides handled.

Type A laboratory	$\geq$ 10 mCi
Type B laboratory	> 10 μCi but < 10 mCi
Type C laboratory	$\leq$ 10 μCi

Radioactivity handling facilities are designed to reduce personnel exposures and to provide containment to the activity handled.

Handling Facilities

Fume hoods, glove boxes and hot cells are provided in the amber areas for the purpose of handling open radioactive materials. A considerable degree of built-in safety is provided in these facilities. Generally, for ease of decontamination the surfaces of these facilities have smooth finish.

Fume Hood

Simple chemical operations may be carried out in fume hoods. A fume hood is an enclosure in which the radioactive material is handled. Containment of material is achieved by air suction through the front sliding panel opening of the fume hood. The minimum face velocity maintained through the opening of the fume hood is 0.5 m/s (100 ft/min.). It should be ensured that the front panel is maintained with a opening of height not more than 0.5m. Activity in milliCurie levels is handled in the fume hood. Services provided in fume hoods are usually water, gas, compressed air, vacuum and electricity.

Glove Box

A glove box is a leak-tight enclosure in which open radioactive sources, mainly alphas, can be handled in isolation from the user's environment. The work is carried out through gauntlet fixed to open ports in the generally transparent walls of the box. The exhaust from the box is adjusted to provide about 1" W.G. negative pressure inside the glove box. In an enclosure, at the inside top of the box, absolute filter is provided for the exhaust air to pass through. Port holes with air-lock is provided for material transfer. Inert atmosphere inside the box may be provided while handling pyrophoric radioactive material, such as plutonium. The normal leakage rate of 0.5% of the box volume per hour provides about 4-5 air changes per hour in the glove box. Services such as electricity, vacuum, compressed air, effluent drainage, etc. are provided inside the box.

Maintenance of adequate negative pressure, and periodic checking of the gloves for pin-hole leaks, are important safety considerations in glove box operations.

Hot-cells

A hotcell is a shielded enclosure fitted with suitable system of manipulation to allow the performance of operations without subjecting the operators to radiation exposure beyond the acceptable dose rates. Kilo Curie levels of high intensity radioactive sources, such as ^{60}Co, and ^{192}Ir can be handled in the hot-cells.

The main construction material for hot cells is concrete (density ~2.4 g/cm^3) which is relatively cheap and density can be varied to suit requirements. The working surface inside the cells is usually stainless steel. Various services such as water, steam, chemicals, viewing window and even welding units are provided inside the hot cells for decontamination and maintenance. Master slave manipulators are used for the remote operations. Doors made of steel are provided for access to the cells for decontamination and maintenance. Special materials such as transparent lead glass windows (density ~ 5.2 g/cm^3), provide unrestricted view to the operators.

Sufficient ventilation (~40-50 air changes per hour), with absolute filters at the exhaust end is provided before the exhaust air is released into the environment. Penetrations of the service lines on the walls are so designed as to prevent any streaming or leakage of radiation to the work areas outside the cell. The cells are kept normally under negative pressure in the order of 10 to 20 mm with respect to the adjoining operating areas.

Based on the experiences of the operation being performed, the form of the material and the type of the facility used for handling radioactive materials, modifying factors (Ref. 3) are used to arrive at the amount of activity that can be handled under normal situations.

Ventilation

Ventilation in a radiation installation should aim at providing comfortable working condition and a continuous air change to ensure the dilution and removal of airborne contaminants. The special features of ventilation in a radiation installation are:

(i) The general space ventilation of the facility should be on once-through basis. A slight negative pressure (-0.02 to -0.08 inches W.G.) is maintained in the active areas.

(ii) Air intake and exhaust points should be well separated to avoid recirculation of contaminated air.

(iii) The flow of air in the working areas is directed in such a way that air flows from non-active or low active areas into active/high active areas.

(iv) The exhaust of general space ventilation is through the fume hood openings.

(v) The fans for air removal should be located on the exhaust side of filters. Stand-by fans are provided.

(vi) Absolute filters are used for particulate removal from the exhaust air. The efficiency of the filters is better than 99.97 percent for 0.3 μm sized particles. Activated charcoal based

Table 3. Doses per unit intake via inhalation (Sv/Bq), and the Derived Annual Limit on Intakes for selected radionuclides (ICRP-68)

Radionuclide	Inhalation Type	Dose per unit intake (5 μm AMAD) Sv/Bq	Derived ALI (ICRP-68) Bq	ALI (ICRP-61) Bq
^{3}H (Tritiated Water vapour)				1×10^{9}
^{60}Co	D(F) Y(S)	7.1×10^{-9} 1.7×10^{-8}	2.8×10^{6} 1.2×10^{6}	2×10^{6} 4×10^{5}
^{90}Sr	D(F) Y(S)	3.0×10^{-8} 7.7×10^{-8}	6.7×10^{5} 2.6×10^{5}	6×10^{5} 6×10^{4}
^{131}I	D(F)	1.1×10^{-8}	1.8×10^{6}	1×10^{6}
^{137}Cs	D(F)	6.7×10^{-9}	3.0×10^{6}	2×10^{6}
^{232}Th	W(M) Y(S)	2.9×10^{-5} 1.2×10^{-5}	690 1660	90 90
^{238}U	D(F) W(M) Y(S)	5.8×10^{-7} 1.6×10^{-6} 5.7×10^{-6}	3.5×10^{4} 1.2×10^{4} 3.5×10^{3}	9×10^{4} 1×10^{4} 600
^{239}Pu	W(M) Y(S)	3.2×10^{-5} 8.3×10^{-6}	625 2400	300 300
^{241}Am	W(M)	2.7×10^{-5}	740	300

Notes : D, W & Y are the types classified on the basis of overall clearance from respiratory track
ICRP-61 values are for workers (AMAD assumed = 1 μm)
ALI values are based on 20 mSv/y effective dose

filters are used to remove iodine vapours and other radioactive gases.

Monitoring

Radiation monitoring forms an important part of radiation safety surveillance. It helps to take steps to control external and internal radiation exposures of the working personnel.

Workplace Monitoring

Working place is monitored regularly for external radiation, for surface contamination and for airborne contamination. Inhalation hazard in the workplaces is evaluated by air sampling technique. Air is sucked through a glassfibre filter paper at the rate of about 40 litres per minute.

Air Monitoring

The particulate radioactivity gets accumulated on the filter, which is counted for alpha and beta gamma activity using alpha counting system (ZnS(Ag) detector), and GM detector, respectively. Knowing the efficiency of the systems, and the quantity of air sampled, air concentration can be calculated. The DAC values, which are currently in use, are given in Table 3 for a few selected radionuclides. Table also shows the most recent DAC values, calculated using the dose coefficient tables given in the ICRP-Publication 68 (1993), for an annual dose limit of 20 mSv. The recent values are less stringent as compared to the ICRP-61 (1991) values.

If a worker has to work in a contaminated atmosphere having activity concentration higher than the DAC values, he is required to wear a suitable

Table 4. Protection factors (PF) for respirators

Sr.No.	Type of respirator	PF
1.	Particulate removing half-face piece	10
2.	Particulate removing full face-piece	50
3.	Gas and vapour removing half-face piece	10
4.	Gas and vapour removing full face-piece	50
5.	Self contained breathing apparatus (SCBA), full-face piece (Pressure inside mask : Negative)	50
6.	Self contained breathing apparatus (SCBA), full-face mask (Pressure inside mask : Positive)	10,000
7.	Supplied air, half-face piece	1000
8.	Supplied air, full-face piece	2000

respirator which affords him maximum respiratory protection. Table 4 shows the protection factors for some types of respirators.

Individual Monitoring

External exposures of the radiation workers are monitored routinely (on monthly or quarterly basis) for all types of radiations handled using appropriate dosimeters, viz., thermoluminescent dosimeters (TLDs), pocket dosimeters, neutron badges, etc.

All the radiation workers after their work monitor the hands, clothing and shoes using appropriate instruments. This is to make sure that no loose contamination is carried outside the controlled areas. The derived values for fixed alpha and beta contamination are given in Table 5.

Any internal contamination or intakes by the workers can be measured by bioassay (analysis of excretion samples) or by whole body counting. Whenever such direct measurement is not feasible assessment of internal exposure is possible using air

Table 5. Derived Working Levels (DWLs) for Radioactive Contamination
(AERB Manual, 1996)

Surface	Beta emitters Bq/cm^2	Alpha emitters Bq/cm^2
Skin	1.5	1.0
Hands	350	250*
Cloths		
Plant	6	2
Personal	2	0.5
Shoes		
Plant	37	3.7
Personal	0.37	0.037
Floor	3.7	0.37

*Total contamination

monitoring data, and the occupancy factor of the worker.

External Exposure/Contamination

The dosimeter is worn by the radiation worker during any work involving ionizing radiation or when he enters area where the radiation emitting sources are located or operated. The cumulative exposure over a specified time period can thus be determined.

External contamination of workers occurs due to deposition of radioactive material on the skin. The skin contamination and the associated radiation dose is one of the risks involved in handling radioactive material. The extent and level of the contamination on the skin can be measured by using suitable monitoring equipment. Dose rates and doses received by the skin due to the presence of 1 kBq/cm^2 of radioactivity on the surface of the skin is given in Table 6.

Internal Exposure/contamination: Internal contamination results when when any radioactive material enters the body either through inhalation, ingestion or injection route. The skin may also absorb it. Tritiated water is an example of a radionuclide, which is absorbed by the skin. Radionuclide undergoes metabolism in the body depending on its chemical and physical properties.

Table 6. Residual skin contamination dose rates and dose received by the based layer of the skin (IAEA TECDOC-869, 1996)

Radionlclide	Dose rate mSv h^{-1} per kBq/cm^2	Dose* mSv per kBq/cm^2
^{14}C	3.2×10^{-1}	1.7×10^{2}
^{32}P	1.9	4.8×10^{2}
^{35}S	3.5×10^{-1}	1.5×10^{2}
^{54}Mn	6.2×10^{-2}	3.1×10^{1}
^{59}Fe	9.6×10^{-1}	3.7×10^{2}
^{60}Co	7.8×10^{-1}	4.0×10^{2}
^{63}Ni	6.5×10^{-2}	3.4×10^{-4}
^{90}Sr	1.6	8.3×10^{2}
^{99m}Tc	2.5×10^{-2}	2.1
^{125}I	2.1×10^{-2}	8.8
^{131}I	1.6	2.9×10^{2}
^{137}Cs	1.6	8.2×10^{2}

*Dose corresponds to self-cleansing of the skin.

