Chapter 12

RADIATION PROTECTION

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12. RADIATION PROTECTION

12.1. Ensuring that occupational radiation exposures are As Low As Reasonably Achievable (ALARA)

This chapter describes the radiation protection measures of the Plant design and the operating policies to ensure that internal and external radiation exposures to Plant personnel, contractors, and the general population due to Plant conditions, including anticipated operational occurrences, will be within applicable limits, and furthermore, will be as low as is reasonably achievable (ALARA).

Radiation protection measures include : separation of radioactive components into separately shielded cubicles ; use of shielding designed to adequately attenuate radiation emanating from pipes and equipment which are sources of significant ionizing radiation ; use of remotely operated valves or handwheel extensions ; ventilation of areas by systems designed to minimize inhalation and submersion doses ; installation of permanent radiation monitoring systems ; control of access to the site and to restricted areas ; training of personnel in radiation protection ; and development and implementation of administrative policies and procedures to maintain exposures ALARA.

12.1.1. Policy considerations

The management of the Korea Electric Power Corporation is committed to keeping occupational exposure to ionizing radiation as low as reasonably achievable (ALARA). Accordingly, the Health Physics program is prepared and conducted in conformance with the recommendations contained in the RCC-P Chapter 5 and Regulatory Guides 8.8 and 8.10. Title 10 Code of Federal Regulations, part 20, provides the regulatory framework under which the ALARA Philisophy is implemented.

The Manager, Radiological Control Division, is responsible for the radiation protection program at KNU 9 and 10. The program consists of written health physics procedures, intensive training of radiation protection fundamentals, and periodic reviews by Plant management. Section 13.1 presents the radiation protection staffing for KNU 9 and 10. 1

The Manager, Radiological Control Division, is responsible for the initial radiation protection program formulation, reports directly to the Plant superintendent, and exercises supervisory control over the Health Physics Group. The health physicist reports to the Manager, Radiation Control Division, and exercises supervisory control over the health physics technicians and instrument repairman. The health physics technicians are responsible for radiation surveys, contamination surveys, and air sampling. Plant personnel are either trained or indoctrinated in radiation protection by the Manager, Radiological Control Division. The radiation protection functions of the Health Physics Group include : 1 1 1

a) Assuring that dose limits established in accordance with the approved radiation program are not exceeded by Plant personnel or visitors and that any dose received is as low as reasonably achievable.

b) Controlling radiation exposure by :

- evaluating radiological conditions and taking precautionary measures,
- controlling personnel and equipment movement into and out of controlled areas,
- ensuring proper use and care of special protective clothing and equipment,
- conspicuously posting each area within the controlled area with appropriate caution signs,
- administering and controlling conditions of radiation work permits for work in areas having high radiation and/or contamination levels in accordance with approved procedures.

c) Determining requirements for and the extent of use of personnel monitoring devices and maintaining records of personnel exposure.

d) Controlling and accounting for all radioactive material entering or leaving the Plant site.

e) Establishing procedure for dealing with potential or actual emergency conditions.

f) Training the Plant staff and visitors in radiation protection policy and procedures, as required.

g) Obtaining assurance that Plant is designed for maintaining personnel radiological doses ALARA.

Sections 12.1.2 and 12.1.3 describe the criteria and methods used in the design of the shielding, ventilation and radiation monitoring systems, including equipment and Plant arrangements and access control provisions, to keep occupational exposures ALARA.

All efforts has been sustained for the design and the design review of KNU 9 and 10, reference Plant in view to ensure that occupational exposures are ALARA. The design of KNU 9 and 10 Plant is the same as that of the reference Plant and consequently may be considered as complying with ALARA principles and the Regulatory Guides 8.8 without specific design review.

The KNU 9 and 10 reference Plant has been designed according to French regulations (cf. RCC-P Chapter 5). French regulations reflect recommendations on radiation protection established by the International Commission on Radiation Protection (ICRP). This regulation specifically in the Decree No. 66-450 of June 20, 1966, specifies maximum allowable levels for occupational radiation exposure, and it states that, the exposure of personnel and the number of persons exposed to ionizing radiation shall be kept as low as possible (ALAP) and shall in no case exceed the maximum values in the regulations.

In accordance with French authorities the meaning of this statement as applied in design, construction, and operation of the KNU 9 and 10 reference Plant is that all efforts should be made to maintain exposures low. Attaining this goal requires good engineering judgement on a case-by-case basis, and does not embody cost benefit analysis.

This practice may be considered as an application together of ALAP and ALARA concepts, with the use of the restrictive approach (qualitative but no quantitative), proposed by NRC in the Regulatory Guides 8.8, in that concerns the ALARA concept.

The licencee of KNU 9 and 10 reference Plant is EDF (Electricité de France) which acts in the same time as architect engineer and utilities. This double role does easy for interfacing between the engineering department and the operating department.

The two departments form a committee which is in charge to evaluate experience and data from operating Plants. The information resulting from the study of operating Plant experience and design is used to establish written criteria and design guide lines to ensure that operator exposures are as low as reasonably achievable (ALARA).

Design reviews by nuclear engineers belonging to operating department ensure conformance with the established criteria. Further more these persons suggest improvements of equipment or facility designs as requirements or recommendations. Subsequent reviews of the revised designs check for conformance with these requirements.

The EDF personnel responsible for ALARA design and review are competent in the application of radiation protection principles, including radiological dose assesment, shielding design, and radwaste system design.

The central health physics group (Radiation Protection Department) which belongs with the operating department of EDF is given the task of assisting the design and review effort. Furthermore it contributes in the reviewing of operating Plant exposures and designs to establish criteria and guidelines for the purposes of radiation protection. This group reviews various aspects of the Plant design to ascertain conformance with the established criteria of maintaining exposures ALARA and recommends design modifications as necessary.

Since a major portion of the occupational radiation dose is received during maintenance, inservice inspection, refuelling, and non-routine operations, these activities receive special attention during design to ensure radiation exposures are ALARA.

Among other responsibilities, the engineering department of EDF represents the utility department interests in dealing with NSSS and components designer in view to ensure that occupational exposures are ALARA.

During the construction of the KNU 9 and 10 reference Plant, field changes submitted by the utilities to the engineering department have been studied and reviewed like the first design to ensure that occupational exposures are ALARA.

Framatome is committed to ensuring that occupational radiation exposures are ALARA in KNU 9 and 10 by providing design, system and components identical to the reference Plant.

Changes in relation to the reference Plant design submitted by Framatome or by the utilities for KNU 9 and 10, were studied and reviewed by competent nuclear engineering personnel of Framatome to ensure that occupational exposures are ALARA. In keeping with this commitment Framatome defined the responsibilities of the radiation protection group and provided an environment in which the radiation protection functions performed their duties properly.

While Framatome does not have responsibility for policy considerations related to the operation of systems and components designed and supplied by Framatome, a commitment had been made to gather operational information related to radiation protection aspects of Framatome systems and components. This operational information was then utilized by the radiation protection staff in working with the designers thereby assuring that operational aspects related to the ALARA philosoply were considered during the design stage.

The policy considerations outlined above ensure that occupational radiation exposures are ALARA in compliance with Regulatory Guides 8.8, revision 3, "information relevant to ensuring that occupational radiation exposures at nuclear power stations will be as low as is reasonably achievable", and 8.10, revision 1, "operating philosophy for maintaining occupational radiation exposures as low as is reasonably achievable", and 10 CRF, part 20, (Section 20.1 (c)).

12.1.2. Design considerations

This subsection discusses the methods and features by which the policy considerations of subsection 12.1.1 are applied. Provisions and designs for maintaining personnel exposures ALARA.

12.1.2.1. General design considerations for as-low-as-reasonably achievable exposures

General design considerations and methods employed to maintain in-Plant radiation exposures ALARA have two objectives :

- minimizing the necessity for an amount of personnel time spent in radiation areas,

- minimizing radiation levels in routinely occupied Plant areas and in the vicinity of Plant equipment expected to require personnel attention.

Equipment and facility designs are considered in maintaining exposures ALARA during Plant operations including : normal operation, maintenance and repairs, refuelling operations and fuel storage, inservice inspection and calibrations, radioactive waste handling and disposal, and other anticipated operational occurrences. The actual design features employed are described in general in Section 12.3.

Experiences and data from operating plants are evaluated to decide if and how equipment or facility designs could be improved to reduce overall Plant personnel exposures. In particular, profit has been taken from experience of 900 MW French Nuclear Power plants. Methods to mitigate such exposures are implemented wherever possible and practical.
In designing radioactive systems, many factors must be considered. Important considerations are to reduce the need for equipment maintenance, to minimize necessary maintenance times, and to lower the radiation levels in which maintenance and other operational activity are performed.

12.1.2.2. Equipment general design considerations

Equipment general design considerations to minimize the necessity for the amount of personnel time spent in a radiation area include :

- reliability, durability, construction, and design features of equipment, components, and materials to reduce or eliminate the need for repair or preventive maintenance,

- servicing convenience for anticipated maintenance or potential repair, including ease of disassembly and modularization of components for replacement or removal to a lower radiation area for repair,

- redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high and when no feasible method is available to reduce radiation levels,

- provisions, where practicable, to remotely or mechanically operate, repair, service, monitor, or inspect equipment.

Equipment general design considerations directed toward minimizing radiation levels proximate to equipment or components requiring personnel attention include :

- provisions for draining, flushing, or, if necessary, remote cleaning of equipment containing radioactive material,

- design of equipment, piping, connections, and valves to minimize the buildup of radioactive material,

- provisions for isolating equipment from radioactive process fluids, 320

- provisions for minimizing the spread of contamination into equipment service areas including direct drain connections,

- utilization of high quality valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials.

12.1.2.3. Facility layout general design considerations

Facility general design considerations to minimize the amount of personnel time spent in a radiation area include :

- locating equipment, instruments, and sampling stations which will require routine maintenance, calibration, operation, or inspection, for ease of access and minimum required occupancy time in radiation areas,

- laying out Plant areas to allow remote or mechanical operation, service, monitoring, or inspection of highly radioactive equipment,

- providing, where practicable, for transportation of equipment or components requiring service to a lower radiation area.

Facility general design considerations directed toward minimizing radiation levels in Plant access areas and in the vicinity of equipment requiring personnel attention include :

- separating radiation sources and occupied areas where practicable (e.g., pipes or ducts containing potentially highly radioactive fluids do not pass through occupied areas),

- providing adequate shielding between radiation sources and access and service areas,

- where appropriate, separating equipment or components in service areas with permanent shielding,

- locating equipment, instruments and sampling sites in the lowest practicable radiation zone,

- providing means and adequate space for utilizing movable shielding for sources within the service area when required,

- providing means for decontamination of service areas,

- providing means to control contamination and to facilitate decontamination of potentially contaminated areas.

12.1.2.4. Framatome design considerations

The basic philosophy embodied in pressurized water reactor design considerations to ensure that occupational radiation exposures are as low as reasonably achievable can be expressed as :

- design of systems and components to ensure increased reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components,

- design of systems and components to reduce the radiation fields to ensure that operation, maintenance, and inspection activities are performed in the minimum radiation field feasible,

- design of systems and components to reduce the time spent in radiation fields during operation, maintenance, and inspection,

- design of systems and components to accomodate remote and semi-remote operation, maintenance, and inspection procedures.

In translating this design philosophy into practice, Framatome and Electricite de France (EDF) have used experience from past designs operating in the field and upon other relevant field experience as well as laboratory tests. The feedback and radiation data from operating plants is used to construct a model to predict occupational radiation exposure patterns for various operation, maintenance, and inspection activities. This model and exposure patterns are further described in Section 12.4. In consequence, the potential for improvements in areas such as reliability, repair time, and operational techniques related to occupational radiation exposures can be identified for further study.

Recommended design practice and design considerations are communicated to the system and component designers of EDF and Framatome.

12.1.3. Operational considerations

Radiological health and safety procedures will be developed and continually reviewed to assure that occupational radiation exposures are as low as reasonably achievable. These procedures will be based on experience gained from operating plants. Improvements suggested during operation will be incorporated and implemented to continually update the program to enlarge protection of the workers.

The operational plans and procedures influence the design of the facility. Wherever possible, structures, equipment, and shielding minimize the duration of work in a radiation area and reduce the radiation levels in areas requiring normal access. Design considerations are based on data and reports from operating plants as discussed in Subsection 12.1.2.

FIGURES

12.2. RADIATION SOURCES

The radioactive sources described in this section are those taken into account for design calculations of biological shieldings. Sources during normal operating conditions are given in 12.2.1.1. and 12.2.1.2. and the detailed study of these sources can be found in Reference 1.

Assumptions which are discussed in Paragraph 12.2.1.3. are used for evaluation of accessibility in main rooms under a post-accidental situation (LOCA).

12.2.1. Contained sources

The radiation sources which provide the necessary design data for determining the location and specifications of biological shields are based upon three Plant conditions of normal full power operation, shut-down and design basis event (for example a loss of coolant accident - LOCA).

12.2.1.1. Sources for normal full power operation

The main sources of activity during normal full power operation are the reactor core, the reactor coolant, the nuclear auxiliary systems, and spent fuel stored in the fuel building.

The values of the sources from various nuclear auxiliary systems depend on design and operating conditions of the corresponding systems.

The main hypotheses postulated for the calculation of the sources related to the core, the reactor coolant system and miscellaneous auxiliary components are those in Reference 2.

Radioactive sources hypotheses are based on the design equivalent cladding failure rate of 1 %.

Radiation levels are calculated at equilibrium (at the end of the third cycle) on the basis of the following conservative assumptions :

- RCV purification efficiency is low,

- the RCV tank gaseous phase is not scavenged throughout the whole cycle,

- no reactor coolant is taken off towards the TEP system to follow the fluctuation in boron concentration.

The computer code ACTIVI (see Reference 1) is used to compute gamma ray radiation energy emitted, and divide it into energy bands.

The chief activity sources in the reactor building are neutron and gamma fluxes from the operating reactor, Nitrogen-16 produced by activation of oxygen in the reactor coolant, the fission products resulting from cladding failure and corrosion products activated during their passage through the core.

12.2.1.1.1. Neutron and gamma fluxes

Neutron and gamma fluxes in the reactor internals, the pressure vessel and the primary concrete shield have been calculated by means of the DOT computer code (See Reference 3).

This two-dimensional computer code has been used in the configuration (R, Θ) which gives the neutron and gamma fluxes as radial variations at 0°, and in the (X, Y) configuration which gives neutron and gamma fluxes as azimuthal variations at different radii in the reactor pit.

Figure F-12.2-1 shows a horizontal section view of one-eighth of the reactor at the core midplane.

12.2.1.1.1.1. Neutron flux

The radial variations of neutron flux in the reactor internals and pressure vessel (at angle 0°) are shown in Figure F-12.2-2. Figure F-12.2-3 shows the radial variations of the neutron flux in the primary concrete shield (at angle 0°).

Figure F-12.2-4 gives the azimuthal variations of fast neutron flux in the reactor internals and the pressure vessel. This information is given for thermal neutron flux in Figure F-12.2-5.

12.2.1.1.1.2. Gamma flux

Gamma flux is calculated for 13 energy bands, ranging from 0 to 10 MeV.

The main feature of gamma flux is heat generation.

Figure F-12.2-6 shows the evolution of gamma heating in the reactor internals and the reactor pressure vessel at azimuthal angles of 0°, 22° and 45°. Figure F-12.2-7 shows the evolution of gamma heating in the primary shield concrete (at angle 0°).

12.2.1.1.2. Reactor coolant water activity

The source of radiation associated with the reactor coolant water has been divided into three separate groups of activity. They are the primary coolant Nitrogen-16 (N-16) activity, the acti vity of the primary coolant excluding N-16 and the pressurizer activity.

12.2.1.1.2.1. Primary coolant Nitrogen-16 activity

When the reactor coolant water passes through the core, the three natural isotopes of oxygen, O-16, O-17, and O-18 undergo a capturing process which produces N-16, N-17 and O-19 respectively.

Examination of the isotopic abundances and the absorption cross sections of the various natural isotopes of oxygen show that the reaction O-16 (n, p), N-16 is the most common. The N-16 radionuclide is a gamma emitter with a half life of T = 7.14 s. When disintegration of a N-16 nucleus occurs, out of 75 gamma rays produced, 69 have an energy of 6.13 MeV, 5 have an energy of 7.11 MeV and 1 has an energy of 2.75 MeV ; thus, the average energy is 6.15 MeV with a yield of 0.75.

Nitrogen-16 activity of coolant water is the predominant radiation used in the design of the main reactor coolant equipment shielding. However, in the case of the pressurizer, N-16 is not taken into account, since the pressurizer water inventory makeup rate is low, and the radioactive half life of B-16 is relatively short.

Nitrogen-16 activity is plotted in Figure F-12.2-8 as a function of the primary coolant water transport time in a reactor coolant loop.

The maximum nitrogen-16 activity, which is 4.033×10^{12} Bq/m^3 (109 Ci/m^3) is reached at the core outlet. Table T-12.2-1 indicates the energy from Nitrogen-16 in the reactor coolant system.

12.2.1.1.2.2. Activity of primary coolant excluding Nitrogen-16

The sources of radiation in the primary coolant other than N-16 are fission products released from fuel and corrosion products which are circulated in the reactor coolant.

a) Fission products

Fission products (gases and solids) are produced in the fuel elements (UO_2). Fission products migrate in the elements, and a certain amount accumulates in pellet-cladding gaps and is retained inside the Zircaloy-4/ZIRLO cladding. If there are cladding defects, fission products are released 116
into the reactor coolant water

Radiation shielding calculations are based on a cladding failure rate of 1 % (See Reference 2) ; the maximum fission product activity in the reactor coolant is 1,03 x 10^{13} Bq/t (279 Ci/t) of which 1,07 x 10^{12} Bq/t (29 Ci/t) is due to solid fission product activity, and 9,25 x 10^{12} Bq/t (250 Ci/t) to gaseous fission product activity (8,51 x 10^{12} Bq/t (230 Ci/t) for xenon-133).

The isotopic inventory of fission products present in the primary coolant differs slightly from that indicated in the PSAR (Chapter 11) ; this evolution is due to the fact that fissions of plutonium are now taken into account in theoretical calculations (see Reference 2).

b) Corrosion products

Radioactive corrosion products are formed in two ways. One way is by the corrosion of materials outside the core. These materials release particles which are carried by the reactor coolant and are activated as the coolant circulates through the core. The second way is by activation of the constituent materials of the core itself followed by corrosion.

These radioactive corrosion products may be conveyed by the reactor coolant, in a soluble or insoluble form, or form deposits.

The activity due to corrosion products has been calculated by means of the PACTOLE 8 computer code (see Reference 2) for the five isotopes whose activity is not negligible (Cr 51, Mn 54, Co 58, Fe 59 and Co 60).

The reactor coolant equilibrium fission and corrosion product activities are listed in Table T-12.2-2.

Energies emitted due to fission and corrosion products in the reactor coolant (except for Nitrogen-16) are given in Table T-12.2-2.

12.2.1.1.2.3. <u>Pressurizer activity</u>

The pressurizer activities for the liquid and vapor volumes are given in Table T-12.2-2.

12.2.1.1.3. <u>Sources in other systems</u>

The main nuclear systems the activity of which is to be taken into account under full power conditions include :

- chemical and volume control system (RCV),
- boron recycle system (TEP),
- waste treatment systems (TEU, TES, TEG),
- steam generator blowdown system (APG),
- vent and drain system (RPE).

Activities in other nuclear auxiliary systems of lesser importance from a radiation point of view (such as the reactor boron and water makeup system (REA), the nuclear sampling system (REN) and the liquid waste discharge system (TER)) are not computed, but they can be appreciated on the basis of the sources described in the previous paragraph and depend on the operating conditions of the various systems.

Radiation sources in the various pumps, in terms of energy per unit of active source volume, are considered identical to the liquid sources in the tanks from which the pumps take suction.

A minimum transit time of 60 seconds from the reactor coolant system to the point at which the letdown line exits from the reactor building, ensures that Nitrogen-16 activity has dropped to a negligible level compared to that of other radionuclides.

Sources of activity are discussed for the following nuclear systems :

a) RCV

The RCV provides continuous purification of reactor coolant water and consists of regenerative, excess and letdown heat exchangers, mixed bed and cation bed demineralizers, letdown primary and resin retention primary filters, a volume control tank, and charging pumps.

Except for regenerative heat exchanger, in which the main activity source is nitrogen 16 (see Reference 1), the RCV source activity is the reactor coolant inventory with 1 % defective fuel cladding. Table T-12.2-3 gives the radiation energy emitted by radionuclides (without Nitrogen-16) contained in letdown coolant sources.

Table T-12.2-3 indicates the major radiation sources for the RCV.

b) TEP

The TEP processes the reactor coolant by means of demineralization and gas stripping before evaporation, which separates and reclaims the boric acid and the reactor coolant makeup water. The system collects, decontaminates, deaerates, and pumps the effluent to the evaporator package, which separates the effluents into two liquid phases : low activity distillates and concentrates which are conveyed to the REA system (to the makeup water storage tank and the 4 % boric acid tanks).

Radiation sources in the TEP are listed in Table T-12.2-4.

c) TEU

The liquid waste treatment system (TEU) stores and processes non-reusable liquid effluents from the reactor coolant system, which are mainly collected by the vent and drain system (RPE) in the two reactor Units.

There are four categories of liquid effluents :

- process drains,
- floor drains,
- service effluents,
- chemical effluents.

Drains (floor, miscellaneous and process drains) are the most radioactive effluent stored in the TEU head storage tanks, and process drain activity is equivalent to 32 % of deaerated reactor coolant activity without decay. Effluents treated by the TEU are processed in the waste evaporator and the liquid concentrates routed toward the solid waste treatment system (TES) storage tanks to be drummed.

Radiation sources in the TEU system are tabulated in Table T-12.2-5.

d) TES

The solid waste treatment system (TES) processes four categories of active waste : spent resins, evaporator concentrates, spent filter cartridges and miscellaneous solid wastes which are directly drummed. The most radioactive resins are produced by the RCV demineralizers ; concentrates come from TEU system and in exceptional circumstances from the TEP system.

Radiation sources in the TES system are tabulated in Table T-12.2-6.

e) TEG

Hydrogen, radioactive gases and nitrogen are collected by the vent and drain system RPE or extracted from the reactor coolant by gas stripping in the volume control tank or in the TEP gas stripper.

These gases pass through a buffer tank before being stored in storage tanks for decay and further release to the ventilation system DVN.

Aerated vents collected by the vent and drain system RPE are filtered by TEG filters before being released.

Preliminary radiation sources values in the TEG system are tabulated in Table T-12.2-7.

f) APG

Steam generator blowdown system radiation sources are based on the removal of radioactive products contained in the steam generator liquid phase. It is assumed that noble gases are in totality carried away by the steam.

