UK EPR	Title: PCSR – Sub-Chapter 14.6 – Radiological consequences of design basis accidents
	UKEPR-0002-146 Issue 06

Total number of pages: 91	Page No.: I / V
Chapter Pilot: F. CERRU *Name/Initials* [signature] *Date 14-11-2012*	
Approved for EDF by: A.MARECHAL *Name/Initials* A.Je.Maechal *Date 14-11-2012*	Approved for AREVA by: G. CRAIG *Name/Initials* G Craig *Date 14-11-2012*

REVISION HISTORY

Issue	Description	Date
00	First issue for INSA review	04.02.08
01	Integration of technical, co-applicant and INSA review comments	29-04-08
02	Integration of SGTR calculations	28-06-08
03	June 2009 PCSR update including: - clarifications of text - addition of references	29-06-09
04	Removal of RESTRICTED classification	10-06-10
05	Consolidated Step 4 PCSR update: - Minor editorial changes - Update of references - Clarification of text (§15.5, §15.6)	29-03-11
06	Consolidated PCSR update: - References listed under each numbered section or sub-section heading numbered [Ref-1], [Ref-2], [Ref-3], etc - Minor editorial changes - Statements added that the revised SGTR analyses do not impact the radiological consequences (§15.5)	14-11-2012

For information address:

AREVA NP SAS
Tour AREVA
92084 Paris La Défense Cedex
France

EDF
Division Ingéniérie Nucléaire
Centre National d'Equipement Nucléaire
165-173, avenue Pierre Brossolette
BP900
92542 Montrouge
France

TABLE OF CONTENTS

SUB-CHAPTER 14.6 – RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

0. SAFETY REQUIREMENTS

0.1. SAFETY OBJECTIVES

The EPR design is based on a deterministic safety approach, complemented by probabilistic analyses, based on the concept of defence in depth. In the approach followed, representative conditions that envelope situations that could be encountered during reactor operation are identified and grouped into 4 categories according to their frequency of occurrence (PCC-1 to PCC-4). The approach results in a system design in which initiating events are controlled, and consequential releases of radioactive substances into the installation and the environment are limited.

The aim in calculating the radiological consequences of the various transients (PCC-2), incidents (PCC-3) and standard accidents (PCC-4) that are significant from the point of view of radiological releases is to verify that the systems are properly designed and operated. It also aims to show that the discharge of radioactive products outside the plant having consequences for the public remain within the limits set out below.

0.2. RADIOLOGICAL OBJECTIVES

The general principle applied is that, for the transients, incidents and accidents considered in the design, the more frequent the event the lower the radiological consequences must be.

A principal objective of the EPR, in line with the above concepts, is to significantly reduce discharges in transients, incidents and accidents considered in the design.

The following radiological objectives are associated with each of the four event categories identified:

- Type PCC-1 operating conditions (normal operating conditions) and type PCC-2 transients, which must not result in normal operating limits being exceeded, are therefore not addressed in this sub-chapter. These transients are covered by the overall limit of 0.3 mSv/year, that applies to normal plant operation.

- The radiological objectives associated with PCC-3 and PCC-4 conditions are identical and, as specified in the Technical Guidelines [Ref-1], are based on the principle that protection measures for the population neighbouring the plant must not be necessary to meet the appropriate release limits. (However, restrictions on the consumption of certain foods produced in surrounding areas are not excluded.)

The population protection measures considered are those envisaged in the ICRP 63 [Ref-2] for the short-term phase of an accident, namely, sheltering and evacuation and distribution of iodine pills. The ICRP associated with these dose protection measures is considered to be optimal. For sheltering, the range proposed by the ICRP (for dose avoidance) is between 5 and 50 mSv effective dose and, for distribution of iodine pills, between 50 and 500 mSv effective thyroid dose. Long-term protection measures such as the temporary or permanent relocation of the population are excluded for this type of accident.

French government regulations use dose thresholds of 10 and 50 mSv (effective dose) for population shelter and evacuation, respectively and a threshold of 100 mSv (equivalent thyroid dose) for the administration of iodine pills. These intervention levels are used by the public authorities to implement population protection measures in urgent radiological situations.

The limits for food restrictions are the same as European Community limits [Ref-3].

The numerical target needed to comply with the EPR safety objectives may be adapted to site regulations and guidance. For the Generic Design Assessment, the proposed targets are based in the lower part of the ICPR dose targets, that is to say, for PCC-3 and PCC-4:

Effective dose	**10 mSv**
Equivalent thyroid dose	**100 mSv**

0.3. DESIGN REQUIREMENTS

The assessment of the radiological consequences of PCC-2 to PCC-4 accidents must demonstrate that the criteria in section 0.2 within this sub-chapter are met for each set of operating conditions. It thus forms part of the verification of the installation design.

1. GENERAL STATEMENTS AND ASSUMPTIONS

1.1. INTRODUCTION

The purpose of section 1 is to present all the data and assumptions common to the evaluation of the radiological consequences of the accidental sequences investigated in this sub-chapter.

The data and assumptions identified in this way concern the following parameters:

- The radioactive inventories involved.
- The physical and chemical forms of these inventories.
- The release mechanisms resulting from the above inventory forms.
- The retention mechanisms due to natural phenomena or to plant unit systems dedicated to retaining the contamination.
- The models and parameters used as a basis for calculating the atmospheric dispersion of radioactive substances as well as radiation exposures.

- The site specific dispersion conditions.
- The release height.

The assumptions detailed in the following paragraphs of this section are those used in the EPR reference calculations. They were established during in the Basic Design Phase of the EPR project and are partly based on German regulations.

In fact, despite design improvements such as design measures against severe accidents, in terms of its functional principles the EPR reactor is close to the reactors currently in operation in France. Constructing the EPR in France made it necessary to adopt a common methodology applicable to EPR and reactors operating in France.

Methods and assumptions adopted for evaluating the radiological consequences of Design Basis Accidents as well as severe accidents for the French Nuclear Power Plants (NPPs) in operation are applied, except for some identified design differences.

A simplified implementation of this common assessment method has been used for this safety report. Corresponding results are shown in Section 15. This approach will be reviewed to address UK specific requirements later in the licensing process.

The reference calculations (with the assumptions retained during the Basic Design phase of the EPR), for the chosen operating conditions, are presented in sections 2 to 14.

As required by paragraphs C.2.1 ("Single failure criterion and preventive maintenance") and D.2.4 ("Radiological consequences") of the Technical Guidelines [Ref-1], two specific sensitivity studies were carried out and are presented in section 16.

1.2. GENERAL STATEMENTS

1.2.1. Radiological Representative Design Basis Accidents

Radiological representative design basis accidents are chosen so that they are characteristic for initial and boundary conditions such as:

- type of release,
- release paths,
- height of release,
- operational modes.

The design basis accidents are also chosen to cover the various locations of possible leakage on the site. These include:

- containment,
- safeguard building,
- fuel building,

- nuclear auxiliary building,
- main steam and feedwater system,

Representative design basis accidents are chosen.

The selection of potential accidents giving rise to activity releases to the environment and checked on the basis of the above mentioned criteria lead to the following radiologically representative events:

1)	Large Break Loss of coolant accident (LOCA)	PCC-4
2)	Small break LOCA	PCC-3
3)	Rupture of a line carrying primary coolant outside containment	PCC-3
4)	Failure in liquid or gaseous waste system	PCC-3
5)	Failure of an equipment containing radioactivity in Nuclear Auxiliary Building	PCC-4
6)	Steam Generator Tube Rupture of 1 tube	PCC-3
7)	Steam Generator Tube Rupture of 2 tubes	PCC-4
8)	Fuel Handling accident	PCC-4
9)	Long-term Loss of offsite Power	PCC-3
10)	Main Steam System Depressurisation	PCC-2
11)	Loss of condenser vacuum	PCC-2
12)	Multiple failure of systems in Nuclear Auxiliary Building under earthquake boundary conditions	Equivalent to PCC-4
13)	RRA [RHRS] failure outside containment	PCC-4

1.2.2. General Methodology

The central idea behind the creation of the data to be used in the accident-related release calculation is to provide data leading to reasonably conservative results of the radiological consequences of the basic design accidents.

The pre-accident situation is normal power operation.

A final result is generally obtained as a function of a large number of parameters. Some of these parameters may vary widely, depending on the specific circumstances. If only extreme values were used (i.e. with a near zero probability of being exceeded in any given specific accident situation) the final result would be overly conservative. A sufficiently conservative result is obtained if one uses, for each of the parameters that can assume different values, a value which has only a 5% probability of being exceeded. In the unlikely cases where one or two of those parameters would exceed this value, it can be expected that the other parameters will have values below those taken for the analyses. Therefore even in these unlikely cases the overall result can be expected to be conservative.

In order to identify a value for a given parameter which is only exceeded with a probability of 0.05, the probability distribution fraction (in the area of "high values") must be known. This has been evaluated for the specific activity of fission products in the primary coolant as well as for the atmospheric dispersion and deposition conditions.

This procedure has been applied in determining the following data:

- Fission product activity inventory:
    - in the primary coolant,
    - in the adjacent plant systems taking into account the specific system data,
- Atmospheric dispersion and deposition conditions.

One important exception to this procedure arises when determining the specific activities of the secondary system (see section 1.3.3 within this sub-chapter).

A limited nuclide spectrum is assumed for the basis of the dose calculation. These nuclides were selected such that the calculated doses can be expected to be at least 95% of the values obtained by taking all the nuclides into account. This procedure leads to a situation in which different nuclide spectra are significant for the various accident scenarios.

Sensitivity analysis for the main key parameters is given in section 15 within this sub-chapter.

1.3. ACTIVITY INVENTORIES

1.3.1. Activity Inventory in the Reactor Core

The reactor core activity inventory is determined assuming a reactor thermal output of 4900 MWth (the actual power rate is only 4500 MWth, therefore the results are conservative), an equilibrium core with 5% U-235 enrichment and an average burn up of 43 MWd/kg. The precise boundary conditions for calculation of the activity inventories are shown in Sub-chapter 14.6 - Table 1, and the activity inventories of a series of nuclides which are radiologically relevant for the Large Break LOCA are compiled in Sub-chapter 14.6 - Table 2. These calculations were performed using the ORIGEN-S computer program [Ref-1].

The impact of a core with MOX-fuel elements will be discussed with respect to the calculated doses in the case of a Large Break LOCA and in the case of a fuel handling accident in section 15 within this sub-chapter.

1.3.2. Primary Coolant Activity

Comprehensive operational experience has been gained on the radiologically relevant nuclides of the primary coolant. It is therefore possible to determine a value which covers 95% of operational time with respect to the measured values of these nuclides.

For accident analyses the design values are used (See Sub-chapter 14.6 - Table 3).

With respect to the dose calculation all nuclides are selected except N-16 (very short half-life) and H-3 (minor radiological relevance).

1.3.3. Secondary Coolant Activity

The steam generator tubes are leak tight under normal conditions, and the secondary system is mostly free of activity. However, very small leaks in the steam generator tubes can be allowed to continue during power plant operation. To characterise such leaks for normal plant operation a primary/secondary side leakage rate of 3 l/h per steam generator has been assumed. Taking into account the 95-percentile values for the activity of the primary coolant as well as these allowable steam generator leakage rates, the potential secondary side specific activity values ("Design values") have been evaluated.

For the radiological evaluations of design basis accidents in which secondary coolant is released, but for which no break is postulated in the steam generator tubes, secondary-side activity values are assumed which should cover the full duration of the given relevant accident sequence. Thus, the basis for calculations is not the "operational specific activity" calculated under operating boundary conditions, but rather considerably higher values. These values are derived considering all phenomena (with the exception of steam generator tube breaks) which could possibly lead to an increase in the specific activity of the secondary system during the full duration of such a design basis accident. e.g.:

- Expansion of small existing holes or cracks caused by the increased pressure differential between the steam generator primary and secondary sides.

- Increase, due to spiking effects, in the specific activity of the reactor coolant leaking into the secondary system via these small holes or cracks.

The values of the specific activity derived in this manner for the secondary circuit are given in Sub-chapter 14.6 - Table 4. Since the fraction of the calculated dose due to noble gases is negligible, the noble gas activities are not included in this table.

1.4. MECHANISMS OF ACTIVITY RELEASE

1.4.1. Activity Release from Failed Fuel Rod Cladding in the Case of Reactor Shutdown and Other Transients (Spiking)

In the case of reactor shutdown and other transients, an increase in activity concentration (especially for the iodine and caesium nuclide groups) in the primary coolant may take place.

the time behaviour of the increase of activity concentration in the primary coolant is of special interest for the evaluation of radiological consequences of several Design Basis Accidents followed by a plant shutdown. This is especially true for the long life nuclides I 131, Cs 134 and Cs 137. The spiking behaviour of these nuclides is given in Sub-chapter 14.6 - Figure 1.

The spiking factors, which lead to the highest values of nuclide specific concentrations in the primary coolant after shut down, are given in Sub-chapter 14.6 - Table 3.

1.4.2. Activity Release Due to Discharge of Liquids

1.4.2.1. Discharge of Liquids with Evaporation

The data given in this section are related to reactor coolant characterised by a temperature above 100°C and discharged into the containment or building atmosphere.

Noble gases: 100% of the noble gases contained in the discharged reactor coolant are released into the compartment atmosphere and become entrained with the arising steam.

It is assumed that the weight-related activity concentration in the steam of iodine and other solids (caesium), including steam moisture, is 10% of the concentration of the out flowing coolant [Ref-1]. This is a conservative value which is backed by experiments.

1.4.2.2. Discharge of Liquids without Evaporation

Under blow down conditions, in which the temperature of the discharged liquid is significantly below its boiling temperature, it is assumed that the noble gas activity contained in the liquid is released entirely into the surrounding atmosphere.

Under the existing reducing conditions in the reactor coolant, any release of iodine and other solids is only possible via the liquid droplets (aerosols) arising in the course of the discharge sequence. The fraction of aerosols arising depends on the given spill height of the discharging liquids.

Experimental investigations of release fractions into the ambient atmosphere during liquid discharge sequences yielded the following results. These are based on aerosols having an aerodynamic equivalent diameter of ≤ 10 µm [Ref-1]:

- For a spill height of 1 m, the release fraction determined over several experiments is 1.2×10^{-6}; the highest value was:
  3×10^{-6}.

- For a spill height of 3 m, the release fraction determined over several experiments is 3.4×10^{-6}; the highest value was:
  5×10^{-6}.

For the radiological analyses of design basis accidents, release fractions are assumed in a conservative approach which are greater than the average of the above values by a factor of approximately 10:

- Given a spill height of approximately 1 m : 1×10^{-5}.

- Given a spill height of approximately 3 m : 3×10^{-5}.

For spill heights greater than these, an bounding release fraction of 10^{-4} is applied.

1.4.3. Activity Carryover into Steam Generated in the Steam Generator

1.4.3.1. General

A release of activity to the environment via the steam generator is dependent on the status of the steam generator tubes. Two cases are considered:

- Activity release to secondary side via normal operational leakages.

- Activity release to secondary side following a steam generator tube rupture (SGTR).

The first case is relevant for the three following design basis accidents:

- Long-term loss of offsite power (PCC-3).
- Main Steam System Depressurisation (PCC-2).
- Loss of condenser vacuum (PCC-2).

The second case is relevant for the two following design basis accidents:

- Steam Generator Tube Rupture of 1 tube (PCC-3).
- Steam Generator Tube Rupture of 2 tubes (PCC-4)

1.4.3.2. Activity Carryover in the Case of Operational Leakage

Due to the chemical composition of the reactor coolant, in particular the use of an alkalising agent as well as the reducing conditions caused by the addition of hydrogen, it can be assumed that all iodine present in the reactor coolant during reactor operation is in the form of water soluble iodide (I^-).

For the secondary system, as well, it is recommended that an alkalising and reducing agent be added to prevent corrosion (All Volatile Treatment or AVT, e.g. by hydrazine).

Therefore the chemical state of the iodine will not change if it leaks to the secondary side: the iodine remains in the form of iodine and is thus, under the operating conditions prevailing in the steam generator, dissolved in the liquid phase. It can therefore only be transported with the steam moisture (water droplets) into the steam outlet plenum of the steam generator. The steam moisture also includes water droplets arising from the bursting of steam bubbles transported with the steam-water mixture from the steam generator.

Based on a specified maximum steam moisture content of 0.25%, iodine carryover likewise amounts to 0.25% [Ref-1]. This value is thus identical to the carryover value for other solids (such as caesium).

1.4.3.3. Activity Carryover in the Case of SGTR

In this second case a rupture of a SG tube is considered. The consequences depend on the steam production rates and the location of SG rupture.

- **Case of High Steam Production**

a) Rupture Located at Tube Sheet:

In the event of tube breaks in steam generators operating at high steam production rates, such as during normal power operation, no change in droplet separation efficiency is incurred: the additional steam-water mass flows assumed to originate from a tube break are well within the separator and dryer design margins. The value of 0.25% specified for the steam moisture content can thus also be assumed to apply for carryover of iodine, caesium and other solids under these conditions.

As was noted in the case of normal steam generator operation, this steam moisture value also contains water droplets arising from the bursting of steam bubbles transported with the steam-water mixture from the steam generator.

A mixing of the discharged reactor coolant with the secondary water is assumed for the postulated break location.

b) Rupture Located at Water-Covered Tube U-bend:

This break location is expected to lead to an inhomogeneous activity concentration on the secondary water side. A multi-zone assessment of mixing has therefore be considered. It is assumed that iodine is released with the steam moisture taking into account the resulting iodine concentration in the upper part of the steam generator (secondary side).

c) Rupture Located at Uncovered Tube U-bend:

The primary water enters the steam generator secondary side in the form of a jet. Part of the water is evaporated directly (ca. 40%). In a similar situation, which was extensively analysed in experimental testing, the fraction of iodine which became airborne was always less than 1% [Ref-1] [Ref-2] [Ref-3] [Ref-4]. It is clearly dominated by the droplet-borne release fraction.