The internally deposited, radioactive material continue to irradiate the tissues until it decays to stable isotope or excreted by some physiological process. The extent of internal contamination can be assessed by (i) measurement of radioactivity in the excretion, i.e. urine and feces and/or (ii) in-vivo measurement of penetrating radiation by whole body counter, or any other suitable instrument. For example, hand held scintillation detector can be used to measure radioactive iodine in thyroid.

Waste Management

Radioactive wastes generated are not to be disposed off into the environments directly. Regulatory system is applicable to such disposals if the radioactivity content of the waste exceeds 70 Bq/g. The various categories of wastes and the waste management aspects are discussed in detail elsewhere (INACAS Bulletin, 1997).

A system of waste collection and its transfer for storage or disposal is worked out. Solid wastes are segregated at the collection point as flammable/compressible or non-flammable/non-compressible type. Solid wastes are properly packed in PVC bags, monitored for radiation dose rate and tagged. Shielded waste storage facility is provided for wastes awaiting disposal after monitoring.

Small quantities of low-level liquid wastes can be directly discharged into the sink which drains to liquid waste holdup tanks which after monitoring, is sent to centralised waste treatment facility. Liquid wastes with activity level greater than 10^{-4} µCi/ml (α) and 10^{-3} µCi/ml (β) are collected in polyethylene carbuoys. Radioactive oganic waste is collected in glass bottles. After monitoring and tagging these containers are sent for disposal to waste management facility.

Emergency Planning and Preparedness

Comprehensive emergency plans are prepared for a possible emergency or emergency-like situations in radioactive facility. Guidance levels, called Intervention Levels for immediate protective actions in case of an emergency are given in the International Basic Safety Standards of IAEA. Action levels in specified supplies of food and drinking water are also specified in the BSS and should be included in the emergency plans. Typical intervention levels and generic action levels for food stuffs are given in Table 7 and 8. The plans give in details the actions required to be taken by various agencies in the facility.

Periodic drill helps in making the concerned agencies understand the responsibilities during an emergency.

General considerations

Survey and Monitoring Instruments

(a) Radiation survey instruments, with appropriate measurement range, are used for radiation monitoring of areas for controlling external exposures.

(b) Area monitors are strategically located in areas to indicate the status of radiation background. Alarm may be set a predetermined level.

Table 7. Intervention levels for wholebody and thyroid exposures and the countermeasures (AERB, 1996)

Countermeasure	Range of Intervention Levels (mSv)		
	Domain 1	Domain 2	Domain 3
Control on food stuff	-	5 - 20	1 - 5
Administration of stable iodne (Thyroid Exposure)	500 - 2500	50 - 500	Not anticipated
Sheltering	20 - 100	5-20	-
Evacuation	100 - 500	-	-

Note: Domains 1,2,& 3 are off-site areas/regions affected by the accidental radioactive releases; within each domain, radiological characteristics, severity are nearly the same.

Table 8. Generic Action Levels for Foodstuffs (BSS-SS-115, 1996)

Radionuclide	Foods distined for General consumptionn (kBq/kg)	Milk, infant food and drinking water (kBq/kg)
^{134}Cs, ^{137}Cs, ^{103}Ru, ^{106}Ru, ^{89}Sr	1	1
^{131}I	-	0.1
^{90}Sr	0.1	-
^{241}Am, ^{234}Pu, ^{239}Pu	0.01	0.001

(c) Hand monitors (alpha and beta) are provided at the barriers to monitor hands after the radioactive work.

(d) Alpha and beta contamination monitors are provided to measure the level of loose or fixed contamination in work places.

(e) Continuous air monitor system for alpha and beta activity is desirable in work areas to indicate the airborne activity level on continuous basis during working hours.

Decontamination

Generally, skin on the hands is the target tissue for contamination while carrying out radioactive work. Decontamination is the removal of radionuclide to reduce the dose. Care should be taken to avoid injury on the skin during the decontamination process. Soaps and detergents can emulsify and dissolve contamination and are adequate for the external decontamination of the skin. Water should be used for decontamination. Chelating agents such as EDTA (10%) or DTPA (1%) in aqueous solution can be used for skin contamination by transuranics, lanthanides or metals such as cobalt, iron, zinc and manganese. Additional chemical techniques, such as use of oxidizing solution of $KMnO_4$ and subsequent application of sodium bisulphite to neutralise the permanganate, should be used with utmost care and expert advice. Any residual fixed contamination on the skin may be allowed to be reduced by the self-cleansing of the skin within a matter of days.

Internal decontamination of radionuclides is more complex and should be attempted under expert/medical guidance. Uranium contamination is removed preferentially by intravenous administration of 1.4% bicarbonate solution in saline. Similarly, Ca or Zn–DTPA is administered intravenously for decorporation of materials such as thorium, plutonium, americium, etc. The decorporation is enhanced if it is administered immediately after the intake.

Dose Records

The dose records of all the occupational workers should be maintained by the management of the facility for the duration given elsewhere in this issue.

Temporary Workers

Temporary workers should be employed after giving them adequate training in radiation protection work methods and emergency procedures in a given facility. For temporary workers employed by contractor and those employed by installation, the dose received should not exceed 15 mSv in a year. Management shall maintain record of the dose received by the workers. (AERB Safety Manual, Rev. 3, 1996).

Good House Keeping and Safety Culture

A good house keeping helps in control of contamination levels in work places to a great extent. Training and education of the work with respect to Industrial Hygiene and Safety are considered as essential elements in overall safety in any given facility. Development of Safety Culture at all levels of organisation, such as understanding the individual and collective commitment to safety, avoiding complacency with respect to safety, etc. should be promoted to maintain an environment of safety in radiation facility.

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Radiation Protection and Safety Aspects in the Use of Radiation in Medicine, Industry and Research

Dr. B.C. Bhatt joined BARC in the year 1969 after graduating from the 12th batch of BARC Training School. He has about 30 years of experience in the field of thermoluminescence dosimetry, personal monitoring, medical dosimetry and radiation safety. He has participated in a number of IAEA expert meetings and international scientific meetings. He is also a member of various DST and AERB committees dealing with radiological testing and safety. Presently, he is Head, Radiological Physics and Advisory Division which is responsible for providing radiation safety related advisory services in the country for medical, industrial and research institution for handling radiation sources. He has more than 150 publications to his credit. He has served as convener as well as Course Director for IAEA courses on Implementation of Basic Safety Standards and Radiation Protection & Safety in Medicine held at BARC, Mumbai. He is a guide for M.Sc. and Ph.D. in Physics for the University of Mumbai.

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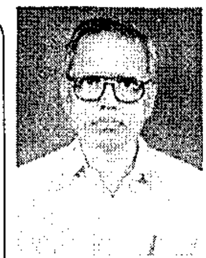

Introduction

The use of radiation sources of various types and activities is widespread and increasing steadily in medicine, industry and research, all over the world. Nearly 90% of the man-made component of radiation is due to use of diagnostic X rays. Radiation therapy ranks with surgery and chemotherapy as one of the most powerful and effective techniques in the treatment of cancer. Recent reports suggest that more than 60% of cancer cases can be treated by radiation. Nuclear medicine has become an established practice during the last three and a half decades and is increasingly employed for the detection and treatment of diseases and evaluation of physiological and pathological behavior not detected by any other way.

Among the peaceful uses of atomic energy, industrial applications of radiation sources form a major branch. Non-destructive testing of vital components by industrial radiography is widely used in industries. Nucleonic control systems, popularly known as nucleonic gauges, are used in several industries for measurement and control of process parameters such as thickness, density, level and composition. High intensity irradiators are used in radiation processing, food preservation, sterilisation of medical products etc. Small quantities of radionuclides are used as tracers in flow and leakage measurements, physico-chemical and industrial research and for the study of reaction pathways in biological processes.

While ionizing radiations have significant and indispensable uses in several fields, it must be borne in mind that it may be harmful to the radiation workers and public if used indiscriminately and without due caution. It is therefore necessary to ensure safety of radiation workers, patients

Table 1. Medical facilities in India

A.	**Teletherapy Facilities**	
	Number of Teletherapy Centres	155
	Number of Teletherapy Units	250
	Cs-137	11
	Co-60	201
	Linear Accelerators	34
	X-knife	1
	Gamma knife	3
B.	**Brachytherapy Facilities**	
	Brachytherapy Centres	102
	Remote Afterloading applicators	52
	Manual Afterloading applications	47
	Centres using Ir-192	22
	Centres using discrete Sources	35
C.	**Nuclear Medicine Laboratories**	125
D.	**RIA Laboratories**	500

undergoing radiation diagnosis and treatment, public and environment so that maximum benefit is derived from the use of radiation with minimum and acceptable risk.

Overview of Applications

Medical Applications

Medical applications of radiation differ from other uses of radiation because it involves patient exposure. Diagnostic radiology using conventional X-ray units is in vogue ever since the discovery of X-rays. There are several tens of thousands of X-ray machines in use all over the country. Computerised Tomography (CT) has now become common for precise diagnosis of diseases and disorders.

Teletherapy machines using sealed sources of radioactive material have been in use since 1950's. Most teletherapy machines use ^{60}Co as the source of radiation. During the past 25 years, various models of high energy medical linear accelerators (capable of providing both photon and electron beams) have come to world market. For certain types of cancer, they have become the choice of treatment.

Brachytherapy is radiotherapy in which the radiation sources are virtually in direct contact with tumour either externally or internally, so as to produce carefully defined irradiation at close proximity of the tumour.

Nuclear medicine deals with the diagnostic and therapeutic applications of unsealed radionuclides in clinical practice. In both diagnostic and therapeutic nuclear medicine procedures, radiopharmaceuticals are administered *in-vivo*. Diagnostic applications of Nuclear Medicine include nuclear cardiology, endocrinology, organ imaging, etc. Non-imaging procedures involving *in-vitro* techniques such as radioimmunoassay (RIA) are also widely used.