Gamma energy emitted by the steam generator liquid phase is given in Table T-12.2-9.

12.2.1.1.4. Sources in the main steam supply system

Radioactivity in the main steam supply system may result from a steam generator tube leakage concurrent with a 1 % fuel cladding failure. The assumptions taken into account for the calculation of the secondary side activity are listed in Section 11.1.3. Secondary side activity is assumed to be negligible with regards to radiation shielding.

12.2.1.2. Sources for shutdown conditions

In the reactor shutdown condition, the only additional significant sources requiring shielding consideration are the spent reactor fuel, the residual heat removal system (RRA), and the reactor cavity and spent fuel pit cooling system (PTR).

- Deposited crud material is more predominant in systems which operate at high temperature (reactor coolant system, letdown line of the chemical and control volume system), than in other systems. In systems which operate at high temperatures, radiation sources are bounded by the sources given for full power operation with the exception of a short time period (i.e., less than 24 hours) following shutdown during which crud bursts can result in increased radiation.

- When a pressurized water reactor containing defected fuel cladding has been operated for a time and the power is decreased, the activity of certain fission products (iodine, cesium , xenon, krypton) in the reactor coolant is often observed to increase. This radioisotope behavior is commonly referred to as the iodine spiking phenomenon, because the most important of these isotopes, from a biological point of view, are the iodines (especially I-131) (See Reference 4).

  Special shielding to accommodate these increases in activity is not necessary due to several factors. These factors are :

  . The spike or crud burst release is of short duration.

  . The letdown flowrate at shutdown of the chemical and control volume system is increased (27,2 m3/h instead of 13,6 m3/h).

  . Measurements taken from operating plants for cycles with important fuel cladding defects recorded (TIHANGE, FESSENHEIM, BUGEY) and analytical results from experimental test loops make it possible to compare the design activity corresponding to an equivalent fuel cladding failure rate of 1 % failed fuel and the fission product reactor coolant activity during steady state operation. A comparison can also be made between the design activity and the activity after a transient with a variation of more than half the reactor power.

Design activity * (Ci/t)		Ratios of activities to the design activity	
		Steady state	Transient
Noble gases	270,2	1/100	1/50
Iodines	10,8	1/20	1/2
Cesiums	2,55	1/10	1/2

* 1 Ci/t = $3{,}7 \times 10^{10}$ Bq/t

The reactor coolant activity is limited by Technical Specifications to less than the design base activity.

12.2.1.2.1. Reactor cavity and spent fuel cooling system (PTR)

Reactor cavity and spent fuel pit cooling system (PTR) maintains pit water chemistry and removes activity released during refuelling operations and the subsequent fuel cooling period by filtration and demineralization for the spent fuel pit and filtration for the reactor cavity.

At shutdown, reactor coolant water is deaerated and run through RCV filters and demineralizers at the maximum flow rate for a period of twenty hours before being added to the decontaminated water from tank PTR 001 BA.

Energy emitted by reactor cavity and spent fuel pit water is indicated in Table T-12.2-8.

12.2.1.2.2. Residual heat removal system (RRA)

During the various cooldown phases of an Unit, the RRA system is aligned to the reactor coolant system.

During phase 1 , from rod drop to the point when P = 28 bars and T = 177 °C, the heat of the reactor coolant is removed by the steam generators. After this point the values of P and T allow the RRA circuit to be used. The duration of phase 1 is longer than 4 hours.

Source strength calculations for the RRA system operation are made using cold shutdown conditions and the following conservative assumptions :

- No dilution of the reactor coolant by the PTR system.

- No purification of the reactor coolant by the TEP system demineralizers and deaerators.

- The reactor coolant is only purified by the chemical and volume control system (mixed and cation bed demineralizers for elimination of solid products, and deaeration by the volume control tank).

The activity of each isotope decreases according to an exponential law given by the formula :

$$Ai\ (t) = Aie^{-Dit}$$

where Ai is the activity at shutdown and, Di the decay constant, which is the sum of the radioactivity constant and of the purification constant.

A natural decay of 4 hours is taken into account (before the RRA system is put into service).

The maximum specific source strengths in the residual heat removal loop are given in Table T-12.2-11.

For lengthy cold shutdowns, the gamma rays emitted have energies below 1,7 MeV.

12.2.1.2.3. Fuel activity after reactor shutdown

Gamma ray sources emitted by spent fuel assemblies come from three parts : fissile part and upper and lower metal structures which are activated by neutron flux in the reactor core.

General assumptions concerning fissile part characteristics and upper and lower metal structures composition are enumerated in Reference 1.

Shielding requirements for spent fuel assembly storage are based on the activity present 4 days after shutdown to conservatively take credit for the time elapsed prior to the initiation of refuelling operations.

The values which are used for calculating the shielding (water height and concrete wall thickness) in the fuel building for handling spent fuel at the time of refuelling are given in Table T-12.2-10.

Gamma ray spectra emitted by the assembly after one year of cooling are also given in this table.

12.2.1.3. Assumptions related to premise accessibility under post-accidental conditions (LOCA)

These assumptions correspond to a hypothetical accidental situation with damaged fuel that leads to high contamination of the primary coolant.

12.2.1.3.1. Activity level of sump fluid and containment atmosphere under post-accidental conditions

- The "source term" taken into account represents a margin of conservatism for activity widespread during hypothetical design basis events taken as design conditions. It is the source term given by the French Basic Safety Rule n° V 1.a.

- The core inventory corresponds to an equilibrium cycle at the end of core life.

- Fractions of this inventory released from the fuel for each family of fission products appear in Table T-12.2-12.

- It is assumed that 90 % of the iodines released in the containment are elemental iodines and 10 % are particulate and organic iodine.

12.2.1.3.1.1. Recirculated coolant activity

The following assumptions are used to calculate the activity of the recirculated fluid from the containment sump to the core.

- Primary coolant is completely deaerated (elemental form).

- 50 % of the iodines are released out of the fuel, i.e. 45 % of the iodines initially present in the fuel, can be found in recirculated coolant.

- The volume of fluid in recirculation is estimated to be 1792 m3 of which :

  1460 m3 is PTR water,
  250 m3 is reactor coolant,
  82 m3 is from the safety injection system (RIS) accumulators.

Specific activity concentrations and gamma energy spectra after 1 hour (as a reminder), 1 day and 1 month of recirculated water are given in Table T-12.2-12.

12.2.1.3.1.2. Containment atmosphere activity

The following assumptions are taken into account for containment atmosphere activity calculation :

- 100% of noble gases released from the fuel (i.e. 50% of noble gases initially present in the core), are directly available in the containment atmosphere,

- 10% of iodines(particulate and organic form) released out of fuel, i.e. 5% of iodines initially present in the fuel can be found in the containment atmosphere,
- containment free volume is estimated to be 50,000m^3 81

  Specific activity concentrations and gamma energy spectra of the containment atmosphere are given in Table T-12.2-12.

12.2.1.3.2. Control room sources

The shielding requirements for the control room are dictated by the post-LOCA dose to control room personnel from direct gamma radiation from the containment, the external radioactive cloud, and control room airborne activity. The source terms are discussed in Subsection 15.6.5 and Appendix 15.A.

12.2.2. Airborne radioactive material sources

Airborne contamination is due to : 320

- leakages from material and systems conveying radioactive fluids.
- activation of Argon contained in the air of the reactor pit ventilation (EVC),
- steam leakages in the turbine hall in the case of secondary side contamination by primary coolant in the steam generators.

This subsection deals with the models, parameters and sources required to evaluate airborne concentrations of radionuclides during plant operations in various plant radiation areas.

Airborne radioactive sources have been considered in Chapter 11 for their contribution to the plant effluent releases.

Specific activity of the isotopes which take part in different building's airborne activity are listed in Chapter 11.

Two sets of assumption are taken into account in this evaluation, according to whether the unit operates under design conditions of activity or under average conditions (see Chapter 11).

When the activity of the reactor coolant equals design values, measures will be taken in order to reduce the amount of primary coolant leakages.

An example of an airborne activity calculation in the reactor building is given in Table T-12.2-14, taking into account a 6.5 kg/h reactor coolant leakage rate within the containment. 320

For any isotope, the differential equation governing concentration versus time is expressed as follows :

$$\frac{dc}{dt} = S - \lambda C - \lambda' C$$

Where:

C is the concentration in Ci/m^3 in the region,

S is the source expressed in $Ci/m^3/s$,

λ is the radioactive constant in s^{-1},

λ' is the ventilation removal constant. This is the constant of radioactive constant, expressed in s^{-1} 320

The source is expressed as :

$$S = \mu \frac{q}{3,600} \frac{A}{V}$$

Where :

μ is the isotope partition factor,

q is the leak rate in the region(kg/h),

A is the activity concentration of the leaking isotope(Ci/kg),

V is the volume of the region(m^3).

In the case of Argon-41, radioactivity in the atmosphere is due to a different process(See Appendix 12A-1).

12.2.2.1. Reactor building

Containment cleanup(EVF) and mini-sweeping by the containment atmosphere control(ETY) system are useful to reduce contamination levels in the containment atmosphere. One or both may be used prior to a personnel intervention inside the containment annulus.

During normal operation, the EVF system is started-up if iodine contamination rises above a preset threshold, whose value depends on the type and the duration of intervention inside the containment. This value could be 5,55 x 10^3 Bq/m3 (1,5 x 10^{-7} Ci/m3).

Startup of the mini-sweeping system (ETY) must be considered as soon as the activity concentration of noble gases exceeds a set limit. The containment clean-up system (EVF) must have been put into service previously, thereby guaranteeing a low activity concentration in iodine isotopes (lower than 3,7 x 10^3 Bq/m3 I-131 (10^{-7} Ci/m3)), and reducing release of iodines to outside atmosphere.

Note that the maximum permissible concentration in I-131 is 7,4 x 10^1 Bq/m3 (2 x 10^{-9} Ci/m3) for a person who must stay during 168 hours in a contaminated atmosphere.

Conservative airborne activity concentrations in the containment atmosphere under stable conditions, with permanent clean-up ventilation and no mini sweeping, are given in Table T-12.2-14.

In the event of modulated operation of mini-sweeping, the dose rate in the reactor containment varies between a maximum value corresponding to startup of mini-sweeping, and a minimum value for shut-off of mini-sweeping. The maximum value is greatly reduced by permanent clean-up ventilation, whereas the minimum value only varies slightly.

12.2.2.2. Nuclear auxiliary building (NAB)

By concentrating all the activity of the leaks in the ventilation ducts, the average air activity in the ventilation ducts is 10^5 Bq/m3 of Xe-133 (2,7 microcurie/m3) and 4,03 x 10^1 Bq/m3 of iodines (1,09 x nanocurie/m3), taking into account primary coolant design activities (1 Ci/t equivalent of I-131).

Under average operating conditions, the average air activity becomes : 3 x 10^3 Bq/m3 of Xe-133 (8,1 x 10^{-8} Ci/m3) and 1,21 Bq/m3 of iodines (3,27 x 10^{-11} Ci/m3).

Note that radioactive decay is not considered in these values.

12.2.2.3. Turbine hall

For the primary-to-secondary leakage in the steam generators noted in Table T-12.2-13, the activity of secondary steam is 4,33 x 10^2 Bq/kg (1,17 x 10^{-8} Ci/kg).

Assuming a secondary leak of 22 t/h in the form of steam, completely located in the turbine hall, I-131 concentration in the turbine hall atmosphere under stable conditions equals 9,09 Bq/m3 (2,46 x 10^{-10} Ci/m3). These values are calculated on the basis of design values of primary coolant activity. Under average operating conditions, I-131 concentration in the turbine hall becomes 2,73 x 10^{-1} Bq/m3 (7,37 x 10-12 Ci/m3).

12.2.3. References

1 - Radiation shielding design report - 12 CDR 04

2 - Note TP/SS/DC.0071 Revision E
900 MWe Nuclear Reactors - Radiation source term data at power and reactor shutdown.

3 - F. R. Mynatt, W. W. Engle, M. L. Gritzner, W. A. Rloades and R. J. Rogers. The DOT-III Two-dimensional Discrete Ordinates Transport code, ORNL-TM-4280 (1973).

4 - R. Leroy, G. Beuken, C. Collette, C. Dehon - "Transient Iodine Behavior in the Primary Coolant of the Tihange Pressurized Water Reactor" ENS/ANS International Topical Meeting on Nuclear Power Reactor Safety - Brussels, October 16-19, 1978.

APPENDIX 12A-1

MEAN ACTIVITY OF ARGON-41 IN THE REACTOR BUILDING ATMOSPHERE

The major activity resulting from exposure of air to neutrons is the Argon-41 (A-41) isotope formed from the Argon in air immediately surrounding the reactor vessel.

The Argon is activated by thermal neutrons through the reaction :

A-40 (n, γ) A-41

with a microscopic absorption cross section of 0,53 bars, according to : the following equation

$$P = \frac{\Sigma a \quad \theta tn \quad Vi \quad \lambda}{3{,}7 \times 10^{+4}}$$

Where :

P = A-41 production rate (Bq/s),
Σ a = macroscopic cross section of Argon in air (cm^{-1}), (air contains 0,924 % by volume of Argon or 1,8 x 10^{+17} atoms of a cubiccentimetre of air),
Σa = 1,8 x 10^{+17} x 0,53 x 10^{-24} = 9,54 x 10^{-8}
tn = thermal neutron flux (n/cm2/s) = 2 x 10^{+9},
Vi = irradiated volume air between vessel insulation and prima shield wall,
Vi = 10^{+7} cm3,
λ = decay constant = 1,055 x 10^{-4} s^{-1}.

Equilibrium activity is establish when the production in Vi equals the decay in the entire containment atmosphere (assuming uniform mixing) according to the following equation :

$$A = \frac{P}{\lambda V}$$

Where :

A = A-41 equilibrium concentration in the containment (Bq/m3)

V = volume of the containment atmosphere (m3) : 5,04 x 10^{+4} m3

A = 3,77 x 10^4 Bq/m3 (1,02 x 10^{-6} Ci/m3)

TABLE T-12.2-1

ENERGIES EMITTED BY NITROGEN-16
IN THE REACTOR COOLANT SYSTEM

Location	Average radiation intensity (MeV/cm3/s)
Hot leg pipe	$1,82 \times 10^{+7}$
Steam generator channel head inlet plenum	$1,71 \times 10^{+7}$
Steam generator U-tubes	$1,44 \times 10^{+7}$
Steam generator channel head outlet plenum	$1,19 \times 10^{+7}$
Between steam generators and reactor coolant pumps	$1,13 \times 10^{+7}$
Reactor coolant pump	$1,07 \times 10^{+7}$
Cold leg pipe	$1,02 \times 10^{+7}$

TABLE T-12.2-2 (1/11)

CHARACTERISTICS OF PRIMARY COOLANT SOURCES

a) Reactor coolant water source strength (except Nitrogen 16)

Average band energy (MeV)	Energy emitted per unit of time and mass (MeV/g/s)
0,07	$5,51 \times 10^{+5}$
0,61	$3,22 \times 10^{+5}$
0,99	$3,18 \times 10^{+5}$
1,53	$1,25 \times 10^{+5}$
1,91	$1,04 \times 10^{+5}$
2,32	$1,45 \times 10^{+5}$
2,72	$3,72 \times 10^{+4}$

TABLE T-12.2-2 (2/11)

b) Specific activities of the radionuclides contained in the reactor coolant (except Nitrogen 16)

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Cr 51	27,8 d	1,78 x 10^{-4}
Mn 54	280 d	1,84 x 10^{-5}
Fe 59	45 d	4,47 x 10^{-6}
Co 58	71 d	2,08 x 10^{-3}
Co 60	5,24 yr	1,42 x 10^{-3}
Br 83	2,3 h	7,82 x 10^{-2}
Br 84	32 mn	3,57 x 10^{-2}
Br 87	56 s	2,28 x 10^{-3}
Kr 83 m	1,86 h	4,11 x 10^{-1}
Kr 85 m	4,4 h	1,63
Kr 85	10,4 yr	8,4 x 10^{-1}
Kr 87	78 mn	1

TABLE T-12.2-2 (3/11)

b) Specific activities of the radionuclides contained in the reactor coolant (except Nitrogen 16)

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Kr 88	2,8 h	2,82
Rb 88	18 mn	2,8
Rb 89	15 mn	8,12 x 10^{-2}
Sr 90	28 yr	6,34 x 10^{-5}
Sr 91	9,7 h	1,53 x 10^{-3}
Sr 92	2,6 h	7,18 x 10^{-4}
Y 90	64,2 h	1,25 x 10^{-4}
Y 91	58 d	4,63 x 10^{-3}
Y 92	3,5 d	9,04 x 10^{-4}
Y 93	10,3 h	6,20 x 10^{-4}
Y 94	20 mn	2,05 x 10^{-5}
Zr 95	65 d	6,32 x 10^{-4}
Nb 95	35 d	6,14 x 10^{-4}

TABLE T-12.2-2 (4/11)

b) Specific activities of the radionuclides contained in the reactor coolant (except Nitrogen 16)

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Nb 97 m	1 mn	9,60 x 10^{-7}
Nb 97	72 mn	7,44 x 10^{-5}
Mo 99	67 h	5,75
Tc 99 m	6 h	3,11
Ru 103	40 d	5,61 x 10^{-4}
Ru 105	4,44 h	1,12 x 10^{-4}
Rh 106	30 s	1,2 x 10^{-4}
Te 129	1,1 h	7,31 x 10^{-3}
Te 131	25 mn	8,49 x 10^{-3}
Te 132	78 h	2,94 x 10^{-1}
Te 134	44 mn	2,87 x 10^{-2}
I 129	1,6 x 10^{7}yr	3,10 x 10^{-8}
I 131	8,05 d	2,82

TABLE T-12.2-2 (5/11)

b) Specific activities of the radionuclides contained in the reactor coolant (except Nitrogen 16)

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
I 132	2,3 h	1,07
I 133	21 h	4,28
I 134	53 mn	6,09 x 10^{-1}
I 135	6,7 h	2,42
Xe 131 m	12 d	2,52
Xe 133 m	2,3 d	3,43
Xe 133	5,27 d	2,3 x 10^{2}
Xe 135 m	15,6 mn	4,94 x 10^{-1}
Xe 135	9,2 h	6,39
Xe 137	3,9 mn	1,92 x 10^{-1}
Xe 138	17 mn	6,45 x 10^{-1}
Cs 134	2,2 yr	1,31
Cs 136	13 d	1,31

TABLE T-12.2-2 (6/11)

b) Specific activities of the radionuclides contained in the reactor coolant (except Nitrogen 16)

* 1 Ci/t = $3,7 \times 10^{10}$ Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Cs 137	30 yr	1,02
Cs 138	32,2 mn	$9,67 \times 10^{-1}$
Ba 137 m	2,6 mn	1,02
Ba 139	85 mn	$1,85 \times 10^{-2}$
Ba 140	12,8 d	$4,19 \times 10^{-3}$
La 140	40,2 h	$1,39 \times 10^{-3}$
Ce 141	32,5 d	$6,37 \times 10^{-4}$
Ce 143	33 h	$4,69 \times 10^{-4}$
Ce 144	285 d	$3,02 \times 10^{-4}$
Pr 144	17,3 mn	$3,02 \times 10^{-4}$
Nd 147	11,1 d	$2,47 \times 10^{-4}$
Sm 153	46,2 h	$2,63 \times 10^{-5}$
Eu 156	15 d	$7,15 \times 10^{-6}$

TABLE T-12.2-2 (7/11)

c) Specific activities of the radionuclides contained in the pressurizer

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Cr 51	27,8 d	1,78 x 10^{-4}
Mn 54	280 d	1,84 x 10^{-5}
Fe 59	45 d	4,47 x 10^{-6}
Co 58	71 d	2,08 x 10^{-3}
Co 60	5,24 yr	1,42 x 10^{-3}
Br 83	2,3 h	5,89 x 10^{-3}
Br 84	32 mn	6,3 x 10^{-4}
Br 87	56 s	1,25 x 10^{-6}
Kr 83 m	1,86 h	2,31 x 10^{-2}
Kr 85 m	4,4 h	2,20 x 10^{-1}
Kr 85	10,4 yr	8,02 x 10^{1}
Kr 87	78 mn	3,84 x 10^{-2}
Kr 88	2,8 h	2,38 x 10^{-1}

TABLE T-12.2-2 (8/11)

c) Specific activities of the radionuclides contained in the pressurizer

* 1 Ci/t = $3{,}7 \times 10^{10}$ Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Rb 88	18 mn	$2{,}78 \times 10^{-2}$
Rb 89	15 mn	$6{,}90 \times 10^{-4}$
Sr 90	28 yr	$6{,}31 \times 10^{-5}$
Sr 91	9,7 h	$3{,}73 \times 10^{-4}$
Sr 92	2,6 h	$6{,}05 \times 10^{-5}$
Y 90	64,2 h	$1{,}05 \times 10^{-4}$
Y 91	58 d	$4{,}53 \times 10^{-3}$
Y 92	3,5 h	$1{,}51 \times 10^{-4}$
Y 93	10,3 h	$1{,}59 \times 10^{-4}$
Y 94	20 mn	$2{,}18 \times 10^{-7}$
Zr 95	65 d	$6{,}20 \times 10^{-4}$
Nb 95	35 d	$6{,}14 \times 10^{-4}$
Nb 97 m	1 mn	$4{,}88 \times 10^{-10}$

TABLE T-12.2-2 (9/11)

c) Specific activities of the radionuclides contained in the pressurizer

* 1 Ci/t = $3,7 \times 10^{10}$ Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Nb 97	72 mn	$2,97 \times 10^{-6}$
Mo 99	67 h	3,97
Tc 99 m	6 h	3,43
Ru 103	40 d	$5,44 \times 10^{-4}$
Rh 105	4,44 h	$1,47 \times 10^{-5}$
Rh 106	30 s	$1,19 \times 10^{-4}$
Te 129	1,1 h	$2,78 \times 10^{-4}$
Te 131	25 mn	$1,18 \times 10^{-4}$
Te 132	78 h	$2,14 \times 10^{-1}$
Te 134	44 mn	$6,65 \times 10^{-4}$
I 129	$1,6 \times 10^{7}$yr	$3,8 \times 10^{-8}$
I 131	8,05 d	2,44
I 132	2,3 h	$2,76 \times 10^{-1}$

TABLE T-12.2-2 (10/11)

c) Specific activities of the radionuclides contained in the pressurizer

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
I 133	21 h	1,77
I 133	53 mn	1,83 x 10^{-2}
I 135	6,7 h	4,42 x 10^{-1}
Xe 131 m	12 d	2,06 x 10^{1}
Xe 133 m	2,3 d	5,53
Xe 133	5,27 d	8,67 x 10^{2}
Xe 135 m	15,6 mn	3,80 x 10^{-3}
Xe 135	9,2 h	1,74
Xe 137	3,9 mn	3,7 x 10^{-4}
Xe 138	17 mn	4,61 x 10^{-3}
Cs 134	2,2 yr	1,30
Cs 136	13 d	1,18
Cs 137	30 yr	1,01