These tests yielded steam-borne carryover values of approximately 0.3% when the discharge flow into the space under consideration was unobstructed. In cases in which an impingement plate was located in the path of the discharging fluid jet, carryover values (concentration ratio) of approximately 0.8% were obtained, the maximum value being 0.84%.

It should be noted that no separation equipment was used in these tests such as a separator or a dryer.

A jet released from a ruptured tube can not penetrate the active part of a separator. For this to happen, 3 conditions would have to be met:

- The break must occur on a peripheral tube.
- It must be vertical and axial with a riser pipe.
- It must occur in the central part of the bundle where the risers are straight (peripheral risers are bent).

But even if all these three conditions are met, the jet would only reach the first stage of vanes but cannot go through because the vanes overlap each other. There is no direct way up and out. The jet will burst on the vanes and the very small amount of water which goes through is left with no axial momentum. It is deflected towards the wall of the separator. Clearly a jet cannot flow through the separators.

A jet could not go through the dryers: The steam has to make a 90° turn to flow through any drying module.

Due to this situation only a very small portion of the steam-water mixture discharged from a broken steam generator tube will reach the outlet nozzle.

It is therefore reasonable to assume that the steam moisture content for the carryover of iodine and caesium is no more than 0.25%.

- **Case of Low Steam Production.**

During periods of low steam production in the steam generator (such as those occurring after reactor trip), the separation efficiency of the separators and dryers caused by impact reduces due to the low velocity of the steam-water mixture. However, in this case additional separation due to gravity forces (so-called natural separation) takes place.

The fraction of water separated by gravity in the free space between the top of the separators and the dryers is a function of:

- separator design and size,
- the height of the free space,
- the mean velocity of the steam.

The velocity of the steam at residual power is in the range of 2 to 3 cm/sec. (0.9 to 1.0 m/sec at full power). For this low velocity and taking into account the free height between the separators and the dryers of a least 0.75 m, the efficiency of the gravity separation is similar to that occurring during power operation from centrifugal and inertial forces. Thus the moisture entrained in the steam exiting the steam generator will be below the value of 0.25%.

Therefore, in this case it is also reasonable to assume that the value for carryover of iodine, caesium and other solids is less than 0.25%.

1.5. RETENTION MECHANISMS

1.5.1. Retention of Radioactivity on Pathway within the Primary Circuit to the Leak (in the Case of Large Break LOCA)

In the case of a cold leg LOCA, the steam/water mixture has to pass through a long path from the reactor core to the leak location. Specifically, the steam/water mixture has to pass through the steam generator tubes.

As a consequence of the above characteristics of the pathway and of the fact that the circuit is partly filled with coolant, only a small fraction of the steam moisture and molecular iodine (I_2) will reach the containment atmosphere. Therefore retention of aerosols, aerosol-entrained iodine and molecular iodine within the primary system can be assumed to be 90%. In addition, it is very conservatively assumed that all radioactive substances which are transported by the steam being discharged into the containment atmosphere will also reach the containment atmosphere.

Following a Large Break LOCA the containment is isolated, and the radiological assessment considers the leakage from the primary containment to the secondary containment. The secondary containment is vented to atmosphere through the “Operational” and “Emergency” filters.

1.5.2. Filter Efficiencies

Where exhaust air filters are used to reduce the activity release, the following retention efficiencies are taken into account (see Sub-chapter 9.4) :

"Operational" HEPA-filters:

Noble gases	0	%
Aerosols	99.9	%
all other substances	0	%

"Operational" HEPA- and activated charcoal filters:

Noble gases	0	%
Iodine in organic compounds	90	%
Elemental iodine	99	%
Aerosols and aerosol-entrained iodine	99.9	%

"Emergency" HEPA- and activated charcoal filters:

Noble gases	0	%
Iodine in organic compounds	99	%
Elemental iodine	99.9	%
Aerosols and aerosol-entrained iodine	99.9	%

1.6. CALCULATION OF RADIATION EXPOSURE

The calculation of radiation exposure is performed for members of the population 500 m and more from the site boundary. All relevant pathways are taken into account. Radiation from the cloud, radiation from the soil due to deposited radioactive particles, inhalation and ingestion of food which is produced in the vicinity of the plant are all considered. The dose factors used are taken from the ICRP recommendations [Ref-1]. The models and parameters are taken from the German Accident Calculation Basis as this document describes all the relevant parameters necessary to calculate the doses. Consumption rates are taken from the German Radiation Protection Ordinance and are listed in Sub-chapter 14.6 - Table 36.

Food consumption restrictions are assumed to take place 24 hours after the beginning of the accident and within a radius of 2 km from the release point. It is assumed that the food produced in this area is not used for the first year after the accident. Outside this area no mitigation measures are assumed. Therefore for the ingestion pathway the dose at 2 km distance may be greater than the dose at 1 km distance.

1.7. SELECTION OF APPROPRIATE WEATHER CONDITIONS

Depending on the prevailing weather conditions, the dispersion and the deposition of the released radionuclides is subject to a wide variability. Consequently, that the results of the evaluated radiation exposure are influenced to a high degree by the assumed meteorological conditions.

For design purposes, it is necessary to select an appropriate value from this wide variety of data.

For the radiological calculations described in this sub-chapter, a probabilistic approach is used. The real weather situations of a representative year (hourly documented weather data) are used to calculate the doses. Using this approach, the value which covers 95% of the cases is judged to be adequately conservative.

1.8. ATMOSPHERIC DISPERSION

The atmospheric dispersion is calculated using a Gaussian Model.

1.9. FALLOUT AND WASHOUT

Fallout and washout factors are used to calculate the amount of radioactive substances that have been deposited during a dry weather period or during a period of precipitation respectively. The fallout and washout factors are not only a function of atmospheric parameters, e.g. wind velocity or precipitation rate. They also depend on the individual substance groups, such as elemental iodine, organically bound iodine and aerosols (including the aerosol-entrained iodine). The characteristic parameters used to describe these dependencies are the deposition velocity for the fallout and the washout coefficient for the washout [Ref-1].

Substance group	Deposition velocity, in m/s	Washout coefficient in s^{-1}
Elemental iodine	10^{-2}	7×10^{-5}
Organically bound iodine	10^{-4}	7×10^{-7}
Aerosols	1.5×10^{-3}	7×10^{-5}

1.10. RADIATION EXPOSURE

The following exposure pathways are considered in the dose calculation for design basis accidents:

- gamma-radiation from the passing plume,
- inhalation of radioactive substances by persons affected by the plume for the time during which the plume passes,
- gamma-radiation from radioactive substances deposited on to the ground surface,
- ingestion of foodstuff contaminated by radionuclides.

The exposure period for to the radioactive substances deposited on the ground surfaces as well as the ingestion of foodstuff is assumed to be the whole life of the individual. This is assumed to be 70 years for infants and 50 years for adults. In addition, the portions of the dose related to internal radiation exposure, due to inhalation and ingestion cover the same time period (so-called committed dose).

The dose factors are taken from the recent ICRP publication [Ref-1]. These factors are consistent with the dose factors given in the Council Directive 96/29/Euratom [Ref-2].

The "Council Directive 96/29/Euratom" lays down basic safety standards against the dangers arising from ionising radiation [Ref-2]. It emphasises that only the effective body dose and the exposure of the skin and the eye lens have to be limited. In the case of activity releases and radiation exposure due to design basis accidents the dose of the skin and the eye lens is only of minor importance. Therefore, these doses are not given in this report.

2. LARGE BREAK LOCA (PCC-4)

2.1. JUSTIFICATION OF THE SELECTION OF A COLD LEG BREAK (RIS [SIS] LINE)

The LOCA case of a break of the RIS [SIS] line at the nozzle in the cold leg of the RCP [RCS] is selected as the bounding case for the following reasons.

In LOCA cases the calculated effective dose results mainly from iodine exposure of persons living in the plant environment. Therefore the assessment focuses on the iodine release.

Compared to equivalent medium hot leg breaks, a break of an RIS [SIS] line leads:

1) to a bigger core heat up,

2) to a considerably longer steaming period within the reactor core,

3) to a higher containment pressure increase,

4) to the radionuclides released from the defective fuel being concentrated in a comparably small coolant volume.

2.2. MAIN RESULTS OF THERMAL-HYDRAULIC ANALYSES RELEVANT FOR RADIOLOGICAL ANALYSIS

The main results of the Thermal-Hydraulic (T/H) analysis of the Large Break LOCA and specific assumptions for the radiological analysis are summarised below. The LOCA case of the break of the RIS [SIS] line at the nozzle in the cold leg of the RCP [RCS] is selected as the bounding case:

- During the first 300 seconds, the heat up phase of the fuel rod cladding, about 40% of the break flow is steam.

- The subsequent phase lasts until steaming to the containment finishes, i.e. until 90 minutes.

In this phase (90 minutes):

- the average reactor coolant mass in the active core region is about 12.5 t and the core is always cooled by a two-phase mixture

- 200 t of steam are produced in the core, but only 120 t of the 200 t are discharged to the containment atmosphere.

After 90 minutes the steam release to the containment is stopped and in the long term the total coolant mass for cooling the core is approximately 2000 t.

The T/H analyses do not predict any core damage (due to the slow temperature rise of the fuel rod cladding). Nevertheless, to obtain clearly conservative results, the radiological analysis considers a cladding failure fraction of 10% at 300 sec.

All other activity releases into the containment are negligible in comparison with the activity released from the fuel rods due to cladding failure. This assessment therefore focuses only on the activity released from the failed fuel rods. As stated above, a fuel rod cladding failure rate of 10% is assumed.

The following fractions of the activity inventory of the failed fuel rods [Ref-1] are released into the reactor pressure vessel (RPV) from the burst fuel rods and spontaneously into the containment atmosphere, respectively:

	Release into the RPV within 90 minutes (% of core inventory)	**Spontaneous release into the containment (% of core inventory)**
Kr-85	5	5
Xe-133 and all other noble gases	2	2
Iodine (non-volatile)	2	0.2
Iodine (volatile: elemental iodine)	-	0.04
Caesium	2	0.2

The release fractions given for noble gases bound the experimental data; a spontaneous release (dry release) of these noble gases is assumed.

Iodine and caesium may be released during the total accident sequence in additional to the release from the failed fuel pins,.

During the 90 minute period the reactor core is cooled by a two-phase mixture. The situation within the reactor core is very similar to the normal operation of a boiling water reactor. The exception is that in the case of the boiling water reactor, the steam/liquid velocities are much higher and separators and dryers are provided. During operation and for some time after shutdown, the temperature of the fuel is so high that liquid water cannot penetrate a defective fuel rod. During this period, a steam cushion is formed as the temperature of the fuel is above the boiling point of the liquid. Once the temperature decreases because of the decrease of the decay heat, the fuel surface temperature is insufficient to prevent a penetration of the liquid coolant through the cladding leaks. These effects result in a low leaching effectiveness in the early phase of the accident. Consequently, a leaching rate of almost 2% for iodine and caesium may be assumed for this time period to 90 minutes after the start of the accident. [Ref-2]

In order to simplify the calculation it is assumed that leaching occurs at a constant rate during this time period and any decay which would actually occur is not taken into account. During this period steam produced in the core is discharged to the containment atmosphere. Thus, this period is important for the activity release into the containment atmosphere and, due to any containment leak rate, to the environment.

The activity release with the steam in these circumstances is very similar to the situation in the reactor pressure vessel of a boiling water reactor during normal operation. Therefore one can expect that the same release mechanisms concerning the carryover by steam moisture are relevant for this period as those in the boiling water reactor during normal operation.

Operational experience of boiling water reactors show that the iodine carry over is composed of steam moisture at about 2%, when the steam dryer is removed. In addition, carry over of volatile iodine occurs, preferentially in the form of I_2. This is also at a rate of order of 2%. This second component is mostly formed by radiolytic oxidation of I^- to I_2 under the oxidising conditions of a boiling water reactor which is operated without hydrogen addition. Under the condition of a PWR with a PH value greater than 7 no release fraction of volatile iodine is assumed.

The carry over of the leached caesium is also only related to steam moisture (2%).

It is also assumed, that all radioactive substances, which were transported by the steam (including the steam moisture) and are discharged through the break will also reach the containment atmosphere.

After this period the core is completely covered with liquid coolant. Thus the steam production in the core region and the steam release to the containment is stopped. The radionuclides in the coolant are gradually distributed into the total water mass of about 2000 t. The temperature of the coolant reduces and is soon significantly below the boiling temperature of water.

In this phase the airborne concentrations of the radionuclides falls. Molecular iodine and aerosol-entrained iodine will deposit on surfaces. The quantities of both of these iodine forms in the containment atmosphere will therefore diminish considerably with time.

The decrease of aerosols and thus also of the aerosol-entrained iodine occurs in particular through sedimentation and diffusiophoresis.

On the other hand the discharge of the coolant out of the break leads to aerosol entrainment, even if the temperature of the coolant is significantly below the boiling temperature of said liquids.

It can be assumed that in this phase the effects of sedimentation and diffusiophoresis dominate in the case of a large break LOCA. Nevertheless, a decrease in the concentration of aerosols in the containment atmosphere is not assumed in the calculations.

For elementary iodine (I_2) a similar behaviour to that of aerosols is assumed for the reduction in the containment atmosphere until the equilibrium by volume is achieved between the total iodine concentration of the sump water and the I_2 concentration of the containment atmosphere. This is reached at a ratio of 10^4 to 1. [Ref-3]

A half-life of seven hours is assumed for the rate of decrease of elementary iodine in the containment atmosphere [Ref-3].

Some of the elementary and particulate iodine may be transformed into organically-bound iodine through chemical reactions with organic materials within the containment (e.g. protective coatings). It is assumed that 0.3% of the inorganic iodine released to the containment is converted into organic iodine over the long term. To simplify the calculation it is assumed in the radiological calculation that this portion of iodine is converted into organic iodine during the early phase of the accident sequence.

The calculations were performed using the ACARE computer program. [Ref-4]

The release fractions from the core inventory which escape into the containment atmosphere are given in Sub-chapter 14.6 - Table 6.

2.3. ACTIVITY RELEASE FROM THE CONTAINMENT TO THE ENVIRONMENT

As a result of the pressure rise and history inside the containment, very small fractions of the containment atmosphere reach the annulus and finally the environment. The route is via the emergency filter system, with HEPA and charcoal filters, and via the stack. To simplify the calculation no account has been taken of the variation of the containment leakage rate due to the variation of pressure with time. The design leak rate of 1% Vol. per day is conservatively assumed to be constant for 24 hours. Note that this value is very conservative as the maximum leak rate of a containment with a steel liner is only 0.3% Vol. per day.

The ventilation rate of the annulus is assumed to be 200 m^3/h. This is consistent with the present design of the blowers. The following efficiencies are assumed for the annulus exhaust air emergency filters:

- 99.9% of elementary iodine (I_2).
- 99.9% of aerosol-entrained iodine and of other aerosols.
- 99.0% of organic iodine.

Sub-chapter 14.6 - Table 5 contains the activity release values. The calculations were performed using the ACARE computer program. The release increase between 24 hours and 168 hours is due to the annulus inventory (there is no further leakage from the containment after 24 hours).

After 168 hours there is, in theoretically, amount of the potential inventory (about 20%) not released from the annulus. However, depletion of aerosols that occurs also in the annulus has been conservatively ignored. The calculated release pattern is therefore conservative even when this remaining 20% inventory is not taken into account.

2.4. POTENTIAL RADIATION EXPOSURE

The calculations were performed using the PRODOS computer program. [Ref-1]

The resulting potential effective dose values for adults and infants including all pathways are given in Sub-chapter 14.6 - Table 7. Sub-chapter 14.6 - Table 8 shows the thyroid dose for adults and infants due to inhalation. All of the values are below 0.1 mSv.

3. SMALL BREAK LOCA (PCC-3)

A Small Break LOCA is a much less severe accident when compared to a Large Break LOCA. No fuel damage occurs during this accident. The pressure in the Containment will also only increase to a very low value. Therefore the release of radioactive nuclides from the containment will be essentially zero. The radiological consequences of small break LOCA cases are therefore much lower than the radiological consequences of the representative large break LOCA. Therefore no specific calculations for these minor radiological consequences were carried out.

4. RUPTURE OF A LINE CARRYING PRIMARY COOLANT OUTSIDE CONTAINMENT (PCC-3)

During normal plant operation there are several potential events which could be selected as the representative accident sequence. It is therefore necessary to select a representative event. A representative loss of coolant accident outside the containment during the operation of the residual heat removal system is dealt with in section 14 of this sub-chapter.

4.1. THERMAL-HYDRAULIC DATA

The loss of primary coolant may result from the failure of:

- either the Chemical and Volume Control System (RCV [CVCS]), including the RCV [CVCS] connecting lines,
- or the Nuclear Sampling System (REN [NSS]).

The most onerous pipe break is respectively:

- the 2A rupture of the RCV [CVCS]-line, located between the Volume Control Tank (VCT) outlet and the VCT suction valves,
- the 2A rupture of a RCV [CVCS] connecting line, located in the nuclear auxiliary building,
- the 2A rupture of the pressuriser liquid phase sampling line, located between the containment penetration and the heat exchanger.

In each case the leak isolation time is conservatively assumed to be 60 minutes, from the pipe break occurrence, as the result of the two following components:

- 30 minutes to reach the signal actuation (sump level high, activity high...),
- 30 minutes to perform the operator action (manual leak isolation from the main control room).