Therapeutic applications of nuclear medicine include treatment of thyrotoxicosis, polycythemia vera and thyroid cancer. Radioisotopes generally used in nuclear medicine departments are ^{131}I, ^{99m}Tc, ^{32}P, ^{201}Tl and ^{67}Ga. Table 1 gives details of the medical facilities available in the country.

Industrial Applications

Industrial radiography is widely in use for NDT and quality control. For example, the construction of petrochemical installations will involve the use of portable radiographic sources of activity up to 5 TBq for testing of welds in pipes and tanks. Sources will most often be ^{192}Ir or ^{60}Co, but ^{169}Yb or ^{170}Tm may also be used. In heavy industries such as steel foundries or fabrication, portable, mobile or fixed radiographic equipment incorporating ^{192}Ir, or ^{60}Co are installed in specially built enclosures. In oil exploration, mining industry and construction work, neutron and gamma sources are used for determination of density, porosity and moisture or hydrocarbon content of geological structures or building materials. The most common neutron sources employed are ^{241}Am/Be of up to 800 GBq. The gamma sources most frequently employed are ^{137}Cs of 50-100 GBq.

In many industries, it is necessary to measure thickness, density or moisture content of a material. Use of radioactive sources enables non-contact measurements to be made. Many different radionuclides, of a wide range of source strengths, are used. Beta sources are used for measuring thickness of paper, plastic and thin light metals, while gamma sources are used in situations where steel plate is being manufactured or the density of coal, rock or oil well fluids is to be determined.

Table 2. Radiation Sources in Industrial Applications

A.	**Industrial Radiography**	
	Number of radiography institutions	366
	Number of radiography cameras	947
	Ir-192 source	876
	Co-60 source	45
	Tm-170 source	2
	High energy accelerators	9
	Industrial X-ray machines	200
B.	**Nucleonic Control Systems**	
	Number of institutions using nucleonic gauges	830
	Level gauges	3426
	Density and moisture gauges	830
	Thickness gauges	395
	Well logging tools	470
	Betascopes	150
	Others	430
C.	**Unsealed sources used in industry**	
	Radiotracer applications	50
	Gasmantle industries	95
	Dial painting industries	20
	Fluorescent lamp starter industries	55
D.	**Other Industrial Applications**	175

Similar sources are used in other industries to measure the level of material in a vessel or tank, normally using ^{137}Cs. Neutron sources, normally ^{241}Am/Be, are used to measure the moisture content of coke or coal prior to loading furnaces. The electronic industries use small sources to measure the thickness of plating of precious metals on circuit boards or electrical contacts. Where blast furnaces are employed in steel industry, ^{60}Co sources are often used to gauge the wear of the refractory lining of the bottom hearth. Very large (several TBq) ^{60}Co and ^{137}Cs sources are used for radiation sterilization of medical products, such as sutures and gloves, and for food preservation. Radionuclides are added in consumer products to make use of the ionising radiation emitted by them to achieve a particular performance (e.g.radioluminescent devices, ionisation chamber smoke detectors) or to make use of some other property of the material where the presence of radioactivity in the final product is merely adventitious (e.g. incandescent gas mantles). Table 2 gives details of radiation sources in industrial applications in India.

Philosophy of Protection and Safety

The cardinal principle underlying the philosophy of protection and safety is to ensure that there exists an appropriate standard of protection and safety for humans without unduly limiting the benefits of practices giving rise to exposure or incurring disproportionate costs in interventions. To realize these objectives, International Commission on Radiological Protection (ICRP-60) and IAEA's Safety Series (IAEA Safety Series 120, 1996) have enunciated the criteria of (a) Justification of practices; (b) Optimization of protection; (c) Dose limitation and (d) Safety of sources. It is worth mentioning here that the dose limitation principle does not apply to medical exposures.

Justification of Practices

Any practice, involving radiation exposure, shall be justified on the grounds that it produces sufficient benefit to the exposed individual and to the society at large to offset the radiation detriment that it may cause. However, decisions on justification are invariably influenced by the political, social and economic concerns prevalent at that time.

Medical exposures shall be justified by weighing the diagnostic or therapeutic gains expected to be obtained vis-à-vis likely radiation detriment.

Limitation of Doses and Risks

Even if sufficient justification is available for a practice, there is a need to restrict the dose that an individual may incur in order to ensure that no person is subjected to unacceptable levels of risk attributable to radiation from all sources. These limitations do not apply to medical exposures because they are intended to yield a net benefit to the patient. Also, dose limits do not apply directly to potential exposures. Because the same individual may be exposed to more than one source, the dose limit applies to the total dose from exposures to all sources .

Optimization of protection

The exposure from a source within a practice shall be optimized in order that the magnitude of the individual doses, the number of people exposed and the likelihood of incurring exposure be kept As Low As Reasonably Achievable (ALARA), social and economic factors be taken into account, with the further provision that doses delivered to individuals by the source be subject to dose constraints. A dose constraint is the value of an individual dose not to be exceeded in the individual dose distribution considered in the optimization process.

Safety of sources

The principle underlying safety of sources states that all reasonably practical measures shall be taken to enhance operational safety, to prevent radiation accidents and to mitigate their consequences, should they occur. These measures for prevention and mitigation may be identical in some cases to those providing protection during normal operations but in other cases, different or additional measures may be needed. Source safety procedures include consideration of the location of the source, its design and construction to ensure the integrity of the source, and operational procedures and standards to ensure that abnormal situations will not arise.

Facility for locating a source shall be so designed that the exposure levels at occupied areas are within the stipulated dose limits for occupational workers and outside the facility, the levels are consistent with those applicable to the members of public.

General Principles of Radiation Protection

Radiation hazards can be broadly classified under 'external hazards' and 'internal hazards'.

External hazards are those caused when the source of radiation is outside the body, e.g., X-rays and sealed (encapsulated) radioactive sources as in the case of telecobalt, radiography and nucleonic gauges. From external hazards point of view alphas are the least hazardous. Neutrons interact with the biological tissue and produce secondary charged particles or gamma rays. Thus, neutrons from sealed sources also constitute a source of external hazard.

Internal hazards are those that arise either due to the actual entry of radioactive material inside the body or due to the presence of radioactivity in the environment. The main routes of intake of radioactive materials leading to internal hazards are:

(a) inhalation of contaminated air,

(b) ingestion of contaminated food and water,

(c) entry of radionuclide directly into blood stream, e.g., through cuts and wounds on skin.

Internally, alpha radiation is most hazardous followed by beta, X- and gamma radiations.

Protection against External Radiation Hazards

When an external radiation hazard is involved, the three basic principles which may be invoked, either singly or jointly, to protect persons from harmful effects are : minimum time, maximum possible distance and adequate shielding. These principles are adopted while designing and operating a facility/device.

Protection against Internal Radiation Hazards

Protection against internal radiation hazards can be achieved by :

(a) provision of proper containment and confinement of unsealed radioactive sources,

Table 3a. Radiation Safety Training Programmes - Medical/Research Applications

Sl.No.	Program	Duration	Number of Programs	Candidates Trained
1.	Diploma in Radiological Physics (D.R.P.)	One year	36	545
2.	Radiation Safety in Research Application of Ionizing Radiations (RA)	8 days	20	443
3.	Radiation Safety for Radiation Therapy Technicians (RTT)	8 days	22	360
4.	Radiation Safety in Servicing of Radiotherapy Equipment (SRT-1)	6 days	4	66

Table 3b. Radiation Safety Training Programmes - Industrial Applications

Sl.No.	Program	Duration	Number of Programs	Candidates Trained
1.	Certification Course for Industrial Radiographers	10 days	62	2147
2.	Radiography Testing Level 1 (RT-1)	15 days	19	582
3.	Radiography Testing Level 2 (RT-2)	4 weeks	18	522
4.	Operators Course for Food Irradiation Plants	30 days	1	18
5.	Radiation Safety Aspects of Nucleonic Gauges (NG)	7 days	45	1307
6.	Operators Course for Industrial Irradiation Plants	14 days	4	64
7.	Familiarisation Programmes	1 to 3 days	24	590

(b) contamination control by regular contamination monitoring surveys, correct use of protective clothing, immediate decontamination of spillage, and good housekeeping,

(c) prohibition of smoking, eating, drinking, use of cosmetics and mouth pipetting in a potentially contaminated area,

(d) personal contamination monitoring on leaving the potentially contaminated area and immediate personal decontamination as necessary.

With normal care, the radioactive material contained within a sealed source will not leak out to give rise to a radioactive contamination problem. However, it is a good practice to subject each sealed source to a regular leak test to ensure that any leak which develops is detected as early as possible.

Effective Radiation Protection Programme

Personnel Training

Organisation of training concerning protection and safety will enable the staff to conduct their work in accordance with the requirements of the safety standards. Periodic refresher courses / retraining programmes are necessary to update the knowledge of the personnel. Radiation safety training programmes conducted in India are listed in Table 3a and 3b.

Radiation Surveillance

A regular feature of surveillance programme is the drawing up of predetermined schedules, rigid adherence to safety recommendations and reviews of periodic safety status. This includes adequate monitoring of the workplace, appropriate personal dosimetry, complete assessment and analysis of data; implementation and follow up of good work practices. All unusual and abnormal situations are identified and investigated.

Safe Transport of Radioactive Material and Safe Disposal of Radioactive sources

Safe transport of radiation sources is essential to ensure that workers and general public are not at risk. Thousands of radioactive consignments are transported via air and road in India with a good safety record. Administrative procedures, packaging, labelling and contigency plans should conform to prevailing national/international regulations. The packages are transported in Type A or Type B packages. Type A packages are designed to withstand normal conditions of transport whereas Type B packages are designed to withstand accident conditions of transport. The packages are further classified as White, Yellow-II and Yellow-III categories depending on the surface dose rate and transport index (dose rate in mR/h at 1 metre from surface). The class and oategory of package are decided by the activity content, surface dose rate and transport index. Many of the low activity radioactive sources used in nuclear medicine and research institutions and some of the sources used in industries are transported as excepted packages which are exempted from the design requirements prescribed for Type A and Type B packages.