TABLE T-12.2-2 (11/11)

c) Specific activities of the radionuclides contained in the pressurizer

* 1 Ci/t = 3,7 x 10^{10} Bq/t

Isotope	Half-life (T)	Activity in the reactor coolant (Ci/t)*
Cs 138	32,2 mn	1,73 x 10^{-2}
Ba 137 m	2,6 mn	1,01
Ba 139	85 mn	3,98 x 10^{-3}
Ba 140	12,8 d	3,82 x 10^{-3}
La 140	40,2 h	2,42 x 10^{-3}
Ce 141	32,5 d	6,14 x 10^{-4}
Ce 143	33 h	2,47 x 10^{-4}
Ce 144	285 d	3 x 10^{-4}
Pr 144	17,3 mn	3 x 10^{-4}
Nd 147	11,1 d	2,22 x 10^{-4}
Sm 153	46,2 h	1,61 x 10^{-5}
Eu 156	15 d	6,62 x 10^{-6}

TABLE T-12.2-3 (1/5)

RADIATION SOURCES : CHEMICAL AND VOLUME CONTROL SYSTEM (RCV)

a) Letdown coolant sources

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	5,51 x 10^{+5}
0,61	3,22 x 10^{+5}
0,99	3,18 x 10^{+5}
1,53	1,25 x 10^{+5}
1,90	1,04 x 10^{+5}
2,32	1,45 x 10^{+5}
2,70	3,72 x 10^{+4}

b) Mixed bed demineralizer

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,33	1,27 x 10^{8}
0,61	8,64 x 10^{7}
1,03	2,79 x 10^{7}
1,53	1,2 x 10^{7}
2,06	2,12 x 10^{6}
2,38	2,47 x 10^{4}
2,61	8,7 x 10^{4}

TABLE T-12.2-3 (2/5)

RADIATION SOURCES : CHEMICAL AND VOLUME CONTROL SYSTEM (RCV)

c) Cation bed demineralizer

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,12	1,1 x 10^{7}
0,68	2,1 x 10^{9}
0,9	1,72 x 10^{8}
1,37	5,91 x 10^{7}
1,91	4,88 x 10^{1}
2,21	2,71 x 10^{4}
2,67	1,81 x 10^{4}

d) Volume control tank

Steam phase

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,06	5,93 x 10^{+6}
0,49	1,16 x 10^{+4}
0,95	2,99 x 10^{+5}
1,53	1,51 x 10^{+5}
1,89	6,42 x 10^{+5}
2,33	9,37 x 10^{+5}
2,73	1,42 x 10^{+5}

TABLE T-12.2-3 (3/5)

RADIATION SOURCES : CHEMICAL AND VOLUME CONTROL SYSTEM (RCV)

Liquid phase

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	$5,51 \times 10^{+5}$
0,61	$3,22 \times 10^{+5}$
0,99	$3,18 \times 10^{+5}$
1,53	$1,25 \times 10^{+5}$
1,90	$1,04 \times 10^{+5}$
2,32	$1,45 \times 10^{+5}$
2,70	$3,72 \times 10^{+4}$

e) Letdown primary filter
(upstream of demineralizers)

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,32	$1,12 \times 10^{4}$
0,51	$0,92 \times 10^{6}$
1,01	$2,73 \times 10^{7}$
1,33	$1,98 \times 10^{7}$

TABLE T-12.2-3 (4/5)
RADIATION SOURCES : CHEMICAL AND VOLUME CONTROL SYSTEM (RCV)

f) Resin retention primary filter (downstream of the demineralizers)

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,31	$6,15 \times 10^{9}$
0,67	$5,23 \times 10^{8}$
0,91	$5,25 \times 10^{7}$
1,39	$1,92 \times 10^{7}$
2,06	$5,86 \times 10^{5}$
2,29	$1,77 \times 10^{4}$
2,62	$4,46 \times 10^{4}$

g) Regenerative heat exchanger

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	$4,24 \times 10^{+5}$
0,61	$2,48 \times 10^{+5}$
0,99	$2,45 \times 10^{+5}$
1,53	$9,63 \times 10^{+4}$
1,90	$8 \times 10^{+4}$
2,32	$1,12 \times 10^{+5}$
2,70	$2,87 \times 10^{+4}$
6,15	$7,62 \times 10^{+5}$ for module 1* $1,32 \times 10^{+4}$ for module 2*

Note : The regenerative heat exchange has two distinct modules with a different source strength in each module (see Reference 1 for more details)

TABLE T-12.2-3 (5/5)

RADIATION SOURCES : CHEMICAL AND VOLUME CONTROL SYSTEM (RCV)

h) Excess letdown heat exchanger

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	$1,05 \times 10^5$
0,61	$6,12 \times 10^4$
0,99	$6,05 \times 10^4$
1,53	$2,37 \times 10^4$
1,90	$1,98 \times 10^4$
2,32	$2,76 \times 10^4$
2,70	$7,07 \times 10^3$

i) Letdown heat exchanger

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	$9,31 \times 10^4$
0,61	$5,44 \times 10^4$
0,99	$5,37 \times 10^4$
1,53	$2,11 \times 10^4$
1,90	$1,75 \times 10^4$
2,32	$2,45 \times 10^4$
2,70	$6,28 \times 10^3$

TABLE T-12.2-4 (1/4)

RADIATION SOURCES : BORON RECYCLE SYSTEM (TEP)

a) Front holdup tank

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	$5,51 \times 10^{+5}$
0,61	$3,22 \times 10^{+5}$
0,99	$3,18 \times 10^{+5}$
1,53	$1,25 \times 10^{+5}$
1,90	$1,04 \times 10^{+5}$
2,32	$1,45 \times 10^{+5}$
2,70	$3,72 \times 10^{+4}$

b) Mixed bed demineralizers

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,33	$2,65 \times 10^{+6}$
0,58	$1,24 \times 10^{+6}$
1,07	$7 \times 10^{+5}$
1,43	$4,45 \times 10^{+5}$
2,10	$2,23 \times 10^{+4}$

TABLE T-12.2-4 (2/4)

RADIATION SOURCES : BORON RECYCLE SYSTEM (TEP)

c) Cation bed demineralizers

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,11	$3{,}19 \times 10^{+6}$
0,66	$9{,}4 \times 10^{+7}$
0,96	$1{,}26 \times 10^{+7}$
1,38	$3{,}74 \times 10^{+4}$

d) Head filters

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,21	$5{,}84 \times 10^{+6}$
0,66	$9{,}5 \times 10^{+7}$
0,96	$1{,}33 \times 10^{+7}$
1,43	$4{,}8 \times 10^{+5}$
2,10	$2{,}23 \times 10^{+4}$

TABLE T-12.2-4 (3/4)

RADIATION SOURCES : BORON RECYCLE SYSTEM (TEP)

e) Resin retention filters

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,32	$1,87 \times 10^{+2}$
0,51	$1,42 \times 10^{+4}$
1,06	$6,31 \times 10^{+5}$
1,33	$5,44 \times 10^{+5}$

f) Stripped water

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,2	$5,43 \times 10^{4}$
0,62	$1,43 \times 10^{5}$
1	$1,41 \times 10^{5}$
1,53	$4,85 \times 10^{4}$
2	$3,36 \times 10^{3}$
2,22	$1,34 \times 10^{4}$
2,73	$1,11 \times 10^{4}$

TABLE T-12.2-4 (4/4)

RADIATION SOURCES : BORON RECYCLE SYSTEM (TEP)

g) Evaporator activity

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,21	$4,06 \times 10^{+5}$
0,63	$9,2 \times 10^{+5}$
0,99	$7,76 \times 10^{+5}$
1,59	$1,26 \times 10^{+5}$
2,1	$1,71 \times 10^{+4}$
2,22	$2,55 \times 10^{+3}$
2,71	$1,92 \times 10^{+3}$

TABLE T-12.2-5

RADIATION SOURCES : LIQUID WASTE TREATMENT SYSTEM (TEU)

a) Process drain tanks

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,07	$1,76 \times 10^5$
0,61	$1,03 \times 10^5$
0,99	$1,02 \times 10^5$
1,53	$4,00 \times 10^4$
1,90	$3,33 \times 10^4$
2,32	$4,64 \times 10^4$
2,70	$1,19 \times 10^4$

b) Evaporator

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,22	$5,85 \times 10^5$
0,64	$1,06 \times 10^6$
0,97	$1,25 \times 10^6$
1,59	$8,35 \times 10^4$
2,1	$1,07 \times 10^4$
2,22	$1,55 \times 10^3$
2,70	$1,24 \times 10^3$

TABLE T-12.2-6

RADIATION SOURCES : SOLID WASTE TREATMENT SYSTEM (TES)

a) Spent resin tanks

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,31	$8,85 \times 10^{7}$
0,67	$7,52 \times 10^{8}$
0,93	$7,55 \times 10^{7}$
1,41	$2,76 \times 10^{7}$
2,05	$1,43 \times 10^{6}$
2,31	$2,55 \times 10^{4}$
2,61	$6,44 \times 10^{4}$

b) Concentrate tank

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,22	$5,85 \times 10^{5}$
0,64	$1,06 \times 10^{6}$
0,97	$1,25 \times 10^{6}$
1,59	$8,35 \times 10^{4}$
2,10	$1,07 \times 10^{4}$
2,22	$1,55 \times 10^{3}$
2,70	$1,24 \times 10^{3}$

TABLE T-12.2-7

RADIATION SOURCES : GASEOUS WASTE TREATMENT SYSTEM (TEG)

a) Buffer tank

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,055	$1,31 \times 10^6$
0,52	$1,18 \times 10^4$
0,95	$5,39 \times 10^4$
1,53	$2,89 \times 10^4$
1,88	$9,05 \times 10^4$
2,33	$1,75 \times 10^5$
2,80	$1,90 \times 10^4$

b) Storage tanks

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,054	$4,04 \times 10^6$
0,53	$2,61 \times 10^2$
0,95	$5,59 \times 10^1$
1,53	$5,64 \times 10^4$
1,88	$1,75 \times 10^1$
2,33	$3,42 \times 10^1$
2,76	$3,29 \times 10^1$

TABLE T-12.2-8

RADIATION SOURCES : REACTOR CAVITY AND SPENT FUEL PIT COOLING SYSTEM (PTR)

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,14	$0,08 \times 10^4$
0,67	$0,19 \times 10^4$
0,96	$0,16 \times 10^4$
1,47	$0,06 \times 10^3$
2,14	$0,34 \times 10^1$
2,34	0,01
2,53	0,05

TABLE T-12.2-9

RADIATION SOURCES : STEAM GENERATOR BLOWDOWN SYSTEM (APG)

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,21	$2{,}34 \times 10^{1}$
0,63	$5{,}31 \times 10^{1}$
0,99	$4{,}57 \times 10^{1}$
1,59	7,82
2,09	1,09
2,22	0,22
2,72	0,16

TABLE T-12.2-10

SPENT FUEL ASSEMBLY SPECIFIC ACTIVITY

Number of γ/cm3/s emitted by the assembly after 4 days of cooling

Energy band (MeV)	Fissile part	Lower structure	Upper structure
0,5 - 0,75	$7,62 \times 10^{11}$	$2,51 \times 10^{9}$	$2,72 \times 10^{9}$
0,75 - 1,25	$3,87 \times 10^{11}$	$1,34 \times 10^{10}$	$1,43 \times 10^{10}$
1,25 - 1,75	$1,98 \times 10^{11}$	$2,17 \times 10^{9}$	$1,90 \times 10^{9}$
1,75 - 2,25	$7,22 \times 10^{9}$	$3,47 \times 10^{4}$	$4,10 \times 10^{4}$
2,25 - 2,75	$7,16 \times 10^{9}$	0	0
2,75 - 3,5	$1,34 \times 10^{8}$	0	0

Number of γ/cm3/s emitted by the assembly after one year of cooling

Energy band (MeV)	Fissile part	Lower structure	Upper structure
0,5 - 0,75	$2,72 \times 10^{10}$	$4,97 \times 10^{7}$	$4,04 \times 10^{7}$
0,75 - 1,25	$2,12 \times 10^{10}$	$3,32 \times 10^{9}$	$3,8 \times 10^{9}$
1,25 - 1,75	$1,15 \times 10^{9}$	$1,5 \times 10^{9}$	$1,73 \times 10^{9}$
1,75 - 2,25	$5,28 \times 10^{8}$	0	0
2,25 - 2,75	$1,14 \times 10^{7}$	0	0
2,75 - 3,5	0	0	0

TABLE T-12.2-11

RADIATION SOURCES : RESIDUAL HEAT REMOVAL SYSTEM (RRA)

Average band energy (MeV)	Specific source strength (MeV/cm3/s)
0,06	$3,77 \times 10^{+5}$
0,63	$1,54 \times 10^{+5}$
0,99	$1,35 \times 10^{+5}$
1,58	$2,87 \times 10^{+4}$
1,91	$2,17 \times 10^{+4}$
2,33	$3,56 \times 10^{+4}$
2,76	$4,67 \times 10^{+3}$

TABLE T-12.2-12 (1/5)

ASSUMPTIONS RELATED TO ACTIVITY CHARACTERISTICS UNDER POST ACCIDENTAL CONDITIONS (LOCA)

a) Fractions of core inventory released from the fuel for each family of fission products in the event of a LOCA

Family of fission products	Core inventory percentage directly available out of fuel
Noble gases (Kr, Xe)	50
Halogens (I, Br)	50
Alkali metals (Cs, Rb)	50
Other products	0

TABLE T-12.2-12 (2/5)

ASSUMPTIONS RELATED TO ACTIVITY CHARACTERISTICS UNDER POST ACCIDENTAL CONDITIONS (LOCA)

b) Specific activity concentrations of coolant in recirculation (Ci/cm3)*

Isotope	1 hour	1 day	1 month
Cs 134 m	$1,06 \times 10^{-3}$	$4,0 \times 10^{-6}$	-
Cs 134	$4,15 \times 10^{-3}$	$4,14 \times 10^{-3}$	$4,03 \times 10^{-3}$
Cs 136	$1,10 \times 10^{-3}$	$1,05 \times 10^{-3}$	$2,19 \times 10^{-4}$
Cs 137	$2,23 \times 10^{-3}$	$2,23 \times 10^{-3}$	$2,22 \times 10^{-3}$
Cs 138	$1,07 \times 10^{-2}$	-	-
Cs 139	$3,74 \times 10^{-4}$	-	-
I 129	7×10^{-10}	7×10^{-10}	7×10^{-10}
I 131	$2,24 \times 10^{-2}$	$2,06 \times 10^{-2}$	$1,64 \times 10^{-3}$
I 132	$2,40 \times 10^{-2}$	$2,0 \times 10^{-5}$	-
I 133	$4,38 \times 10^{-2}$	$2,03 \times 10^{-2}$	-
I 134	$2,21 \times 10^{-2}$	$2,0 \times 10^{-10}$	-
I 135	$3,82 \times 10^{-2}$	$3,54 \times 10^{-3}$	-
I 136	-	-	
Kr 83 m	$2,22 \times 10^{-6}$	$6,0 \times 10^{-10}$	-
Kr 85 m	$7,7 \times 10^{-6}$	$2,0 \times 10^{-7}$	-
Kr 85	$2,0 \times 10^{-7}$	$2,0 \times 10^{-7}$	$2,0 \times 10^{-7}$
Kr 87	$1,0 \times 10^{-5}$	-	-
Kr 88	$1,85 \times 10^{-5}$	$6,0 \times 10^{-8}$	-
Xe 131 m	$3,5 \times 10^{-7}$	$3,3 \times 10^{-7}$	$5,85 \times 10^{-8}$
Xe 133 m	$2,95 \times 10^{-6}$	$2,2 \times 10^{-6}$	-
Xe 133	$1,4 \times 10^{-4}$	$9,2 \times 10^{-5}$	$1,9 \times 10^{-6}$
Xe 135 m	$1,34 \times 10^{-6}$	-	-
Xe 135	$2,35 \times 10^{-5}$	$4,15 \times 10^{-6}$	-
Xe 138	$5,0 \times 10^{-6}$	-	-

* 1 Ci/cm3 = $3,7 \times 10^{10}$ Bq/cm3

TABLE T-12.2-12 (3/5)

ASSUMPTIONS RELATED TO ACTIVITY CHARACTERISTICS UNDER
POST ACCIDENTAL CONDITIONS (LOCA)

c) Energy emitted by iodines and cesiums contained in recirculated coolant after a LOCA

Iodines

1 hour		1 day		1 month	
Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)
0,3404	$3,15 \times 10^{8}$	0,3565	$2,50 \times 10^{8}$	0,3575	$1,96 \times 10^{7}$
0,6259	$2,81 \times 10^{9}$	0,5481	$4,43 \times 10^{8}$	0,6506	$3,54 \times 10^{6}$
0,9522	$2,50 \times 10^{9}$	1,0010	$1,05 \times 10^{8}$	0,8874	$2,52 \times 10^{-3}$
1,3099	$9,68 \times 10^{8}$	1,2665	$8,29 \times 10^{7}$	1,2741	$2,14 \times 10^{-3}$
1,6668	$9,37 \times 10^{8}$	1,6436	$7,34 \times 10^{7}$	-	-
2,1253	$1,79 \times 10^{8}$	2,1415	$9,95 \times 10^{6}$	-	-
2,7884	$2,76 \times 10^{-5}$	-	-	-	-

Cesiums

1 hour		1 day		1 month	
Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)
0,2446	$1,72 \times 10^{7}$	0,2554	$1,30 \times 10^{7}$	0,2439	$2,86 \times 10^{6}$
0,6408	$3,44 \times 10^{8}$	0,6886	$2,66 \times 10^{8}$	0,6885	$2,60 \times 10^{8}$
0,9792	$2,39 \times 10^{8}$	1,9400	$8,89 \times 10^{7}$	0,9258	$3,09 \times 10^{7}$
1,4308	$4,57 \times 10^{8}$	1,2901	$1,64 \times 10^{7}$	1,3357	$8,70 \times 10^{6}$
1,8652	$2,81 \times 10^{5}$	-	-	-	-
2,2089	$1,47 \times 10^{8}$	2,2085	$1,83 \times 10^{-5}$	-	-
2,6856	$9,83 \times 10^{+7}$	2,6839	$1,23 \times 10^{-5}$	-	-

TABLE T-12.2-12 (4/5)

ASSUMPTIONS RELATED TO ACTIVITY CHARACTERISTICS UNDER POST ACCIDENTAL CONDITIONS (LOCA)

d) Specific activity concentrations in containment atmosphere after a LOCA
(Ci/cm3)*

Isotope	1 hour	1 day	1 month
Kr 83 m	$6{,}20 \times 10^{-5}$	$1{,}41 \times 10^{-8}$	-
Kr 85 m	$1{,}67 \times 10^{-4}$	$4{,}45 \times 10^{-8}$	-
Kr 85	$7{,}14 \times 10^{-6}$	$7{,}14 \times 10^{-6}$	$7{,}10 \times 10^{-6}$
Kr 87	$2{,}18 \times 10^{-4}$	$1{,}03 \times 10^{-9}$	-
Kr 88	$3{,}97 \times 10^{-4}$	$1{,}33 \times 10^{-6}$	-
Xe 131 m	$5{,}43 \times 10^{-6}$	$5{,}13 \times 10^{-6}$	$9{,}11 \times 10^{-7}$
Xe 133 m	$4{,}40 \times 10^{-5}$	$3{,}28 \times 10^{-5}$	$3{,}96 \times 10^{-9}$
Xe 133	$1{,}56 \times 10^{-3}$	$1{,}38 \times 10^{-3}$	$2{,}87 \times 10^{-5}$
Xe 135 m	$2{,}11 \times 10^{-5}$	-	-
Xe 135	$3{,}39 \times 10^{-4}$	$6{,}00 \times 10^{-5}$	-
Xe 137	$2{,}99 \times 10^{-8}$	-	-
Xe 138	$7{,}16 \times 10^{-5}$	-	-
Total	$2{,}89 \times 10^{-3}$	$1{,}49 \times 10^{-3}$	$3{,}67 \times 10^{-5}$
I 129	$2{,}36 \times 10^{-13}$	$2{,}26 \times 10^{-13}$	$2{,}36 \times 10^{-13}$
I 131	$7{,}72 \times 10^{-5}$	$7{,}11 \times 10^{-5}$	$5{,}65 \times 10^{-6}$
I 132	$8{,}29 \times 10^{-9}$	$8{,}09 \times 10^{-8}$	-
I 133	$1{,}51 \times 10^{-4}$	$7{,}01 \times 10^{-5}$	-
I 134	$7{,}64 \times 10^{-5}$	$7{,}84 \times 10^{-13}$	-
I 135	$1{,}32 \times 10^{-4}$	$1{,}22 \times 10^{-5}$	-
Total	$5{,}19 \times 10^{-4}$	$1{,}53 \times 10^{-4}$	$5{,}65 \times 10^{-6}$

* 1 Ci/cm3 = $3{,}7 \times 10^{10}$ Bq/cm3

TABLE T-12.2-12 (5/5)

ASSUMPTIONS RELATED TO ACTIVITY CHARACTERISTICS UNDER POST ACCIDENTAL CONDITIONS (LOCA)

e) Energy emitted by noble gases and iodines in containment atmosphere after a LOCA

Noble gases

1 hour		1 day		1 month	
Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)
0,1723	$7,74 \times 10^6$	0,1310	$2,19 \times 10^6$	0,1310	$2,39 \times 10^4$
0,4567	$2,82 \times 10^6$	0,5906	$4,02 \times 10^4$	0,5140	$5,81 \times 10^2$
0,8742	$4,68 \times 10^6$	0,8690	$1,43 \times 10^4$	-	-
1,4258	$4,75 \times 10^6$	1,4326	$1,50 \times 10^4$	-	-
1,8178	$7,91 \times 10^6$	1,8278	$2,22 \times 10^4$	-	-
2,2761	$2,27 \times 10^7$	2,2962	$6,98 \times 10^4$	-	-
2,6591	$4,52 \times 10^6$	2,8241	$5,86 \times 10^3$	-	-

Iodines

1 hour		1 day		1 month	
Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)	Average band energy (MeV)	Emitted energy (MeV/s/cm3)
0,3403	$1,09 \times 10^6$	0,3565	$8,60 \times 10^7$	0,3575	$6,76 \times 10^4$
0,626	$9,71 \times 10^6$	0,5481	$1,52 \times 10^8$	0,6506	$1,22 \times 10^4$
0,9521	$8,62 \times 10^6$	1,0011	$3,61 \times 10^7$	-	-
1,3100	$3,34 \times 10^6$	1,2665	$2,86 \times 10^7$	-	-
1,6668	$3,23 \times 10^6$	1,6436	$2,53 \times 10^5$	-	-
2,1252	$6,17 \times 10^6$	2,1415	$3,43 \times 10^4$	-	-

TABLE T-12.2-13 (1/2)

PARAMETERS AND ASSUMPTIONS FOR CALCULATING
AIRBORNE RADIOACTIVITY CONCENTRATIONS

- Building airborne radioactive concentrations are at equilibrium
- Airborne particulates are negligible in all buildings
- Radioactivity in reactor coolant
  . Design operating conditions : 3,7 x 10^{10} Bq/t (1 Ci/t) of equivalent I-131.
  . Average operating conditions : 1,11 x 10^{9} Bq/t (0,03 Ci/t) of equivalent I-131.