In the case of a RCV [CVCS] line break, the primary coolant leak rate consists initially of fluid drained from the VCT and subsequently from the RCV [CVCS] letdown flow rate entering the VCT. The letdown flow is conservatively assumed at its maximum constant value until the isolation occurs after the 60 minute delay. Two cases are considered depending on the initial content of the VCT:

- Case A: VCT initially full of water (20 m^3 water)

the RCV [CVCS] leak rate is,

  - 225 t/hour at 1 bar/50°C for 8 minutes (before VCT emptying),
  - 72 t/hour water from RCP [RCS] at 1 bar/50°C for 52 minutes (after VCT emptying),

- Case B: VCT initially empty (20 m^3 H_2 or N_2 at 7 bar)

the RCV [CVCS] leak rate is,

  - 120 m^3 H_2 or N_2 at 1 bar /50°C instantaneously released (VCT emptying),
  - 72 t/hour water from RCP [RCS] at 1 bar/50°C for 60 minutes (after VCT emptying)

In the case of a RCV [CVCS] connecting line break the leak rate is assumed to be 72 t/hour of water from the RCP [RCS] at 1 bar/50°C for 60 minutes.

In the case of a REN [NSS] line break, the REN [NSS] maximum leak rate is assumed at the critical break flow rate:

- 1.5 t/hour water from the pressuriser at 1630 kJ/kg (155 bar/345°C) for 60 minutes .

4.2. SELECTION OF REPRESENTATIVE ACCIDENT

In order to determine the radiologically bounding accident, the boundary conditions for the accident sequences must be investigated in greater detail.

The important variables influencing the activity release are the following:

- The specific activity of the discharged liquid,
- The mass of discharged liquid,
- The spill height of the discharged liquid,
- The thermodynamic boundary conditions which influence the release of activity into the ambient atmosphere.

as well as:

- The assumed boundary conditions for the heating, ventilation and air-conditioning systems, including the exhaust air filtering in operation under accident conditions.

The above-mentioned three cases are based on the same specific activity in the discharged liquid (unpurified primary coolant water) as well as the same (or at least very similar) boundary conditions for the heating, ventilation and air-conditioning systems.

As a result of the assumed leak isolation time reactor coolant can flow from the pipe break for a maximum of 60 minutes. In the event of a guillotine break in the line at the Volume Control Tank, 92 t coolant will be discharged, assuming the tank was initially full. In the event the tank was empty, the discharged mass is 72 t. A mass of 72 t is also discharged in the cases of a guillotine break in a RCV [CVCS] connecting line. Approximately 1.5 t of liquid are discharged over the 60-minute period from the postulated break in the Nuclear Sampling System.

Two nuclide groups, which have very different release behaviour into the ambient atmosphere, are responsible for the potential radiation exposure of persons living in the plant environment as a result of these accidents:

- noble gases,
- all other nuclides, which are released in the aerosols (liquid droplets, which are generated during the reactor coolant discharge process.

The release of noble gases is directly proportional to the discharged coolant mass, whereas the release of the other nuclides depends on the amount of aerosol formation during the discharge process.

Therefore, the aerosol formation is of particular importance in addition to the discharged mass of reactor coolant. The aerosol formation is influenced by the thermodynamic boundary conditions, 'hot' or 'cold' water, and by the spill height of the discharged water. The detailed issues for the release mechanism are described in detail in section 1.4.2 as they apply to the various release locations. They cover the following aspects:

- The variable thermodynamic boundary conditions to which the reactor coolant is subjected in the systems under consideration
- the influence of the spill height at the break location.

Therefore the spill heights at the break locations considered also need to be determined.

It is planned to locate the RCV [CVCS] lines in the Fuel Building slightly above the floor of the affected room (in general: 0.5 m or less). The RCV [CVCS] connected lines will be located partly in the vertical direction within the nuclear auxiliary building. The greatest spill height inside the nuclear auxiliary building is 5.8 m.

These spatial differences of the planned run of piping have the following consequences for aerosol formation following a REN [NSS] leak. For a spill height for the discharge of "cold" reactor coolant of approximately 0.5 m, the aerosol formation in the case of the REN [NSS] leak is higher by a factor of approximately 4000 due to spontaneous vaporisation of the discharged reactor coolant.

For a spill height of more than 3 m and up to 5.8 m, the corresponding value is 400. This effect leads to a greater aerosol formation in the case of the REN [NSS] leak compared to the cold discharge. This is because the reactor coolant mass in the case of the RCV [CVCS] leak is greater than that of REN [NSS] by only a factor of between 50 and 60. This assumes that the leak is isolated, 60 minutes after it occurred in both cases.

The results of this analysis can be summarised up as follows:

The RCV [CVCS] leak or a leak in a RCV [CVCS] connecting line is the more onerous case for radiation exposure due to noble gases. The REN [NSS] leak is the more onerous case for radiation exposure due to other nuclides (iodine and caesium nuclides).

Therefore, both accident cases have been analysed. The guillotine break of a nuclear sampling system line in the Fuel Building is the conservative case of a rupture of a line carrying primary coolant outside containment. This is based on the boundary conditions for the Heating, Ventilation and Air-Conditioning (HVAC) System given in the next section.

4.3. ASSUMED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

The assumed scenario of the "Rupture of a line carrying primary coolant outside containment " is as follows:

Prior to the isolation (assumed at 60 minutes) approximately 1.5 t of reactor coolant are discharged from the leak of a Nuclear Sampling System line. Of this approximately 0.6 t evaporates instantaneously during the discharge.

Information concerning the activity inventory of the reactor coolant used in the accident analysis is presented in Sub-chapter 14.6 - Table 3.

The break of a line of the Nuclear Sampling System in the Fuel Building does not cause the reactor protection system to respond. However, it is conservatively assumed that the reactor is manually tripped when the signal actuation occurs. This occurs 30 minutes after the start of the accident. Following the reactor trip there is an increase in the release of radioactivity from failed fuel rods due to temperature and pressure transient (spiking effect) Consequently, an elevated radioactivity concentration in the coolant must be assumed. The increased radioactivity concentration is assumed for the remainder of the reactor coolant discharge after the shutdown of the reactor. For the radiologically relevant nuclides iodine-131, caesium-134, and caesium-137 an exponential activity increase is assumed as described in section 1.4.1.

It is assumed that the entire noble gas inventory of the discharged coolant is released into the building atmosphere. A portion of the iodine and the solids, discharged with the reactor coolant, is released with the moisture entrained in the steam flow and hence into the building atmosphere,. It is assumed that the radioactive concentration by weight of iodine and solids in the steam, including the moisture, is 10% of the concentration of the discharged coolant which has not evaporated. The release of iodine and other solids into the building atmosphere is only associated with the water droplets entrained in the steam.

The form of iodine may be subject to changes. In general, these transformations are very slow processes. In the German Accident Calculation Basis these transformations are assumed for activity releases over short periods: It is assumed, that 10% of the released iodine is transferred immediately into elemental iodine.

The air exchange rate of the sampling system rooms is approximately 4 Vol./hour. The air from these rooms is discharged via HEPA-filters and the stack to the environment. However, no credit has been claimed for the retention in these HEPA filters in the activity release calculation. This is because these filters are not claimed to be effective under such accident conditions.

Sub-chapter 14.6 - Table 9 presents the activity release values. The calculations were performed using the ACARE computer program.

4.4. POTENTIAL RADIATION EXPOSURE

It is conservatively assumed that the release occurs at ground level.

The calculations were performed using the PRODOS computer program. The different forms of iodine are only used for the dispersion calculation. For the dose calculations, the most conservative dose factor of any form of iodine is used for the total amount of iodine.

The resulting potential effective dose values for adults and infants are presented in Sub-chapter 14.6 - Table 10. Sub-chapter 14.6 – Table 11 shows the thyroid dose for adults and infants due to inhalation. All values are lower than 0.1 mSv.

5. FAILURE IN LIQUID OR GASEOUS PROCESSING SYSTEM (PCC-3)

5.1. SYSTEM DATA AND SELECTION OF THE POSTULATED ACCIDENT

The Failure in the Liquid Waste Processing System is presented in section 6 of this sub-chapter.

The activity inventories of the gaseous waste processing system are taken from Chapter 11. For a leak in this system, two scenarios for the leak are possible:

- a leak in the sub atmospheric pressure part of the system (in the flushing section or in the recombination section),

or:

- a leak in the positive pressure part of the system (in the section between the waste gas of the system compressors and the reducing stations, in the delay unit with the gel drier and the delay beds).

Following a leak in the sub-atmospheric pressure part of the system, the pressure in this part increases and the pressure reducing station 2 tries to maintain a constant pressure. Consequently, only small volumes of the ambient air in the building atmosphere are drawn in at the delivery rate of one waste gas compressor. These are then released to the environment in the long term via the waste gas compressors and the delay unit.

This increase discharge will be detected by either an increase of the oxygen concentration in the waste gas or by an increased release of gas to the stack. The release can be stopped within 30 minutes by closing the release valve.

The main consequence is that, due to the delivery rate of the gas compressor, the pressure in the delay unit increases. Thus, any gas present in the delay unit will be released. The amount of radioactive gases in this released gas volume is extremely small, as the noble gases are delayed owing to the transport of these gases through the delay beds.

Such activity releases are part of normal plant operation. However, a leak in the positive pressure part of the gaseous waste processing system leads, as a consequence of the pressure differential between the system pressure and the building atmosphere, to a significant discharge of radioactive noble gases.

Therefore, only a leak in the positive pressure part of the system is assessed for its potential radiological consequences.

5.2. ACTIVITY RELEASE

Before the accident occurs, the noble gases inside the positive pressure part of the system are present in two very different states:

- a portion of the noble gases are in the piping and component volume of this part of system,
- another portion of the noble gases is within the activated charcoal beds, mostly adsorbed within the charcoal.

The activity inventories of the noble gases contained in these two area are given in Chapter 11.

Whereas the activity inventory contained in the activated charcoal is also given in Chapter 11, the activity inventory of the free volume in this portion of the system is evaluated on the basis of the free volume:

- of the delay tanks,
- of the gel drier,
- of the gas filter,
- of the piping,

by taking into account a system pressure of 9 bara, which is only used during plant shut down and which is a highly conservative assumption. In normal operation, the system pressure is 1.5 bar).

The calculation shows an activity inventory which is in the order of 2% of that of the purge gas circuit. However it is conservatively assumed for the radiological calculation that in the above mentioned free volume there is an activity inventory of noble gases of 3% of that of the purge gas circuit.

It is assumed that the total gaseous activity inventories of the free volume and of the charcoal beds are discharged into the atmosphere of the nuclear auxiliary building. This assumption is very conservative for the noble gases adsorbed on the activated charcoal. It is also assumed that the total Rb-88- and Cs-138-inventory of the above mentioned free volume of the positive pressure part of the system is also released into the building atmosphere.

The air changing rate of the rooms of the delay unit (positive pressure part of the system) is about 4 Vol/h. The air of these rooms is ventilated via HEPA-filters and the stack to the environment. Conservatively, the retention of aerosols by these HEPA-filters has not been considered in the activity release calculations. Sub-chapter 14.6 - Table 12 contains the activity release values. The calculations were performed by using the ACARE computer program.

5.3. POTENTIAL RADIATION EXPOSURE

It is conservatively assumed that the release occurs as a ground release.

The calculations were performed by using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 13. The maximum effective dose is 2.4 µSv for adults and 3.9 µSv for infants. The thyroid dose due to inhalation for adults and infants is less than 0.1 µSv.

6. FAILURE OF A PRESSURE BOUNDARY IN THE NUCLEAR AUXILIARY BUILDING (PCC-4)

6.1. POSTULATED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

The assumed scenario for the "Water Leak in the Nuclear Auxiliary Building" is based on a guillotine break of a Chemical and Volume Control System (RCV [CVCS]) connected line in the pipe duct between level – 6.50 m and level 0 m. From the break to the time of isolation, assumed after 60 minutes, approximately 72 t of reactor coolant are discharged out of the leak into the pipe duct. The discharged water then flows down into the rooms at the -9.60 m level.

The assumed spill height is more than 3.00 m. (A spill height up to 5.80 m is possible in this area).

Information on the activity inventory of the reactor coolant used in the accident analysis is given in Sub-chapter 14.6 - Table 3.

The break of a line of the chemical and volume control system in the nuclear auxiliary building does not cause the reactor protection system to respond. However, it is conservatively assumed in the PSA that the reactor is tripped when the signal identifying the event occurs, 30 minutes after the onset of the accident. The temperature and pressure transient, the spiking effect, on reactor trip leads to an increase in the release of radioactivity. Consequently, an elevated radioactivity concentration in the coolant must be assumed for the assessment. The increased radioactivity concentration is considered in relation to the duration of reactor coolant discharge after the shutdown of the reactor. For the radiologically significant nuclides iodine-131, caesium-134, and caesium-137 an exponential activity increase is assumed as shown in Sub-chapter 14.6 - Figure 1.

For noble gases it is assumed that the entire noble gas inventory of the discharged coolant is released into the building atmosphere. A portion of the iodine and the solids, discharged with the reactor coolant, is released into the building atmosphere. A proportion of the water flowing out may become airborne water droplets (10^{-4}) due to the spill height of more than 3 m. The release of iodine and other solids into the building atmosphere is only associated with the water droplets. It is assumed that 10% of the released iodine is transformed instantaneously into elemental iodine.

The air exchange rate for the pipe duct is assumed to be about 2 Vol./h. The air of this pipe duct is ventilated via HEPA-filters and the stack to the environment. However, it is conservatively assumed in the activity release calculation that there is no retention of aerosols by the HEPA-filters .

Sub-chapter 14.6 - Table 14 contains the activity release values. The calculations were performed using the ACARE computer program.

6.2. POTENTIAL RADIATION EXPOSURE

It is conservatively assumed that the release occurs as a ground release.

The calculations were performed using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 15. Sub-chapter 14.6 - Table16 shows the thyroid dose for adults and infants due to inhalation.

7. STEAM GENERATOR TUBE RUPTURE OF 1 TUBE (PCC-3)

This section, as well as section 8 which covers the **Steam Generator Tube Rupture of 2 tubes** accident (PCC-4), shows the reference radiological consequence calculations which have been carried out with the assumptions made during the Basic Design Phase of the EPR project. Some parameters, like power level, steam generators size or operating procedures, which are particularly significant in calculating the radiological consequences of a SGTR accident, have been changed since that time. Nevertheless, the re-evaluated steam releases remain quite close to the previous values and thus the calculations remain applicable.

For the most recent results presented in Sub-chapter 14.4, which take into account the UK EPR operating data, the French assessment method has been used, discussed in section 15 in this sub-chapter.

7.1. THERMAL-HYDRAULIC DATA

The accident examined is either due to longitudinal crack or to the complete severance of one single steam generator tube rupture (SGTR - 2A).

The main consequences of this accident arise from the contamination of the secondary side inventory, mainly the affected SG by the leakage of radioactive coolant from the primary side. It is assumed that the primary coolant is contaminated by activated corrosion and fission products at a level corresponding to continuous operation with a limited number of defective fuel rods. The possible discharge of activity to the atmosphere via the steam generator safety and/or power operated relief valves following reactor and turbine trip would lead to the release of activity to the environment.

To simplify the calculational route, this transient is divided into two main phases:

- First release (from tube rupture up to leak termination) : 102% power.
- Second release (from leak termination up to Low Head Safety Injection Train In Residual Heat Removal Mode (LHSI/RHR) connection) : 2% power.

More margin exist to fill-up the affected SG at 102% power as the SG water level after the trip is below the level for 2% power. The assumption of 2% power for the long-term phase is thus conservative for release calculations. This is consistent with the assumption used in the derivation of the thermal-hydraulic results related to the SGTR 1 tube event. Thus:

- 93.0 t of steam released to the atmosphere for first phase of the release,

- 41.2 t of steam released to the atmosphere for the second phase of the release. .

Activity release and radiation exposure calculations are performed on the basis of the T/H results of those two cases.

7.2. ACTIVITY RELEASE

The activity release to the environment is calculated for each of the two cases defined in the previous sub-section:

- first phase of the release,
- second phase of the release.

The isotopes considered are the most significant for the radiological consequences of their release into their environment. Their specific activities in the primary and secondary coolants are taken from Sub-chapter 14.6 - Tables 3 and 4.

During the first phase of the release, starting from initial full power, the spiking effect is taken into account at the time of reactor trip, as described in section 1.4.1.

In the second phase of the release, the T/H input data have been chosen to maximise the steam release to the atmosphere. Liquid discharge is prevented in the EPR by appropriate design. This assumes draining of the affected SG before depressurisation of the RCP [RCS] and the affected SG is undertaken. In particular a low initial power, corresponding to hot standby conditions, has been considered from the start of the transient. The spiking effect is also considered for this case in a conservative way to maximise the radiological consequences. This is superimposed on the T/H transient at the time of the reactor trip signal.

The transfer rate of the radioactive species considered, from the water into the steam, depends on the vaporisation rate and on the carry-over factor as discussed in section 1.4.3.

Sub-chapter 14.6 - Table 17 shows the total released activity during the first phase and the second phase.

7.3. POTENTIAL RADIATION EXPOSURE

7.3.1. Release Height

The release height is assumed to be 34 m, corresponding to the height of the upper edge of the silencer on the main steam relief valves.

Prior to leak termination the mass of steam released from the affected steam generator to the atmosphere is 93 t during a period of about 48 minutes, the first release. In the following phase up to the LHSI/RHR connection the mass of steam released from the defective steam generator is 41.2 t over a period of approximately 28 minutes, the second release.

Expansion in the atmosphere causes the steam pressure to reduced to atmospheric pressure. Consequently, the equilibrium conditions of saturated steam are approximated to 373K and 1013 hPa.

7.3.2. Results

It is assumed that 10% of the released iodine is transferred instantaneously into elemental iodine in the radiological release calculation.

The different forms of iodine are only used for the dispersion calculation. For dose calculations, the most conservative dose factor is used for the total amount of iodine. This conservative approach takes into account possible transformations into other iodine species that take place during transport in the environment.

The calculations were performed using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 18. The maximum effective dose is 38 µSv for adults and 77 µSv for infants. Sub-chapter 14.6 - Table 19 shows the thyroid dose for adults and infants due to inhalation. All values are lower than 0.1 mSv.