For safe disposal of decayed and redundant sources, institutions are advised to return them to the manufacturer / supplier for disposal. However, low activity radiation sources and waste generated during a practice are disposed off locally by the users with prior authorisation from the regulatory authority. Long lived and relatively higher activity radiation sources / wastes are sent to approved radioactive waste storage site such as the Waste Management facility of BARC, for ultimate disposal.

Administrative Controls

A good radiation protection programme (IAEA Safety Series 102, 1990) depends upon administration of sound principles for the designation of working areas and the control of persons working in them. Comprehensive records are maintained in respect of the use, location, storage, movement and disposal of all radioactive materials used in the premises. Appropriate personal monitoring should be provided and comprehensive records of personnel radiation exposures maintained. Annual and cumulative dose records of all occupational workers are maintained throughout the occupational period and for a further period of twenty years after the cessation of work or till the demise of the worker, whichever is shorter.

Accidents and Emergencies

Though the radiation safety records in these applications has been good, there have been a few incidents/accidents during production, transport and use of radioactive materials.

Accident as defined in the glossary in International Basic Safety Standards (IAEA Safety Series 115, 1996) is : "Any unintended event including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety"

In spite of proper control and adequate precautions to avoid accidents, accidents are likely to occur and it is therefore necessary to foresee accident situations and be prepared to cope with them as early as possible in order to control radiation exposures of workers and public and contamination of environment and restore normalcy.

Foreseeable Accident Types in Applications of Radiation Sources

(i) A radioactive source may be misplaced, lost or stolen. A simple example could be the loss or misplacement of a gamma radiography source container with source. It is possible that the container may get into the hands of ignorant

Table 4. Summary of Incidents/Accidents in Applications of Radiation Sources in Medicine, Industry and Research

Sl. No.	Place (Year)	Source & Activity	Brief description of the incident	Consequences
1.	India (1976)	Co-60 111 TBq (3000 Ci)	Source drawer fell down during source loading in a teletherapy unit. Two persons who were present, ran out of the room and locked the room. These two were exposed to a dose of 23 mSv and 130 mSv.	Elaborate planning had to be done for retrieval operations involving lot of facilities, accessories and persons. Seven persons involved in the retrieval operations received exposures in the range of 5-16 mSv. No radiation injuries or effects to any of the persons
2.	Morocco (1984)	Ir-192 (Activity not quoted)	Ir-192 source got detached from the guide tube of a radiography equipment at a construction site. The source dropped to the ground and was picked up by a passerby who took it home.	Eight persons of the entire family got exposed to radiation doses ranging from 8-25 Gy and all of them died.
3.	Mexico (1984)	Co-60 15 TBq (450 Ci)	Old teletherapy source mixed with scrap	1800 tons of table base and 550 tons of rebar contaminated
4.	Brazil (1987)	Cs-137 50.9 TBq (1375 Ci)	Source from a teletherapy machine from an abandoned hospital ruptured leading to external and internal exposure and contamination.	Four persons died, 28 people suffered radiation burn. Seven residences demolished. 3500 M^3 radwaste generated.
5.	El Salvador (1989)	Co-60 666 TBq (18000 Ci)	Operator along with two others entered the irradiation cell and released the source rack which was entangled with objects.	Three persons exposed to radiation doses of 8.7, 3.1 and 2.7 Gy. One person (18.7 Gy dose) died. Legs amputed for another person (3.1 Gy dose).

persons who may dismantle and damage the source thereby exposing themselves and others to unshielded sources or giving rise to contamination. Goianian accident involving caesium source in a teletherapy unit is an example of external exposure and contamination pertaining to the use of a sealed source.

(ii) A radioactive source becomes unshielded as a result of failure of drive mechanism during routine operations. Inadvertent exposure to gamma radiography source due to equipment failure or to an irradiator source due to lapse in operational control, are examples under this category.

(iii) A radioactive material may be dispersed. This could be due to breakage of vial containing a radioactive solution or damage to a sealed source as mentioned above.

In all the above cases, there is a possibility of uncontrolled exposure of persons unless appropriate protective measures are taken. The radiological consequences due to the above possible accidents could be :

- internal exposure
- external exposure
- potential to produce a significant collective dose

Some incidents/accidents involving applications of radiation sources used in medicine, industry and research are listed in Table 4 (IAEA SS 91, 1989).

Depending on the scale and magnitude of operation in any application and the hazard potential in the event of an accident, detailed emergency planning and preparedness programmes have to be worked out.

Conclusions

Radiation safety practices, if followed as per the recommended standards, could ensure radiation protection and safety for radiation workers and general public. Availability of trained manpower is very important in any radiation safety programme. A well established radiation safety set up and administrative control are essential to fulfill the radiation protection requirements in any practice/application to derive optimum benefits from use of radiation sources.

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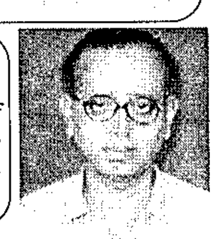
Safety in Nuclear Reactor Operation - Indian Scenario

Shri G.S. Jauhri joined Health Physics Division of BARC in 1963 after obtaining M.Sc. degree from D.A.V. College, Kanpur. He received his initial training in Health and Safety aspects of Fuel Reprocessing Plants in the Plutonium Plant in Trombay and worked as a health physicist in this plant for a number of years. He joined the Health Physics group of PREFRE Plant, Tarapur in 1973 and was also associated with various Waste Management Facilities at Tarapur including the Waste Immobilisation Plant. He was associated with planning for radiological protection activities of KARP, Kalpakkam and WIP Trombay. He led the Health Physics Group of Research Reactors at Trombay for a number of years. He has been associated with writing of Safety Guides on Decommissioning of small Nuclear Facilities (IAEA) and AERB guide on Treatment of Radioactive Waste from Indian PHWR Programme. Shri Jauhri has been associated with the various Safety Committees of AERB. Currently he is heading the Radiation Hazards Control Section of Radiation Safety Systems Division, BARC.

Shri N. Kalyanasudaram joined Health Physics Division of BARC in 1976 after graduation from Loyola College, Madras and is presently with the Radiation Safety Systems Division. He has a degree in Chemical Engineering and many years of experience in the operational aspects of radiation protection in nuclear power reactor and research reactors. Areas of special interest are computerisation of radiological data, training and public awareness.



Introduction

Safety is of paramount importance in all spheres of industrial activity. Nuclear Industry, from its inception has accorded topmost priority not only to individual safety but also to environmental impact and to the safety of public at large. This is exemplified by the fact that the frequency of severe accidents in nuclear power plants is low by 2 to 3 orders of magnitude compared to natural mishaps or accidents in other man made activities considering world as a whole.

Types of Reactors

Presently, around 430 reactors are operating in the world generating about 362,000 MWe of electrical power and 36 more reactors are under various stages of construction. Most of these reactors belong to the class of Pressurized Light Water Reactor (PWR), Boiling Water Reactor (BWR) and CANDU Pressurized Heavy Water Reactor (PHWR) types. PWR and BWR type reactors use enriched uranium oxide as fuel and light water as moderator and coolant. Canadians are the pioneers in PHWR type reactor, which uses natural uranium oxide as fuel and heavy water as coolant and moderator. Indian nuclear power programme started in 1969 with Tarapur Atomic Power Station (TAPS) reactor, a BWR type, at Tarapur. Power Reactors at Kota, Kalpakkam, Narora and Kakrapar are of PHWR type. Presently in India 10 reactors are operating and 4 more are under advanced stage of construction at Kota and Kaiga. During the year 1997 Indian reactors operated with a capacity factor of 73% and generated 9800 million Units of electricity. Indian scientists and engineers have standardized the design of Pressurized Heavy Water Reactor. Now we can construct, commission and operate safely, at international standards, PHWR type of reactor with indigenous technology. Research Reactors are under operation with the objective of nuclear research, material testing, isotope production for industrial and medical applications. APSARA, a swimming pool type of reactor has completed over 40 years of successful operation. CIRUS, 40 MW, natural uranium fueled reactor has completed 35 years of successful operation. Dhruva, a 100 MW, natural uranium fuelled, heavy water moderated and cooled research reactor having over 15 years of operating experience has enabled scientists to carry out many neutron beam studies and produce a variety of isotopes for commercial and medical use. The utilization of APSARA and CIRUS for new research and development works by DAE facilities remains high even now.

Regulatory Agency

All activities associated with nuclear energy are carried out under the Department of Atomic Energy. All research reactors are under Bhabha Atomic Research Centre, Mumbai and Indira Gandhi Centre of Atomic Research, Kalpakkam. Nuclear Power Corporation has the responsibility for design, construction and operation of nuclear power stations set up in the country. An independent agency, namely, Atomic Energy Regulatory Board (AERB) reporting to Atomic Energy commission has been set up as the regulatory body. AERB was constituted in 1983. The Board has a full time chairman, vice chairman, 3 Members and a Secretary. The Safety Review Committee for Operating Plants (SARCOP), Safety Review Committee for Applications of Radiation (SARCAR) and Advisory Committees for Project Safety Review support AERB. The SARCOP carries out safety surveillance and enforces safety stipulations in the operating units of nuclear power plants and research reactors in India. AERB dose limits are based on the recommendations of the International Commission on Radiological Protection (ICRP) as embodied in the Basic Safety Standards of radiation protection adopted jointly by IAEA, WHO and ILO.

Design Concepts

Starting from site selection for a nuclear power plant, enormous amount of data are collected and analyzed to check whether the site is suitable for a nuclear power plant operation, keeping in mind the safety of the public at large. Safety features are incorporated at the design stage itself to ensure that the risk to public at large is kept minimum under all eventualities. Design Safety Committee set up by AERB reviews design aspects of all systems and components relevant to the safety of plant. Quality Assurance programme exists at all the phases like procurement of equipment, manufacture, and site; construction activity, commissioning stage and during operation.

Nuclear Safety Principles

The following are the safety principles as outlined in the Basic Safety Standards and are fully implemented in all the nuclear reactor facilities in India.