  Iodine and noble gas activity spectrum of reactor coolant is given in Chapter 11.
  . 3,7 x 10^{10} Bq/m3 (1 Ci/m3) of tritium in the reactor coolant system.
- Leakages of reactor coolant water
  . In the containment building :

  design operating conditions : 6,5 kg/h
  average operating conditions : 90 kg/h
  . In the nuclear auxiliary building (NAB) :

  hot water : 2 kg/h per Unit
  cold water : 31 kg/h per Unit
  . Primary-to-secondary leakage : increasing from 0 to 72 kg/h within two months for a single steam generator
- Secondary system leakage (in the form of steam) : 22 t/h per Unit.
- Fission product partition factor.
  . Noble gases : 1,0
  . Iodines : hot reactor coolant water : 10^{-1}
  cold reactor coolant water : 10^{-4}
  . Other radioactive products : 0

TABLE T-12.2-13 (2/2)

PARAMETERS AND ASSUMPTIONS FOR CALCULATING
AIRBORNE RADIOACTIVITY CONCENTRATIONS

- Reactor building ventilation data
  - Free volume of reactor building : 50 400 m3
  - Internal filtration in the reactor building (EVF) :

    flowrate : 20 000 m3/h
    decontamination factor (iodine isotopes) : 10
    ventilated capacity of the reactor vessel pit : 10 m3
  - Mini-sweeping flowrate : 1 500 m3/h (ETY)
  - Characteristics of reactor building atmosphere :

    humidity : 60 %
    temperature : 30 °C
    water coolant : 20 g/m3
  - Water content in sweeping air : 20 g/m3
- Nuclear auxiliary building ventilation data (DVN)
  - NAB ventilation normal exhaust flowrate : 184 000 m3/h
  - NAB ventilation iodine exhaust flowrate : 21 800 m3/h
- Turbine hall ventilation data (DVM)
  - Ventilation flowrate of turbine hall : 1 048 000 m3/h per Unit
  - Iodine deposit in turbine hall : 0,5

TABLE T-12.2-14 (1/2)

AIRBORNE ACTIVITY CONCENTRATIONS IN THE CONTAINMENT ATMOSPHERE UNDER NORMAL OPERATING CONDITIONS

a) Design values

- $3,7 \times 10^{10}$ Bq/t (1 Ci/t) of equivalent I-131 for fission products
- $3,7 \times 10^{10}$ Bq/m3 (1 Ci/m3) of tritium
- Permanent clean-up ventilation
- No mini-sweeping

Isotope	Activity (Ci/m3)*
Kr-85 m	$4,81 \times 10^{-7}$
Kr-85	$6,67 \times 10^{-4}$
Kr-87	$2,28 \times 10^{-7}$
Kr-88	$9,37 \times 10^{-7}$
Xe-133 m	$3,5 \times 10^{-5}$
Xe-133	$3,52 \times 10^{-4}$
Xe-135	$5,89 \times 10^{-6}$
Xe-138	$4,53 \times 10^{-7}$
I-131	$1,77 \times 10^{-8}$
I-132	$1,89 \times 10^{-8}$
I-133	$5,36 \times 10^{-8}$
I-134	$7,05 \times 10^{-9}$
I-135	$3,04 \times 10^{-8}$
H-3	2×10^{-5}

* 1 Ci/m3 = $3,7 \times 10^{10}$ Bq/m3

TABLE T-12.2-14 (2/2)

AIRBORNE ACTIVITY CONCENTRATIONS IN THE CONTAINMENT ATMOSPHERE UNDER NORMAL OPERATING CONDITIONS

b) Average values

- $1,11 \times 10^{9}$ Bq/t (0,03 Ci/t) of equivalent I-131
- Permanent clean-up ventilation
- No mini-sweeping

Isotope	Activity (Ci/m3)*
Kr-85 m	2×10^{-7}
Kr-85	$2,77 \times 10^{-4}$
Kr-87	$9,47 \times 10^{-8}$
Kr-88	$3,89 \times 10^{-7}$
Xe-133 m	$1,45 \times 10^{-5}$
Xe-133	$1,46 \times 10^{-4}$
Xe-135	$2,45 \times 10^{-6}$
Xe-138	$1,88 \times 10^{-7}$
I-131	$7,35 \times 10^{-9}$
I-132	$7,85 \times 10^{-9}$
I-133	$2,23 \times 10^{-8}$
I-134	$2,93 \times 10^{-9}$
I-135	$1,26 \times 10^{-8}$

* 1 Ci/m3 = $3,7 \times 10^{10}$ Bq/m3

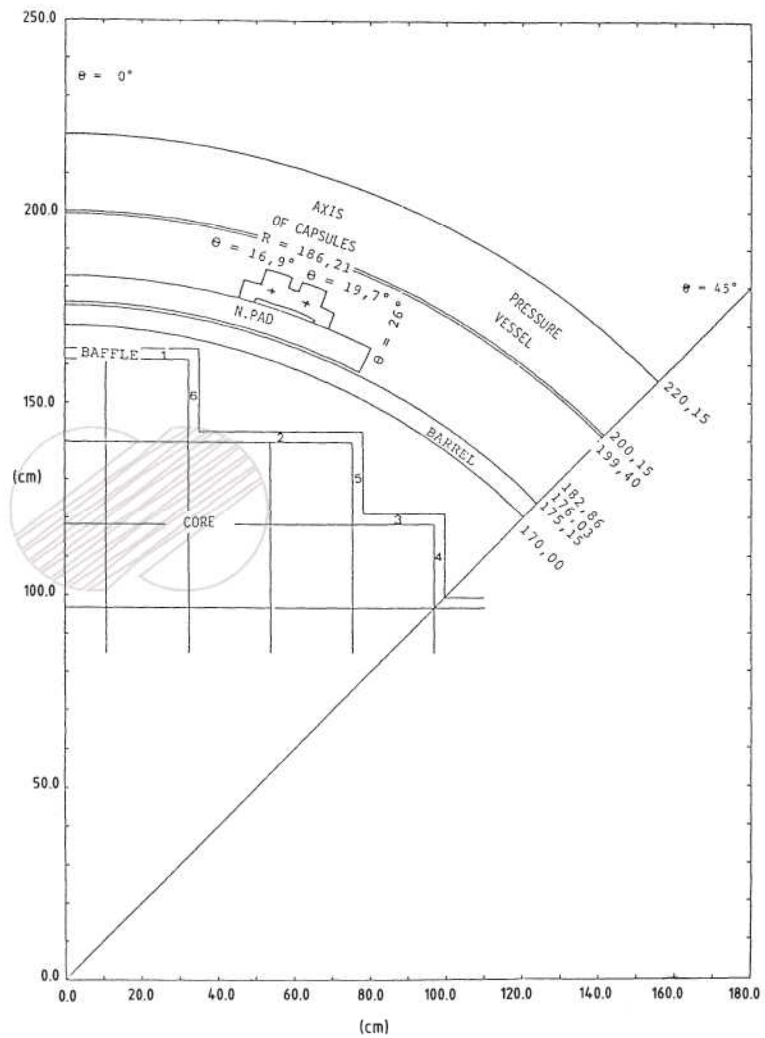

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FIGURE 12.2-1
Geometric layout of 1/8 of the reactor

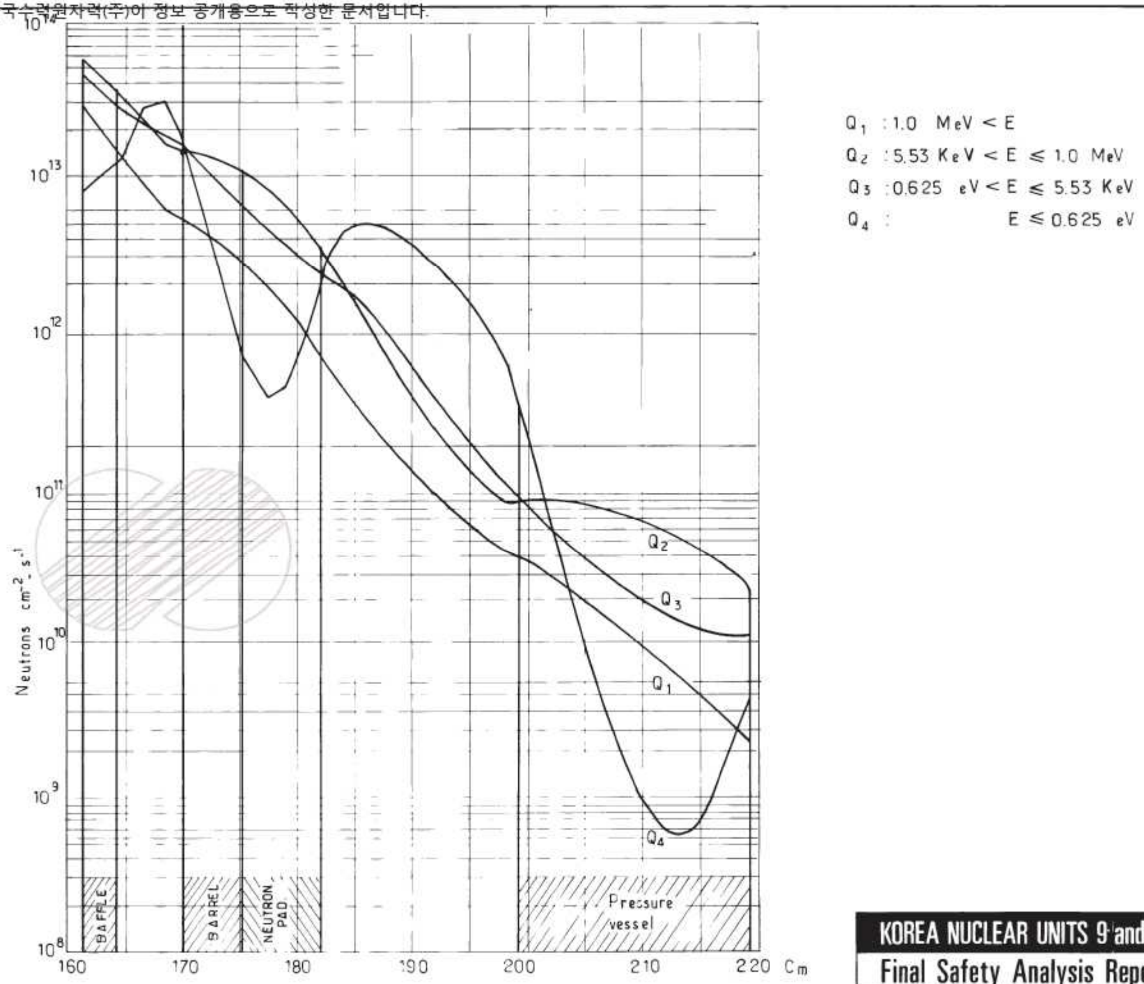

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FIGURE 12.2-2

Radial variations of neutron flux in reactor internals and pressure vessel (at angle 0°)

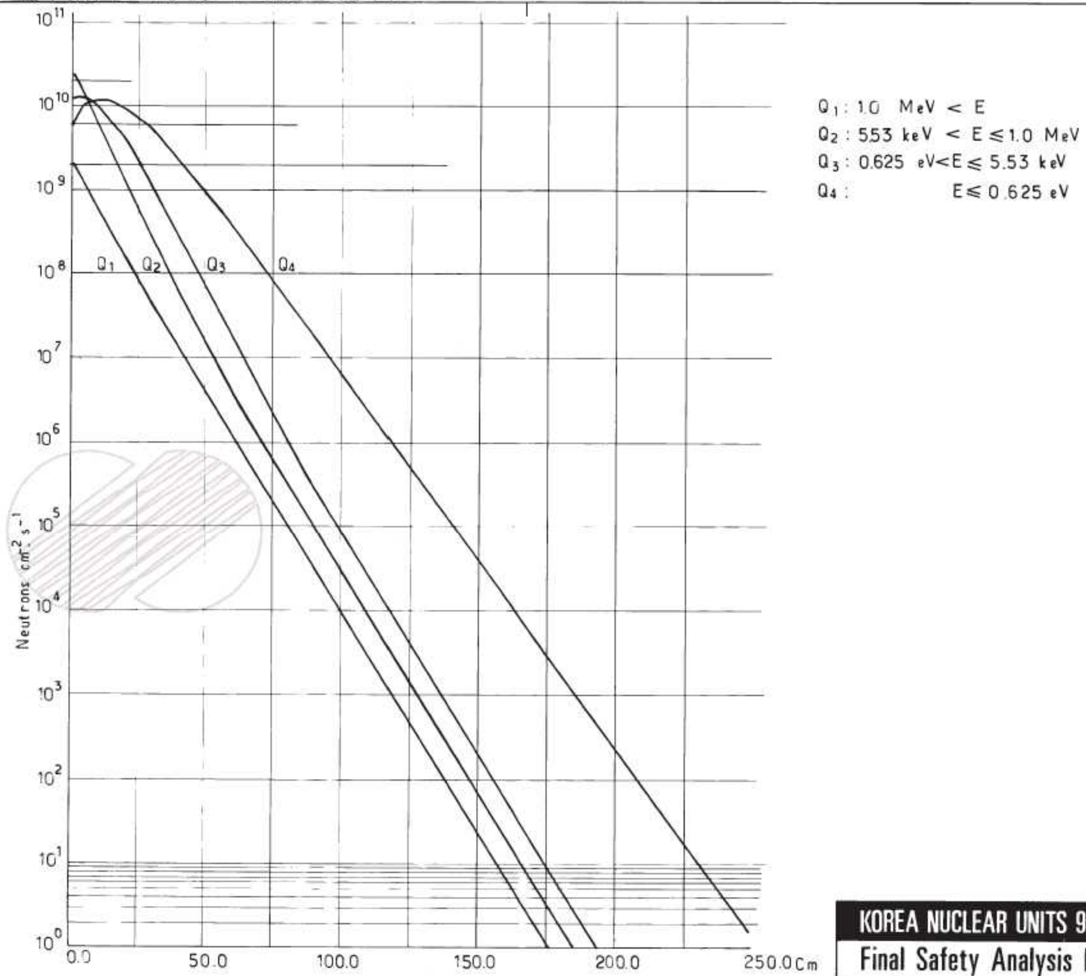

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FIGURE 12.2-3
Radial variations of neutron flux in the primary concrete (at angle 0°)

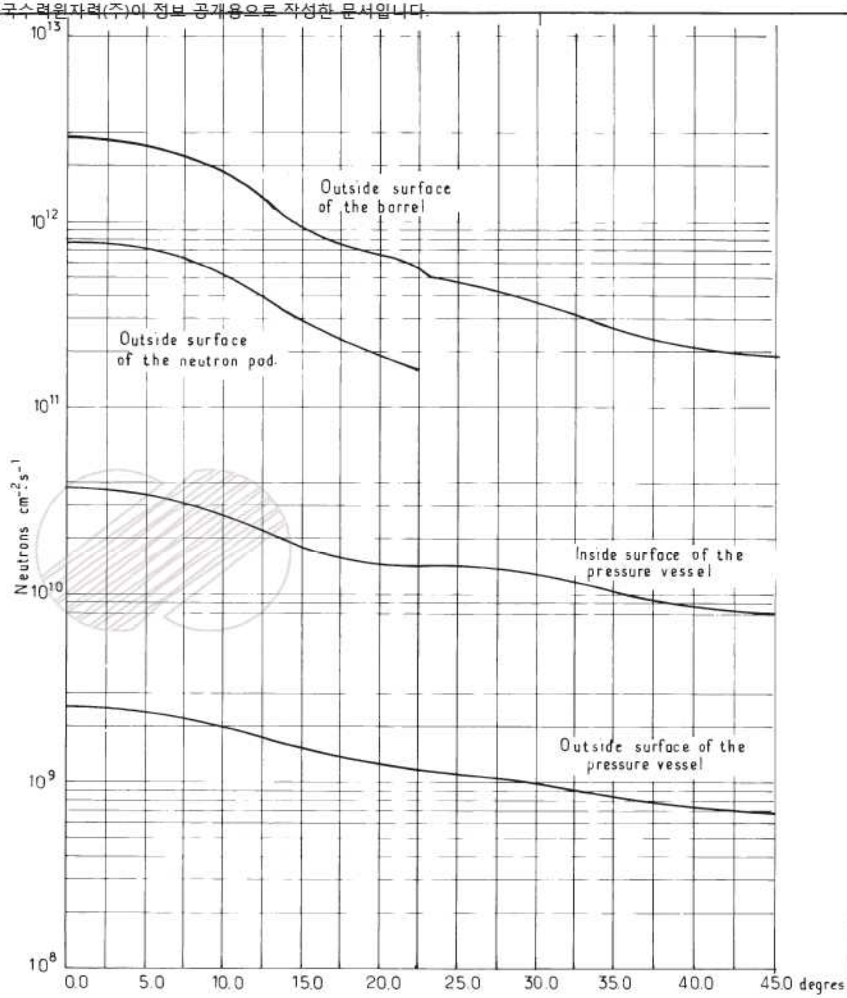

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FIGURE 12.2-4
Azimuthal variations of fast neutron flux in reactor internals and pressure vessel

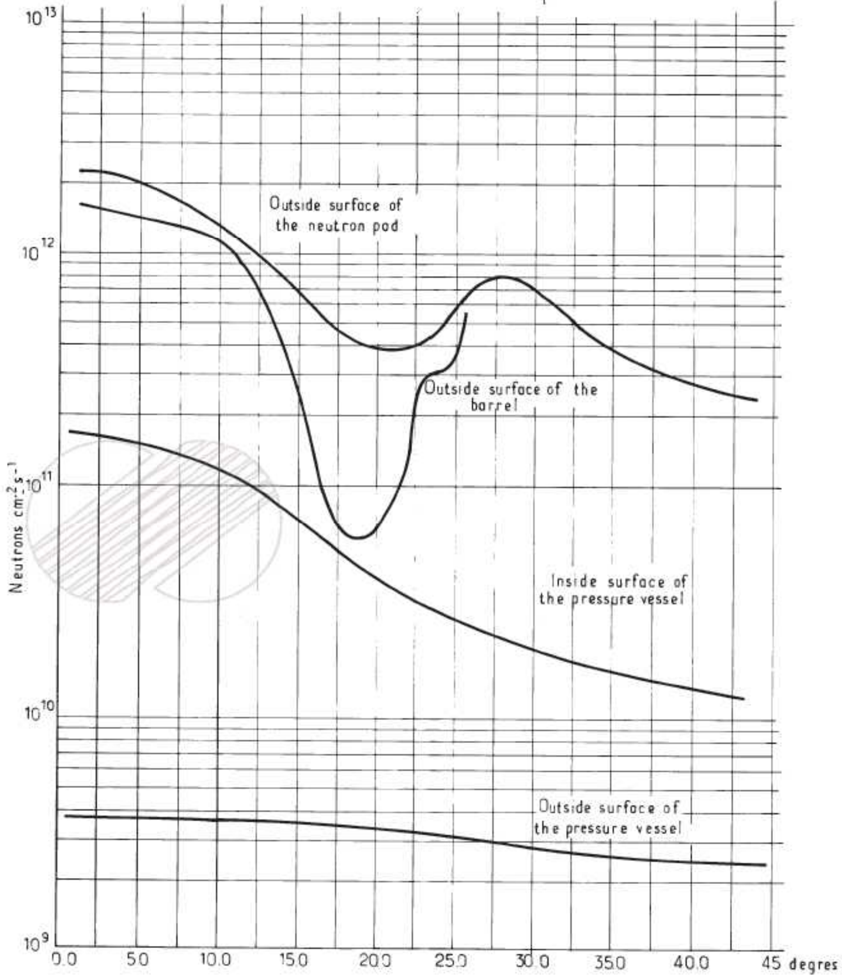

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FIGURE 12.2-5

Azimuthal variations of thermal neutron flux in reactor internals and pressure vessel

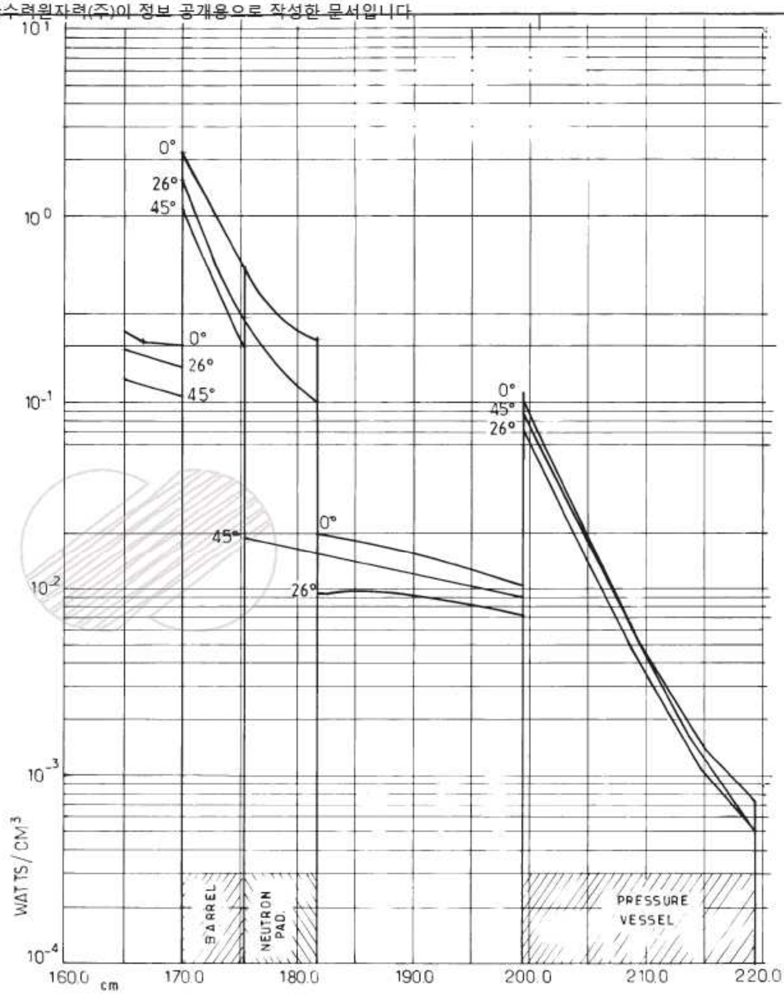

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FIGURE 12.2-6
Radial variations of gamma heating in reactor internals and pressure vessel (at 0°,26°,45°)