8. STEAM GENERATOR TUBE RUPTURE OF 2 TUBES (PCC-4)

This section, as well as section 7 which covers the **Steam Generator Tube Rupture of 1 tube** accident (PCC-3), shows the reference radiological consequence calculations which have been carried out with the assumptions made during the Basic Design Phase of the EPR project. Some parameters, like power level, steam generators size or operating procedures, which are particularly significant in calculating the radiological consequences of a SGTR accident, have been changed since that time. Nevertheless, the re-evaluated steam releases remain quite close to the previous values and thus the calculations remain applicable.

For the most recent results presented in Sub-chapter 14.5, which take into account the UK EPR data, the French assessment method has been used, see section 15 in this sub-chapter.

8.1. THERMAL-HYDRAULICAL DATA

The accident examined is either due to two longitudinal cracks or to the complete severance of two steam generator tubes (SGTR - 4A).

The main consequences of this accident arise from the contamination of the secondary side inventory, mainly the affected SG by the leakage of radioactive coolant from the primary side. It is assumed that the primary coolant is contaminated by activated corrosion and fission products at a level corresponding to continuous operation with a limited number of defective fuel rods. The possible discharge of activity to the atmosphere via the steam generator safety and/or power operated relief valves following reactor and turbine trip would lead to the release of activity to the environment.

To simplify the calculational route, this transient is divided into two main phases :

- First phase of the release (from tube rupture up to leak termination) : 102% power.
- Second phase of the release (from leak termination up to LHSI/RHR connection : 2% power.

More margin exist to fill-up the affected SG at 102% power as the SG water level after the trip is below the level of 2% power. The assumption of 2% power for the long-term phase is thus conservative for release calculations. The following steam masses are released during the accident:

- 79 t of steam released to the atmosphere for the short term.
- 53 t of steam released to the atmosphere for the long term.

Activity release and radiation exposure calculations are performed on the basis of the T/H results of these two calculated releases.

8.2. ACTIVITY RELEASE

The activity released to the environment is calculated for each of the two releases defined in the previous sub-section :

- first phase of the release,
- second phase of the release.

The isotopes taken into consideration are the most significant for their radiological consequence on release into the environment. Their specific activities in the primary and secondary coolants are taken from Sub-chapter 14.6 - Table 3 and Sub-chapter 14.6 - Table 4.

During the first release, starting from initial full power, the spiking effect is taken into account at the time of reactor trip, as described in section 1.4.1.

In the long second phase of the release, the T/H input data have been chosen to maximise the steam release to the atmosphere. Liquid discharge is prevented in the EPR by appropriate design. The affected SG is drained before depressurisation of the RCP [RCS] and the affected SG is undertaken. In particular, a low initial power corresponding to the hot standby condition has been considered for transient., A spiking effect is also considered for this case in a conservative way and superimposed on the T/H transient at the occurrence of the reactor trip signal to maximise the radiological consequences of the second phase release.

The transfer rate of the radioactive species considered, from the water into the steam, depends on the vaporisation rate and on the carry-over factor as discussed in section 1.4.3.

Sub-chapter 14.6 - Table 20 shows the total released activity for the first release, the second release and the total release.

8.3. POTENTIAL RADIATION EXPOSURE

8.3.1. Release Height

The release height is assumed to be 34 m, corresponding to the height of the upper edge of the silencer on the main steam relief valves.

Prior to leak termination the mass of the steam released from the affected steam generator to the atmosphere is 79 t during a period of approximately 16 minutes, the first release. In the following phase up to the LHSI/RHR connection the mass of steam released from the affected steam generator is 53 t during a period of approximately 50 minutes, the second release.

Expansion of the steam in the atmosphere reduces the pressure to atmospheric pressure. The equilibrium conditions for saturated steam are approximated to 373K and 1013 hPa.

8.3.2. Results

It is assumed that 10% of the released iodine is transferred instantaneously into elemental iodine in the radiological release calculation..

The different forms of iodine are used for the dispersion calculation. For dose calculations, the most conservative dose factor is used for the total amount of iodine.

The calculations were performed using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 21. The maximum effective dose is 84 µSv for adults and 170 µSv for infants. Sub-chapter 14.6 -Table 22 shows the thyroid dose for adults and infants due to inhalation.

9. FUEL HANDLING ACCIDENT (PCC-4)

9.1. POSTULATED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

It is assumed that a fuel assembly is damaged during handling in the Fuel Building.

The assumed accident sequence is as follows:

- the accident occurs 60 hours after reactor shutdown as this is the earliest time for the start of refuelling,
- amount of damage. It is assumed that all of the fuel rods located at an outer edge of the affected fuel assembly ,17 fuel rods, are fractured.

The activity inventory has been calculated using the ORIGEN-S [Ref-1] code with the following assumptions:

- The core is at the end of an equilibrium cycle.
- The damaged assembly is the first discharged at 60 hours after shutdown.
- The damaged fuel rods of the assembly have the average burnup of the core.
- It is conservatively assumed that the defective fuel rods have a relative power of 1.8 during the last cycle.

The noble gas and iodine inventory of one fuel rod is given in Sub-chapter 14.6 - Table 23.

It is assumed that 10% of the noble gases from the defective fuel rods are released into the fuel building atmosphere, and 5% of the iodine from the damaged fuel rods are released into the fuel pool water. This iodine is released into the fuel building atmosphere in the long term, due to the slow exchange processes between the gas and water phases. To quantify these processes, a volume-related distribution coefficient of 10^5 is assumed. The airborne iodine is assumed to be I_2. The water volume of the fuel pool is approximately 1425 m^3 and the free air volume approximately 5400 m^3. The volume-related distribution coefficient of 10^5 is based on experiments and on operational data. [Ref-2]

Purification of the fuel pool water is conservatively omitted from the accident sequence calculation. The air changing rate is assumed to be 2 Vol/hour, and the duration of the activity release is assumed to be 7 days. The air from the free atmosphere above the fuel pool is discharged via HEPA-filters and the stack to the environment. Half an hour after the accident occurs there is a switch over of the air flow to the activated charcoal filter device. The retention factor for elementary iodine of this charcoal device is assumed to be 99%.

The radioactive releases to the plant environment calculated in line with the above assumptions are given in Sub-chapter 14.6 - Table 24.

The calculations were performed using the ACARE computer program.

9.2. POTENTIAL RADIATION EXPOSURE

It is conservatively assumed that the release occurs as a ground release.

The different iodine forms are only used for the dispersion calculation. For the dose calculations, the conservative dose factor is used for the total amount of iodine.

The calculations were performed using the PRODOS computer program.

The resulting potential effective dose values for adults and infants are given in Sub-chapter 14.6 - Table 25. The maximum effective dose is 71 µSv for adults and 310 µSv for infants. Sub-chapter 14.6 - Table 26 shows the thyroid dose for adults and infants due to inhalation.

10. LONG-TERM LOSS OF OFFSITE POWER (PCC-3)

This case is covered by the loss of condenser vacuum case (section 12).

11. MAIN STEAM SYSTEM DEPRESSURISATION (PCC-2)

This case is covered by the loss of condenser vacuum case (section 12).

12. LOSS OF CONDENSER VACUUM (PCC-2)

This accident is representative of events with secondary side depressurisation, e.g. long term loss of emergency power. This accident is also combined with the single failure of a stuck open main steam control valve and with RCP [RCS] pumps running.

12.1. POSTULATED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

It is conservatively assumed for the evaluation of the radiological consequences of this accident that during the accident sequence all the water in the steam generator (87.6 t) is evaporated within a period of less than 18 minutes. Therefore, the release of all the activity in the steam generator water is assumed. The specific activity values for this water are given in Sub-chapter 14.6 - Table 4. The resulting activity release is given in Sub-chapter 14.6 - Table 27.

12.2. POTENTIAL RADIATION EXPOSURE

It is assumed that 10% of the released iodine is transferred spontaneously into elemental iodine. The different iodine forms are used only for the dispersion calculation. For dose calculations, the conservative dose factor is used for the total amount of iodine.

12.2.1. Release Height for Calculation of the Radiation Exposure

The release height is assumed to be 34 m, corresponding to the height of the silencer on the affected main steam relief valve.

87.6 t of steam is released to the atmosphere during a time period of about 18 minutes.

The pressure of the steam will be reduced to atmospheric pressure due to expansion in the atmosphere. The equilibrium conditions of saturated steam are approximately 373K and 1013 hPa.

For this calculation a conservative approach is adopted, assuming the release occurs at ground level.

12.2.2. Results

The calculations were performed using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 28. The maximum effective dose for adults is 34 µSv, for infants 69 µSv. Sub-chapter 14.6 - Table 29 shows the thyroid dose for adults and infants due to inhalation.

13. MULTIPLE FAILURE OF SYSTEMS IN THE NUCLEAR AUXILIARY BUILDING UNDER EARTHQUAKE BOUNDARY CONDITION (PCC-4)

13.1. POSTULATED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

To assess the potential radiological consequences of an earthquake related to the nuclear auxiliary building, all the tanks containing radioactive contaminated water are assumed to fail or breaks of the connecting lines occur.

It is assumed that the entire inventory of the systems containing contaminated water spills out into the building. Additionally it is assumed that the gaseous inventory is completely released into the room atmosphere. The activity which is bound on filters, or resins, is assumed to remain fixed.

These assumptions are very conservative. It is known from real earthquake events in other countries with higher intensities than expected in the UK, that industrial buildings and technical systems show only minor damage after earthquake events.

The assessment also assumes that the activity released into the room atmosphere is transported to the environment with a ventilation rate of 5 Vol/hour.

The resulting activity release to the environment is given in Sub-chapter 14.6 - Table 30.

13.2. POTENTIAL RADIATION EXPOSURE

13.2.1. Release Height for Calculation of the Radiation Exposure

The emission height is assumed to be at ground level for the whole release period.

13.2.2. Results

The calculations were performed by using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 31. The maximum effective dose for adults is 28 µSv, for infants 42 µSv. Sub-chapter 14.6 - Table 32 shows the thyroid dose for adults and infants due to inhalation.

14. RRA [RHRS] FAILURE OUTSIDE CONTAINMENT (PCC-4)

14.1. POSTULATED ACCIDENT SEQUENCE

The scenario of the "RRA [RHRS] failure outside containment” is assumed to be as follows:

Approximately 10 hours after reactor trip the Residual Heat Removal conditions of the Reactor Coolant System (120°C/30 bar) are reached and further cool down of the reactor coolant will be performed by the RIS/RRA [LHSI/RHR]-System. Immediately following the switch-over to the RRA [RHRS] a break of a main system line is assumed in one of two Safeguard Buildings 1 and 4.

A 2A break (DN 250) was considered in the following calculations to cover the different possible breaks sizes,.

A water mass up to 43 t is discharged as a consequence of this event. Approximately 2 t of steam is released due to flashing of the reactor coolant. It is conservatively assumed that the total steam volume and all radioactive substances released into the building atmosphere are directly released to the environment at ground level without filtration.

14.2. ACTIVITY RELEASE TO THE ENVIRONMENT

The specific activity of the reactor coolant used in the accident analysis is given in Sub-chapter 14.6 - Table 3. The specific activity of the discharged coolant takes into account.

- spiking of fission products due to the normal plant shutdown procedure (see section 1.4.1) A maximum for 2 hours after shutdown is assumed,

- purification and degasification of the coolant for the time period after the assumed spiking maximum of fission products,

- shutdown spiking of activated corrosion products. The maximum value is assumed at the time of the RRA[RHRS] break.

the entire noble gas inventory of the discharged coolant is assumed to be released into the building atmosphere of one of Safeguard Buildings 1 and 4. In addition, a fraction of the iodine and the solids discharged with the coolant is released with the steam flow into the building atmosphere. The concentration by weight of radioactive iodine and solids in the steam formed during the discharge including moisture, is assumed to be 10% of the concentration of the discharged coolant which has not evaporated. The release of iodine and other solids into the building atmosphere is contained within the water droplets entrained in the steam.

All radioactive substances released into the building atmosphere are conservatively assumed to be immediately released to the environment.

It is assumed that 10% of the released iodine is transferred instantaneously into elemental iodine. The different iodine forms are only used for the dispersion calculation. For the dose calculations, the most conservative dose factor is used for the total amount of iodine.

The release is assumed to occur from one of the above mentioned safeguard buildings at ground level.

Sub-chapter 14.6 - Table 33 contains the values of activity released. A time dependent release pattern was not modelled as the period of release is small. The calculations were performed using the ACARE computer program.

14.3. POTENTIAL RADIATION EXPOSURE

The calculations were performed using the PRODOS computer program.

The resulting potential effective dose values of adults and infants are given in Sub-chapter 14.6 - Table 34. The maximum effective dose for adults reaches 172 µSv, and for infants 243 µSv. Sub-chapter 14.6 - Table 35 shows the thyroid dose for adults and infants due to inhalation.

15. EVALUATION OF THE POTENTIAL RADIOLOGICAL CONSEQUENCES USING THE FRENCH NPPS ASSESSMENT METHOD

15.1. PRINCIPLES

The principles for assessing radiological consequences are summarised below:

- the transients for which the radiological consequences are calculated are the same as the transients used in the safety design, analysed with the same conservative assessment rules.
- the assessment of released activity is based on conservative methods and assumptions (initial primary activity, rate of cladding failures, etc).
- the assumptions for calculating the radiological consequences, the evaluation of doses, are set realistically [Ref-1]. This results in a reasonably conservative assessment of the radiological consequences of the transients studied.

The calculation of the effective dose includes all potential exposure routes. These are external exposure to radiation plumes and deposits, internal exposure by inhalation and ingestion of contaminated products. The effective dose is assessed over a 50 year period.

The calculational results are as follows:

- At 7 days: The doses in this phase correspond to the exposure of an individual located in the immediate vicinity of the site at the time of the release. The effective dose received via inhalation and external exposure to the plume and to the deposits on the ground are calculated 500m away from the site boundary. In addition, the dose taken up by the thyroid from inhalation is evaluated for both an adult and a 1 year old infant.
- At 50 years: The dose represents the effects integrated over the life of an individual. In addition to the dose received during the passage of the radioactive cloud, the doses are received from the contamination deposited on the ground. Individuals living close to the power plant are subjected to external exposure to deposits on the ground and to internal exposure from ingestion of contaminated foodstuffs. This exposure is calculated over a period of 50 years. These doses are evaluated 2 km from the point of release.

15.2. DOSE CALCULATION ASSUMPTIONS

<u>Atmospheric diffusion of fission products released into the environment:</u>

The atmospheric concentration integrated during the passage of the plume is obtained using differential equations for atmospheric diffusion. The model used is a Gaussian plume model with a standard deviation evaluated by the 2 class Doury model. [Ref-1]

The calculations are performed taking into account standard meteorological conditions which provide a broad coverage of their effects on atmospheric dispersion. The conditions of weak diffusion with a wind of 2 m/sec are assumed. The conditions cover around 90% of the conditions encountered for French NPP sites (including coastal sites such as Flamanville). Changes in meteorological conditions such as wind speed, wind direction and diffusion are taken into account to cover the effects of the release duration via a correction factor. This factor ranges from 1 to 5 and is applied to the horizontal standard deviation.

Conversion factors for dosage, respiratory flow:

A respiratory volume of 29 m^3 per day has been used for adults, and 5 m^3 per day for the 1 year old child. [Ref-2]

The conversion factors into doses are as follows (cf. [Ref-3] [Ref-4]):

- FD_{plume}: Federal Guidance
- $FD_{inhalation}$: ICRP 71 and European Directive 96/29/EURATOM
- $FD_{inhalation\ thyroid}$: ICRP 71
- $FD_{deposits:}$ Federal Guidance
- $FD_{ingestion}$: ICRP 72 and European Directive 96/29/EURATOM

Realistic life habits, exposure conditions, integration time and transfer of radionuclides into the environment, are taken in the EDF methodology, representative of French Nuclear sites.

The preliminary dose calculations from applying the EDF methodology presented here are considered to be generic in nature.

15.3. ACCIDENTS USED

This design and safety report presents an analysis of the radiological consequences of the main transients for the various situations considered (PCC-2, PCC-3, PCC-4). This analysis is undertaken to verify that the radiological release limitation objectives are met, as defined in section 0.2.

The representative operating conditions for radiological consequences assessment are selected from all the accidents studied, based on initial conditions and limiting factors such as release type, release height, as well as plant operating modes.

Various possible locations for leaks inside and throughout the buildings are considered in calculating the releases. The containment, safeguard buildings, fuel building, auxiliary nuclear building, primary steam and feedwater systems, steam generators, etc are all considered to establish the representative cases.

This produces the following list of standard events, used for the analysis of radiological consequences:

- **Category 2 Transients** :
    - Loss of condenser vacuum

- **Category 3 Incidents :**
  - Break of primary pipe work outside the containment
  - Rupture of a steam generator tube
- **Category 4 Accidents:**
  - Large break LOCA during at power operation
  - LOCA in shutdown state
  - Multiple failures in the nuclear auxiliaries building in an earthquake
  - Rupture of two steam generator tubes with a pre-existing iodine peak
  - Accident during fuel handling

15.4. PRINCIPAL ASSUMPTIONS

The main assumptions used in assessing the radiological consequences of the above events are as follows [Ref-1]:

- **Activity in the primary coolant and secondary coolant**
  - *Activity in primary coolant*

    The activity used for the primary coolant is calculated for the maximum values used in the technical specifications applied in French NPPs. For the method applied here, the following values have been used:

    - Primary activity in stable operations: 20 GBq/t of equivalent iodine-131(*),
    - Primary activity after a power transient (iodine peak): 150 GBq/t of equivalent iodine-131 (*),

    (*) (I131eq = I131 + I132 / 30 + I133 / 4 + I 134 / 50 + I135 / 10)

  - *Activity in secondary coolant*

    The maximum activity in the secondary coolant of the steam generators is calculated based on the following assumptions:

    - Primary water activity corresponding to the maximum values in the technical specifications
    - A primary-secondary leak flow rate of 20 l/hr
    - A purge rate from all the steam generators corresponding to operation at nominal power

- Entrainment factors used for the transfer of activity via the steam phase of the SGs are as follows:

  - All the noble gases present in the SG coolant are assumed to be transferred in gas phase

  - For the other radionuclides, two cases are identified :

    - For a SG with a tube rupture, an entrainment factor of 1% is assumed.