(a) Defense in depth

(b) Fail safe principle

(c) Containment principle

(d) Concept of design basis accident

(e) Emergency preparedness and planning

(f) Comprehensive radiation protection programme

(g) Concept of ALARA

(h) Safety culture

(i) Safety organization

(j) International cooperation and exchange of ideas

Defense in Depth

Defense in depth consists in a hierarchical deployment of different levels of equipment and procedures to maintain effectiveness of physical barriers placed between radioactive materials and workers, the public and environment.

Objectives

- To compensate for potential human and component failure.
- To maintain effectiveness of the barriers by averting damage to the plant and to the barriers themselves.
- To protect the public and environment from harm in the event that these barriers are not fully effective.

Strategy

- To prevent accidents and if prevention fails, to limit their potential consequences and prevent any evolution to more serious conditions and mitigation of radiological consequences.

Multiple Barriers which contain radioactivity within the reactor

- Fuel matrix
- Fuel cladding
- Boundary of reactor coolant system
- Containment

Levels of defense

Level 1: Prevention of abnormal operation and failure

Achieved by: Design and operating procedures

Level 2: Control of abnormal operation and detection of failures

Achieved by: Reactor regulating system, Protective system, Technical specification, engineered safety features, Redundancy and diversity

Level 3: Control of Accidents within the design basis

Achieved by: Emergency core cooling system, double containment

Level 4: Control of severe conditions involving prevention of accident progression and mitigation of the consequences of a severe accident

Achieved by: Exclusion Boundary, sterile and low population zone

Level 5: Mitigation of radiological consequences of significant external release of radioactive materials

Achieved by: Emergency preparedness, Planning and Training

Fail safe principle

All nuclear instrumentation is based on the fail safe principle. All critical equipments are provided with triplicate instrumentation and independent power supply. If due to any reason any of these instruments do fail, this arrangement will ensure the safety of the plant and provide the necessary feed back to the operator.

Containment Principle

All nuclear power reactors are provided with a double walled prestressed concrete containment. During normal operation the reactor building is kept at negative pressure compared to the outside and the gaseous release from the reactor is exhausted from the reactor building after filtration through a bank of HEPA filters, through a tall stack which provides ample dilution and dispersion. In case of any nuclear accident, the resulting radioactive gaseous effluents would be contained within the reactor building and released at a controlled rate after passing through a post accident clean up system to effectively reduce radioactive iodine discharge to the environment. Leak Test of this containment building is carried out at test pressure annually to check whether the containment is healthy and that the leak rate is within the design based value (Generally allowed leak rate is 0.15% volume/hr).

Design Basis Accident

Nuclear power plant is unique in its way of planning and providing for mitigation of an accident, which has a very low probability of one in million at the design stage, itself! Normally a 'Loss of Coolant Accident' coupled with the failure of shut down system is taken as the design basis accident scenario. By taking into account the 'SOURCE TERM' and worst meteorological conditions, design safety features are incorporated so that the releases from the plant and the resulting exposure to public domain will be within the acceptable levels in an emergency situation.

Emergency Preparedness and Planning

It is the principle of remaining in readiness for a very low probable nuclear accident, which may affect the station workers and the public, at large. A detailed manual is prepared by the Station Management, which clearly spells out the various types of emergency situations anticipated, action and the responsibilities of personnel in the organization. Annual mock drills/exercises are carried out to look for any deficiency in the emergency planning and corrective steps are taken. To handle the 'Off Site Emergency' situation, periodically police, revenue and district authorities are trained and briefed about their roles during an emergency situation in a nuclear power plant.

Comprehensive Radiation Protection Programme

The Divisions in Health, Safety and Environment Group of Bhabha Atomic Research

Centre, at all nuclear reactors, carry out radiation monitoring and control. This includes Area monitoring, air sampling, contamination control, dose control, exhaust air monitoring and personnel monitoring. All radioactive jobs are covered under Radiological Work Permit system. Health physics personnel provide information in the permit about the radiological conditions at the work place, dose details of the personnel involved in the work and protective equipment and maximum allowed time at the work spot. All station personnel are imparted basic training in radiation protection and procedures to be followed while working in radioactive area, by the Health Physics group. Health Physics group interacts with operating personnel during major jobs which result in significant manrem expenditure, in planning and control of exposure and for any modification or improvement in dose reduction techniques.

Sources of Occupational Exposure

(a) Fuel clad failure

(b) Corrosion and activation products in coolant

(c) Internal exposure due to tritium in PHWR

(d) Spent fuel ion exchange resin handling

Measures Undertaken to Reduce Exposure

Improved fuel design, better fuel management and improved testing of fuel bundles prior to loading in the reactor has reduced significantly fuel clad failure. This results in lower coolant water activity and the consequent system contamination.

Among a number of corrosion and activation products, ^{60}Co, an activation product, is the single most important contributor to external occupational exposure in reactor. Selection of better material, avoiding Stellite component and dilute chemical decontamination campaign of primary coolant system has resulted in lowering of Station collective dose in power station.

Zircaloy - 2 coolant tubes undergo creep and periodic creep adjustment results in significant manrem consumption. Presently, Zirconium - Niobium tubes are used (RAPS 2) which show very little or no radiation induced creep and hence a significant reduction in occupational exposure in future PHWRs is expected due to this modification.

Tritium (^{3}H) an Isotope of hydrogen in the form of Tritiated vapour is the main source of internal exposure to occupational workers in PHWR type reactors. For makeup of system water whenever possible virgin heavy water is added. Attempts are in progress to detritiate the active coolant and moderator. Improvements are made in the ventilation system to keep the air activity level low and leaky valves are replaced immediately to prevent escape of tritium into the working atmosphere.

Sources of Population Exposure

Sources of public exposure are Argon-41, tritium and iodine-131. Improved design changes made since NAPS have helped in removing the component of ^{41}Ar as source of exposure to public. Better fuel performance, has kept the activity level of iodine-131 in coolant water to very low level and hence release of ^{131}I from stack is generally below detectable levels. Continuous monitoring and control of heavy water leak from the system also has resulted in very low release of tritium through the stack.

Concept of ALARA

ALARA means "As Low As Reasonably Achievable" by taking into consideration economic and social factors. ALARA concept was introduced by International Commission on Radiological protection to keep exposure to radiation workers and public from gaseous and liquid discharges as low as possible by enforcing the operator to exercise the options of optimization of all his resources effectively. Over the years this concept has helped a great deal in reducing significantly, individual exposures, Station collective dose and effluent discharges from the plant due to the efforts of ALARA committees operating in Power Reactors.

Safety Culture

Safety culture as defined in IAEA's International Nuclear Safety Advisory Group (INSAG - 4) report: "Safety culture is that assembly of characteristics and attitudes in organization and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance". A good safety culture has a framework for managing safety and staff at all levels within an organization that share values, perceptions and attitudes for safety. Human factor is one of the most vital links in establishing a good safety culture.

Key factors, which enhance the level of safety in an organization are:

- Individual awareness
- Knowledge and competence
- Commitment
- Motivation
- Supervision and
- Responsibility

Safety Organization

Atomic Energy Regulatory Board authorizes the license for operation after satisfying itself that all the required mandatory safety regulations are adhered to by the operating unit. The strength of a power plant safety lies in the multilayered safety set up in place. Each unit has its own Station Operation Review Committee which meets periodically and discusses all the operational and maintenance problems encountered and arrive at solution through root cause analysis. Safety Related Unusual Occurrences are fully analyzed by the Station and reports are sent to AERB for information and necessary action. Inspection team from AERB inspect nuclear establishments every year and this practice has been an important contributor in ensuring a high level of safety in plant.

International Cooperation and Exchange of Information

India is a founder member of IAEA and has participated in many of IAEA nuclear safety related programmes. This forum provides a valuable source for information and knowledge for the participants on nuclear safety related issues. India is also a member of World Association of Nuclear Operators (WANO). WANO collects data regarding nuclear reactor operation from all Nuclear power stations of the member country, which serve as "performance indicators" of the plants. These indicators provide a basis of comparison about the safety performance of the reactor in comparison with others and each unit learns from good practices followed at other stations, which in turn help in improving their performance. WANO also conducts 'Peer Review' of nuclear plant by qualified and experienced nuclear experts from various fields in member countries and objectively evaluates the reactor safety aspects. It provides the plant management an opportunity to bring out the good practices followed at their station to outside world and at the same time Peer Review Committee also points out to the management areas where improvements and upgradation are required to enhance the overall safety performance. India also participated recently in Peer Review of one of its Power Station. India is an active participant in Incident Reporting System (IRS) of IAEA. In the year 1990, International Nuclear Event Scale (INES) was introduced by IAEA to facilitate communication between nuclear community and public on events occurring at nuclear installations. The scale runs from zero for events with no safety significance to seven for a major accident. Lower levels (1 to 3) are termed as incidents and levels (4 to 7) as accidents. IAEA receives information from member countries regarding any nuclear event, which has occurred in the nuclear establishment. Exchange of this information to other member countries help in the upgradation of the existing equipment and procedure with the aim to improve safety performance.

Conclusion

In addition to the basic requirements of mankind, namely, "Food/Dress/Shelter", electricity too has become very basic necessity in modern times. Per capita consumption of electricity is considered to be one of the yardsticks of development of a country. Nuclear energy has great potential to supply electric power at economic levels. Even one or two incidents at nuclear power stations make headlines in the newspapers and hence it is all the more essential that nuclear power plant operates efficiently and performs at a high level of safety to get public acceptance as a safe source of power. With the level of inputs into the nuclear industry both in terms of technology and trained manpower this task is well within our reach. Till date, the safety record of Indian nuclear power plants and research reactors is excellent as demonstrated by the fact that no incident has occurred in any of the reactors which can be classified above 3 on the INES.

Safety Aspects of Nuclear Fuel Reprocessing

Shri V.N. Venkitaraman joined Health Physics Division of Bhabha Atomic Research Centre in 1963, after graduation in Physics from University of Madras. Presently, he is in Radiation Safety Systems Division. He is associated with all health physics activities connected with the plutonium plant. He has actively contributed towards the development of proper air monitoring devices suitable to the plant. Currently, he is involved in the development of instrumentation for safety surveillance capable of estimating both alpha and beta activities of plutonium plant. His latest development is the indigenisation of high volume air sampler by modifying a local vacuum cleaner and successfully put into operational use in the plant.