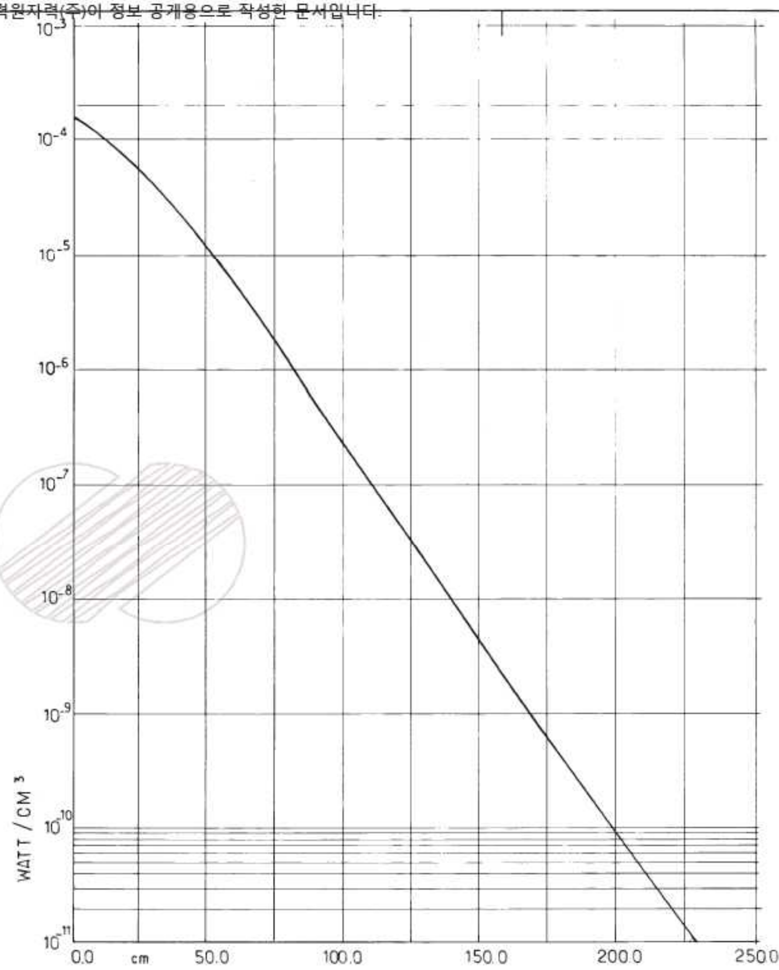

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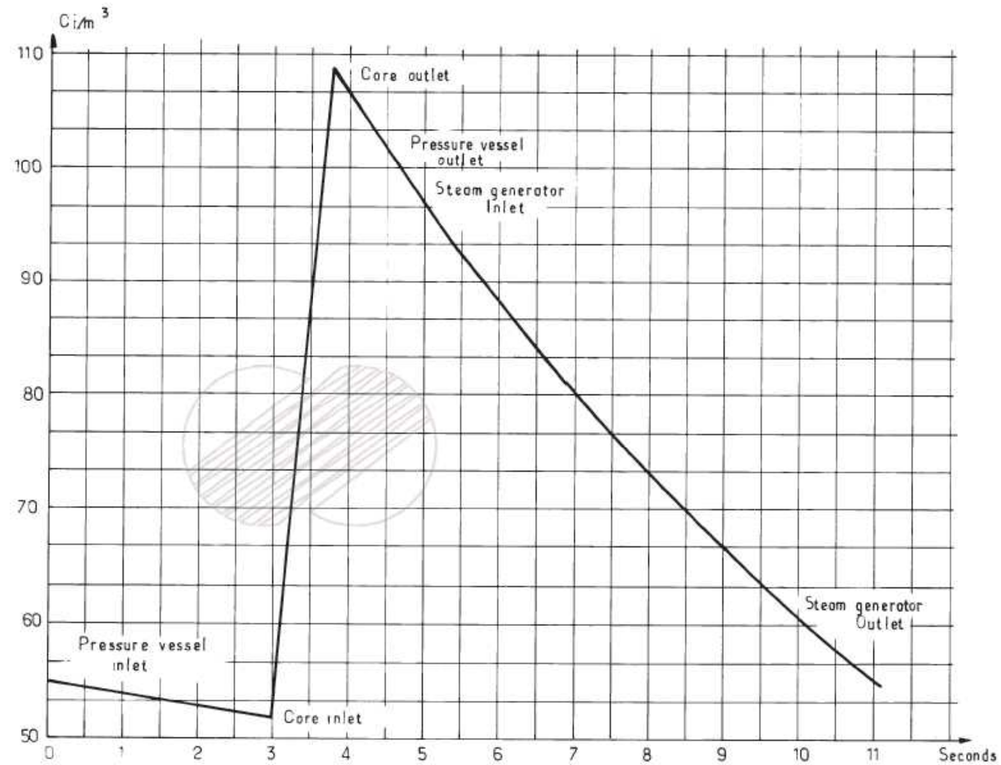
FIGURE 12.2-7

Radial variations of gamma heating in the primary concrete (at 0°)



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Final Safety Analysis Report

FIGURE 12.2-8

Nitrogen 16 activity in a reactor coolant loop

TABLE

FIGURES

12.3. RAOIATION PROTECTION DESIGN FEATURES

Policy considerations have been described in Section 12.1. Taking into account radiation sources described in Section 12.2., this section deals with radiation protection features which are designed in order that personnel doses shall be kept as low as reasonably achievable.

12.3.1. Facility design features

Protection against radiation during general operation of the Plant is achieved :

- at the design stage, by the sizing of shields and of ventilation, by the quality classification of equipment and their appropriate location in the Plant,

- in operation, by following radiation protection instructions, training of personnel, analysis of tasks, and by adequated operating conditions such as chemical processing, decontamination, etc. (See Section 12.5).

Special attention is paid to the maintenance and inspection since most of the total personal collective dose is due to these operations (See Section 12.4).

12.3.1.1. Layout arrangements related to health physics at the design stage

Careful attention is paid to layout arrangement in order to reduce doses. In particular, advantage has been taken from previous PWR plant experience (Chooz, Tihange, Fessenheim, Bugey) and from numerous operating plant data including the Reference Plant.

The following examples can be mentioned.

- Reactor buildings

  . For Fessenheim and Bugey, the polar crane of the reactor building rests on a circular skirt, inside the reactor containment, thus creating an annular area 3m wide, on the building periphery and the same height as the reactor building. In the Reference Plant, and thus in KNU 9 and 10, this skirt has been suppressed above the operating deck, the polar crane then rests on consoles which form an integral part of the containment. This arrangement leaves considerable more space on the operating deck, space which is very useful for setting up a servicing area for equipment which has been dismounted (stud tensioning machine). The greater ease of work and the reduction in the number of handlings which results from this extra space has enabled the exposition time in irradiation zones to be reduced.

- In the fuel building, profiting from Tihange, Fessenheim and Bugey experience, the Reference Plant and KNU 9 and 10 are provided with a spent fuel loading cask preparation area, and also a washing area thus reducing dose absorbed by the personnel near the cask.

- In the nuclear auxiliary building, servicing of the filters and the demineralizers can result in heavy irradiation. Arranged in independent cells, aligned and delimited by very thick reinforced concrete walls, these tanks are accessible from the top by means of biological plugs. All the connecting pipes and isolating valves are fitted towards the bottom. At Tihange, Fessenheim, and Bugey pipes and valves are grouped together in a central corridor between the lines of tanks. The valve controls are brought up into a second corridor located on top of the first. In normal operation, the operator seeking to isolate a tank is not in direct sight of any equipment. But during maintenance of the valves, he needs to have access to the lower corridor where he is then in direct contact with the pipes in use serving the other tanks. At the Reference Plant and thus KNU 9 and 10, the valves are no longer arranged in the same corridor as the connecting pipes. All the valves are grouped together in a corridor designed for this purpose (See Figure F-12.3-1).

12.3.1.2. <u>Design considerations for the components related to health</u>

Protection of Plant personnel against ionizing radiation, and equipment accessibility and testability conform to RCC-P requirements (Section 3.1.3).

Whenever possible, the following engineering rules are applied :

- separation of active equipment items from non-active equipment (e.g., filter and demineralizer bay),

- separation of active items from one another (e.g., RRA pumps),

- separation of items needing little or no servicing from those requiring frequent attention (e.g., tanks from their associated pumps),

- separation of items requiring frequent attention from one another,

- maximum possible distance between sources and personnel,

- shielding.

The above design considerations as related to components can be stated as follows :

a) <u>Pipes</u>

- insure that the gravitational flow is sufficient to prevent water stagnation,

- minimize the high and low points requiring drains and vents, in order to reduce the number of valves and as a consequence, to reduce maintenance.

- it is forbidden to route active and inactive pipes in the same layer,

- length of very active pipes must be reduced as much as possible, in zones of circulation and in rooms containing equipment requiring servicing or inspection,

- for transfer of resins, it is recommended that dismountable hoses and couplings be avoided and that every precaution should be taken in conveying the resins (diameter, valves, slope),

- pipes conveying fluids containing insoluble bodies in large quantities must be capable of being rinsed with water.

b) Valves

- Means of handling valves of more than 25 kg, must be provided.

- Avoid a geographical concentration of valves.

- For double tightness valves with intermediate leak recovery, the recovery circuit for stuffing box leaks must have :

  . good resistance to pressure,

  . free expansion,

  . rapid means of detecting valves which are not tight,

  . an estimate of the flow rate of the leak.

- In the case of a high ambient dose rate at a valve which is frequently operated, remote control by means of a linkage may be envisaged.

c) Pumps

The level of activity at a pump is a function of the liquid conveyed. However, the dose rate at the body of a pump is, generally speaking, lower than that measured at the suction or discharge pipe associated with the pump.

In these circumstances, the body of a motor driven pump unit fitted with its suction and discharge pipes is not the principal source, since they are the major factors in the ambient dose rate of the room.

That is why a screen located between the motor and the pump, intended to protect the pump when work is being carried out on the motor is rarely an efficient protection : in addition it would hinder certain work such as, for example, alignment. However, an arrangement whereby the pump motor and the pipes are installed in two adjoining rooms is acceptable from this point of view, at the price of greater technical difficulties :

- if there is a high dose rate in the room, pump unit replacement should be envisaged,

- the drainage of the body of the pump must be connected to the RPE tank for contaminated fluids,

- the base frame of the pump must be designed to allow leaks from the shafts or the two flanges of the suction and discharge pipes to be collected and discharged,

- the electrical connection boxes and the instrumentation sensors must be located outside high activity zones,

- the ventilation of the room must be designed to avoid contamination being conveyed from one compartment to a less contaminated one.

- the valve controls (suction, discharge, drains, etc.) must be located outside rooms with high activity rates.

d) <u>Tanks</u>

- Tanks within the buildings, containing radioactive products are generally in compartments.

- External tanks containing a contaminated fluid are surrounded by a retention concrete work.

- Tanks are designed to allow for easy mechanical decontamination.

- The bottom of tanks containing radioactive products constitute a zone for deposit of contaminated insolubles. Under this conditions it is prudent to :

  . install drain valves at a distance from the bottom of the tank,

  . ensure that the watchman only has access to the upper part of the tank.

e) <u>Filters and demineralisers of circuits for contaminated fluids must in general be in compartments</u>

f) Evaporators - Gas strippers

Figure F-12.3-2 shows the principle of compartmenting an evaporator. These rules are identical for both evaporators and gas strippers and require :

- the installation in compartments of all very active equipment relating to the Plant,
- the fitting of valves of active circuits in rooms adjoining the compartments,
- the grouping into non-active tunnels of pipes for the auxiliary circuits together with their valves, measuring and monitoring instruments as well as the remote controls for manual valves.

g) NSSS main equipment design consideration

Radiation protection design features of NSSS main equipment reflect the goals and objectives to maintain occupational radiation exposures ALARA.

12.3.1.3. Premises classification

a) Classification

The nuclear power plant site and buildings are broken down into controlled areas and noncontrolled area.

The access to the controlled areas is regulated for reasons of protection against ionizing radiations.

A controlled area is created wherever exposure conditions are such that persons (directly or indirectly connected with work under ioinizing radiation) are liable to receive dose equivalents greater than 1,5 x 10^{-2} Sv (1,5 rem) per year over the organism as a whole. Persons entering such areas are subject to dosimetric checking and followup. This value corresponds to a presence of 2000 hours per year in an area in which the dose rate does not exceed 7,5 x 10^{-3} Sv/h (0,75 mrem/h).

Within a controlled area, the boundaries and the indications of specially regulated zones or those to which access is forbidden, are defined as follows :

	d* < 25 μ Sv/h (2,5 mrem/h)	green zone
25 μ Sv/h (2,5 mrem/h)	< d* < 2 m Sv/h (200 mrem/h)	yellow zone
2 m Sv/h (200 mrem/h)	< d* < 0,1 Sv/h (10 rem/h)	orange zone
0,1 Sv/h (10 rem/h)	< d*	red zone

Doses received by persons working under radiation must not exceed
1 5×10^{-2} Sv (5 rem) per year ($1,5 \times 10^{-2}$ Sv (3 rem) per quarter). This value corresponds to a presence of 2000 hours per year for a worker staying in a zone where the dose rate does not exceed 25 μ Sv/h (2,5 mrem/h). Exposure of persons and the number of persons exposed must be reduced as far as possible.

The non-regulated zone - within the controlled area - where the dose rate is less than 25 μ Sv/h (2,5 mrem/h) is commonly referred to as the green zone.

At the site boundary, the dose rate shall remain below 0,57 m Sv/h (0,057 mrem/h), given the limit of 5 m Sv (0,5 rem) per year for the population.

b) Bases for premises classification established by the designer

- The doses rates which fix the classification of premises are due to external irradiation caused by radiation from contained sources. No dose rate is calculated for contamination. At the design level, everything must be done to ensure that atmospheric contamination is negligible, and zero in a green zone. The hypotheses used for the calculation of irradiation sources must be sufficiently conservative.

- The premises classification are calculated on the bases of assumptions made for the Reference Plant :

  . the unit is operating at full power,

  . the "equivalent fuel cladding failure rate" is conservatively assumed to be 1 %,

  . no deposits and no peaks are taken into account,

  . systems are operating at rated pressure and temperature,

  . the containment spraying and safety injection systems are not contaminated.

(*) d = equivalent dose rate

c) Boundary marking

- Zone limits (See Figures F-12.3-3 to F-12.3-10 and F-12.3-15) 224

  Each room has a zone classification corresponding to a mean ambient dose rate. High pin-point dose rates can be accepted if mobile shields can be installed. Every effort is made so that the zone limits coincide with those for the rooms.

- Arrangement of the rooms

  Access to the main nuclear island controlled area is only possible during normal operation from the "transit zone", comprising changing rooms, together with equipment for checking body and tools contamination.

The arrangement of the rooms is designed so as to conform to the following rules :

- to enter a green zone, it shall not be necessary to cross a yellow zone, an orange zone or a red zone,
- it is possible to monitor equipment from a gallery corridor with a low dose rate (i.e., green zone for the nuclear auxiliary building, dose rate below 0.15 mSv/h (15 mrem/h) for the reactor building annular area, when the reactor is in operation).

- Access to equipment

  Equipment is laid out so as to ensure accessibility for preventive and occasional maintenance, as well as inservice inspection where appropriate. The layout allows to provide :

  · sufficient temporary storage areas,
  · easy disassembly capability for equipment,
  · easy personnel access to the maintenance area,
  · appropriate handling facilities.

d) Accessibility

All persons entering controlled areas are subject to strict authorization, instructions and controls. Wearing of a personal indirectly readable dosimeter (film dosimeter, for example) is mandatory. A contamination and irradiation check is compulsory at all controlled area exit points. The flow of personnel is shown in figure F-12.3-11. 1

Access to the different zones within the controlled area is bound to following principles : (See table T-12.3-2) 1

- Green zone : permanent access area

- Yellow zone : limited access area

  Provisions for protection against contamination and monitoring of exposure is taken. Additional clothing may be necessary in case of contamination.

- Orange zone : regulated access area

  Access, stay and egress are governed by strict rules and persons entering these zones must obtain prior approval from a qualified health physics officer. Permanent presence of an attendant is desirable.

  Special clothing and equipment may be required.

- Red zone : exceptional access area

  Access to these zones is padlocked and normally prohibited. Exceptional entry authorization has to be delivered by the unit supervisor. The same conditions as for the orange zone apply for access, stay and egress ; only the exposure time is more limited and each operation has to be prepared with the utmost care.

  Special clothing and equipment may be required.

12.3.2. Shielding

Bases for shielding design are described in Paragraph 12.3.2.1, shielding calculational methods in Paragraph 12.3.2.2, and shielding configurations in Paragraph 12.3.2.3.

Methodology for shielding calculation is described with more details in the document in Reference 1.

12.3.2.1. Bases for shielding design

The basic objective of the Plant radiation shielding is to reduce personnel and population exposures, in conjunction with a program of controlled personnel access and occupancy of radiation areas, to levels that are within the dose regulations of RCC-P, Chapter 5.

The shielding design source terms are based upon the three situations of normal full power operation, shutdown and loss of coolant accident (See Section 12.2).

Biological shielding assures sufficient personnel access and occupancy time to allow normal anticipated operation and maintenance, inspection and testing required, in accordance with premises classification given in Paragraph 12.3.1.3.

12.3.2.2. Calculational methods

a) Materials

Ordinary concrete (theoretical density of 2,35 g/cm3) is the most widely used material for radiation shielding in nuclear power plants.

Whenever poured-in-place concrete has been replaced by concrete blocks or other materials, design assures protection on an equivalent shielding basis as determined by the characteristics of the material selected.

- High density concrete (3,4 g/cm3) is used when the biological shield required is restricted in terms of space, namely for the NAB filter cartridge transfer tube between +13,15m and +5,00m and for the floor of the TEG buffer tank room at level +8,00 m.

- Other materials used include cast iron (e.g., slabs above ducts in the NAB), iron (e.g., NAB steel shield doors) or in exceptional cases lead (e.g., reactor pit door) and neutron absorber concrete (reactor cavity bottom at +10,86m) (See Subparagraph 12.3.2.3.1).

- Water is used as biological shielding in reactor cavity and spent fuel pit.

b) Mathematical models

The main mathematical models used to evaluate shielding are :

- the codes used to evaluate neutron and gamma dose rates due to reactor core fluxes out of which :

  . the ANISN code which uses core physics and transport theory methods to give a set of cross sections condensed into groups for DOT calculations,

  . the DOT Code for radiation protection opposite the core. Neutron and gamma-ray flux across the vessel internals, the reactor vessel, and primary shield concrete are calculated using this two-dimension discrete-ordinates transport code,

  . the MERCURE 4 Code (CEA code) which calculates dose rates integrating neutron and gamma-ray line of sight point attenuation kernels using the Monte-Carlo method in three-dimension geometrical configurations and multigroup approximations,

- the codes used to evaluate gamma dose rates due to tanks, heat exchangers, filters, demineralizers, evaporators or any finite cylindrical volume source :

  . the ACTIVI Code (EDF code) computes the gamma energy groups emitted by a 3-dimensional source composed of a mixture of radio nuclides, of which isotopic concentrations are known.

The results serve as input data for the MERCURE 4 or CYLIND shielding codes.

- the CYLIND code (EDF code) for radiation protection opposite equipment conveying radioactive fluids. This code calculates dose rates using the point kernel method for cylindrical volume sources.

12.3.2.3. Shielding configurations

12.3.2.3.1. Reactor building

- Containment shielding

  During reactor operation, the containment protects personnel occupying adjacent Plant structures from radiation originating in the reactor vessel and primary loop components.

  For a loss-of-coolant accident, the containment reduces the Plant radiation intensities emitted by fission products inside the containment. Biological shielding doors are located at level +20,00m and +0,00m in front of the equipment hatch and the personnel airlock, where the containment shielding is reduced.

- Primary shield around the reactor pressure vessel (pressure vessel pit)

  The primary shield surrounding the reactor pressure vessel supports the vessel and forms a biological shield around it (See Figure F-12.3-11). It is continued above the perimeter wall of the reactor cavity and supports the service floor from which access to the primary pumps is gained and on which the rails for the fuel manipulator crane are fixed.

  The primary shield isolates the adjacent steel structures from excessive radiation during operation (primary circuits, heat exchangers, etc.) and provides biological protection for the personnel inside the reactor building during shutdown.

- Secondary shield surrounding the primary circuits

  The secondary shield consists of a circumferential wall with radial walls such that all elements of the primary circuit are shielded.

  The secondary shield supports and encloses, in subcompartments, the pressurizer, the primary circuit motor-pump assemblies, the primary circuit piping and the steam generators.

- Reactor cavity floor shielding

Neutrons issuing from the reactor core escape into the reactor building through shielding discontinuities of the reactor pit (see Figure F-12.3-12)

- at level -3,50 m through the access bunker to the reactor pit,
- at level 8,10 m in the steam generator bunkers through the gap between primary loops and concrete wall.
- at level 10,80 m at the edge of the reactor pit through the gap between the reactor vessel and the concrete shielding and through the 8 openings of RPN measuring chambers.

Neutron shielding is necessary at level +10,862m to minimize the leakage of neutrons onto the operating deck. The reactor vessel is therefore surrounded by a 30 cm thick layer of limonite concrete (density of 2,12) (or a material equivalent to limonite concrete from a neutron attenuation point of view) to which is attached the seal ring between the reactor vessel and cavity floor. This is surrounded by a second layer of 50 cm thick limonite concrete. This layer has shield discontinuities at ionization chamber access ports. To maintain protection, these openings will be filled with a light weight, protective material, thus avoiding missile generation during a LOCA. This material is a polyester resin compound including polyethylene resin, alumina trihydrate and borated colemanite.

The document in Reference 2 gives several examples of neutron streaming calculations and show that the dose rate calculated on the operating deck due to neutron escaping from reactor cavity is of the order of 8 m Sv/h (0,8 rem/h) under full power operating conditions.

The neutron dose rates measured at the same point on French Standard 900 MW PWR under full power conditions range between 1,5 to 9 m Sv/h (0,15 to 0,9 rem/h).

An assessment of the efficiency of biological plugs at ionization chamber access ports has been made at the Blayais-4 unit in which measures at 30 % PN shows a neutron attenuation factor of 8 to 10 due to this material.

On the KNU 9 & 10 Plant, a reinforced door filled with a 10 cm thick neutron attenuating compound implemented at the entrance of the reactor pit access bunker at level - 3,40 m, permits the reduction of total dose rate (neutron and gamma) to a value consistent with premises classification.