    - For a SG without tube rupture, an entrainment factor of 0.25% is assumed

- **Release of activity following cladding failure**

In some accidents (mainly LOCAs), the thermal-hydraulic transient sustained by the fuel cladding can cause cladding failures. The fission products contained in the pellet to cladding gap can be released into the primary system.

The release rate refers to the fraction of the fuel rod fission product inventory assumed to be released into the system following the loss of cladding integrity.

The release rate envelope used for the failed fuel assemblies is provided in Sub-chapter 14.6 - Table 37.

This envelope is also used in the case of the fuel handling accident.

- **Deposition of fission products**

The equations for aerosol and molecular iodine deposition in the containment assume an exponential decay. The deposition constants for this are 0.035 h^{-1} and 0.014 h^{-1}, respectively.

- **Leak rate from the containment**

The overall leak rate across the internal EPR containment (with a steel liner) is 0.3% of volume per day at the design pressure of 5.5 bar.

- **Filter efficiency**

The retention efficiency of the extraction filters used to reduce radiological release are as follows : [Ref-2]

  - **High efficiency filters:**

    Noble gases 0%
    Aerosols on which iodine precipitates 99.9%
    all the other substances 0%

- **High efficiency filters + iodine trap:**

  Noble gases 0%
  Iodine in organic form 99%
  Molecular iodine (I_2) 99.9%
  Aerosols on which iodine precipitates 99.9%

15.5. SPECIFIC ASSUMPTIONS RELATING TO THE ACCIDENTS CONSIDERED

In addition to the general assumptions described above, specific assumptions for the accidents are presented below:

- **Loss of vacuum in condenser (PCC-2)**

This accident is representative of the events involving a depressurisation of the secondary system, with steam discharge into the atmosphere. To assess the radiological consequences of this accident, it is conservatively assumed that the steam generator water is completely evaporated and that all the activity in the steam generator water is released into the environment.

- **Break of primary pipe work outside the containment (PCC-3)**

The assumed scenario is the break of a line of the nuclear sampling system connected to the pressuriser. This results in the release of primary coolant outside the containment up to the moment that this line is isolated by operator intervention. The pressuriser fluid is at 155 bar, 345°C. The release continues until isolation, which is assumed to be performed 60 minutes after the break occurs.

- **Rupture of a steam generator tube (SGTR 1 tube, PCC-3)**

The radiological consequences of this accident arise from the release of activity to the environment via the main steam atmospheric dump valves on the affected steam generator. The activity arises from the contamination of the secondary system by the primary system through the flow through the broken SG tube.

New thermal-hydraulic SGTR calculations had been carried out for SGTR, one tube and two tubes, considering the latest design of the UK EPR. These model the UK EPR thermal power, steam generator size and operating procedures. The radiological consequences assessment for these thermal-hydraulic SGTR calculations also uses updated source term data [Ref-1] [Ref-2].

The revised SGTR analysis in Sub-chapter 14.4, which takes into account ALARP considerations for PCC-3 events, does not impact the radiological consequences assessment.

The activity peak in the primary system, the iodine spike, is caused by the release of activity from the fuel pellet/cladding gap into the primary fluid following an automatic reactor shutdown. This is taken into account in assessing the radiological consequences of this accident.

- **SGTR 2 tubes (PCC-4)**

The calculation of the radiological consequences of this accident is very similar to that of the SGTR in category 3. In addition to modelling the rupture of two tubes instead of one, it is conservatively assumed that the iodine peak is at its peak at the time of the shutdown.

The revised SGTR analysis in Sub-chapter 14.5, which takes into account the new RCV [CVCS] assumptions for PCC-4 events, does not impact the radiological consequences assessment.

- **Large break LOCA in normal operation (PCC-4)**

The category 4 LOCA is defined as a break of a safety injection line in the pipe work to the cold leg of the primary system.

In this accident it is assumed that the reactor core is emptied, with the failure of 10% of the fuel rods. The release considered in this accident comes partly from containment leakage, and partly from the leakage assumed to occur from the reactor cooling circuits outside the containment, in the ventilated and filtered auxiliary safeguard buildings.

- **LOCA in shutdown state (PCC-4)**

This accident is defined as a failure of the cooling system at shutdown (RRA[RHRS]) outside the containment. It occurs after the operating conditions of this system are reached, approximately 10 hours after reactor shutdown. The break diameter is assumed to be equal to the largest diameter of the pipework considered (250 mm) to cover all possible break sizes,.

- **Multiple failures in the nuclear auxiliaries building during an earthquake (PCC-4)**

To assess the potential radiological consequences of an earthquake affecting the nuclear auxiliary building, it is assumed that all the tanks containing radioactive contaminated fluid are fractured or that a break occurs in their connection line.

The entire inventory of the systems containing contaminated water is assumed to spill into the building. In addition, it is assumed that the gas inventory of the various systems is completely released into the free volume of the building. Only activity already retained by the filters or resins is assumed to remain within the systems.

The activity released to the building is assumed to be released to the environment without being filtered. This is assumed to occur at a leakage rate of 5 Vol/day. This higher rate of air renewal than the normal static containment rate for the building takes account of any damage incurred by the building.

- **Accident during fuel handling (PCC-4)**

The accident studied is the dropping of a fuel assembly of maximum burnup in the spent fuel pool. All the fuel rods of this assembly are assumed to be fractured.

The assembly drop accident is assumed to occur 60 hours after reactor shutdown. This corresponds to the minimum time between reactor shutdown and the start of fuel handling.

It is assumed that the radioactive isotopes released from the spent fuel pool are dispersed immediately and uniformly throughout the free volume of the fuel building.

The signal to automatically close the isolation devices of the main extraction circuit is generated by a high-activity sensor located on the service floor of the fuel storage pond. This isolation confines the activity by switching on the ventilation with iodine filters at a reduced flow rate (DWK [FBVS]).

15.6. RESULTS

The radiological consequences of the representative accidents, calculated using the methods and assumptions described in section 15.1 [Ref-1], are summarised in Sub-chapter 14.6 - Table 38.

As mentioned in section 15.5, new thermal-hydraulic SGTR calculations have been carried out for the SGTR of one tube and two tubes for the latest design of the UK EPR. The radiological consequences of both accidents have been evaluated using the associated steam releases. The doses from the detailed evaluations are shown in Sub-chapter 14.6 - Table 38 [Ref-2].

The results show that the radiological release limits, as defined in section 0.2, are met.

16. SENSITIVITY STUDIES REQUIRED BY THE TECHNICAL GUIDELINES

The Technical Guidelines applicable to the EPR reactor [Ref-1] require two sensitivity studies for the loss of primary coolant accident (LOCA) PCC-4 to be carried out. These are identified in paragraphs C.2.1: "Single failure criterion and preventive maintenance" and D.2.4 "radiological consequences",.

The first study considers the possibility of a leakage from the reactor building to a peripheral building. The sensitivity assumes the leak tightness of this building, but takes no credit for any filtration of the discharge from that building.

The results show a significant increase of the released activity. Considering that the entirety of the leak from the containment to the peripheral buildings is not filtered (60% of the total containment leak rate), the effective short term dose and the equivalent thyroid dose will be multiplied by factors of 20 and 200 respectively. However, the radiological objectives given in section 0.2 of this sub-chapter are still met, even with these increased factors.

The second sensitivity study concerns the possibility of a passive failure in the operation of safety systems in the long term. This is assumed to be after 24 hours, during recirculation mode., A larger leak rate than the design value of 200 l/min has been considered as required in the Technical Guidelines (Section C.2.1). A leak rate of up to 5100 l/min, corresponding to the rupture of an connected pipe with an inner diameter of 50 mm, has been modelled. In addition, a passive single failure has been assumed in the short term, before 24 hours. The impact on short term doses the effective dose and the equivalent thyroid dose, is limited to a factor 4 increase. The radiological objectives given in section 0.2 of this sub-chapter for PCC-4 events are met and no cliff edge effects are identified.

SUB-CHAPTER 14.6 - TABLE 1

Data for the Calculation of the Activity Inventories of the EPR Reactor Core with Uranium Fuel Elements

Subject	Assumption. Data
Thermal Output	4900 MW
Type of Core	Equilibrium core
Average U-235 enrichment	5 %
Cycle length	321.35 Full-load days
Uranium content per fuel assembly (weight)	515 kg
Average burnup of equilibrium Core	43 MWd/kg
Average burnup of discharge	63.57 MWd/kg

Fuel cycle Characterisation	Specific power MWd/kg	Number of fuel assemblies per batch
First Cycle	49.94	48
Second Cycle	50.91	48
Third Cycle	43.63	48
Fourth Cycle	30.44	48
Fifth residence time	22.89	49

SUB-CHAPTER 14.6 - TABLE 2

Activity Inventory of the Reactor Core at the End of the Fuel Cycle

Nuclide	Activity Inventory of the Reactor Core (Bq)	Nuclide	Activity Inventory of the Reactor Core (Bq)	Nuclide	Activity Inventory of the Reactor Core (Bq)
Kr-83m	6.0E+17	Ru-106	2.6E+18	Ba-137m	6.1E+17
Kr-85	5.7E+16	Rh-103m	7.4E+18	Ba-139	8.9E+18
Kr-85m	1.3E+18	Rh-105	4.6E+18	Ba-140	8.9E+18
Kr-87	2.5E+18	Rh-105m	1.4E+18	Ba-141	8.0E+18
Kr-88	3.5E+18	Rh-106	2.8E+18	Ba-142	2.1E+18
Xe-133	9.7E+18	Rh-107	2.8E+18	La-140	9.4E+18
Xe-133m	3.1E+17	Pd-109	1.7E+18	La-141	8.1E+18
Xe-135	3.0E+18	Pd-112	2.1E+16	La-142	7.9E+18
Xe-135m	2.1E+18	Ag-109m	1.7E+18	La-143	7.6E+18
Xe-138	8.6E+18	Ag-111	2.3E+17	Ce-141	8.1E+18
Br-83	6.0E+17	Ag-112	2.1E+16	Ce-143	7.6E+18
Br-84	1.1E+18	Sb-127	4.0E+17	Ce-144	6.1E+18
Rb-88	3.6E+18	Sb-128m	7.0E+17	Ce-146	4.1E+18
Rb-89	4.7E+18	Sb-129	1.5E+18	Pr-142	2.7E+17
Sr-89	4.9E+18	Sb-130	5.0E+17	Pr-143	7.4E+18
Sr-90	4.7E+17	Sb-131	3.8E+18	Pr-144	6.2E+18
Sr-91	6.1E+18	Te-127	3.9E+17	Pr-145	5.2E+18
Sr-92	6.4E+18	Te-129	1.4E+18	Pr-146	4.1E+18
Y-90	4.9E+17	Te-129m	2.9E+17	Pr-147	2.4E+18
Y-91	6.3E+18	Te-131	4.1E+18	Nd-147	3.3E+18
Y-91m	3.5E+18	Te-131m	9.2E+17	Nd-149	1.9E+18
Y-92	6.5E+18	Te-132	6.9E+18	Pm-147	8.6E+17
Y-93	4.9E+18	Te-133	5.4E+18	Pm-148	8.2E+17
Y-94	7.8E+18	Te-133m	4.5E+18	Pm-149	2.9E+18
Y 95	4.4E+18	Te-134	8.9E+18	Pm-151	9.5E+17
Zr-95	8.3E+18	Sn-128	6.7E+17	Sm-153	2.2E+18
Zr-97	7.9E+18	I-130	2.2E+16	Eu-156	1.1E+18
Nb-95	8.3E+18	I-131	4.8E+18	U-237	4.8E+18
Nb-97	7.9E+18	I-132	7.0E+18	U-239	9.2E+19
Nb-97m	7.5E+18	I-133	1.0E+19	Np-238	2.1E+18
Mo-99	9.1E+18	I-133m	7.4E+17	Np-239	9.2E+19
Mo-101	8.2E+18	I-134	1.1E+19	Np-240	2.2E+17
Mo-102	7.7E+18	I-135	9.5E+18	Pu-238	9.0E+15
Tc-99m	8.1E+18	Cs-134	9.3E+17	Pu-241	7.6E+17
Tc-101	8.2E+18	Cs-134m	1.4E+17	Pu-243	2.1E+18
Tc-102	7.7E+18	Cs-135m	1.0E+17	Am-242	4.1E+17
Tc-104	6.0E+18	Cs-136	2.3E+17	Am-244	8.1E+17
Ru-103	7.4E+18	Cs-137	6.4E+17	Cm-242	2.4E+17
Ru-105	5.0E+18	Cs-138	9.3E+18	Cm-244	1.7E+16

SUB-CHAPTER 14.6 - TABLE 3

Nuclide Specific Concentrations in the Primary Coolant (Bq/Mg) – Shutdown Spiking Factors

NUCLIDE	SPECIFIC ACTIVITY (Bq/Mg)		SHUTDOWN
	DESIGN	TYPICAL	SPIKING RATIO
Mn-54	4.0E+06	2.0E+06	300
Co-58	1.6E+07	8.0E+06	1000
Fe-59	1.0E+06	5.0E+05	300
Co-60	1.0E+06	5.0E+05	500
Ar-41	1.0E+09	3.0E+08	1.0
Kr-85m	5.5E+09	2.0E+08	2.3
Kr-85	5.2E+08	1.9E+07	1.0
Kr-87	1.0E+10	3.6E+08	2.3
Kr-88	1.4E+10	5.0E+08	2.3
Xe-133m	1.7E+09	1.1E+08	2.3
Xe-133	8.0E+10	5.0E+09	1.9
Xe-135	1.8E+10	1.1E+09	1.4
Xe-138	1.4E+10	8.5E+08	2.9
Sr-89	4.9E+06	3.0E+05	1.0
Sr-90	3.0E+04	1.9E+03	1.0
I-131	1.6E+09	1.0E+08	30
I-132	2.8E+09	1.8E+08	12
I-133	4.9E+09	3.1E+08	7.6
I-134	1.7E+09	1.1E+08	14
I-135	3.3E+09	2.0E+08	7.1
Cs-134	3.2E+08	4.0E+07	30
Cs-137	3.2E+08	4.0E+07	30
Cs-138	1.4E+10	8.5E+08	2.9
N-16	5.7E+12	4.3E+12	0.0
H-3	3.7E+10	3.7E+10	1.0

SUB-CHAPTER 14.6 - TABLE 4

Specific Activity of the Secondary Coolant Circuit, when only the Secondary Coolant Activity is Affected by an Accident

Nuclide	Specific Activity of the Main Steam (Bq/Mg)	Specific Activity of Steam Generator Water (Bq/Mg)
I-131	3.7E+04	1.5E+07
I-133	1.0E+05	4.1E+07
I-135	5.8E+05	2.3E+07
Sr-89	1.1E+02	4.6E+04
Sr-90	7.0E-01	2.8E+02
Cs-134	7.4E+03	3.0E+06
Cs-137	7.4E+03	3.0E+06

SUB-CHAPTER 14.6 - TABLE 5

Large Break LOCA -Cumulative activity release to the environment of the plant, Bq