Dr. Pushparaja joined the Health Physics Division of Bhabha Atomic Research Centre in 1968, after graduation in chemistry from the University of Bombay. At present, he is in Radiation Safety Systems Division. He obtained Ph.D. degree in 1977 in chemistry, Currently, he is heading the RHC group which covers radiochemistry and fuel chemistry divisions at south site and at radiological laboratories, and plutonium plant, purnima reactor and waste immobilisation plant, at Trombay. He has been carrying out R&D connected with the operational radiation protection, particularly in monitoring systems and high level dosimetry. He has been supporting the activities of Atomic Energy Regulatory Board in various capacities, such as member of safety committees, inspection teams, etc. He was a memeber of the IAEA Advisory Committee, which met in 1996 at Vienna, to review and revise the Safety Guide on Occupational Radiation Protection based on the recent International Basic Safety Standards. Dr. Pushparaja is in the Editorial Board of the Journal, Radiation Protection and Environment. He has over 70 publications to his credit in national/international journals and symposium/conference proceedings.

Introduction

Indian nuclear power reactors are of Pressurised Heavy Water Reactor Type (PHWR), where natural uranium is used as fuel. Uranium, as found in nature, is called natural uranium and it consists of three isotopes, viz., ^{238}U (99.28%), ^{235}U (0.72%) and ^{234}U (0.0056%), with a specific activity of 0.67 μCi/g. The ^{235}U present in the natural uranium is a fissile material and it undergoes fission by capture of neutrons at thermal energies in the nuclear reactor. The products of this reaction are: fission products, a lot of energy and a couple of neutrons which help in sustaining the fission chain reaction in the reactor. In these reactors, some ^{238}U captures fast neutrons and is converted to a very important, fissile isotope of plutonium, i.e., ^{239}Pu. Fuel reprocessing is one of the operations in back-end of the nuclear fuel cycle, which follows reactor operations.

The spent fuel taken from the reactors, after a certain burn-up, calculated in terms of MegaWatt Days per tonne (MWd/t) contains: (i) unused uranium isotopes, (ii) valuable quantity of ^{239}Pu and (iii) highly radioactive fission products. By chemical reprocessing of the fuel one can recover the uranium, called depleted uranium, separate plutonium and isolate the fission product waste materials from the fuel. Because of the radiation properties of the material, specially designed enclosures called hot cells, are used for the reprocessing of the fuel. In this article, safety aspects of the fuel reprocessing are discussed.

Process

The highly radioactive fuel bundle from the reactor is transported to a reprocessing plant in a well shielded container called Shipment Cask and stored under water in Fuel Handling Area (FHA). The fuel is then introduced into chopper-cum-dissolver cell.

The cell is shielded against radiation and it has thick concrete walls. Using a shearing machine, the fuel bundles are cut into pieces and the cut fuel is dissolved in concentrated nitric acid. The zircalloy cladding used on the fuel is not soluble in the nitric acid and forms part of the solid waste inventory for disposal.

The solution containing uranium, plutonium and fission products in nitric acid medium, is then subjected to separation processes such as solvent extraction with tributylphosphate (TBP), and ion-exchange. During the solvent extraction, only plutonium and uranium will get extracted into the organic phase. The fission products are left behind in the aqueous phase. The plutonium and uranium are then chemically separated from each other. The plutonium is further purified by ion-exchange process. The fission product remaining in the solution is concentrated to reduce volume and is stored in underground stainless steel tanks.

Devoid of most of the fission products and uranium, plutonium nitrate is then converted to plutonium oxalate which on calcination gives PuO_2 and stored in stainless steel container. This container is kept in a specially designed storage unit called Bird Cage, to ensure that the fissile plutonium does not go critical while in storage leading to criticality accident. Table 1 shows typical values for critical mass for plutonium and some of its compounds.

Table 1. Critical mass for selected plutonium and its compounds

Compounds	Mass (bare) kg	Mass (water reflected) kg
Pu (solution)	-	0.510
Pu (metal)	-	5.6
PuO_2	24.50	12.2
$Pu(C_2O_4)_2$	152.00	66.87
$Pu(NO_3)_4$	103.96	54.94
PuC	17.93	9.07

Process Cells

Process equipment that handle the radioactive material such as shearing machine, dissolvers, extraction columns, evaporators, etc. are located inside special enclosures called Cells. The cells have concrete walls of thickness in the range 900 mm to 1800 mm. During plant operations, the cells are inaccessible. The process equipment are remotely operated from the control room. The operating gallery houses the control valves meant for various liquid transfer operations.

Hazards

The hazards, external and internal radiation exposures, associated with the fuel reprocessing are due to:

(i) large inventory of long-lived fission products, in dispersable form,

(ii) handling of high activity (Ci/ml) of acidic/organic solutions,

(iii) handling plutonium in solution/solid form in significant quantities, with potential for criticality, and

(iv) storage and disposal of high level radioactive wastes.

Safety Aspects

Plutonium (Pu) is the important radioactive material that is handled in reprocessing plant. ^{239}Pu is the isotope of interest as a fuel in fast breeder reactors, as mixed oxide fuel in thermal reactors and as a strategic material in nuclear weapons. Hazards associated with the handling of plutonium are essentially due to its radiotoxicity and its distribution within human body after its intake via., inhalation, ingestion and injection through cuts/wounds.

The criticality hazard in plutonium handling is avoided by proper design of the equipment and strict administrative control over the amount of the material handled and stored.

Protection Standards

The isotopes of Pu which are of interest, viz., ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{242}Pu are alpha active, and with their associated low energy x-rays/ or low yield gammas are not considered as external hazard unless present in massive form. However, due to the small range of the alphas in tissue and its high ionising power, plutonium is considered as an internal hazard. One of the isotopes of plutonium, i.e., ^{241}Pu is a beta emitter. It decays to ^{241}Am which buildsup with

Table 2. Biokinetic parameters for plutonium

Organ of deposition	ICRP-48, 1986		ICRP-67, 1993	
	Percent deposition	Removal half time	Percent deposition	Removal half time
Skeleton	45	50 y	50	See ref. 2
Liver	45	20 y	30	10 y
Soft tissues Comp. 1 Comp. 2 Kidney path Kidney tissue			12.5 2 1 0.5	2 y 100 y 50 d 500 d
Testes	0.035	∝	0.035	10 y
Ovaries	0.011	∝	0.011	10 y
Early excretion	10			

∝ - Infinite retention

time. The gamma dose given by Pu is mainly due to low energy (59 keV) photons from ^{241}Am.

Over the years, the protection standards for plutonium has changed considerably as recommended by the International Commission on Radiological Protection (ICRP). The most recent standards are based on stochastic considerations and takes into account the recent metabolic model [ICRP-67, 1993]. The metabolic parameters are listed in Table 2, and the derived values for the Annual Limit on Intake (ALI) for ^{239}Pu using the committed effective dose coefficients [ICRP-68, 1994] are listed in Table 3. The currently used values [ICRP-61, 1991] are also given in the same Table as a footnote for comparison.

Hazard Control Methods

The nature of radiation exposure control methods in a reprocessing plant is somewhat different from that in a nuclear reactor. Table 4 gives the comparison of the relevant features.

The process equipment are located within well shielded cells and the plant is operated from a control room. Hence, not many operators are required to work in radiation areas. The fuel reprocessing is a chemical process and since the fuel cladding is removed there exists high radiation fields, airborne activities & contamination in the process cells. Any breach in the containment, leak from valves, spills and frequent cell entries for maintenance works pose challenges to the radiation protection professionals. The exposures resulting from this can be external, internal or both.

Table 3. Effective inhalation dose coefficient (Sv/Bq) for Pu-Isotopes (ICRP-68, 1994)

Radio-nuclide	Absorp-tion type	Dose coeff. Sv/Bq	ALI Bq	DAC Bq/m^3
Pu-238	M S	4.3×10^{-5} 1.5×10^{-5}	465 1330	0.19 0.55
Pu-239	M S	4.7×10^{-5} 1.5×10^{-5}	425 1330	0.18 0.55
Pu-240	M S	4.7×10^{-5} 1.5×10^{-5}	425 1330	0.18 0.55
Pu-241 (β-emitter)	M S	8.5×10^{-7} 1.6×10^{-7}	2.4x104 1.25×10^{5}	10 52
Pu-242	M S	4.4×10^{-5} 1.4×10^{-5}	450 1430	0.19 0.60

$f_1 = 5\times10^{-4}$ (M); 1×10^{-5} (S) - recent values
1×10^{-3} [ICRP-48, 1986]
Current value for ^{239}Pu: ALI (M&S) =300 Bq [ICRP-61, 1991]

Table 4. Comparison of important safety features of nuclear reactor and reprocessing plant

	Feature	Reactor	Reprocessing Plant
1.	Inventory of long-lived fission products in dispersible form	Low	High
2.	Containment of radioactive material	Well contained	Containment is less than that of a reactor
3.	Specific activity of solutions	mCi/ml	Ci/ml
4.	Type of liquids handled	Water	Acid/organic; chemicals are reactive and corrosive
5.	Potential for internal exposure	Not significant (except HWRs)	Significant
6.	Problem of handling α emitters	None	Significant
7.	Potential for skin dose	Little	Significant

Control of External Exposures

In a fuel reprocessing plant the main source of exposure is the process cell which is shielded by thick layer of concrete. Other sources like wastes, pumps etc., are also either shielded or isolated in safer places. Leaks and spills often results in hot spots showing radiation fields of few Rad/h.

The three cardinal principles of external exposure control, that is , reduction in exposure time, increased distance between the source and operator and use of shielding (Time, Distance & Shielding), are judiciously employed in reprocessing facilities. Extensive shielding is provided on spots showing high radiation levels. Workers are advised to use remote handled instruments/ tongs to carry out work related to high radiation fields. Warning signs are displayed at places of high radiation to caution workers to keep off the areas. On occasions, if necessary, the exposure control is done by the control of time. Special Work Permit (SWP) is in force for maintenance jobs or special operations so that specific instructions are given to ensure safety and minimise personnel exposures. To identify the high risk areas, extensive detailed surveys are carried out and judicial planning done to execute a potential high radiation job.