- Reactor cavity shielding

The reactor cavity is filled with water during refueling. When the reactor internals are completely immersed in water, the dose rate at any point on the cavity surface due to deposits on (and activation of) vessel structures is negligible.

Reactor upper internals are stored in the cavity and are completely immersed in water.

- Transfer tube shielding

  Spent fuel is the primary source of radiation during refueling. Because of the extremely high activity of the fission products contained in the spent fuel elements, extensive shielding has been provided for areas surrounding the fuel transfer tube to ensure that radiation levels remain below zone levels specified for adjacent areas.

  The need is to ensure an equivalent concrete protection of 150 cm thick when an irradiated fuel assembly is in the tube. This is done via concrete walls built vertically at level +8,00m, on both sides of the tube. Floors above and under the tube are completed with lead bricks.

  In the area between the containment and fuel buildings, a lead protection is built in a metallic frame hung on springs. It is planned to plug the spaces inside the reactor building and between the metallic frame and the containment wall with lead wool.

  Biological shieldings implemented around the transfer tube within the reactor building and in the space between the reactor building and the fuel building are fully discussed in Reference 1.

12.3.2.3.2. Nuclear auxiliary building

Shielding is provided for each piece of equipment consistent with its postulated maximum activity (Subsection 12.2.1) and with the access and zoning requirements of adjacent areas (Figures F-12.3-3 to F-12.3-10).

Auxiliary shielding is designed to protect operating and maintenance personnel working near the various auxiliary system components such as those in the chemical and volume control system, the boron recovery system, the vent and drain system, the liquid, solid and gaseous waste treatment systems, etc.

In certain cases, shielding walls consist of removable concrete blocks or slabs to facilitate access to the equipment during maintenance periods.

12.3.2.3.3. Spent fuel pit building

The bottom and the walls of the spent fuel pit are made of concrete. Shielding above spent fuel assemblies when they are handled or stored, is formed by borated water.

12.3.2.3.4. Control room shielding

Figures F-12.3-13 and 14 show the layout of the control rooms and their relationship to the containment. The design basis loss of coolant accident (LOCA) dictates the shielding requirements for the control rooms. Shielding is provided to permit access and occupancy of the control room under LOCA conditions with low radiations doses. A minimum of 30 cm of concrete is estimated to be adequate for this purpose.

The thickness of the walls and the ceilings of the control rooms necessary to ensure protection against airplane crash is between 80 and 100 cm. Thus, the walls and the ceilings of the control rooms constitue a sufficient shielding against radiations emitted by airborne fission products contained inside the containment and against radiation emitted by the plume. 320

12.3.3. Ventilation

The ventilation systems servicing potentially contaminated rooms are designed to minimize the circulation of radioactive products inside the Plant and their releases to the environment (See RCC-P, Subsection 2.3.3.3). 320

The function and design bases of these ventilation systems are given in Sections 6.4 and 9.4. Consistent with these, the following specific objectives pertain to radiation protection and the commitment that occupational radiation exposures will be as low as possible.

12.3.3.1. Main functions

Each ventilation system has four main functions :

- the collection of gaseous radioactive leakage,
- the reduction of the equilibrium contamination in rooms and other areas to acceptable levels,
- the limitation of the spreading of radioactivity from source rooms to others with lower acceptable contamination limits. In this respect, design is such that air flow is always from rooms with lower contamination towards rooms with higher contamination,
- efficient filtration of gaseous effluents (particulate filters and iodine traps), before release to the environment.

12.3.3.2. Design guidelines

Each ventilation system has four main series of design factors, related respectively to the air supply, the ductwork, the flow rate served, and the exhaust.

12.3.3.2.1. Air supply

Each air supply unit comprises air inlet fixtures, a defroster (for safety related systems only), a prefiltering assembly, main filters, and supply fans. One of the functions of the supply filters is to maintain high air purity within the rooms, to protect the radioactive exhaust filters.

12.3.3.2.2. Ductwork

The main characteristics of the ductwork are : decontaminable (use for removable duct liners) ; fire resistant (PCV and other inflammable materials are prohibited) ; corrosion resistant (stainless steel within the reactor building : tinned sheet metal cannot be used for ducts within the reactor building, for example, because of the high hydrogen levels generated during a LOCA) ; leak-proof, to prevent the spreading of contamination ; and pressure resistant, the design pressure being determined by calculated overpressure conditions after trip of a supply fan or exhaust fan.

12.3.3.2.3. Flow rate

Renewal of air in the rooms depends on the level of irradiation ; e.g., a minimum of four removals per hour for orange and red areas, and two for green and yellow areas.

However, abnormal conditions (large leaks, maintenance periods, etc.) may lead to intermittent deficiencies, since it is not realistic to design and size the normal ventilation plant to take care of all such conditions. During some periods of abnormal operation, it is possible to use additional mobile ventilation units to remedy deficiencies, in which case the design of the Plant is such as to allow connection of the mobile units to ductwork.

Moreover, the ducts run systematically from less contaminated to more contaminated rooms, and the exhaust flowrate is set greater than the supply flowrate, to maintain negative pressure within the rooms (except in the case of the control room).

12.3.3.2.4. Exhaust

Finally, each exhaust unit is composed of exhaust fans, prefiltering assemblies (designed according to the air dust content, and whose characteristics determine the frequency at which absolute filter media must be renewed), absolute filters, and activated charcoal filters (for portions of ventilation system designed to trap iodines).

Operation of activated charcoal filters is generally intermittent, to avoid too rapid degradation, starting upon the detection of iodine, or during periods of potential risk (fuel handling, for example).

The loading of the filters and adsorbers with radioactive material during normal Plant operation is a slow process ; therefore, in addition to monitoring for pressure drop, the filters are checked for radioactivity on a scheduled maintenance basis with portable equipment, and the filter elements are replaced before the radioactivity level is of sufficient magnitude to create a personnel hazard.

An order of magnitude of renewal frequency is :

- once or twice per year for prefilters,
- once every two years for absolute filters,
- once every one to four years for activated charcoal filters (when operation is intermittent).

12.3.3.3. Personnel protection features

In order to protect personnel from radiation, the following features can be quoted :

- each filter bank is housed in a shielded compartment, room, or cubicle,

- adequate aisle space is provided for both personnel and equipment adjacent to the service side of the filter trains,

- convenient and accessible passageways and corridors from the filter trains to the elevators and equipment hatches are provided for transport of replaceable filter train components and the equipment used in accomplishing their replacement,

- replaceable elements are designed for easy removal and minimal radiation exposure of personnel. It is not necessary for workers to handle filter units immediately after an accident so that exposures can be minimized by allowing short lived isotopes to decay before changing the filter,

- rigid, hinged access doors are provided,

- filter support and alignment cradles are used for aligning and supporting filter elements during filter change, except for side replaceable filters,

- permanent test fitting are provided for initial and periodic field testing. The ventilation system performances are checked during Plant startup and inspected annually. The iodine filters are subject to performance specifications. They are subject to annual onsite tests to check their capability.

12.3.4. Area radiation and airborne radioactivity monitoring instrumentation

Area radiation and airborne radioactivity monitoring equipment is provided to assess radiation levels at various in-plant locations.

12.3.4.1. Area radiation and airborne radioactivity monitoring system design objectives

The area radiation and airborne radioactivity monitoring system is part of the radiation monitoring system (KRT). It is designed to provide indication of radiation levels in selected Plant areas and provide alarms when a radiation level exceed a preset value. This monitoring system consists of multiple channels that accomplish one of the following design objectives :

- warn of abnormal conditions in the Plant,
- assist Plant operator in decision on deployment of personnel in the event of an accident resulting in a release of radioactive material in the Plant,
- assist Plant operator in decision on operation to prevent further exposure of Plant personnel,
- provide local alarms where a substantial change in radiation levels might be of immediate importance to personnel frequenting or working in an area,
- provide automatic changeover of ventilation system to reduce the risk of Plant personnel contamination or of external releases,
- provide post accidental information,
- provide monitoring for use in assuring that solidified radwaste containers meet the requirements of shipping and storage.

The selection of monitors and of monitor locations to satisfy the above objectives is based on the following considerations.

In the controlled area of a nuclear power plant, there is no fixed and permanent place of work. Access to the zone is afforded only for rounds, short-term inspections or action connected with Plant operation, and for maintenance or repair work. However, some areas are an exception to this rule (laboratory, transit area). In these areas the risks of exposure are limited and must be specifically investigated : moreover, they affect only a limited number of suitably qualified personnel.

Due to the complexity of the installation, it is not possible to continuously monitor all possible sources of contamination in each room. Therefore, during the periodic rounds of the controlled area, portable instruments are used to measure the values of the dose rate (gamma and exceptionally neutron radiation) and the contamination prevailing in each room. These values are then posted at the entrance to the room or the controlled area. Such data is sufficient for estimating the permissible exposure time and for deciding on the protective clothing or respiratory apparatus to be used as long as there is no change in the level of the activity sources.

Areas where exposure is most intense (orange area above 200 mrem/h and red above 10 rem/h) are not accessible until after assessment of exposure conditions by means of portable apparatus at the time of and during access and the issuance of a special permit signed by an authorized person properly informed about radiological protection.

In places where activity sources are liable to slow or sudden variations, it is advisable to analyse the causes of such variations in order to determine suitable means of detecting them.

During maintenance periods, the best way to ensure a good protection for workers is to control the work place by means of movable detectors.

Outside the maintenance period, a variation in external irradiation may be due to either a contained source or to an atmospheric contamination by noble gases, which are a major air contaminant in a power Plant in operation (See Subsection 12.3.4.2.).

In the case of a contained source, the variation may be localized at some of the hot points, which are known a priori : filters, resins, tanks. These points are always located in shielded places, most often inaccessible to personnel except during the maintenance period. Consequently, there is no need for continuous monitoring of these areas for direct protection of workers.

The solid waste hall to which radioactive waste is brought at irregular intervals may cause a variation of the ambiant dose rate of which personnel are warned before entering the room. This function is fulfilled by an area radiation monitor.

If there is wide-spread variations, following a clad failure for example, they are detectable at any point in the reactor coolant system. Therefore, the continuous monitoring of the reactor coolant, which is described in Section 11.5., is sufficient for detection, and information of Plant personnel.

In the second case, atmospheric contamination by noble gases, the variation may be either slow or sudden.

If it is slow, following a small leak in the system, account is taken of the fact that the main contaminant is easily vectored by the ventilation systems without retention on the surfaces or the filters in order to use the measurement of gas in a ventilation system as a means for detecting the leakage variation. In this respect, depending on whether the leak is located in the reactor building or in an area, the measurement of gaseous activity in the containment air or the measurement of gaseous activity in the stack is available. These measurement channels which are part of the process and effluent radiological monitoring system (See Section 11.5) contribute to the area radiation and airborne radioactivity monitoring.

A sudden variation in atmospheric contamination by noble gases liable to cause a variation in external irradiation, outside maintenance periods, can only be due to significant leakage from the gas storage tank, a fuel handling accident or a main accident in the reactor building. The occurence of significant leakage from the gas storage tank in non-accessible areas does not constitue a risk to personnel and is detected by measurement at the stack (See Section 11.5).

In case of a fuel handling accident, the measurement of the dose rate resulting from atmospheric contamination by noble gases would appear to be the quickest way to detect this type of accident (a measurement of atmospheric contamination, which should be made in a ventilation duct to be representative, results in a significant response time).

Taking into account the importance of the risk, this measurement is
320 paralleled above the spent fuel storage pool as well as below the reactor
cavity. The detectors are located in areas which are well spaced out in order to detect incidents more efficiently.

In case of a main accident in the reactor building, gamma dose rate inside the containment is monitored by means of two channels, which provide information to the PAMS. Dose rate in the control room produced by radiations or by airborne contamination due to the containment leaks are monitored by two detector located in the control room air inlet fixture and in the duct.

Indirect personnel protection is ensured by monitoring spent resins and concentrate activity before packing into drums for determining how to divide up the resins and concentrated loads in the drums to limit the activity contained in each concrete block in order to satisfy the conditions for transportation. Compliance with these conditions is verified by measuring radiation emitted by the drum being filled. This function is complied by dose rate monitors located in front of filters, spent resin or concentrate tanks and of the filled drum.

Channels monitoring dose rate around the reactor pit, the spent fuel pool and the control room perform an additional function. A high radiation signal initiates normal ventilation isolation and start up of iodine filtration.

As a general statement, the design objectives of the area radiation and airborne radioactivity monitoring system are complied by monitoring channels which are listed in Table T-12.3-1 and by some monitoring channels which are part of the process and effluent radiological monitoring system (See Section 11.5). Specifically, all the airborne radioactivity monitoring channels belong to that system.

12.3.4.2. Design bases

Design bases are as follows :

- channels of area radiation and airborne radioactivity monitoring system belong to the radiation monitoring system (KRT). They are organized like the other channels of this system (See Section 11.5) :
  - respective ratemeters (6 digits) are located in the radiation monitoring system cabinets,
  - the channels have three alarm functions, the first one to detect signal failure and power or circuit failure, the two others to alert that specified radioactivity levels have been exceeded,
  - alarm level continuously adjustable over the entire range of the channel,
  - the alarm trips initiate alarm lights on the front panel of the ratemeter and on the KRT panel in the control room,
  - electric power supply of channels is fed through the radiation monitoring system cabinets,
  - unit channels are supplied by seismically qualified electric sources and emergency supplied,
- channels monitoring the dose rate around the reactor pit, the spent fuel pool and in the solid waste hall actuate warning beacoms that give sound and light alarms,
- warning beacon are auxiliary supplied by internal electrical batteries,
- detectors are designed to function properly in ambient conditions, including exceptional ambient conditions,
- detectors provided to monitor the dose rate inside the containment after an accident are linked to PAMS and are designed according to the following criteria :
  - application of single failure criterion,
  - physical and electrical separation (electrical power supplied by reactor protection system power supplies, 1E),

· signal protection,

· qualification (for LOCA and seismic conditions),

· tests.

Furthermore, according to Authorities' requirements, the monitors have an extended gamma response range from 10^{-2} up to 10^{5} Sv/h (1 to 10^{7} rad/h) ; they detect gamma photons with energies as low as 60 keV and have a flat response for gamma energies between 100 keV and 3 MeV, as demonstrated by type-test and each detector calibration.

These monitors are located in containment in a manner as to provide a reasonable assessment of area radiation conditions inside containment, nevertheless they are accessible for maintenance or calibration.

In situ calibrations are made by means of calibrated radiation sources (one decade) and by electronic signal substitution for other decades.

- channels monitoring the dose rate around the reactor pit and the spent fuel pool or the control room air are paralleled and qualified for seismic conditions,

- internal long half-life radiation sources permit continuous control of the operation of the channels,

- when possible, external radiation test sources permit checking the operation of the channels and one point calibration.

12.3.4.3. System description

General system description is given in Section 11.5.

12.3.4.3.1. Gamma warning beacon

Gamma warning beacons are listed in Table T-12.3-1. They operate as follows.

The dose rate is measured by an ionization chamber. The current is processed by the electronic device included in the warning beacon box. This box also contains sound and light alarm devices, the alarm threshold
320 adjustment, the electrical battery which permits an operation of 2 hours (monitoring) and 1 hour (alarm operating) without external power supply.

The processed information is send to a simplified electronical set fitted in the radiation monitoring system cabinets.

The external test source is operated from the centralized KRT cabinets. Performances of these monitors are given in Table T-12.3-1.

Gamma warning beacons are qualified Seismic 1.

12.3.4.3.2. High radiation contained source monitor

This kind of monitor measures dose rates due to the activity contained in filters, tanks, or drums. They are listed in Table T-12.3-1. This monitoring channel is of the same type as that described in Paragraph 11.5.2.4.

The detector is an ionization chamber located in the concrete shielding surrounding the source. Performances of these channels are given in Table T-12.3-1.

12.3.4.3.3. Control room dose rate monitor

These channels monitor the gamma dose rate inside the ventilation rooms of the control room.

Radiation is detected by an ionization chamber. The detector contains an internal radioactive test source and may be checked using an external test source operated from the centralized KRT cabinet. Performances of these channels are given in Table T-12.3-1.

These channels are qualified Seismic 1.

12.3.4.3.4. Gamma exposure rate monitor in reactor containment under accident conditions

These channels consist of ionization chambers designed to support LOCA ambient conditions. They are qualified Seismic 1.

The detectors are located inside the reactor building.

The measurement boxes (preamplifiers) are located outside the containment in the nuclear auxiliary building near the containment.

The detectors contain internal radioactive sources.

Measurement signals give alarm signal and are recorded in the PAMS. Performances of these channels are given in Table T-12.3-1.

12.3.4.3.5. Airborne radioactivity monitoring

The airborne radioactivity monitoring consists of the following monitors :

- reactor containment air activity measurements,

- stack discharge air activity measurements,

- nuclear auxiliary building air activity monitors.

These monitors are discussed in Section 11.5.

12.3.4.4. Alarm set points, in-service test and calibration

Area monitor alarm set points are dependent upon the normal background radiation at the detector location and the limits for personnel exposure in restricted areas. Alarm set points are adjusted by a health physist responsible according to actual background radiation dose rates.

All the area radiation and airborne radioactivity monitors are designed for testing during operation.

Some detectors have an internal radioactive source that makes a continuous background noise. The related signal is used in a way as a "watch dog", meaning that the electronic associated to the detector control the signal on a continuous basis and set off an alarm in the event of any change. The background noise is automatically substracted from the measurement values but can also be read so as to check that there is no shift.

Proper alarm operation and automatic action triggering can be tested by adjusting the alarm threshold to a value under that of the background noise.

The other detectors are equipped with a device consisting of a radioactive source that can be shifted in front of the detector from a shielded position to an open one.

The background noise created by the source can be compared to the initial values and can also be used to test the alarms and the proper automatic action operation.

KRT activity monitoring devices are not required to measure accurately activity or dose rate, this is the purpose of portable equipment or laboratory measurements. The function of these devices is to provide an alarm when levels exceed preselected values to detect the variations not the values per se. As a consequence, in-service calibrations of radiation monitors are not required.

Nevertheless, accuracy of fixed apparatus may be regularly deduced from the comparison with accurate measurement device measures. Single point calibration may be performed by actuating the external radioactive test source.

Gamma exposure rate monitors in the reactor containment under accidents conditions (containment high range monitor) are regularly calibrated according to means described in Paragraph 12.3.4.2.

12.3.5. References

1 - Radiation shielding design report
12 CDR 04

2 - SFL/NG/SU - 82.1674 Rev. A
Neutron and gamma dose rates in the reactor building of a standard 900 MWe Power Plant.

TABLE T-12.3-1
IN-PLANT AREA RADIATION MONITORS

Area monitored	Quantity		Type	External check source	seismic category	Range	Specific information treatment, action initiated by high level alarm	
	per unit	interunit						
Reactor cavity	2		Ionization chamber	1	1	1×10^{-5} to 1 Gy/h	Local alarm + action on EBA system	
Spent fuel pit	2		Scintillator	1	1	1×10^{-6} to 1 Gy/h	Local alarm + action on EBA system	373
Solid waste hall		1	Ionization chamber	1	1	1×10^{-5} to 1 Gy/h	Local alarm	
Control room ventilation duct	2		Ionization chamber	1	1	1×10^{-6} to 1 Gy/h	Measyre recored + action on DVC system	
Containment after accident	2		Ionization chamber		1	1×10^{-2} to 1×10^{-5} Gy/h	Mesure recorded	
TEP filter room	1		Ionization chamber		No	1×10^{-5} to 1×10^{-1} Gy/h	Mesure recorded	
RCV filter room	1		Ionization chamber		No	1×10^{-5} to 1×10^{-1} Gy/h	Mesure recorded	
Spent resin tank room		2	Ionization chamber		No	1×10^{-5} to 1×10^{-1} Gy/h	Mesure recorded	
Concentrate tank room		1	Ionization chamber		No	1×10^{-5} to 1×10^{-1} Gy/h	Mesure recorded	
TES, drum filling		1	Ionization chamber		No	1×10^{-5} to 1×10^{-1} Gy/h	Mesure recorded	
OSG storage 1 area		1	Scintillator		No	1×10^{-3} to $1\times10^{+3}$MSV/h	Local alarm	224
OSG storage 2 area		1	Scintillator		No	1×10^{-3} to $1\times10^{+3}$MSV/h	Local alarm	

Table 12.3-2

Radiation Zone Designations.