Time(h):	1	4	8	16	24	48	120	168
Kr83m	2.1E+09	1.7E+10	3.0E+10	3.7E+10	3.7E+10	3.7E+10	3.7E+10	3.7E+10
Kr85m	5.2E+09	6.1E+10	1.7E+11	3.3E+11	4.0E+11	4.4E+11	4.4E+11	4.4E+11
Kr87	8.1E+09	4.9E+10	7.2E+10	7.6E+10	7.7E+10	7.7E+10	7.7E+10	7.7E+10
Kr88	1.4E+10	1.4E+11	3.1E+11	4.7E+11	5.1E+11	5.1E+11	5.1E+11	5.1E+11
Rb88	1.2E+07	1.5E+08	3.4E+08	5.2E+08	5.6E+08	5.7E+08	5.7E+08	5.7E+08
Rb89	8.5E+05	1.1E+06	1.1E+06	1.1E+06	1.1E+06	1.1E+06	1.1E+06	1.1E+06
Sr89	5.4E+05	8.6E+06	3.4E+07	1.3E+08	2.8E+08	7.6E+08	1.6E+09	1.9E+09
Sr90	5.2E+04	8.2E+05	3.2E+06	1.3E+07	2.8E+07	7.4E+07	1.6E+08	1.9E+08
Y90	5.4E+04	8.5E+05	3.4E+06	1.3E+07	2.8E+07	7.6E+07	1.6E+08	1.9E+08
Sr91	6.4E+05	8.8E+06	2.9E+07	7.8E+07	1.2E+08	1.7E+08	1.8E+08	1.8E+08
Y91	7.0E+05	1.1E+07	4.3E+07	1.7E+08	3.7E+08	9.8E+08	2.1E+09	2.4E+09
Sr92	6.0E+05	5.9E+06	1.3E+07	1.9E+07	2.1E+07	2.1E+07	2.1E+07	2.1E+07
Y92	7.1E+05	9.9E+06	3.0E+07	6.2E+07	7.6E+07	8.1E+07	8.1E+07	8.1E+07
Y93	5.2E+05	7.2E+06	2.4E+07	6.6E+07	1.1E+08	1.5E+08	1.6E+08	1.6E+08
Y95	5.4E+04	5.9E+04	5.9E+04	5.9E+04	5.9E+04	5.9E+04	5.9E+04	5.9E+04
Zr95	9.2E+05	1.5E+07	5.7E+07	2.2E+08	4.8E+08	1.3E+09	2.7E+09	3.2E+09
Nb95	9.2E+05	1.5E+07	5.7E+07	2.2E+08	4.8E+08	1.3E+09	2.7E+09	3.1E+09
Zr97	8.5E+05	1.2E+07	4.4E+07	1.4E+08	2.5E+08	4.3E+08	5.1E+08	5.1E+08
Mo99	1.0E+06	1.6E+07	5.9E+07	2.2E+08	4.5E+08	1.1E+09	1.8E+09	1.9E+09
Tc99m	8.9E+05	1.4E+07	5.5E+07	2.1E+08	4.3E+08	1.0E+09	1.7E+09	1.8E+09
Ru103	8.1E+05	1.3E+07	5.1E+07	2.0E+08	4.3E+08	1.1E+09	2.4E+09	2.8E+09
Rh103m	8.2E+05	1.3E+07	5.1E+07	2.0E+08	4.3E+08	1.1E+09	2.4E+09	2.8E+09
Ru105	5.0E+05	5.8E+06	1.6E+07	3.1E+07	3.8E+07	4.1E+07	4.1E+07	4.1E+07
Rh105	5.1E+05	8.0E+06	3.1E+07	1.1E+08	2.3E+08	4.9E+08	7.0E+08	7.3E+08
Ru106	2.9E+05	4.5E+06	1.8E+07	6.9E+07	1.5E+08	4.0E+08	8.6E+08	1.0E+09
Sb127	4.4E+04	6.8E+05	2.6E+06	9.8E+06	2.1E+07	5.1E+07	9.0E+07	9.8E+07
Te127	4.3E+04	6.6E+05	2.5E+06	9.3E+06	1.9E+07	4.7E+07	8.3E+07	9.1E+07
Sb129	1.5E+05	1.8E+06	4.8E+06	9.3E+06	1.1E+07	1.2E+07	1.2E+07	1.2E+07
Te129m	3.2E+04	5.1E+05	2.0E+06	7.7E+06	1.7E+07	4.5E+07	9.3E+07	1.1E+08
Te129	1.6E+05	2.1E+06	6.1E+06	1.5E+07	2.3E+07	4.2E+07	7.3E+07	8.3E+07
Sb131	1.4E+05	2.6E+05	2.6E+05	2.6E+05	2.6E+05	2.6E+05	2.6E+05	2.6E+05
Te131m	1.0E+05	1.5E+06	5.7E+06	2.0E+07	3.8E+07	7.9E+07	1.1E+08	1.1E+08
Te131	3.0E+05	1.1E+06	2.1E+06	5.2E+06	9.3E+06	1.8E+07	2.5E+07	2.5E+07
I131ae	4.3E+06	6.8E+07	2.6E+08	1.0E+09	2.2E+09	5.5E+09	1.1E+10	1.2E+10
I131el	4.3E+05	6.1E+06	2.1E+07	6.5E+07	1.2E+08	2.5E+08	4.5E+08	5.1E+08
Te132	7.6E+05	1.2E+07	4.5E+07	1.7E+08	3.5E+08	8.4E+08	1.5E+09	1.6E+09
I132ae	5.2E+06	4.7E+07	9.5E+07	1.3E+08	1.4E+08	1.4E+08	1.4E+08	1.4E+08
I132el	5.1E+05	4.3E+06	8.1E+06	1.0E+07	1.1E+07	1.1E+07	1.1E+07	1.1E+07

SUB-CHAPTER 14.6 - TABLE 5 (CONT)

Large Break LOCA - Cumulative activity release to the environment of the plant, Bq

Time(h):	1	4	8	16	24	48	120	168
Te133m	3.1E+05	1.4E+06	1.7E+06	1.7E+06	1.7E+06	1.7E+06	1.7E+06	1.7E+06
Te133	1.3E+05	3.3E+05	3.8E+05	3.9E+05	3.9E+05	3.9E+05	3.9E+05	3.9E+05
I133ae	8.8E+06	1.3E+08	4.7E+08	1.6E+09	2.9E+09	5.4E+09	6.7E+09	6.7E+09
I133el	8.7E+05	1.2E+07	3.8E+07	1.0E+08	1.6E+08	2.6E+08	3.1E+08	3.1E+08
Xe133m	1.4E+09	2.2E+10	8.5E+10	3.1E+11	6.4E+11	1.5E+12	2.4E+12	2.5E+12
Xe133	4.4E+10	7.0E+11	2.7E+12	1.0E+13	2.2E+13	5.5E+13	1.0E+14	1.2E+14
Xe133e	3.2E+06	1.8E+08	1.2E+09	7.3E+09	1.9E+10	6.1E+10	1.4E+11	1.6E+11
Te134	5.2E+05	1.8E+06	2.0E+06	2.0E+06	2.0E+06	2.0E+06	2.0E+06	2.0E+06
I134ae	6.3E+06	2.9E+07	3.5E+07	3.6E+07	3.6E+07	3.6E+07	3.6E+07	3.6E+07
I134el	5.9E+05	2.4E+06	2.8E+06	2.8E+06	2.8E+06	2.8E+06	2.8E+06	2.8E+06
Cs134	8.4E+05	1.3E+07	5.2E+07	2.0E+08	4.4E+08	1.2E+09	2.5E+09	3.0E+09
I135ae	8.0E+06	1.0E+08	3.1E+08	7.5E+08	1.1E+09	1.3E+09	1.3E+09	1.3E+09
I135el	7.9E+05	9.4E+06	2.6E+07	5.2E+07	6.6E+07	7.5E+07	7.6E+07	7.6E+07
Xe135m	3.0E+09	2.0E+10	5.7E+10	1.3E+11	1.8E+11	2.2E+11	2.2E+11	2.2E+11
Xe135	1.4E+10	2.0E+11	7.2E+11	2.2E+12	3.7E+12	5.6E+12	6.0E+12	6.0E+12
Cs137	5.8E+05	9.1E+06	3.6E+07	1.4E+08	3.1E+08	8.2E+08	1.7E+09	2.1E+09
Xe138	7.2E+09	9.1E+09	9.1E+09	9.1E+09	9.1E+09	9.1E+09	9.1E+09	9.1E+09
Cs138	1.2E+07	3.8E+07	4.0E+07	4.0E+07	4.0E+07	4.0E+07	4.0E+07	4.0E+07
Ba140	9.8E+05	1.6E+07	6.1E+07	2.3E+08	5.0E+08	1.3E+09	2.6E+09	3.0E+09
La140	1.0E+06	1.6E+07	6.4E+07	2.5E+08	5.4E+08	1.4E+09	2.9E+09	3.3E+09
Ce141	9.0E+05	1.4E+07	5.6E+07	2.2E+08	4.7E+08	1.3E+09	2.6E+09	3.0E+09
Ce143	8.3E+05	1.3E+07	4.7E+07	1.6E+08	3.2E+08	6.8E+08	9.6E+08	9.8E+08
Pr143	8.2E+05	1.3E+07	5.1E+07	2.0E+08	4.3E+08	1.1E+09	2.3E+09	2.7E+09
Ce144	6.8E+05	1.1E+07	4.2E+07	1.6E+08	3.6E+08	9.6E+08	2.0E+09	2.4E+09
Nd147	3.6E+05	5.7E+06	2.2E+07	8.5E+07	1.8E+08	4.8E+08	9.5E+08	1.1E+09
Pu238	9.9E+02	1.6E+04	6.2E+04	2.4E+05	5.2E+05	1.4E+06	3.0E+06	3.5E+06
Np239	1.0E+07	1.6E+08	5.9E+08	2.2E+09	4.4E+09	1.0E+10	1.7E+10	1.8E+10
Pu241	8.5E+04	1.3E+06	5.3E+06	2.0E+07	4.5E+07	1.2E+08	2.6E+08	3.0E+08
Cm242	2.6E+04	4.1E+05	1.6E+06	6.3E+06	1.4E+07	3.7E+07	7.8E+07	9.1E+07
Cm244	1.8E+03	2.9E+04	1.1E+05	4.4E+05	9.7E+05	2.6E+06	5.5E+06	6.5E+06
Total	**9.9E+10**	**1.2E+12**	**4.2E+12**	**1.4E+13**	**2.8E+13**	**6.4E+13**	**1.1E+14**	**1.3E+14**

SUB-CHAPTER 14.6 - TABLE 6

Large Break LOCA Release fractions of the core inventory in the containment atmosphere

Nuclide	Release fraction
Kr(Kr85)	0.002(0.005)
Xe	0.002
Rb	3.9E-04
Sr	4.8E-05
Y	4.8E-05
Zr	4.8E-05
Nb	4.8E-05
Mo	4.8E-05
Tc	4.8E-05
Ru	4.8E-05
Rh	4.8E-05
Sb	4.8E-05
Te	4.8E-05
I el.	4.0E-05
I aer.	3.9E-04
Cs	3.9E-04
Ba	4.8E-05
La	4.8E-05
Ce	4.8E-05
Pr	4.8E-05
Nd	4.8E-05
Np	4.8E-05
Pu	4.8E-05
Cm	4.8E-05

SUB-CHAPTER 14.6 - TABLE 7

Large Break LOCA, Effective Doses, all Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	54	80
1	19	28
2	18	32
5	4.2	7.5
10	1.5	2.7

SUB-CHAPTER 14.6 - TABLE 8

Large Break LOCA, Thyroid Doses, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	32	63
1	11	21
2	3.5	7
5	0.8	1.6
10	0.26	0.52

SUB-CHAPTER 14.6 - TABLE 9

Rupture of a line carrying primary coolant outside the containment - Cumulative activity release to the environment of the plant, Bq

Time(h):	1.0E+00	2.0E+00	3.0E+00	6.0E+00
Ar41	1.4E+09	1.4E+09	1.4E+09	1.4E+09
Kr83m	1.1E+08	1.2E+08	1.2E+08	1.2E+08
Kr85m	7.8E+09	7.9E+09	7.9E+09	7.9E+09
Kr85	7.7E+08	7.8E+08	7.8E+08	7.8E+08
Kr87	1.3E+10	1.3E+10	1.3E+10	1.3E+10
Kr88	2.0E+10	2.0E+10	2.0E+10	2.0E+10
Mn54	3.9E+05	4.0E+05	4.0E+05	4.0E+05
Co58	1.6E+06	1.6E+06	1.6E+06	1.6E+06
Fe59	9.8E+04	1.0E+05	1.0E+05	1.0E+05
Co60	9.8E+04	1.0E+05	1.0E+05	1.0E+05
Rb88	7.0E+09	7.3E+09	7.3E+09	7.3E+09
Sr89	4.8E+05	4.9E+05	4.9E+05	4.9E+05
Sr90	3.0E+03	3.0E+03	3.0E+03	3.0E+03
I131ae	3.2E+08	3.2E+08	3.2E+08	3.2E+08
I131el	3.5E+07	3.6E+07	3.6E+07	3.6E+07
I132ae	2.3E+08	2.3E+08	2.3E+08	2.3E+08
I132el	2.6E+07	2.6E+07	2.6E+07	2.6E+07
I133ae	4.3E+08	4.4E+08	4.4E+08	4.4E+08
I133el	4.8E+07	4.9E+07	4.9E+07	4.9E+07
Xe133m	2.5E+09	2.5E+09	2.5E+09	2.5E+09
Xe133	1.2E+11	1.2E+11	1.2E+11	1.2E+11
I134ae	1.3E+08	1.3E+08	1.3E+08	1.3E+08
I134el	1.4E+07	1.4E+07	1.4E+07	1.4E+07
Cs134	7.1E+07	7.2E+07	7.2E+07	7.2E+07
I135ae	2.9E+08	2.9E+08	2.9E+08	2.9E+08
I135el	3.2E+07	3.2E+07	3.2E+07	3.2E+07
Xe135	2.6E+10	2.7E+10	2.7E+10	2.7E+10
Cs137	7.1E+07	7.2E+07	7.2E+07	7.2E+07
Xe138	1.2E+10	1.2E+10	1.2E+10	1.2E+10
Cs138	3.9E+09	4.0E+09	4.0E+09	4.0E+09
Total:	**2.1E+11**	**2.1E+11**	**2.1E+11**	**2.1E+11**

SUB-CHAPTER 14.6 - TABLE 10

Rupture of a line carrying primary coolant, Effective Doses, all Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	9.3	18
1	3.6	6.7
2	4.3	8.3
5	1.2	2.3
10	0.5	0.88

SUB-CHAPTER 14.6 - TABLE 11

Rupture of a line carrying primary coolant outside the Containment, Thyroid Doses, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	7.8	16
1	2.8	0.58
2	1	0.21
5	0.27	0.56
10	0.1	0.2

SUB-CHAPTER 14.6 - TABLE 12

Failure in the Gaseous Waste Processing System, Cumulative activity release to the environment of the plant

Nuclide	Cumulative Activity Release [Bq]		
	1 hour	2 hours	3 hours
Kr85m	4.4E+10	4.5E+10	4.5E+10
Kr85	1.6E+11	1.7E+11	1.7E+11
Kr87	6.8E+09	6.9E+09	6.9E+09
Kr88	3.8E+10	3.9E+10	3.9E+10
Rb88	3.6E+10	3.7E+10	3.7E+10
Xe133m	3.5E+11	3.6E+11	3.6E+11
Xe133	2.0E+13	2.1E+13	2.1E+13
Xe135	6.7E+11	6.8E+11	6.8E+11
Xe138	4.8E+06	4.8E+06	4.8E+06
Cs138	7.5E+06	7.5E+06	7.5E+06
sum	2.2E+13	2.2E+13	2.2E+13

SUB-CHAPTER 14.6 - TABLE 13

failure in the Gaseous Waste Processing System, Effective Doses for Adults and Infants, All Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	2.4	3.9
1	1.3	2.1
2	0.66	1.1
5	0.26	0.43
10	0.14	0.23

SUB-CHAPTER 14.6 - TABLE 14

Failure of an equipment containing radioactivity in Nuclear Auxiliary Building, Cumulative activity release to the environment of the plant, Bq

Time(h):	**1**	**2**	**3**	**6**
Ar41	5.5E+10	6.0E+10	6.1E+10	6.1E+10
Kr83m	6.6E+09	9.2E+09	9.6E+09	9.6E+09
Kr85m	3.3E+11	3.6E+11	3.7E+11	3.7E+11
Kr85	3.2E+10	3.7E+10	3.7E+10	3.7E+10
Kr87	5.2E+11	5.6E+11	5.7E+11	5.7E+11
Kr88	8.0E+11	8.9E+11	9.0E+11	9.0E+11
Mn54	2.5E+04	2.8E+04	2.9E+04	2.9E+04
Co58	9.9E+04	1.1E+05	1.2E+05	1.2E+05
Fe59	6.2E+03	7.1E+03	7.2E+03	7.2E+03
Co60	6.2E+03	7.1E+03	7.2E+03	7.2E+03
Rb88	3.9E+11	4.7E+11	4.8E+11	4.9E+11
Sr89	3.1E+04	3.5E+04	3.5E+04	3.5E+04
Sr90	1.9E+02	2.1E+02	2.2E+02	2.2E+02
I131ae	2.0E+07	2.3E+07	2.3E+07	2.3E+07
I131el	2.2E+06	2.5E+06	2.6E+06	2.6E+06
I132ae	1.4E+07	1.6E+07	1.6E+07	1.6E+07
I132el	1.6E+06	1.7E+06	1.8E+06	1.8E+06
I133ae	2.7E+07	3.1E+07	3.1E+07	3.1E+07
I133el	3.0E+06	3.4E+06	3.5E+06	3.5E+06
Xe133m	1.1E+11	1.2E+11	1.2E+11	1.2E+11
Xe133	5.0E+12	5.6E+12	5.7E+12	5.7E+12
I134ae	7.4E+06	7.8E+06	7.9E+06	7.9E+06
I134el	8.2E+05	8.7E+05	8.7E+05	8.7E+05
Cs134	4.5E+06	5.1E+06	5.2E+06	5.2E+06
I135ae	1.8E+07	2.0E+07	2.0E+07	2.0E+07
I135el	2.0E+06	2.2E+06	2.3E+06	2.3E+06
Xe135	1.1E+12	1.2E+12	1.3E+12	1.3E+12
Cs137	4.5E+06	5.1E+06	5.2E+06	5.2E+06
Xe138	4.1E+11	4.1E+11	4.1E+11	4.1E+11
Cs138	1.4E+11	1.6E+11	1.6E+11	1.6E+11
Total:	**8.8E+12**	**9.9E+12**	**1.0E+13**	**1.0E+13**

SUB-CHAPTER 14.6 - TABLE 15

Failure of an equipment containing radioactivity in Nuclear Auxiliary Building, Effective Doses, all Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	6.2	9.6
1	3.2	4.8
2	1.8	2.7
5	0.65	1
10	0.32	0.47

SUB-CHAPTER 14.6 - TABLE 16

Failure of an equipment containing radioactivity in Nuclear Auxiliary Building, Thyroid Doses, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	0.56	1.1
1	0.2	0.41
2	0.07	0.15
5	0.02	0.04
10	0.01	0.01

SUB-CHAPTER 14.6 - TABLE 17

Steam Generator Rupture of 1 Tube, Cumulative Activity Release to the Environment of the Plant

Nuclide	First Release [Bq]	Second Release [Bq]	Total Release [Bq]
Kr85m	1.2E+09	3.1E+07	1.2E+09
Kr85	1.3E+08	1.1E+07	1.5E+08
Kr87	1.5E+09	2.1E+06	1.5E+09
Kr88	2.8E+09	3.6E+07	2.8E+09
Xe135	4.0E+07	6.5E+06	4.6E+07
Xe133m	4.3E+08	3.2E+07	4.7E+08
Xe133	2.1E+10	1.6E+09	2.2E+10
I131ae	8.3E+08	4.6E+08	1.3E+09
I131el	9.3E+07	5.2E+07	1.4E+08
I132ae	1.1E+09	7.9E+07	1.2E+09
I132el	1.3E+08	8.8E+06	1.4E+08
I133ae	2.5E+09	1.1E+09	3.6E+09
I133el	2.8E+08	1.3E+08	4.0E+08
I134ae	4.6E+08	1.1E+06	4.7E+08
I134el	5.2E+07	1.3E+05	5.2E+07
I135ae	1.6E+09	4.4E+08	2.0E+09
I135el	1.8E+08	4.9E+07	2.3E+08
Sr89	1.3E+06	1.0E+05	1.4E+06
Sr90	7.8E+03	6.2E+02	8.4E+03
Cs134	1.9E+08	1.1E+08	2.9E+08
Cs137	1.9E+08	1.1E+08	2.9E+08
Total:	**3.4E+10**	**4.3E+09**	**3.9E+10**