Internal Exposure Control

In the fuel reprocessing plants where spills and leaks are common due to use of valves and pumps, the risk of internal contamination is quite high if proper care is not taken. The possibilities of internal exposures are due to:

(i) by inhalation - breathing of airborne radioactive contaminants

(ii) by ingestion - eating food using contaminated hands

(iii) by absorption through skin or wounds due to accidental injury during work.

Internal exposure can be avoided or controlled by carefully executing the following:

(a) proper containment of the equipment or source,

(b) providing good ventilation and

(c) using appropriate personnel protective equipment

Personnel protection equipment include wearing protective clothing, cap, handgloves, shoes, respirators, etc. The above protective measures will prevent personal contamination of the worker thus avoiding internal exposures from these routes. Workers are advised to handle instruments with care to avoid injuries/wounds. Over and above, good house keeping coupled with clean habits are tutored to achieve total control of internal exposures.

Basic radiation protection standards as incorporated in the annual dose limits stipulated by the AERB, are strictly adhered for occupational exposures to avoid stochastic and deterministic effects.

Waste Management

Radioactive wastes are the major concern in reprocessing plants. They are generated during processing as well as while cleaning, maintenance and upkeep of the plant.

The wastes generated in fuel reprocessing are of three kinds, i.e., solid, liquid and gaseous.

Solid Waste

The solid wastes are generated during clean up work, decontamination of leaks, spills or after maintenance work. These wastes are packed in polythene bags as per the type of radiation (alpha or beta-gamma) and also by the radiation levels such as high, medium or low. The bags are then appropriately tagged, packed in drums and sent to storage building or disposal site.

Liquid Waste

Process radioactive liquid wastes are of three categories. They are:

(i) high level acidic wastes (stored in underground stainless steel tanks),

(ii) medium level alkaline & organic wastes (stored in underground carbon steel tanks), and

(iii) low level wastes (collected in Delay Tank and sent to Effluent Treatment Plant (ETP))

The high level liquid wastes will be processed, immobilised and vitrified in waste immobilisation plant (WIP), medium level liquid wastes are treated, conditioned and stored for future processing and the low level liquid wastes are analysed, diluted if required and discharged to sea via., Effluent Treatment Plant (ETP).

Particulate and Gaseous Wastes

These wastes are generated during the processing of the spent fuel rods and also during maintenance work. All particulate matters and gases discharged during the process operation are passed through deep-bed glasswool filters followed by prefilters and high efficiency particulate air filters (HEPA). Gaseous wastes are: radioactive ^{85}Kr released during dissolution process of the fuel and oxides of nitrogen generated during dissolution and acid killing operation. The exhaust air containing oxides of nitrogen are passed through scrubbers to minimise the concentration before releasing it to atmosphere, while ^{85}Kr, being inert gas is not amenable for treatment and hence is released through 135 metres high stack which provides enough dilution. The discharge levels are controlled well below the stipulated regulatory limits.

The discharges through the stack are monitored continuously and record is maintained about the quantity released.

Conclusion

In addition to the design based safety, radiological safety in nuclear fuel reprocessing plants is ensured by an effective safety surveillance programme which is aimed at minimising the external exposures and preventing internal contamination as low as reasonably achievable. The control is achieved by adopting the following:

(i) Identification of sources of radiation exposures

(ii) Control of the source (decontamination/ shielding)

(iii) System of special work permit

(iv) Job planning for high exposure/ high air-borne activities through Operational Review Committee (ORC)

(v) Training of plant personnel in radiation safety

(vi) Development of safety culture amongst the workers

(vii) Periodic review of radiological status of the plant

All the above practices together with professionalism have helped in maintaining Man-rem for fuel reprocessing facilities low, and in keeping the individual exposures well below the regulatory limits.

Future Scenario

Reprocessing of fuels from Advanced Fuel Cycle (MOX and Th/^{233}U fuel cycle) offers real challenges and tremendous scope for innovative development in automation and robotic to minimise exposure of the personnel. Some of the important considerations in fuel reprocessing in the future should be: process flow-sheet design to minimise generation of radioactive waste, remote maintenance of process equipment, miniaturisation of monitoring/safety systems, and extensive usage of computers for dose management and handling of monitoring data.

Further Reading

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3. Hazards of Plutonium and Fuel Processing, G.M.Watson, Atomic Energy in Australia, Vol. 21(1), (1978), 1-12.

AERB - its Role as Regulator of Nuclear Fuel Cycle Operations in India

Shri A.R. Sundararajan started his career as Health Physicist in Fuel Reprocessing Plant at Trombay after graduationg from the BARC Training School in 1965. He was also associated with the Health Physics design aspects of PREFRE at Tarapur. Later, as Head, Operational Health and Safety Section he was responsible for providing surveillance both for radiation protection and industrial safety at Indira Gandhi Centre for Atomic Research at Kalpakkam. As Head, Health and Safety Division he was closely associated with the design, construction and commissioning of Health and Safety Laboratory at Kalpakkam in particular, responsible for starting a strong research group on internal dosimetry, atmospheric studies, accident source term and aerosol research. He was a member of Kalpakkam Emergency Committee and was principal author of Emergency Preparedness Manual for the Kalpakkam DAE centre. He has been a member of Safety Committees for KARP, KAMINI, PFBR and Waste Management Projects. Currently as Head, Health and Safety Division of AERB, he is responsible for regulatory activities related to radiation safety in nuclear fuel cycle facilities and in application of radiation technology in industry, medicine and research. He has participated in many meetings of technical experts, advisory group and working groups at International Atomic Energy Agency. He has 68 publications to his credit.

Introduction

The main steps in the nuclear fuel cycle comprise mining, milling, fuel fabrication, reactor operation, fuel reprocessing and waste management operations. All these operations involve handling of hazardous radioactive materials in variety of forms and of different radioactive inventories. When Atomic Energy Act of 1962 was formulated to provide the development, control and use of atomic energy, considerable attention was given to radiation safety aspects. This act has enabled framing of several rules related to radiation safety of all installations in the country handling radioactive materials or radiation generating equipment, which include the nuclear fuel cycle facilities of the Department of Atomic Energy. These are listed below:

(i) Radiation Protection Rules, 1971

(ii) Atomic Energy (working of Mines, Minerals and Handling of Prescribed Substances) Rules, 1984.

(iii) Atomic Energy (Safe Disposal of Radioactive Wastes) Rules, 1987

The Atomic Energy Regulatory Board (AERB) was constituted in 1983 by the Government of India with a mandate to carry out certain regulatory and safety functions under the Atomic Energy Act, 1962. AERB consists of a Chairman and four members appointed by Government of India. With respect to DAE installations, AERB has the following major functions and responsibilities.

(i) Develop Safety Codes, Guides, Standards and Manuals for siting, design, construction, commissioning, operation and decommissioning of different types of plants of DAE.

(ii) Review of the safety aspects of DAE projects/ plants and issue authorisation/ licences for siting, construction, commissioning, operation and decommissioning of the plants

(iii) Ensure compliance of safety codes, standards etc. by DAE plants

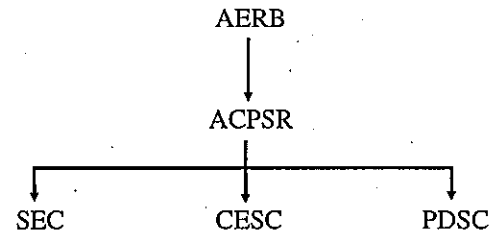
Industrial safety aspects of all DAE installations are covered by Atomic Energy (Factory) Rules, 1996.

Authorization Process for DAE Nuclear Installations

For major nuclear installations such as nuclear power plants, reprocessing plants, heavy water plants etc., AERB issues authorisations at different stages. The regulatory authorisation process followed for a typical PHWR is described here. The major stages of the authorisation process for nuclear power plants are siting, construction, commissioning (at different stages) and power operation (at different stages). Authorisation at each stage is preceded by a detailed review of all safety aspects of the proposal. In general, the safety review process is carried out at three different levels. In the first level the project is reviewed by Site Evaluation Committee (SEC), Project Design Safety Committee (PDSC) and Civil Engineering Safety Committee (CESC). The recommendations of these committees are reviewed at the next level of scrutiny through an Advisory Committee for Project Safety Review (ACPSR). AERB has constituted two ACPSRs for nuclear power plants, one for heavy water plants, one for fuel reprocessing plants and one for nuclear waste management plants. These committees are composed of experts not only from DAE and AERB but also from other governmental agencies and reputed academic institutions.

ACPSR after, its own assessment makes recommendations to AERB which is the statutory authorising agency. The Board can delegate its powers to Chairman, AERB for issue of authorisation for certain project activities. However, all important authorisation stages such as siting, construction, commissioning, first approach to criticality of reactors and first power generation stage are reviewed by the board. AERB while issuing the authorisation for a specific activity may stipulate the requirements and conditions governing the performance of the activity and whereever appropriate specifies a time limit for the validity of the authorisation.

Three Level Safety Review



ACPSR : Advisory Committee for Project Safety Review (Members are from DAE and other Govt. Organisations and academic Institutions)

SEC : Site Evaluation Committee

CESC : Civil Engineering Safety Committee

PDSC : Project Design Safety Committee (PDSC)

Siting

The main objective of the review before issue of authorisation for siting is to ensure that the applicant will be able to construct and operate nuclear power plants safely and to provide protection of the workers and the members of public against radiological impact resulting from releases of radionuclides during normal operations of the plant as well as under accidental conditions.

In evaluating the suitability of a site for locating a nuclear power plant, the following major aspects are considered:

(i) Effect of external events (nature and man induced) on the plant

(ii) Effect of plant on environment and population

(iii) Implementation of emergency procedures in the public domain

Sites for nuclear power plants shall be examined with respect to the frequency and the severity of external events and phenomena, natural and man made, that could affect the safety of the plant. All those events having significant radiological risk should be considered and their design bases determined. The radiological risks associated with external events should not exceed the range of radiological risks associated with the accidents of internal origin. For an external event, design basis should ensure that structures, systems and components important to safety in relation to that event will maintain their integrity and will not suffer loss of function during or after the design basis event.

While considering the natural events, it is important to collect the historical records of the occurrences and severity of the important natural phenomena for the region. The data collected shall be carefully analyzed for reliability, accuracy and completeness.