Zone No.	Nuclear Island Premises Classification	Doss Rate (mrem/h)	
	Noncontrolled area	0.75	
	Controlled area		1
1	Green Zone : Permanent access area	0.75 to 2.5	
2	Yellow Zone : Limited access are	2.5 to 200	
3	Orange Zone : Regulated access area	200 to 10 rem/h	
4	Red Zone : Exceptional access area	10 rem/h <	

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-1

Filter-demineralizer bay

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-2
Elevation views in the NAB of
the installation of TEP 001EV
as regards to fluid activity

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-3.1
Korea Nuclear Island
Level -6,70
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-3.2
Korea Nuclear Island
Level -3,40
premises classification

Ulchin Nuclear Power Plant 1, 2

Final Safety Analysis Report

Figure 12.3−3.3
Korea Nuclear Island
Level +0.00
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-4
Korea Nuclear Island
Level +2,80
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-5
Korea Nuclear Island
Level +5,00
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-6
Korea Nuclear Island
Level +7,30
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-7
Korea Nuclear Island
Level +8,00
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-8
Korea Nuclear Island
Level +11,50
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-9
Korea Nuclear Island
Level +15,50
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-10
Korea Nuclear Island
Level +20,00
premises classification

KOREA NUCLEAR UNITS 9 and 10

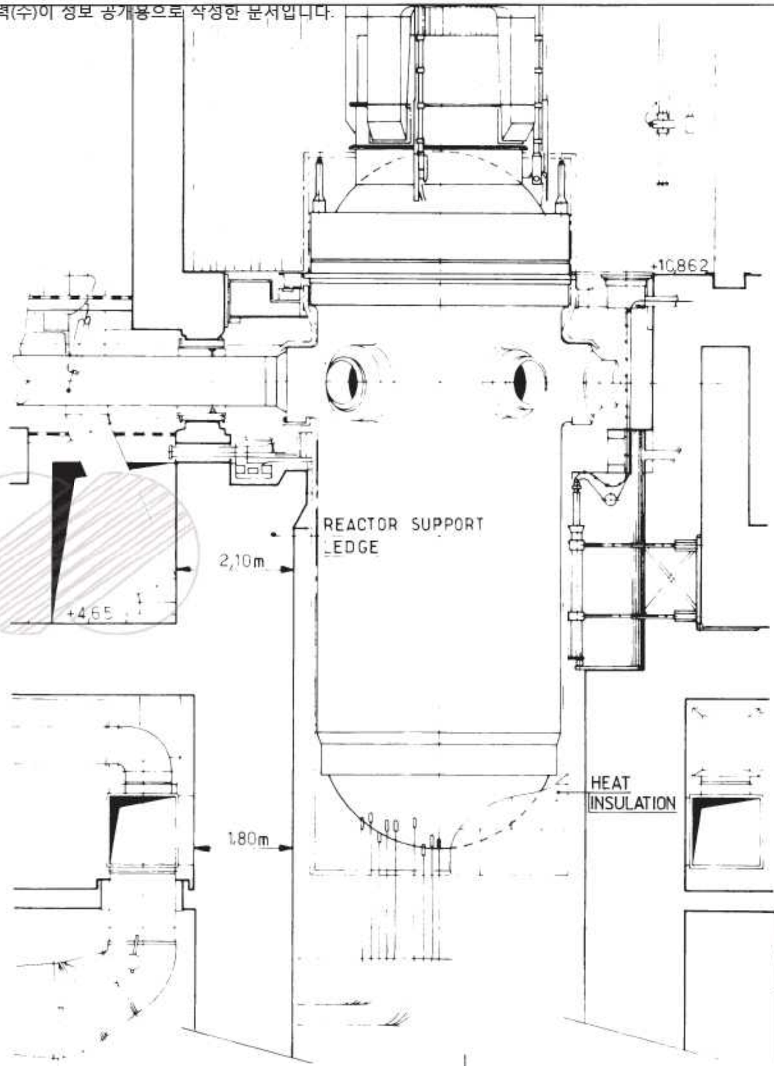
Final Safety Analysis Report

FIGURE 12.3-10.1
Korea Nuclear Island
Level +24.00
premises classification

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-10.2
Korea Nuclear Island
Section A-A
premises classification

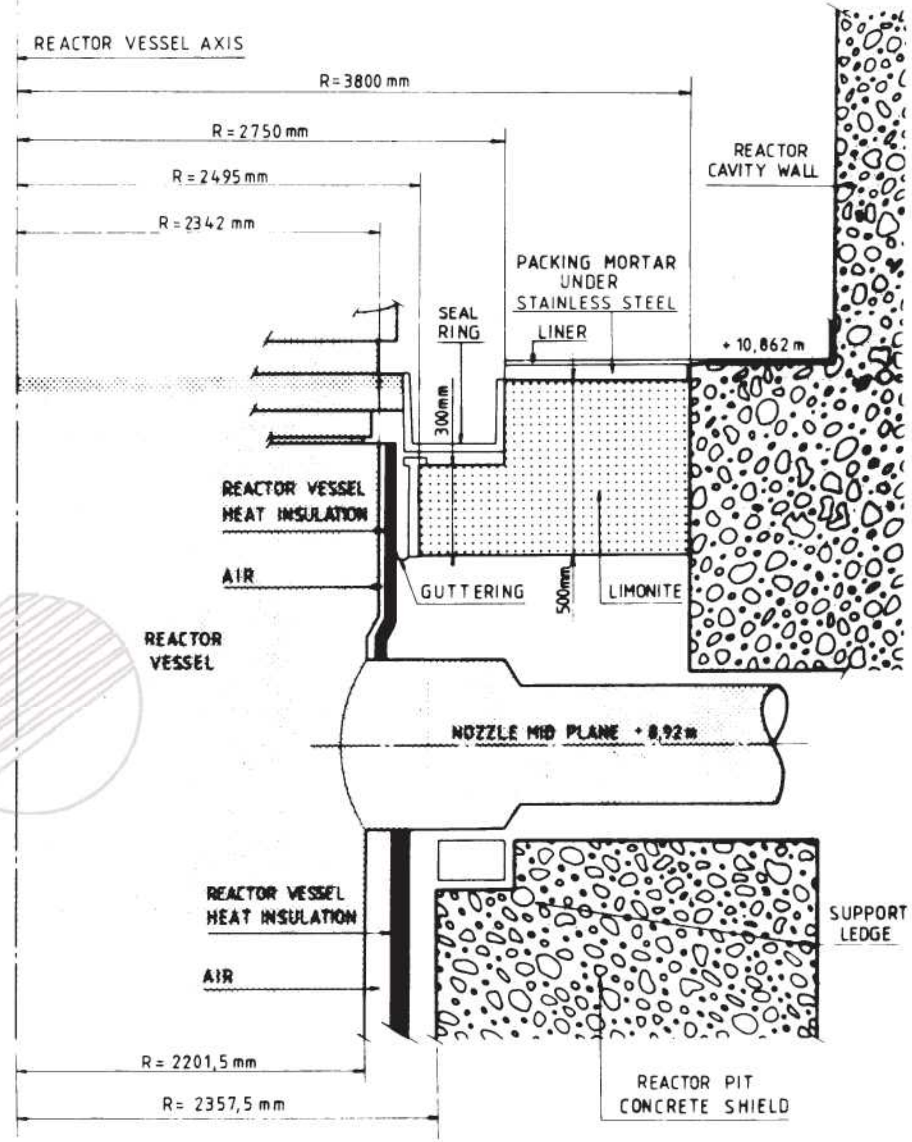


KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-11

Cross-section of the reactor pit



KOREA NUCLEAR UNITS 9 and 10

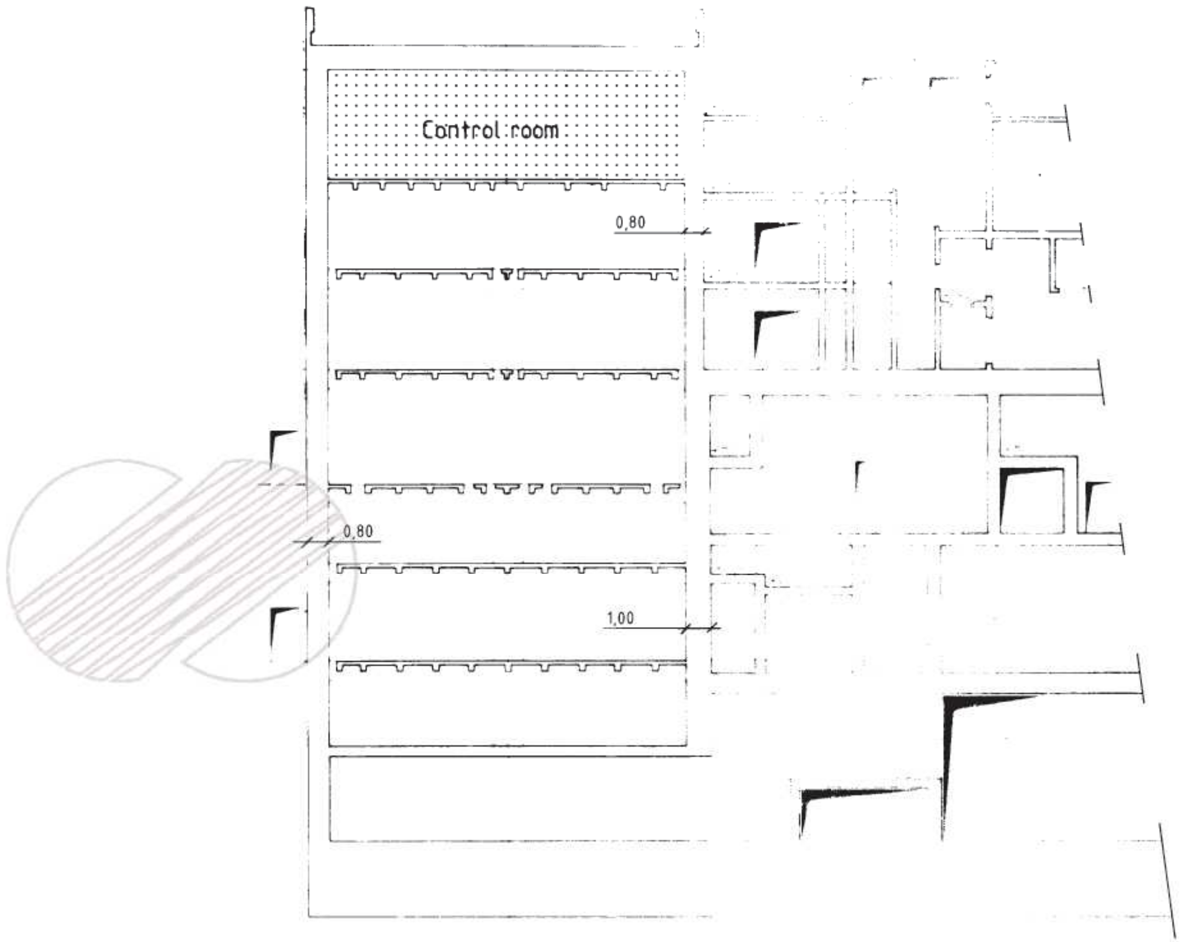
Final Safety Analysis Report

FIGURE 12.3-12
Simplified cross section of the reactor pit above core mid-plane

KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-13
Relation of control
rooms to the containment



KOREA NUCLEAR UNITS 9 and 10

Final Safety Analysis Report

FIGURE 12.3-14

Control room shielding

ULCHIN NUCLEAR UNIT 1 and 2

Final Safety Analysis Report

FIGURE 12.3-15

Old steam generator storage premises classification

TABLES

12.4. DOSE ASSESSMENT

12.4.1. General

The dose received by personnel exposed to ionizing radiations, shall be kept as low as possible and shall in no case exceed the maximum values set forth in Table 12.4-7 75 320

The collective dose resulting from the operation of a nuclear power plant reflects the total radiation detriment to workers. Moreover, depending on, its value, on plant operators and the number of maintenance people, it may entail an increase of this personnel in order to keep individual dose values below regulatory limits.

In consequence, attention has been given in France to the knowledge of the total dose and of detailed dose. This entails a continuous lowering of doses by improvement of design, operation and maintenance.

In particular, a dose assessment has been carried out by Electricite de France (EDF) for the 900 MW PWR FESSENHIM nuclear power plant in 1976.

A dose assessment (as described by R.G. 8.19) for the French reference Plant LE BLAYAIS 3-4 has not been performed. The FESSENHEIM dose 320
assessment was considered as good enough, and it was expected that LE BLAYAIS 3-4 collective doses would be lower since this plant incorporates improvements in design, special tools ... and benefits from prior operating experience.

In view of the number of 900 MW power plants that have now been commissioned in France, and of their similarity in design and operation, it is possible, by analysing the measurement records of each plant. to assess the anticipated dose rate results under the most significant normal operating conditions.

The similarity between UCN 1 & 2 and French power plants is such that the estimations made for the latter may be considered as totally representative of the doses estimated for UCN 1& 2. 320

Nevertheless the actual dose will depend more upon the operating and maintenance modes.

12.4.2. American operating plant data

This Subsection has been replaced by Subsection 12.4.4. in the FSAR.

12.4.3. In-plant dose estimates for FESSENHEIM nuclear power plant

The methodology and results used in the dose assessment at the FESSENHEIM nuclear power plant are presented in this Subsection as an example.

12.4.3.1 Dose assessment

a) Methodology

Dose assessment relies on analysis of the elementary tasks performed by workers. This necessitates the detailed knowledge of the action including the duration, the number of workers, the handling equipment used, and the tools employed. This also necessitates the knowledge of dose rates near the corresponding equipment and in rooms and working areas.

Profit has been taken from experience of French-Belgium PWR power plants, 320 MW CHOOZ and 900 MW TIHANGE. The results of a survey of dosimetric values at TIHANGE during the refuelling period of 1976, has been used for estimated dose rates.

b) Results

Results which are given below do not include :

- doses due to airborne radionuclides,
- doses due to direct radiations outside the nuclear island structures,
- doses to workers in FESSENHEIM Unit 2 during the erection of the unit when FESSENHEIM Unit 1 was operating

This is due to the fact that these doses are negligible with regard to in-plant radiation exposure.

Table T-12.4-4 gives estimated doses and exposure times within the nuclear auxiliary building during reactor operation and during refuelling shutdown. Taking into account the fact that some systems are common to the two Units, whereas others are particular to each Unit, the total exposure time for one Unit is 2,200 hours, with a total of 0.65 man-Sv(65 man-rem).

81 Table T-12.4-5 concerns the reactor building, for which the equivalent totals for one unit are 8,550 hours, and 3.7 man-Sv(370 man-rem). Table T-12.4-6 summarizes dose assessment. The calculated total dose is 4.35 man-Sv(435 man-rem).

12.4.3.2. FESSENHEIM experience

81 In normal operation - excluding shutdown - the total collective dose in 1979 was 0.195 man-Sv (19.5 man-rem) for both Units.

During the first shutdown period, including all regulatory tests and inspections, the collective
81 dose for one Unit was about 3.6 man-Sv(360 man-rem).

The following remarks can be made :

- benefit has been taken from FESSENHEIM 1 shutdown period for FESSENHEIM 2 and has allowed collective dose saving,
- the equipment which caused the highest dose rates was the steam generator (about 25 % of the total dose),
- the operation which caused the highest dose rates was the complete inspection of the main primary system, i.e., about 20 % of the total dose, out of which 60 % was due to access and to heat insulation removal.

  Improvements related to special tools, design features, and training of personnel are described in other parts of Chapter 12.

12.4.4. French operating plant data

12.4.4.1. Methodology

The anticipated annual doses and their estimated evolution up to an equilibrium valve situated at roughly the 10th refuelling cycle, were established on the basis of the following methods :

- The present system (as installed on all French plants since 1982) of automatic individual dosimetry recording consists of a calculator connected to dosimetry readers installed at the exits to all controlled zones. Among other data, it gives the values of the accumulated doses received per person, per operation.

  This system provides a reliable and detailed annual dose assessment and an indication of the comparative importance of the main shutdown operations.

- The collective dosimetry increases with the number of operating years and finally reaches an equilibrium value at around the 10th refuelling cycle. This growth is linked to the increase in activity within the plant systems, largely due to the deposit of corrosion products. The significance of this phenomenon can be expressed by an activity index (mrem/h) which is the mean doserate of the reactor coolant loops. This index was evaluated for the first 10 cycles, on the basis of the results of routine doserate measurements carried out in the vicinity of the main nuclear systems (RCP - RRA - RCV).

  A collective dose assessment for a reactor having reached radiological equilibrium can therefore be made on the basis of the estimated value of the equilibrium activity index.

12.4.4.2. Total dose assessment

a) Total dose for a refuelling shutdown period

The present mean dose value on French 900 MW units, as a whole, is :

- 180 man x rem for the first shutdown including full inspection of the facility,

- 120 man x rem for the second shutdown.

These dosimetric values include over-doses due to generic modifications in the construction of French 900 MW plant series, a large number of which have been incorporated in the KNU 9 & 10 unit design. Such over-doses can be estimated as follows :

- 10 % for the first cycle shutdown periods,
- 20 % for the following shutdown periods.

According to conservative assumptions, this dose increment is taken into account in the values estimated and therefore the man x rem dose assessment is that shown in Table T-12.4-1.

b) Annual dosimetric values

The total annual dosimetry in man x rem/reactor-year can be broken down into 25 % ex-shutdown time and 75 % shutdown time for a year which includes one normal refuelling shutdown period.

For a year which includes full inspection or decennial inspection, this is broken down into 10 % ex-shutdown time and 90 % shutdown time with inspection.

Ex-shutdown doses include all doses required by overall plant operation together with in-service maintenance operations performed within the nuclear island.

c) Equilibrium dosimetry

Once equilibrium has been reached, the mean values over a 10 year period including one decennial inspection, should normally be as shown in Table T-12.4-1, i.e. :

- unit shutdown dose : 228 man x rem,
- total annual dose : 280 man x rem.

It will be noted that the number of hours worked during a normal refueling shutdown period is from 100 000 to 150 000 distributed between the different operating personnel.

These dose values do not include :

- incidents leading to unscheduled overhaul operations, other than those having already occurred on French plants,

- equipment ageing.

12.4.4.3. Distribution of shutdown period doses

a) The dose distribution per group of operation personnel is given in Table T-12.4-2 and represents the national average for 1983 and 1984.

b) The dose distribution per primary shutdown operation is given in Table T-12.4-3.

Maintenance operations on plant system valves and fittings (RCP, RRA, RCV, etc.) represent, when assembled, roughly 18 % of total maintenance operations, whatever the type of shutdown.

12.4.4.4. Conclusions and facilities for the improvemental dose assessment

The estimated dose values quoted here above are closely linked to plant operation organization (work-force, work scheduling, etc.), to personnel qualification (training, practice on simulators etc.), to the techniques and tools employed and to the nature and number of inspections prescribed by the Safety Authorities.

The applicability of the assessment to KNU 9 & 10 is inevitably related to such parameters.

The comparison between the theoretical values of the dose assessment forecast for the Fessenheim plant and the annual equilibrium dosimetry determined on the basis of measurement records, has brought to light the extent of the means set up in France, since 1976, to reduce the doses to the utmost, both by the search for improved operating and organization methods and by the use of new techniques and equipment.

TABLE T-12.4-1

TOTAL ANNUAL DOSES DURING REACTOR OPERATION AND SHUTDOWN

	Cycle number									Equilibrium	
	1	2	3	4	5	6	7	8	9	10	11
Probable activity index in man-rem/h	55	61	65	77	84	100	113	123	127	130	130
Normal shutdown dose in man-rem	83	98	104	123	134	160	180	197	203	208	208
Shutdown dose with full or decennial inspection in man-ren	164										410
Annual mean dose with : 1 normal shutdown 1 decennial shutdown	103 184	120	130	154	168	200	225	236	254	260	260 462

TABLE T-12.4-2

NATIONAL AVERAGE VALUE IN 1983 AND 1984

Work function	Percentage of total dose at shutdown
Maintenance	92 %
Operation management	7 %
Radio protection Security	
Technical service	
Visitors	1 %
	100 %

TABLE T-12.4-3

MEAN VALUE IN RELATION TO THE TOTAL MAINTENANCE

DOSE UNDER NORMAL SHUTDOWN (in %)

Operations	Total maintenance	Normal shutdown
Reactor vessel opening and closure	8	11
Fuel handling	6	7
Maintenance in inspection of steam generators	18	25
Reactor coolant system - maintenance and inspection on pumps, valves and fittings	10	11
Systems RRA and RCV - Pumps, heat exchangers, valves and fittings	12	12
Other systems - valves, fittings and miscellaneous facilities	11	8
Regulatory checks and inspections	9	3
Cleaning, room decontamination, heat insulator removal, site installation and protections	26	23

TABLE T-12.4-4 (1/2)

FESSENHEIM DOSE ASSESSMENT : RESULTS FOR THE NUCLEAR AUXILIARY BUILDING

System	During operation	
	Duration (h)	Man-rems
For Unit 1 or Unit 2		
RCV	8	0,24
RIS	30	1,57
REA	15	0,68
RRI	115	0,24
SED	12	0,04
Total	180	2,77
For the NAB common to the two Units		
RRI	38	3,09
SED	42	1,06
PTR	115	1,42
REN	10	0,05
APG	15	0,05
RPE	90	3,05
TEP	200	28
TEU	310	14,22
TEG	70	0,5
TES	75	2,3
Total	965	53,74
For each Unit	482,5	26,87

TABLE T-12.4-4 (2/2)

FESSENHEIM DOSE ASSESSMENT : RESULTS FOR THE NUCLEAR AUXILIARY BUILDING

System	At shutdown for one Unit	
	Duration (h)	Man-rems
RCV	192	13,95
RIS	170	2,41
REA	120	4,16
RRI	350	1,91
PTR	50	0,72
REN	100	2,41
EAS	40	0,16
APG	60	0,31
ASG	100	0
RPE	140	5,37
TEP	75	4,13
SEB	45	0,28
SAR	20	0,01
SAT	10	0,01
RAZ	20	0,08
Total	1 495	35,91

TABLE T-12.4-5

FESSENHEIM DOSE ASSESSMENT : RESULTS FOR THE REACTOR BUILDING

Operation	Duration (h)	man-rems
General	216	0,65
RC pump seal inspection	586	21,1
RC pump motor inspection	88	1,68
Refuelling	1 675	47,35
Reactor vessel *	44	173,42
Steam generator *	137	31,78
Reactor coolant pump *	39	4
RV head *	38	0,97
Pressurizer *	19	1,76
Air compressor	102	0,28
Ventilation (EVF, RRM)	486	6,29
RRA pumps	14	1,7
Primary drain pump	46	1,78
Valves	5 050	76
Total	8 550	370

(*) yearly inspection

Assumptions :

1 - annual inspection of seals on 2 primary pumps,
- annual inspection of the motor on the third pump,
- complete maintenance on one EVF motor and one RRM motor,
- maintenance of one RRA pump.

2 - use of a MSDG (stud tensioning device),
- use of steam generator nozzle covers,
- use of EDDY current testing system for SG tube inspection

TABLE T-12.4-6

FESSENHEIM DOSE ASSESSMENT : FINAL RESULTS

Location	Operation	Duration	Man-rems
NAB	Plant shutdown	1 500 h	35
	Plant operation	700 h	30
RB	Plant shutdown	8 600 h	370
Total		11 000 h	435

These doses include :

- annual inspection (total 210 man-rems) which includes :
  - . reactor vessel 75 man-rems
  - . steam generator 30 man-rems
  - . reactor coolant pump 4 man-rems
  - . pressurizer 2 man-rems

- cleanup decontamination (total 48 man-rems)
- maintenance (total 100 man-rems) which includes :
  - . primary pumps 25 man-rems
  - . valves 5 man-rems

TABLE T-12.4-7

DOSE LIMIT

		Workers subject to occupational exposure		
		Category 1*	Category 2**	Public***
1. Effective dose limit		Maximum 50mSv/ 1 year 100mSv/5years**** (the five year duration is defined as the duration of every five years from 1998, for example : 1988-2002)	Maximum 6mSv/year	Maximum 1mSv/year*****
2. Equivalent Dose Limit	Crystalline lens	Maximum 150mSv/year	Maximum 15mSv/year	Maximum 15mSv/year
	Hands, Feet, and Skin	Maximum 500mSv/year	Maximum 50mSv/year	Maximum 50mSv/year

* Cat. 1 : regular exposure
** Cat. 2 : occasional exposure
*** Persons not subject to occupational exposure
**** Effective dose limit is limited to 200mSv in five years by Dec. 31. 2002, with an additional limit of 50mSv in any one year.
***** One time over exposure is permitted only for 1 year provided that 5 year-average exposure is within the dose limit.

320

12.5. HEALTH PHYSICS PROGRAM

12.5.1. Organization

12.5.1.1. Program organization

The KNU 9 and 10 organization is shown in Subsection 13.1.2. The plant superintendent is responsible for radiation protection and contamination control for the station. The Manager, Radiological Control Division 1
is responsible for administering the station radiation protection program which encompasses the handling and monitoring of radioactive materials, including special nuclear source, and by-product materials. He is also responsible for assuring that the station operation meets the radiation protection requirements of 10 CFR 19, 10 CFR 20, 10 CFR 50, Appendix I, and US NRC Regulatory Guides that are applicable to the health physics program throughout Section 12.5. The company commitment to the philosophies embodied in the above documents, and the authority to implement them, are discussed in Subsection 12.1.1.