SUB-CHAPTER 14.6 - TABLE 18

Steam Generator Rupture of 1 Tube, Effective Doses for Adults and Infants, All Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	38	77
1	15	29
2	17	35
5	5.1	9.6
10	2	3.7

SUB-CHAPTER 14.6 - TABLE 19

Steam Generator Rupture of 1 Tube, Thyroid Doses for Adults and Infants, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	39	83
1	14	30
2	5.1	11
5	1.4	2.9
10	0.5	1.1

SUB-CHAPTER 14.6 - TABLE 20

Steam Generator Rupture of 2 Tubes, Cumulative activity release to the environment of the plant

Nuclide	First Release [Bq]	Second Release [Bq]
Kr85m	8.0E+08	2.7E+08
Kr85	7.9E+07	6.5E+07
Kr87	1.3E+09	1.7E+08
Kr88	2.0E+09	4.4E+08
Xe133m	2.6E+08	2.0E+08
Xe133	1.2E+10	9.7E+09
Xe135	4.7E+07	1.4E+09
I131ae	2.4E+08	2.6E+09
I131el	2.7E+07	2.9E+08
I132ae	4.1E+08	6.1E+08
I132el	4.5E+07	6.8E+07
I133ae	7.4E+08	6.5E+09
I133el	8.3E+07	7.3E+08
I134ae	2.3E+08	5.7E+07
I134el	2.5E+07	6.4E+06
I135ae	4.6E+08	2.7E+09
I135el	5.5E+07	3.0E+08
Sr89	7.4E+05	6.1E+05
Sr90	4.5E+03	3.8E+03
Cs134	5.4E+07	6.0E+08
Cs137	5.4E+07	6.0E+08
Total:	**1.9E+10**	2.7E+10

SUB-CHAPTER 14.6 - TABLE 21

Steam Generator Tube Rupture of 2 Tubes, Effective Doses for Adults and Infants, All Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	84	170
1	34	64
2	39	79
5	11	22
10	4.6	8.2

SUB-CHAPTER 14.6 - TABLE 22

Steam Generator Tube Rupture of 2 Tubes, Thyroid Doses for Adults and Infants, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	83	176
1	30	64
2	11	23
5	2.9	6.1
10	1.1	2.2

SUB-CHAPTER 14.6 - TABLE 23

Fuel Handling Accident; Activity Inventory of one Fuel Rod

Nuclide	Activity Inventory of a Fuel Rod (Bq)
Kr-85m	2.3E+09
Kr-85	1.4E+12
Kr-88	2.8E+7
Xe-131m	2.1E+12
Xe-133m	5.7E+12
Xe-133	2.2E+14
I-131	1.1E+14
I-133	3.8E+13
I-135	4.7E+11

SUB-CHAPTER 14.6 - TABLE 24

Fuel Handling Accident - Cumulative activity release to the environment of the plant, Bq

Time(h):	**0.5**	**2.5**	**4.5**	**8.4**	**16.4**	**24.3**	**48.1**	**121.5**	**167.1**
Kr85m	2.4E+09	3.6E+09	3.6E+09	3.6E+09	3.6E+09	3.6E+09	3.6E+09	3.6E+09	3.6E+09
Kr85	1.5E+12	2.3E+12	2.3E+12	2.3E+12	2.3E+12	2.3E+12	2.3E+12	2.3E+12	2.3E+12
Kr88	2.8E+07	4.2E+07	4.2E+07	4.2E+07	4.2E+07	4.2E+07	4.2E+07	4.2E+07	4.2E+07
Xe131m	2.2E+12	3.5E+12	3.6E+12	3.6E+12	3.6E+12	3.6E+12	3.6E+12	3.6E+12	3.6E+12
Xe133m	6.1E+12	9.6E+12	9.7E+12	9.7E+12	9.7E+12	9.7E+12	9.7E+12	9.7E+12	9.7E+12
Xe133	2.4E+14	3.7E+14	3.8E+14	3.8E+14	3.8E+14	3.8E+14	3.8E+14	3.8E+14	3.8E+14
I131el	3.6E+09	3.8E+09	3.9E+09	4.2E+09	4.8E+09	5.3E+09	6.8E+09	1.1E+10	1.3E+10
I133el	1.2E+09	1.3E+09	1.3E+09	1.4E+09	1.5E+09	1.6E+09	1.8E+09	1.9E+09	1.9E+09
I135el	1.5E+07	1.5E+07	1.6E+07	1.6E+07	1.7E+07	1.7E+07	1.7E+07	1.7E+07	1.7E+07
Total:	**2.5E+14**	**3.9E+14**	**3.9E+14**	**3.9E+14**	**3.9E+14**	**3.9E+14**	**3.9E+14**	**3.9E+14**	**3.9E+14**

SUB-CHAPTER 14.6 - TABLE 25

Fuel Handling Accident, Effective Doses, all Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	71	310
1	30	113
2	36	207
5	9.3	45
10	3.5	14

SUB-CHAPTER 14.6 - TABLE 26

Fuel Handling Accident, Thyroid Doses, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	60	118
1	20	39
2	6.2	12
5	1.3	2.5
10	0.37	0.73

SUB-CHAPTER 14.6 - TABLE 27

Loss of Condenser Vacuum, Cumulative Activity Release to the Environment of the Plant

Nuclide	Activity Release [Bq]
Sr89	3.93E+06
Sr90	2.43E+04
I131ae	1.17E+09
I131el	1.30E+08
I132ae	1.12E+09
I132el	1.24E+08
I133ae	3.25E+09
I133el	3.61E+08
I134ae	3.78E+08
I134el	4.20E+07
Cs134	2.59E+08
I135ae	1.84E+09
I135el	2.04E+08
Cs137	2.59E+08
Cs138	2.36E+09
Mn54	3.24E+06
Co58	1.30E+07
Fe59	8.10E+05
Co60	8.10E+05

SUB-CHAPTER 14.6 - TABLE 28

Loss of Condenser Vacuum, Effective Doses for Adults and Infants, All Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	34	69
1	13	26
2	16	32
5	4.5	8.6
10	1.8	3.3

SUB-CHAPTER 14.6 - TABLE 29

Loss of Condenser Vacuum, Thyroid Doses for Adults and Infants, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	35	75
1	13	28
2	4.6	9.9
5	1.2	2.6
10	0.45	0.97

SUB-CHAPTER 14.6 - TABLE 30

Multiple failure of systems in Nuclear Auxiliary Building under earthquake boundary conditions, Activity Releases to the Environment, Bq

Time (h):	1	4	8	16	24	48	72	96	120	144	168
Ar41	1.3E+10	2.6E+10	2.9E+10	2.9E+10	2.9E+10	2.9E+10	2.9E+10	2.9E+10	2.9E+10	2.9E+10	2.9E+10
Kr83m	2.3E+09	1.3E+10	1.7E+10	1.8E+10	1.8E+10	1.8E+10	1.8E+10	1.8E+10	1.8E+10	1.8E+10	1.8E+10
Kr85m	2.9E+11	7.0E+11	8.5E+11	9.0E+11	9.0E+11	9.0E+11	9.0E+11	9.0E+11	9.0E+11	9.0E+11	9.0E+11
Kr85	9.9E+10	2.9E+11	4.0E+11	4.8E+11	4.9E+11	4.9E+11	4.9E+11	4.9E+11	4.9E+11	4.9E+11	4.9E+11
Kr87	2.1E+11	3.7E+11	3.9E+11	3.9E+11	3.9E+11	3.9E+11	3.9E+11	3.9E+11	3.9E+11	3.9E+11	3.9E+11
Kr88	5.2E+11	1.2E+12	1.3E+12	1.4E+12	1.4E+12	1.4E+12	1.4E+12	1.4E+12	1.4E+12	1.4E+12	1.4E+12
Mn54	1.3E+05	3.7E+05	5.2E+05	6.1E+05	6.3E+05	6.4E+05	6.4E+05	6.4E+05	6.4E+05	6.4E+05	6.4E+05
Co58	5.2E+05	1.5E+06	2.1E+06	2.5E+06	2.6E+06	2.6E+06	2.6E+06	2.6E+06	2.6E+06	2.6E+06	2.6E+06
Fe59	3.2E+04	9.2E+04	1.3E+05	1.5E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05
Co60	3.2E+04	9.2E+04	1.3E+05	1.5E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05	1.6E+05
Rb88	3.2E+11	1.0E+12	1.2E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12
Sr89	8.2E+04	2.4E+05	3.3E+05	3.9E+05	4.0E+05	4.1E+05	4.1E+05	4.1E+05	4.1E+05	4.1E+05	4.1E+05
Sr90	5.1E+02	1.5E+03	2.1E+03	2.5E+03	2.5E+03	2.6E+03	2.6E+03	2.6E+03	2.6E+03	2.6E+03	2.6E+03
I131ae	8.9E+06	2.6E+07	3.6E+07	4.2E+07	4.4E+07	4.4E+07	4.4E+07	4.4E+07	4.4E+07	4.4E+07	4.4E+07
I131org	1.1E+07	4.3E+07	8.5E+07	1.7E+08	2.4E+08	4.7E+08	6.7E+08	8.6E+08	1.0E+09	1.2E+09	1.3E+09
I131el	1.2E+07	4.6E+07	8.9E+07	1.7E+08	2.5E+08	4.7E+08	6.7E+08	8.6E+08	1.0E+09	1.2E+09	1.3E+09
I132ae	4.3E+06	9.2E+06	1.0E+07	1.0E+07	1.1E+07	1.1E+07	1.1E+07	1.1E+07	1.1E+07	1.1E+07	1.1E+07
I132org	7.3E+06	1.9E+07	2.5E+07	2.7E+07	2.7E+07	2.7E+07	2.7E+07	2.7E+07	2.7E+07	2.7E+07	2.7E+07
I132el	7. 7E+06	2.0E+07	2.6E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07
I133ae	1.6E+07	4.5E+07	6.1E+07	6.9E+07	7.1E+07	7.1E+07	7.1E+07	7.1E+07	7.1E+07	7.1E+07	7.1E+07
I133org	1.7E+07	6.3E+07	1.2E+08	2.0E+08	2.7E+08	3.9E+08	4.5E+08	4.7E+08	4.8E+08	4.9E+08	4.9E+08
I133el	1.9E+07	6.8E+07	1.2E+08	2.1E+08	2.8E+08	4.0E+08	4.5E+08	4.8E+08	4.9E+08	4.9E+08	5.0E+08
Xe133m	2.7E+11	7.8E+11	1.1E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12	1.3E+12
Xe133	1.4E+13	4.0E+13	5.6E+13	6.6E+13	6.7E+13	6.8E+13	6.8E+13	6.8E+13	6.8E+13	6.8E+13	6.8E+13
I134ae	1.8E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06	2.7E+06
I134org	3.4E+06	5.8E+06	6.1E+06	6.1E+06	6.1E+06	6.1E+06	6.1E+06	6.1E+06	6.1E+06	6.1E+06	6.1E+06
I134el	.6E+06	6.1E+06	6.4E+06	6.4E+06	6.4E+06	6.4E+06	6.4E+06	6.4E+06	6.4E+06	6.4E+06	6.4E+06
Cs134	8.1E+07	2.3E+08	3.3E+08	3.9E+08	4.0E+08	4.1E+08	4.1E+08	4.1E+08	4.1E+08	4.1E+08	4.1E+08
I135ae	7.9E+06	2.0E+07	2.6E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07	2.8E+07
I135org	1.0E+07	3.4E+07	5.6E+07	7.9E+07	8.9E+07	9.6E+07	9.7E+07	9.7E+07	9.7E+07	9.7E+07	9.7E+07
I135el	1.1E+07	3.6E+07	5.8E+07	8.2E+07	9.2E+07	9.9E+07	1.0E+08	1.0E+08	1.0E+08	1.0E+08	1.0E+08
Xe135	1.5E+12	4.0E+12	5.2E+12	5.7E+12	5.7E+12	5.7E+12	5.7E+12	5.7E+12	5.7E+12	5.7E+12	5.7E+12
Cs137	8.1E+07	2.3E+08	3.3E+08	3.9E+08	4.0E+08	4.1E+08	4.1E+08	4.1E+08	4.1E+08	4.1E+08	4.1E+08
Xe138	7.7E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10	8.0E+10
Cs138	4.2E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10	6.8E+10
Total:	1.7E+13	4.8E+13	6.6E+13	7.7E+13	7.9E+13	7.9E+13	7.9E+13	7.9E+13	7.9E+13	7.9E+13	7.9E+13

SUB-CHAPTER 14.6 - TABLE 31

Multiple failure of systems in Nuclear Auxiliary Building under earthquake boundary conditions, Effective Doses for Adults and Infants, All Pathways

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	28	42
1	11	17
2	9.7	12
5	2.8	3.1
10	1.1	1.3

SUB-CHAPTER 14.6 - TABLE 32

Multiple failure of systems in Nuclear Auxiliary Building under earthquake boundary conditions, Thyroid Doses for Adults and Infants, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	5.4	11
1	1.7	3.4
2	0.55	1.1
5	0.12	0.24
10	0.04	0.08

SUB-CHAPTER 14.6 - TABLE 33

RRA [RHRS] failure outside Containment - Cumulative activity release to the environment of the plant

Nuclide	Radioactivity [Bq]
Kr85m	4.0E+10
Kr85	5.7E+09
Kr87	3.2E+09
Kr88	5.0E+10
Mn54	2.2E+08
Co58	2.9E+09
Fe59	5.4E+07
Co60	9.0E+07
Sr89	2.3E+05
Sr90	1.4E+03
I131ae	1.5E+09
I131el	1.7E+08
I132ae	1.2E+08
I132el	1.3E+07
I133ae	1.1E+09
I133el	1.2E+08
Xe133m	3.9E+10
Xe133	1.6E+12
I134ae	1.6E+06
I134el	1.7E+05
Cs134	1.4E+09
I135ae	4.0E+08
I135el	4.4E+07
Xe135	1.5E+11
Cs137	1.4E+09
Cs138	2.0E+05
Total:	**1.9E+12**

SUB-CHAPTER 14.6 - TABLE 34

RRA [RHRS] failure outside Containment, effective Dose

Distance	Effective Dose Adult	Effective Dose Infant
km	µSv	µSv
0.5	172	243
1	67	95
2	72	81
5	21	24
10	8.6	9.6

SUB-CHAPTER 14.6 - TABLE 35

RRA [RHRS] failure outside Containment, Thyroid Dose, Inhalation

Distance	Thyroid Dose Adult	Thyroid Dose Infant
km	µSv	µSv
0.5	34	67
1	12	24
2	4.4	8.6
5	1.2	2.3
10	0.43	0.86

SUB-CHAPTER 14.6 - TABLE 36

Consumption Rates (kg/a)

	Infant	Adult
Drinking Water	200	700
Milk, Milkproducts	480	390
Fish	15	37.5
Meat, Sausage, Eggs	26	180
Cereals and Products	60	220
Local Fruits and Fruit Juice	135	105
Potatoes and Root Vegetables	120	165
Leafy Vegetables	18	39
Vegetables and Products, Juices of vegetables	51	120

Source: German Radiation Protection Ordinance

SUB-CHAPTER 14.6 - TABLE 37

Rate of activity release in the case of cladding failure

Elements	Rate used for evaluating the release from UO_2		Rate used for evaluating the release from MOX	
	Burnup ≤ 47 GWd/t	**Burnup > 47 GWd/t**	**Burnup ≤ 33 GWd/t**	**Burnup > 33 GWd/t**
Kr85	8%	25%	8%	50%
other noble gases	2%	8%	2%	15%
Bromium, rubidium, iodine, caesium	2%	8%	2%	15%