For each site, the potential radiological impact on people in the region during operational states and accident conditions shall be assessed. Consideration should be given to possible radiological consequences in the event of locating in the same site, other fuel cycle facilities like fuel fabrication, fuel reprocessing and waste management plants. Low population density in the region will help in achieving reduced population dose. It shall be ensured that effective implementation of emergency counter measures in case of an accident will be possible.

Every nuclear installation is required to submit a Site Evaluation Report to AERB. This report is reviewed by a Site Evaluation Committee constituted by AERB. Table 1 gives the typical contents of a Site Evaluation Report. ACPSR examines the findings of the Site Evaluation Committee and recommends to AERB for issue of regulatory consent to siting of the project with conditions and stipulations required if any.

Table 1. Contents for site evaluation report

I.	**Salient Features of the Proposed Site**
1	Topography
2	Accessibility
3	Industrial Infrastructure and Construction Facilities
4	Availability of Power Supply and Transmission Lines
5	Availability of Cooling Water
6	Township
II.	**Site Characteristics Affecting Safety**
1	Geology
2	Natural events
3	Man-Induced Events Affecting Safety of Plant
4	Meteorology
5	Ultimate Heat Sink
6	Use of Land
7	Use of Water
8	Disposal of Radioactive Solid Waste from the NPP
9	Disposal of Radioactive Liquid Waste from the NPP
10	Radioactive Gas Release
11	Radiological Impact
12	Thermal Pollution
13	Storage and Transportation of Fresh and Spent Fuel
14	Fuel Reprocessing Facility
15	Population Considerations
16	Emergency Preparedness Considerations
17	Additional statutory requirements of the central and state governments

Construction

Like in siting, a three tier regulatory review is carried out for grant of authorisation of construction. The applicant is required to submit a Preliminary Safety Analysis Report (PSAR), a quality assurance program and a construction schedule along with the application for authorisation of construction. The contents of a typical PSAR are given in Table 2. First tier review of the application for permission to construct the plant is conducted by Civil Engineering Safety Review Committee, constituted by AERB. Some of the important aspects of the project that are reviewed by this committee include the following :

Geotechnical investigation data and foundation parameter

- Design basis ground motion
- Plant layout and surface drainage
- Engineering of site against natural and man made hazards
- Construction methodology

The second tier review is by ACPSR and the third tier review is by AERB before the issue of authorisation.

Table 2. Contents of Safety Report for Nuclear Reactors

1.	General Description (Design criteria, Design highlights operating characteristics etc.)
2.	Site and Civil Structures
3.	Reactor
4.	Coolant System
5.	Containment System
6.	Engineered Safety Features
7.	Instrumentation and Control
8.	Electrical System
9.	Auxiliary and Emergency System
10.	Steam and Power Conversion System
11.	Radiation Protection
12.	Radioactive Wastes
13.	Organizational Structure
14.	Accident Analysis

Commissioning

Commissioning is the process by which plant components and systems are made operational and verified to be in accordance with design specifications. The commissioning should also demonstrate that the plant could be operated in safe manner through integrated testing of the plant system. For grant of authorization for commissioning, the plant is required to submit a PSAR, Quality Assurance program documents. In addition, the plant should also submit a list of all tests, related activities in their sequence, results expected, acceptance criteria and their relevance to the proposed operational limits and conditions, if any. First tier review of the application for issue of authorization for commissioning is carried out by the Project Design Safety Committee. The second and third level reviews are carried out by ACPSR and AERB, respectively, before the grant of authorization for commissioning.

For Indian nuclear power plants, the authorization for commissioning is given in several interim stages. AERB Safety Guide SG/0-4 covers all aspects of commissioning procedures for PHWR based nuclear power plants. For PHWRs following are the interim stages at which regulatory reviews are required.

(a) Hot conditioning of the primary heat transport system

(b) Fuel loading of the reactor core

(c) Heavy water addition to moderator system

(d) First approach to criticality and low power physics experiments

(e) Initial power raising and synchronization with grid up to 50% of full power (F.P)

(f) Power operation typically at 50, 75, 90 and 100% F.P. or at power levels stipulated by AERB based on review of performance.

Authorization procedures for chemical and metallurgical plants of the nuclear fuel cycle and heavy water plants are almost identical to those established for nuclear power plants. For fuel reprocessing plants following are the interim stages at which regulatory reviews are called for :

(a) Acid-solvent runs through the process cycles

(b) Dissolution and processing of fresh uranium fuel (cold runs)

(c) Receipt and storage of irradiated fuel

(d) Processing of irradiated fuel (specified inventory)

(e) Regular processing of irradiated fuel (full plant capacity)

Operations

For comprehensive review of safety status and enforcement of safety regulations during the operational phase of nuclear installations, a three-tier review structure has been put in place by AERB. At the plant level, a plant operation review committee reviews all operations and maintenance activities in the plant with potentials for safety problems. This committee reviews all unusual occurrences, deviations from Technical Specifications, modifications in the plant and changes in plant procedures.

Next higher-level review committee is the unit level safety committee. The reports from the plant operation review committee and the health physics unit at the plant provide the input for unit level safety review committee, which in turn reports to Safety Review Committee for Operating Plants (SARCOP) of AERB. Chairman, SARCOP is an AERB official who is an ex-officio member of the Board of AERB. Unit level safety committees are constituted by Chairman, SARCOP and have in their membership experts from AERB. Apart from the report from unit level safety committee, SARCOP receives as inputs for its review, periodic reports from AERB's Directorate of Regulatory Inspection and Enforcement (DRI&E) and Quarterly reports of Health Physics units. In order to cater to the regulatory surveillance requirements of large number of operating plants, SARCOP meets almost every week and some of the meetings are held at the plant sites. SARCOP is empowered to impose restrictions or suspension of operation of the facility under intimation to DAE. Recommencement of operation after suspension following serious violations of safety norms or serious incidents will be permitted only after detailed review and approval by the regulatory board.

Other regulatory requirements to be complied with during the operational stage of nuclear plants include:

- Licensing of operating personnel at various levels through committees constituted by SARCOP
- Issue of authorization for disposal of radioactive wastes
- Maintenance of emergency preparedness
- Compliance with the requirements of Atomic Energy Factory Rules

Decommissioning

Though decommissioning of nuclear facilities is not a problem of immediate concern to DAE, recognizing the importance of this activity, AERB has issued a Safety Manual on Decommissioning of Nuclear Facilities. The manual provides the regulatory framework of safety within which the decommissioning of an operating nuclear facility can be carried out at the end of its service life. It provides information on decommissioning, acceptance criteria and their bases, health physics considerations, waste management aspects, quality assurance practices and documentation requirements. It also includes an outline of design provisions to be made to facilitate decommissioning and recommends an organizational structure for the decommissioning activities.

Documentation

For obtaining the authorizations from the regulatory body at various stages of the plant from siting to decommissioning, it is important that relevant documents are carefully prepared and submitted in time to the appropriate committees. Table 3 gives the list of typical documents to be submitted to AERB and advance periods recommended for submission of these documents. The periods indicated are typical time duration required for completion of the three-tier review before issue of authorization.

Conclusion

Atomic Energy Regulatory Board has been mandated to review, enforce standards and authorize from safety angle, siting, design, construction, commissioning, operation and decommissioning of nuclear installations. Since its inception in 1983, AERB has published 61 safety documents like codes, guides, standards and manuals that provide regulatory guidance to the plant authorities. AERB over the years has put in place a sound regulatory framework and mechanism which permit the Department of Atomic Energy to construct and operate the nuclear power plants and associated fuel cycle facilities without undue risk to the operating personnel and members of the public.

Table 3. Minimum Advance Period for Submission of Supporting Documentation for Authorisation Stages

Sr.No.	Document	Recommended submission time
1.	Site Evaluation Report	Six months prior to start of site construction
2.	Design Basis Information	Six months prior to start of site construction
3.	Preliminary Safety Analysis Report	Six months prior to start of site construction
4.	Quality Assurance Program in Design, Construction, Commissioning	Six months prior to start of site construction or commissioning
5.	PERT chart giving construction schedule	Three months prior to start of site construction
6.	PERT chart for commissioning program	Two months prior to start of commissioning
7.	Technical Specifications for Operation	Three months prior to fuel loading/heavy water addition whichever is earlier
8.	Training and Qualification Program (including schedule for key operating personnel)	Three months prior to fuel loading/ heavy water addition whichever is earlier
9.	Emergency Preparedness Plans	Three months prior to first approach to criticality
10.	Complete set of flow and logic diagrams	Three months prior to relevant authorization stage
11.	Commissioning procedures and test/ inspection procedures	Two months before relevant authorization stage
12.	Pre-requisite commissioning test/ inspection results	One month before relevant authorization stage
13.	In-service Inspection and Testing Program	Two months prior to initial criticality
14.	Final Safety Analysis Report (Operational data and its analysis at each power operation stage i.e. 25%, 50%, 75% and 100% of rated power)	One month prior to request for power operation authorization (as part of request for authorization for next higher power operation)
15.	Design Basis Manuals (DBM)	Two months prior to authorization stage relevant to the DBM document
16.	Detailed design description and flow sheets	Available at site prior to initial criticality
17.	Training documents	Available at site three months prior to fuel loading
18.	Operating Manuals	Available at site prior to fuel loading/ heavy water addition, whichever is earlier
19.	Maintenance Procedures	Available at site prior to initial power operation

References

1. Code of Practice on Safety in Nuclear Power Plant Siting, AERB/SC/S, 1990.
2. Safety Guide for Seismic Studies and Design Basis Ground Motion for NPPs, AERB/SG/S-11, 1990.
3. Safety Manual for Civil Engineering & Building Works for NPPs, AERB/SM-S-1, 1988.
4. Safety Manual Governing Authorization Procedure for Nuclear Power Plant, AERB/M/NSD-3, 1989.
5. Safety Guide for Commissioning Procedures for PHWR Based NPP, AERB/SG/0-4, 1998.
6. Safety Manual Decommissioning of Nuclear Facilities, AERB/SM/DECOM-1, 1998.
7. Safety Guide for Radiation Protection During Operation of Nuclear Power Plant, AERB/SG/0-5.

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