The Assistant Manager, Health physics, who acts as assistant and should 1
be an experienced professional in applied radiation at nuclear facilities dealing with radiation protection problems, reports to the Manager
Radiological Control Division. He prepares necessary reports, per forms 1
shielding calculations for high radiation level jobs as part of the ALARA program, reviews old procedures and writes new ones, and performs other special assignments as directed.

The health physics engineers, who report directly to the Assistant
Manager, Health physics, handle the daily operations of the radiation 1
protection and health physics programs at the individual units. They supervise the health physics technicians.

The health physics technician performs the various surveys for radiation protection and the sample collection and analyses for waste disposal.

For a more detailed discussion of the responsibilities and authority of the supervisory positions mentioned above, and the training and qualifications of the personnel presently holding these positions, refer to Subsections 13.1.2 and 13.1.3, respectively.

12.5.1.2. Program objectives

The objectives of the radiation protection program are :

a) To provide administrative control of persons on the site to ensure that personnel exposure to radiation and radioactive material is within the guidelines of 10 CFR 20 and that such exposure is kept as low as reasonably achievable (ALARA).

   Administrative control directives will be prepared to ensure that all procedures and requirements are followed by all plant personnel.

Limits are developed consistent with the 10 CFR 20 Standard for protection against radiation. The radiation protection manual is designed to ensure that all procedures and requirements relating to radiological protection are uniformly and consistently followed by all plant personnel.

b) To provide administrative control over any plant effluent releases to ensure that these releases are below 10 CFR 20 and 10 CFR 50, Appendix I values.

12.5.1.3. Radiation protection program

The station radiation protection program will be officially initiated at Unit 9, and later at Unit 10, when radioactive material licensed to KNU 9 and 10 is first brought into the respective unit, and will be in effect continuously thereafter until the units are decommissioned. This program consists of rules, practices, and procedures that are used to accomplish the objectives stated above in a practical and safe manner. The program is consistent with the recommendations of US NRC Regulatory Guide 8.2, 8.8, 8.10 and 1.8.

The radiation protection program will ensure that :

- personnel receive appropriate radiation protection training,

- appropriate access control techniques and protective clothing are used to limit external contamination,

- respiratory protection equipment is used where needed to limit radiation exposure,

- radiation areas are segregated and appropriately posted to limit radiation exposure,

- instruments and equipment are properly calibrated so that accurate radiation, contamination, and airborne activity surveys can be performed,

- appropriate personnel dosimetry devices are supplied,

- an internal dose assessment program (whole body counting and/or bioassay) is supplied,

- incoming and outgoing shipments of radioactive materials are properly handled,

- necessary measures are performed to keep exposures ALARA consistent with safely supplying a reliable source of power to the public.

A more detailed discussion of the procedures used to implement this program is contained in Subsection 12.5.3.

The program also assures that appropriate effluent release samples are collected and analyzed consistent with the recommendations of RCC-P and US NRC Regulatory Guide 1.21 and 4.2 to verify that the station has little effect on the environment, and therefore the people offsite. In addition, the program assures that the emergency plans can be properly implemented, if necessary, to limit the consequences of any emergencies at the station as discussed in Section 13.3.

12.5.2. Equipment, instrumentation, and facilities

12.5.2.1. Controlled area

The plant design establishes a controlled area. The controlled area includes all areas in which radioactive materials are present or potentially present in quantities sufficient to require protective measures. The boundaries of the controlled area are defined in Paragraph 12.3.1.3.

The normal access and egress of the main controlled area is through the health physics station set up at the access control point(See Figure F 12.3-3) and is controlled by the radiation protection personnel. All other potential access points to the controlled area are kept locked or sealed. Temporary controlled access areas may be established in the clean areas of the plant and are subject to all rules and procedures of the controlled area. A radiation monitor is provided at the access control point and is used by all persons to check irradiation levels in the premises and to check themselves for contamination when leaving the area.

The controlled area is extended to include portions of the backyard area to allow access for trucks to the fuel and radwaste buildings. The extended controlled-area boundary is maintained by fencing, rope, or other continuous barricades. The backyard areas are not used to store contaminated materials. Personnel requiring entry, such as a truck driver, are provided with protective clothing, radiation dosimetry devices, and continual escort while in the controlled area. Upon leaving, the vehicle and personnel are monitored, and a record of the results is maintained. Only personnel who have received a radiation protection orientation or who are escorted by an individual who has received this orientation are permitted to make access to controlled areas.

Individuals permitted by a radiation work permit(RWP) to enter controlled area are provided with a radiation monitoring device which continuously indicates the dose rate in the area. The | 78
RWP is designed to inform the individual of radiation conditions in and adjacent to the area, specify the protective clothing and monitoring device requirements for entry, and specify the maximum occupancy time permitted. Any pertinent additional information is provided on the RWP.

High radiation areas are conspicuously posted and are maintained with a locked barrier which prevents unauthorized access.

12.5.2.2. Facilities related to radiation protection

Facilities for radiation protection are conveniently located for ingress to and egress from potentially contaminated areas of the plant. These facilities include the following :

- a sampling room where primary coolant samples are drawn,
- a radiochemistry laboratory where radioactive samples are chemically analyzed and/or prepared for radiochemical analyses. The laboratory is maintained at negative pressure to contain any airborne or gaseous radioactive materials released, and is equipped with constant airflow fume hoods, emergency showers, and eyewash stations,
- a counting room where radioactive samples are analyzed for isotopic composition and activity levels. It is shielded and air-conditioned to reduce background levels and fluctuations,
- radiation protection office and access control check point,
- a locker room where personnel change into clean work clothes and special clothing as required,
- personnel decontamination facility with showers and a washroom where personnel are monitored for contamination and appropriate measures are taken,
- a clean cloth storage facility,
- fixed area radiation monitoring system,
- a monitoring station where personnel exiting from the controlled area are checked for radiation contamination levels prior to leaving the area. The monitoring station includes a whole body contamination monitor and a frisking probe,
- an instrument decontamination facility with equipment handling facilities and a hot instrument shop to provide an area for repairing and maintaining contaminated instrumentation.

12.5.2.3. Special shielding

Access to orange and red zones is restricted by a locked door or gate,constructed to allow rapid entry or exit in case of emergency. Special shielding materials, such as lead block or sheet, are provided to personnel working in radiation areas and are used to reduce exposures whenever reasonable and practical.

Special equipment such as remote tools and handling equipment, and shielded transfer casks are provided and used for normal radioactive materials handling (i.e., filter changing). Abnormal and non-routine handling of radioactive materials is planned on a case by case basis and special shielding or tools are utilized to the extent practical to limit radiation exposure to personnel.

12.5.2.4. Radiation protection instrumentation

12.5.2.4.1. Laboratory radiation detection instrumentation

The laboratory-type radiation instrumentation located in the counting room includes the following instruments :

- a Hyper-pure germanium detector provided with conventional lead shielding and a multichannel analyzer,

- G.M. counter with G.M. detector,

- Automatic planchet counting system,

- a liquid scintillation counter, 1

- Automatic TLD reader, 1

12.5.2.4.2. Portable radiation detection instrumentation

The portable radiation detection instruments are stored in the portable instrument storage area in access control. They include :

- neutron detectors having dose-proportional output,

- alpha detectors having count rate output,

- low range and high range ionization chamber instruments having dose-proportional output for beta-gamma radiations, 320

- remote probe wide-range G.M. instruments having count rate output,

- an electrometer and a set of standardized ionization chamber thimbles. 137

12.5.2.4.3. Portable air sampling instrumentation 320

The portable air sampling instrumentation includes :

- Alpha air monitors,
- Beta air monitoring system,
- Regulated air sampler.

The air samplers are used to collect grab of radioactive particulates and halogens for subsequent analysis in the laboratory. These samplers are used for periodic sampling of localized areas prior to entry by operations or maintenance personnel. The continuous air monitoring collects and measures gross activity concentrations of airborne radioactive particulates, halogens, and tritium. These monitors are stationed in airborne radioactivity areas during personnel occupancy, and warn of increasing airborne radioactivity levels. The continuous air monitors can also be employed for routine surveys of gross airborne radioactivity levels throughout the plant.

12.5.2.4.4. Personnel radiation monitoring instrumentation

The radiation monitoring instrumentation includes :

- Beta-gamma dosimeters,
- Neutron dosimeters,
- Alarming pocket digital dosimeters.

12.5.2.4.5. Emergency instrumentation

Portable instruments are kept in the control room and in the visitors
2 center for access in the event of an emergency. These instruments are rotated with plant instruments semiannually to assure their proper functioning. They include :

- a wide-range G-M survey meter,
- a low level contamination detection instrument,
- a self-powered portable air sampler.
- 1 self reading dosimeters,
- various respirators.

12.5.2.4.6. Calibration of radiation protection instrumentation

The following instrumentation is tested and calibrated on the basis of UCN
101 1&2 operating technical specification and after each repair by the radiation protection technicians using a portable or mobile calibration facility. Repairs are performed by the instrument and control technicians.

- Portable radiation detection instruments
- Air samplers
- Personnel monitoring instruments
- Emergency instruments.

12.5.2.5. Equipment decontamination facilities

Decontamination areas are provided in the containment, fuel building, nuclear auxiliary building and in the decontamination shop. In the containment, hose and drain connections are provided for decontaminating the refuelling cavity liner following refuelling. In the fuel building, a cask decontamination pit and permanent spray nozzles are provided to decontaminate the spent fuel shipping cask. A decontamination station is in the solid radwaste area for locally decontaminating the outside of drums which contain solidified waste. In the hot machine shop, a centralized decontamination facility is provided for the decontamination of tools and equipment. The central decontamination facility will contain spray booths, ultrasonic baths, and chemical baths.

Typically, components of process system will be decontaminated in place prior to maintenance or equipment removal. Each process component, e.g., pumps, heat exchangers, and filters which contain a significant quantity of radioactive fluid is provided with flush and drain connections. These components will be flushed in place prior to maintenance. If necessary, a portable pump and drum of decontamination chemicals may be used to circulate the decontamination solutions through the component using the flush and drain connections. If further decontamination of components is necessary, they will be enclosed in poly bags and transported to the central decontamination facility.

Decontamination of large components will have to be treated on a case by case basis due to the varied configurations and locations of equipment. Where necessary, temporary curtains can be placed around components requiring in-place decontamination. Components in potentially radioactive areas are coated with decontaminable coatings to facilitate cleanup. If necessary, solvents or sandblasting will be employed to completely remove coatings for maximum decontamination.

12.5.3. Procedures

12.5.3.1. Radiation and contamination surveys

12.5.3.1.1. Policy

Whenever a potential or existing radiation hazard is present, routine and special surveys are done on a scheduled basis. Unscheduled surveys are conducted at the discretion of the Manager, Radiation Management Section, or at the request of other supervisors, for evaluation of work areas or situations where the exposure rate and/or contamination levels are unknown or are subjected to change. | 1

1993. 5. 10

12.5.3.1.2. Responsibility

The recording of complete and accurate surveys on proper forms is the responsibility of the Health Physics Group. The review, evaluation, and recommended measures to be taken, if any, are the responsibility of the
1 Manager, Radiation Management Section, or his designated alternate.

12.5.3.1.3. Types of surveys

a) Radiation

Periodic general radiation surveys are performed in clean and controlled areas as frequently as necessary, depending on the type, use, and potential hazard of the area ; or whenever radiation conditions are uncertain or changing. Specific radiation surveys are made on request, i.e., for an RWP. Continuing surveys are made in occupied areas when the potential of increasing radiation levels exists, while the area is occupied.

b) Contamination

Periodic contamination surveys (smear surveys), which are an evaluation of the level of removable surface contamination, are made in clean and controlled areas dependent on the type, use, and potential hazard of the area, and also whenever contamination levels are uncertain. Specific smear surveys are made upon request to evaluate and determine safe working conditions for specific jobs.

c) Air

Periodic air surveys, which are an evaluation of the concentration of airborne radioactivity present in any area, are made in clean and controlled areas dependent on the type, use, and potential hazard of the area, or whenever the presence of airborne contaminants are uncertain. Unscheduled air surveys are made upon request.

d) Water

Periodic surveys for determination of radioactivity present are made dependent upon the system function or whenever there is a possibility of the presence of amounts of activity which may either present a personnel hazard or affect plant operation.

12.5.3.2. Procedures and methods to maintain exposures ALARA

Procedures for access control to radiation areas or potential radiation areas are developed by the radiation protection staffs and are incorporated into the plant procedures manual. These procedures require a radiation work permit (RWP), and a survey of any area to be entered to ensure that occupational radiation doses are ALARA.

12.5.3.3. Controlling access and stay time

12.5.3.3.1. General

Persons not thoroughly familiar with the controlled area procedures are escorted by health physics technicians or are given instructions to assure adequate radio logical protection. Upon termination of work within the controlled area, workers leave immediately. Certain areas within the controlled area are posted with signs bearing such words as, "Authorized Entry Only". These areas may be entered only by individuals who have obtained proper clearance in the form of a radiation work permit. The purpose of the RWP is to control access to these areas and to limit exposure and contamination problems by informing the worker of radiation and contamination conditions and of the protective clothing required, or of other requirements to safely perform his job.

12.5.3.3.2. Entry into the controlled area

The following requirements are met prior to entry :

- TLD badges, dosimeters, protective clothing and other needed personal monitors are worn as specified by the appropriate RWP,

- there may be no exposed open wounds present on the body. All open wounds are sealed with a waterproof bandage prior to entry,

- all personnel who nave not received a radiation protection orientation are escorted by someone who has demonstrated his knowledge in this area to the satisfaction of the Manager, Radiation Management Section, or his 1
  designated alternate, 320

- entry normally is through the access control point. Entrance via any other route must be authorized by the plant superintendent or his designee.

12.5.3.3.3. Exit from the controlled area

The following procedure is followed upon exiting from the controlled area :

- exit is made through the access control point only. Exit via other route must be authorized by the plant superintendent or his designee.
- all protective clothing is removed at the step-off area.
- before leaving the monitor room area, personnel monitor themselves for possible contamination.
- all contaminated personnel report to the health physics group. Decontamination is effected prior to leaving the area.

12.5.3.4. Contamination control

12.5.3.4.1. Facility contamination control

Contamination of general plant areas by the movement of personnel between areas is controlled by using the step-off pad technique. A double step-off pad is employed for jobs involving high levels of contamination. plastic bags and absorbent paper are used to carry contaminated tools and equipment between areas. G-M count rate meters(friskers) are located at exit place in which radiation protection clothes are put off in normal operation, and are also located during outage at keypoint step-off pad like personnel hatch so that personnel can check themselves prior to
1 entering another area of the plant. To recheck contamination of the body, hand and foot monitors and portal monitors are also located at exit place before putting on personnel clothes. The final check point for all personnel leaving all restricted areas of the plant is the access control point where a portal monitor is located.

12.5.3.4.2. Personnel contamination control

12.5.3.4.2.1. Protective clothing

Contamination of personnel is controlled by the use of several types of protective clothing when entering contaminated areas :

- Every personnel who enter the radiation control area should wear coveralls, but personnel who enter the radiation control zone for the purpose of simple visit or patrol by guidance
1 of radiation worker, can wear only lapcoat considering "ALARA" concept Radiation Management Section Managers, approval in case of no possibility to take place any contamination or emergency accident.
- cloth shoe covers are worn in areas where dry contamination is encountered. In the case of wet
78 contamination, shoe covers are worn.
- cloth gloves are worn in areas where dry contamination is encountered. Rubber or plastic gloves are worn in the event of wet contamination.
- plastic suits are worn over cloth coveralls in areas where the potential exists for liquid contamination of personnel.

- cloth caps are worn for low-level contamination, cloth hoods for high-level contamination. 78

- protective clothing taped at the joints is worn by personnel when performing maintenance in contaminated areas. In cases where two sets 320 of protective clothing are worn, a double step-off pad is used.

Normally, most of the plant is accessible to personnel in street clothes. As a result, and to minimize the area in which protective clothing must be worn, temporary change areas are set up adjacent to the work areas for special maintenance jobs Also, permanent change areas are established for 320 areas routinely requiring protective clothing.
If at any time the number of separate areas requiring protective clothing becomes large enough to make travel about the plant adjacent to access control becomes the main area for the entire restricted part of the plant.

12.5.3.5. Airborne activity control

Individuals may enter an airborne radioactivity area only after being issued proper responsibility of the health physics group to survey the area and to specify the required devices needed according to the concentration and type of airborne problem is likely to exist, i.e.,
working with radioactive materials. Air contamination is kept to a minimum through the use of proper ventilation and decontamination of equipment and work areas. Respiratory protective devices may be required to prevent internal exposure in situations where airborne radioactivity exists. In such cases, radiation protection personnel sample the air and specify the type of respiratory device to be worn. Respiratory devices provided include half-face respirators with particulate and/or iodine filters. full-face respirators with particulate and/or iodine filters and supplied air breathing apparatus.

12.5.3.6. Personnel monitoring

12.5.3.6.1 Policy

This area of the radiation protection program deals with the wearing of proper personal monitoring devices, accurate recording of dose received proper evaluation of the reading received, and medical and bio-assay examinations and whole body counts as required. Proper personnel monitoring devices for the purpose of this procedure shall mean thermo luminescence dosimetry(TLD), filmbadge, and dosimeters. All personnel, regardless of their job are issued and must wear monitoring devices as specified while within the restricted area of the station.

12.5.3.6.2. Plant personnel

A self-reading, pocket-type dosimeter is provided for indication of external 320
radiation exposure on a day-to-day basis. Personnel assigned dosimeters are required to place them in the designated storage rack at the end of each work period. Determination of accumulated dose received is obtained principally from the interpretation of film badges and TLD's. This is an official and permanent record.

12.5.3.6.3. Personnel exposure investigations

In the event of an actual or suspected over-exposure, an investigation is
1 immediately initiated by the manager radiological control division or his designated alternate. Personnel exposures, TLD badges, radiation surveys, air samples, whole body counts, and special bio-assays are evaluated, or employed, as required for his investigation. If the over exposure is verified, the plant nuclear safety committee is notified. An investigation is carried out as specified by the chairman of the Plant Nuclear Safety Committee(PNSC)

12.5.3.6.4 Plant Personnel Exposure

12.5.3.6.4.1 External Dosimetry

Thermoluminescent dosimeter(TLD) badges are issued to the approved personnel entering in the controlled area of the plant. The badge has its own identification number and is assigned to each personnel. Badges must be worn at all times while within the controlled area and are put in the designated rack upon leaving the plant they may be picked up there again before entering the plant. 137

Badges are normally processed on a monthly basis. They are also processed as required if an individual has been involved in an emergency incident, or any time exposure to an individual is questionable. In this event, the individual is restricted from further exposure until his TLD is read and evaluation of the situation has been made. The skin and whole-body external exposure is logged upon measuring the personnel TLD.

A limited number of film badges are available for use by personnel where conditions of the job necessitate this and they serve as a backup in such situations for lack of TLD badges. All regularly assigned dosimeters are normally read, recorded, and set to zero by the user at least daily.

All radiation workers are given a complete baseline physical examination before commencement of work. All other personnel who randomly enter the controlled area of the plant should have a physical examination before entrance.

12.5.3.6.4.2 Internal Dosimetry

The internal uptake of radioactive materials in personnel working in controlled area of the plant is evaluated by urinalysis. Urinalysis is performed, if necessary, by the chief, radiological control section. 137

The whole-body counting sysem will also be used for the internal dose assessment. The plant procedure will dictate to all radiation workers to be whole-body scanned at least once a year. Special non-routine scanning may be requested by plant H.P. whenever significant internal radiation exposure is suspected. The over-exposure will be reported to the Plant Nuclear Safety Committee that will initiate further detail investigation.

12.5.3.7. Radioactive materials safety program

The storage, handling, transportation, and disposal of radioactive materials is subject to regulations which assure compliance with all applicable regulations so that personnel are not exposed unnecessarily.

12.5.3.7.1. Receiving radioactive material

1 When the ordering of radioactive material by anyone at the plant is anticipated, the health physics group is informed of the type and amount of material requested, the activity, and the physical and chemical form.

Shipments containing radioactive material from offsite sources must comply with all applicable regulations for packaging and labeling. The entrance of radioactive material into the site, the unpacking, and the delivery are authorized and supervised by the health physics group. If the shipment is not destined for immediate use, the manager, Radiological Control Division, has it placed in a suitable location and ensures that it is properly labeled. The manager, radiological Control Division, also assumed responsibility for reporting all instances of broken, leaking, or defective shipping containers to the applicable agencies. When contamination exceeds the limits allowed, the driver is notified and the health physics group supervises the decontamination of vehicles involved in shipment, as requested. Receipt of shipments may be refused for any reason relating to radiological problems, as determined by the Manager, Radiological Control Division.

12.5.3.7.2. Storing radioactive material

The storage of radioactive material is in the area designated by the health physics group. All radioactive materials entering and leaving the area are logged and the storage area itself is labeled so that it is clearly recognized as a hazardous area by all personnel.

12.5.3.7.3. Onsite transfer of radioactive material

Onsite transfer of radioactive material is rigorously controlled to minimize exposure. Radioactive materials that could cause unnecessary exposure to personnel are not moved into a work area unless the personnel in that area have been advised of the move. Containers for onsite transfer are constructed and/or shielded so that leakage or breakage does not readily occur. The shielding must be adequate to protect personnel in the area and those engaged in transporting the material.

All material is tagged and/or labeled prior to transfer, when the possibility of personnel exposure exists.

12.5.3.7.4. Fuel handling, storage, and shipment

The receipt, inventory (including location), disposal, and transfer of all fuel, new and spent, is in accordance with RCC-P and 10 CRF 70, Special Nuclear Material, and applicable regulations. The health physics group is responsible for surveys, both radiation and contamination, of all fuel prior to or during unpacking and storage. Surveys of the shipping containers also are performed before shipment from the site.

12.5.3.8. Radiation protection training

The objective of radiation protection training is to enable all personnel to safely carry out assignments involving potential exposure to radiation. Training programs are designed to cover the subjects in the depth required by various individuals. Each program covers the basic subjects, but additional material is covered according to the level of knowledge required for the individual to accomplish his job assignment safely.

The Manager Radiation Management Section, is responsible for the radiation protection training of KNU 9 and 10 employees and other individuals assigned to the station. It is his responsibility to ensure that all personnel assigned to KNU 9 and 10, other personnel working with radioactive materials or in controlled areas, are adequately trained. A record is kept of all individuals trained. | 1

1993. 5. 10