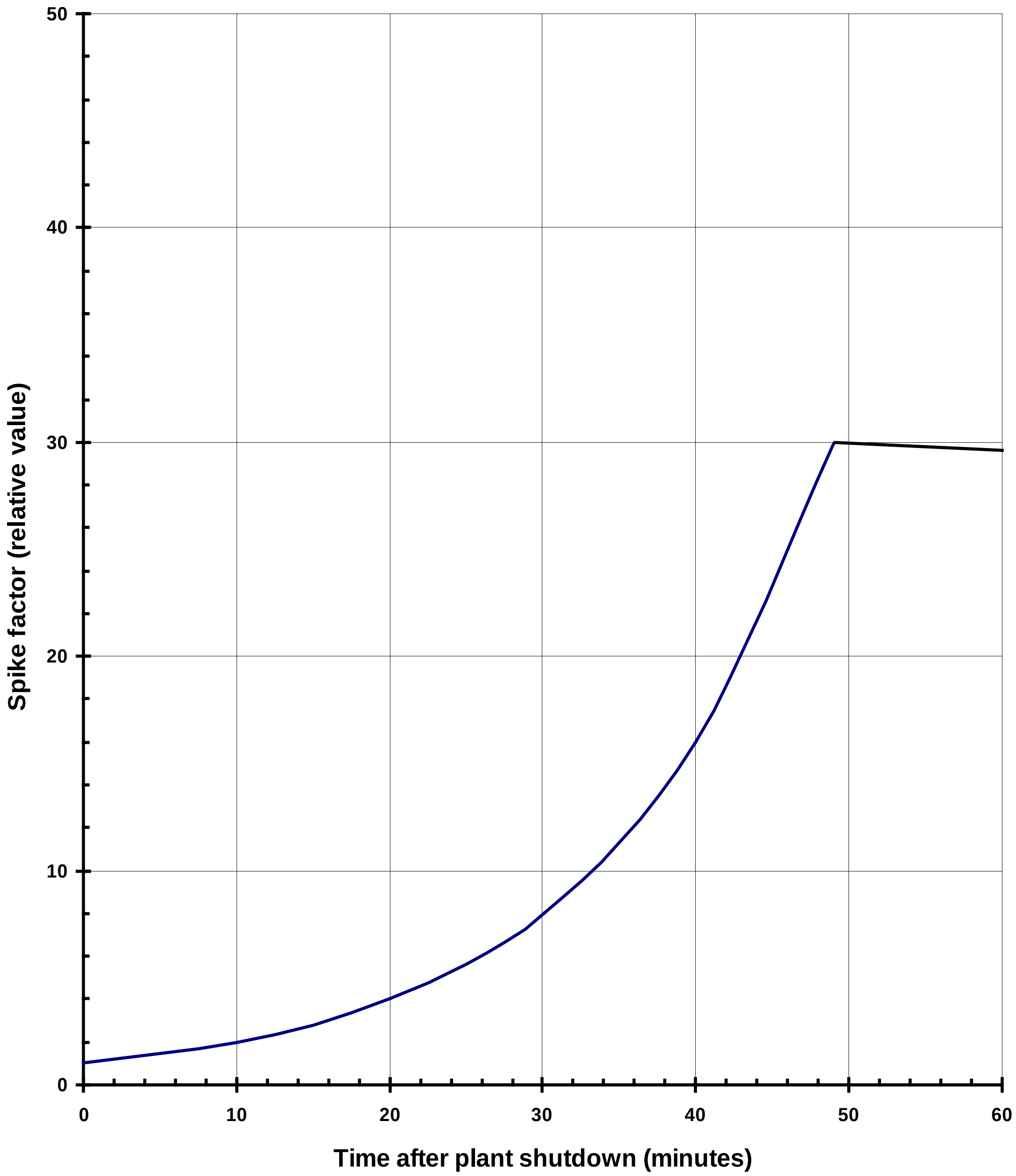
SUB-CHAPTER 14.6 - TABLE 38

Radiological consequences of accidents with single initiating event

		Calculations at 500 m (short-term, 7 days)		Calculations at 2 km (long-term, 50 years)
Transient - Category 2:	**Dose (Sv)**	**Adult (Sv)**	**Child (Sv)**	**Adult (Sv)**
Loss of condenser vacuum	Effective Thyroid	1.9×10^{-5} 2.2×10^{-4}	2.5×10^{-5} 4.0×10^{-4}	6.9×10^{-5} 1.7×10^{-5}
Incidents - Category 3:				
Break of a primary coolant pipe outside of the containment	Effective Thyroid	5.6×10^{-6} 1.7×10^{-5}	6.0×10^{-6} 2.9×10^{-5}	6.8×10^{-6} 1.3×10^{-6}
Steam Generator Tube Rupture – 1 tube	Effective Thyroid	1.4×10^{-3} 2.4×10^{-3}	1.4×10^{-3} 4.1×10^{-3}	1.1×10^{-3} 2×10^{-4}
Accidents - Category 4:				
Large break LOCA in nominal operation	Effective Thyroid	2.9×10^{-4} 2.4×10^{-4}	2.3×10^{-4} 3.9×10^{-4}	1.4×10^{-4} 1.9×10^{-5}
LOCA in shutdown state	Effective Thyroid	2.3×10^{-5} 9.3×10^{-5}	2.2×10^{-5} 1.5×10^{-4}	1.4×10^{-4} 7.0×10^{-6}
Multiple failures in the Nuclear Auxiliaries Building in an earthquake	Effective Thyroid	3.8×10^{-4} 2.1×10^{-4}	3.8×10^{-4} 3.1×10^{-4}	7.3×10^{-5} 1.7×10^{-5}
Steam Generator Tube Rupture – 2 tubes	Effective Thyroid	1.5×10^{-3} 4.8×10^{-3}	1.6×10^{-3} 8.4×10^{-3}	2.1×10^{-3} 4×10^{-4}
Accident during fuel handling	Effective Thyroid	5.5×10^{-3} 1.8×10^{-4}	5.5×10^{-3} 2.7×10^{-4}	6.1×10^{-4} 2.0×10^{-5}

SUB-CHAPTER 14.6 - FIGURE 1

Assumed Relative Increase of Specific Activity of I-131, Cs-134 and Cs-137 in the Primary Coolant After Plant Shutdown



SUB-CHAPTER 14.6 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

0. SAFETY REQUIREMENTS

0.2. RADIOLOGICAL OBJECTIVES

[Ref-1] Technical Guidelines for the design and construction of the next generation of nuclear power plants with pressurised water reactors – Adopted during the GPR/German experts plenary meetings held on October 19th and 26th 2000 (E)

[Ref-2] ICRP Publication 63. Principles for Intervention for Protection of the Public in a Radiological Emergency. A report by a task group of Committee 4 of ICRP, adopted by the ICRP in November 1992, Oxford, New York, Seoul, Tokyo, 1993 (E)

[Ref-3] Council Decree dated December 22, 1987, Specifying Maximum Values for Radioactivity in Foodstuffs and Animal Feeds in the Event of a Nuclear Accident or Other Radiological Emergency, Official Journal of the European Community No. L371/11, amended on July 18, 1989.
Official Journal of the European Community No. L211/1 (E)

1. GENERAL STATEMENTS AND ASSUMPTIONS

1.1. INTRODUCTION

[Ref-1] Technical Guidelines for the design and construction of the next generation of nuclear power plants with pressurised water reactors – Adopted during the GPR/German experts plenary meetings held on October 19th and 26th 2000

1.3. ACTIVITY INVENTORIES

1.3.1. Activity Inventory in the Reactor Core

[Ref-1] C. Gauld, O. W. Hermann, R. M. Westfall, ORIGEN-S: Scale System Module to calculate fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms.
NUREG/CR-200, ORNL/TM-2005/39 Version 6 Vol. II, Sect. F7.
Oak Ridge National Laboratory, 2002 (E)

1.4. MECHANISMS OF ACTIVITY RELEASE

1.4.2. Activity Release Due to Discharge of Liquids

1.4.2.1. Discharge of Liquids with Evaporation

[Ref-1] Incident Calculation Bases for the Guidelines Issued by the Federal Minister of the Interior (BMI) for the assessment of the design of PWR Nuclear Power Plants pursuant to Sec. 28, para. (3) of the Radiological Protection Ordinance (StrlSchV). 1989. (E)

1.4.2.2. Discharge of Liquids without Evaporation

[Ref-1] Sutter, S.L.: Johnston, J.W., Mishima, J.:
Aerosols generated by free fall spill of powders and solutions in static air.
NUREG/CR-2139. 1981. (E)

1.4.3. Activity Carryover into Steam Generated in the Steam Generator

1.4.3.2. Activity Carryover in the Case of Operational Leakage

[Ref-1] Incident Calculation Bases for the Guidelines Issued by the Federal Minister of the Interior (BMI) for the assessment of the design of PWR Nuclear Power Plants pursuant to Sec. 28, para. (3) of the Radiological Protection Ordinance (StrlSchV). 1989. (E)

1.4.3.3. Activity Carryover in the Case of SGTR

- **Case of High Steam Production**

c) Rupture Located at Uncovered Tube U-bend:

[Ref-1] Forschungsvorhaben Reaktorsicherheit Abschlussbericht Förderungsvorhaben BMFT [Research project nuclear reactor safety; Final report of the financially funded project BMFT] 1500 685/5

[Ref-2] K. H. Neeb; W. Morell; F. Richter; E.Haas; W. Kastner; R. Rippel; G. Roebig; R. Ahrens-Botzong Experimentelle Untersuchungen zur Abgabe von Spaltprodukten (insbesondere von Jod und Cäsium) bei Auslegungsstörfällen- Teilvorhaben I
[Experimental investigation on the release of fission products (in particular of iodine and caesium) during design basis accidents – Sub-project I]
R/914/84/008 - Kraftwerk Union AG - April 1984

[Ref-3] Forschungsvorhaben Reaktorsicherheit Abschlussbericht Förderungsvorhaben BMFT [Research project nuclear reactor safety; Final report of the financially funded project BMFT] 1500 770/1

[Ref-4] Fr. Sieghard Hellmann; Alois Bleier; Reinhard Rippel; Joachim Funcke - Experimentelle Untersuchungen zur Abgabe von Spaltprodukten (insbesondere von Jod und Cäsium) bei Auslegungsstörfällen (Wirkleitungsbruch) - Teilvorhaben II
[Experimental investigation on the release of fission products (in particular of iodine and caesium) during design basis accidents (rupture of a differential pressure pipe) - Sub-project II]
E 141/90/200 - Siemens Bereich Energieerzeugung (KWU) - November 1990

1.6. CALCULATION OF RADIATION EXPOSURE

[Ref-1] International Commission on Radiological Protection:
Recommendations of the International Commission on Radiological Protection; Publications No: 56 (Age-Dependent Doses to the Public from Intake of Radionuclides; Part 1; 1990), 60 (1990 Recommendations of the International Commission on Radiological Protection, 1992), 67 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 2; Ingestion Dose Coefficients 1990; 1994), 69 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 3; Ingestion Dose Coefficients ;1995), 71 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 4; Ingestion Dose Coefficients ;1996), 72 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 5; Compilation of Ingestion and Inhalation Dose Coefficients ;1996) (E)

1.9. FALLOUT AND WASHOUT

[Ref-1] [Incident Calculation Bases for the Guidelines Issued by the Federal Minister of the Interior (BMI) for the assessment of the design of PWR Nuclear Power Plants persuant to Sec. 28, para. (3) of the Radiological Protection Ordinance (StrlSchV)] 1989 (E)

1.10. RADIATION EXPOSURE

[Ref-1] International Commission on Radiological Protection:
Recommendations of the International Commission on Radiological Protection; Publications No: 56 (Age-Dependent Doses to the Public from Intake of Radionuclides; Part 1; 1990), 60 (1990 Recommendations of the International Commission on Radiological Protection, 1992), 67 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 2; Ingestion Dose Coefficients 1990; 1994), 69 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 3; Ingestion Dose Coefficients ;1995), 71 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 4; Ingestion Dose Coefficients ;1996), 72 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 5; Compilation of Ingestion and Inhalation Dose Coefficients ;1996) (E)

[Ref-2] Council Directive 96/29/Euratom dated May 13 1996 which sets out the basic standards relating to the health protection of the population and worked against the dangers arising from ionising radiation.
Official Journal of the European Communities, L159; Vol. 39; June 1996 (E)

2. LARGE BREAK LOCA (PCC-4)

2.2. MAIN RESULTS OF THERMAL-HYDRAULIC ANALYSES RELEVANT FOR RADIOLOGICAL ANALYSIS

[Ref-1] Stephenson, W., Dutton, L.M.C., Handy, B. J., Smedley, C.
Realistic Methods for Calculating the Releases and Consequences of a Large LOCA, Commission of the European Communities; Contract No. ETNU-0001/UK; Final Report, EUR 14179; 1992 (E)

[Ref-2] Kabat, M.J.: Chemical behavior of radioiodine under loss of coolant accident condition, 16th DOE Nuclear Air Cleaning Conference, San Diego 1980. (E)

[Ref-3] Incident Calculation Bases for the Guidelines Issued by the Federal Minister of the Interior (BMI) for the assessment of the design of PWR Nuclear Power Plants pursuant to Sec. 28, para. (3) of the Radiological Protection Ordinance (StrlSchV). 1989. (E)

[Ref-4] The computer code ACARE (Activity in interrelated compartments and release to the environment) was formerly used as a pre-program for PRODOS (Probabilistic Dose Calculation, cf. answer below, ch. 3.3) to calculate the activity flow of nuclides in coupled compartments. Both programs were kind of complicated to handle and therefore taken out of use a couple of years ago and are not going to be reactivated.

Radioactive decay and production of daughter nuclides were taken into account. Deposition of aerosols and iodine in the single compartments could be considered in four different ways. The last compartment represented the atmosphere. The results were presented in a time dependent form for the activity inventory in every compartment, the release rates to the atmosphere and the total release.

The essential application was to sum over all nuclides the nuclide specific release multiplied with dose factors. These values were used in PRODOS to calculate probabilistically distributions of radiation doses in the vicinity of nuclear power plants after accidents.

For the calculation of radiation doses the program chain ACARE-PRODOS was replaced by MACCS.

D. Chanin, M.L. Young: Code Manual for MACCS2: Volume 1, User’s Guide, NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, Albuquerque, NM, USA (1998). (E)

2.4. POTENTIAL RADIATION EXPOSURE

[Ref-1] The computer code PRODOS (Probabilistic Dose Calculation) was formerly used to make prediction of radiological impact in the environment resulting from accidental releases into air and for the assessment of emergency countermeasures after postulated severe accidents inside nuclear power plants, using input data from the computer code ACARE (Activity in interrelated compartments and release to the environment; cf. answer ch. 3.2.6). Both programs were kind of complicated to handle and therefore taken out of use a couple of years ago and are not going to be reactivated.

PRODOS calculated probability distributions of consequences to the environment of a plant due to a time depending release of radioactive substances out of this plant using a weather course of a larger time span. The release period was related to each possible weather sequences in the time span.

For the calculation of radiation doses the program chain ACARE-PRODOS was replaced by MACCS.

D. Chanin, M.L. Young: Code Manual for MACCS2: Volume 1, User’s Guide, NUREG/CR-6613, SAND97-0594, Vol. 1, Sandia National Laboratories, Albuquerque, NM, USA (1998). (E)

9. FUEL HANDLING ACCIDENT (PCC-4)

9.1. POSTULATED ACCIDENT SEQUENCE AND ACTIVITY RELEASE TO THE ENVIRONMENT

[Ref-1] C. Gauld, O. W. Hermann, R. M. Westfall, ORIGEN-S: Scale System Module to calculate fuel depletion, actinide transmutation, fission product buildup and decay, and associated radiation source terms.
NUREG/CR-200, ORNL/TM-2005/39 Version 6 Vol. II, Sect. F7.
Oak Ridge National Laboratory, 2002. (E)

[Ref-2] Incident Calculation Bases for the Guidelines Issued by the Federal Minister of the Interior (BMI) for the assessment of the design of PWR Nuclear Power Plants pursuant to Sec. 28, para. (3) of the Radiological Protection Ordinance (StrlSchV). 1989. (E)

15. EVALUATION OF THE POTENTIAL RADIOLOGICAL CONSEQUENCES USING FRENCH NPPS ASSESSMENT METHOD

15.1. PRINCIPLES

[Ref-1] G Fleury. Methodology for evaluating the radiological consequences of an accidental atmospheric release. ENTEAG090116 Revision A. EDF. April 2009. (E)

ENTEAG090116 Revision A is the English translation of ENTEAG030152 Revision B.

15.2. DOSE CALCULATION ASSUMPTIONS

Atmospheric diffusion of fission products released into the environment:

[Ref-1] Doury. Le vadémécum des transferts atmosphériques [Atmospheric Transfer Handbook] Report DSN n° 440. February 1992.

Conversion factors for dosage, respiratory flow:

[Ref-2] G Fleury. Methodology for evaluating the radiological consequences of an accidental atmospheric release. ENTEAG090116 Revision A. EDF. April 2009. (E)

ENTEAG090116 Revision A is the English translation of ENTEAG030152 Revision B

[Ref-3] International Commission on Radiological Protection:
Recommendations of the International Commission on Radiological Protection;
Publications No: 56 (Age-Dependent Doses to the Public from Intake of Radionuclides;
Part 1; 1990), 60 (1990 Recommendations of the International Commission on Radiological Protection, 1992), 67 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 2; Ingestion Dose Coefficients 1990; 1994), 69 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 3; Ingestion Dose Coefficients ;1995), 71 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 4; Ingestion Dose Coefficients ;1996), 72 (Age-Dependent Doses to Members of the Public from Intake of Radionuclides;
Part 5; Compilation of Ingestion and Inhalation Dose Coefficients ;1996) (E)

[Ref-4] Council Directive 96/29/Euratom dated May 13 1996 which sets out the basic standards relating to the health protection of the population and worked against the dangers arising from ionising radiation.
Official Journal of the European Communities, L159; Vol. 39; June 1996 (E)

15.4. PRINCIPAL ASSUMPTIONS

[Ref-1] P Jan. PSAR EPR - Sensitivity analysis about radiological consequences in fault situations. ENTEAG090113 Revision A. EDF. April 2009. (E)

ENTEAG090113 Revision A is the English translation of ENTEAG050091 Revision B.

- **Filter efficiency**

[Ref-2] C Cherbonnel. Hypothesis to be used for evaluating the radiological consequences of design basis accident – Analysis of the ability of the different ventilation systems to reduce the fission products releases. ENFPIN/0100177 Revision A1. EDF. September 2009. (E)

ENFPIN/0100177 Revision A1 is the English translation of ENFPIN/0100177 Revision A

15.5. SPECIFIC ASSUMPTIONS RELATING TO THE ACCIDENTS CONSIDERED

- **Rupture of a steam generator tube (SGTR 1 tube, PCC-3)**

[Ref-1] Primary Source Term of the EPR Reactor. ENTERP090062 Revision A. EDF. March 2009. (E)

ENTERP090062 Revision A is the English translation of ENTERP070147 Revision A.

[Ref-2] G Ranchoux. Primary Source Term of EPR Reactor: Additional Information. ENTERP090128 Revision A. EDF. May 2009. (E)

15.6. RESULTS

[Ref-1] P Jan. PSAR EPR - Sensitivity analysis about radiological consequences in fault situations. ENTEAG090113 Revision A. EDF. April 2009. (E)

ENTEAG090113 Revision A is the English translation of ENTEAG050091 Revision B.

[Ref-2] P Jan. UK-EPR – Radiological consequences for SGTR (PCC3 and PCC4). ENTEAG080029 Revision A. EDF. June 2008. (E)

16. SENSITIVITY STUDIES REQUIRED BY THE TECHNICAL GUIDELINES

[Ref-1] Technical Guidelines for the design and construction of the next generation of nuclear power plants with pressurised water reactors – Adopted during the GPR/German experts plenary meetings held on October 19th and 26th 2000. (E)