

# Safety Standards

of the

Nuclear Safety Standards Commission (KTA)

---

**KTA 1503.3 (6/99)**

**Monitoring the Discharge of Gaseous and Aerosol-bound  
Radioactive Substances;**

**Part 3: Monitoring the Non-stack Discharge of Radioactive  
Substances**

(Überwachung der Ableitung gasförmiger und aerosolgebun-  
dener radioaktiver Stoffe;

Teil 3: Überwachung der nicht mit der Kaminfortluft abgeleiteten  
radioaktiven Stoffe)

---

If there is any doubt regarding the information contained in this translation, the German wording shall apply.

Editor:

KTA-Geschäftsstelle c/o Bundesamt fuer Strahlenschutz (BfS)

Willy-Brandt-Str. 5 • 38226 Salzgitter • Germany

Telephone +49-(0)1888-333-(0)1621 • Telefax +49-(0)1888-333-1625



# KTA SAFETY STANDARD

June 1999

**Monitoring the Discharge of  
Gaseous and Aerosol-bound Radioactive Substances;**

KTA 1503.3

**Part 3: Monitoring the Non-stack Discharge of  
Radioactive Substances**

## CONTENTS

Fundamentals .....	1
1 Scope .....	1
2 Definitions .....	1
3 Determining the Discharge Paths and Monitoring Concept .....	2
3.1 Criteria for Determining the Discharge Paths to be Surveilled .....	2
3.2 Surveillance Concept .....	2
4 Technical Means, Administrative Measures and Surveillance Procedures .....	2
4.1 Discharge Paths .....	2
4.2 Surveillance Measures .....	2
4.3 Surveillance Procedures .....	2
5 Design of Equipment for the Continuous Radioactivity Measurements and Sample Extractions .....	4
5.1 General Requirements .....	4
5.2 Measurement Devices for Continuous Radioactivity Measurements .....	6
6 Maintenance of the Equipment for Continuous Radioactivity Measurements and Sample Extraction .....	10
6.1 Servicing and Repair .....	10
6.2 Tests and Inspections .....	10
6.3 Removal of Defects .....	11
7 Documentation of Measurement Results .....	11
7.1 Flow Chart .....	11
7.2 Extent of Documentation .....	11
Appendix A: Procedures for the Analysis of PWR .....	14
Appendix B: Procedure for the Analysis of BWR of Product Line 72 .....	15
Appendix C: Nuclide Specific Evaluation during Specified Normal Operation of Samples from the Blowdown Water (KONVOI-type Plants) or Main Steam (PWR with Straight-tube Steam Generators of the Type Built by BBR) .....	16
Appendix D: Regulations Referred to in this Safety Standard .....	16

PLEASE NOTE: Only the original German version of this safety standard represents the joint resolution of the 50-member Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA). The German version was made public in Bundesanzeiger BAnz No. 243b of December 23, 1999. Copies may be ordered through the Carl Heymanns Verlag KG, Luxemburger Str. 449, 50939 Koeln, Germany (Telefax +49-(0)221-94373603).

All questions regarding this English translation should please be directed to:

KTA-Geschäftsstelle c/o BfS, Willy-Brandt-Str. 5, 38226 Salzgitter, Germany

### Comments by the editor:

Taking into account the meaning and usage of auxiliary verbs in the German language, in this translation the following agreements are effective:

- shall** indicates a mandatory requirement,
- shall basically** is used in the case of mandatory requirements to which specific exceptions (and only those!) are permitted. It is a requirement of the KTA that these exceptions - other than those in the case of **shall normally** - are specified in the text of the safety standard,
- shall normally** indicates a requirement to which exceptions are allowed. However, the exceptions used shall be substantiated during the licensing procedure,
- should** indicates a recommendation or an example of good practice,
- may** indicates an acceptable or permissible method within the scope of this safety standard.

## Fundamentals

(1) The safety standards of the Nuclear Safety Standards Commission (KTA) have the task of specifying those safety related requirements which shall be met with regard to precautions to be taken in accordance with the state of science and technology against the hazards arising from the construction and operation of the facility (Section 7 para. 2 no. 3 Atomic Energy Act), in order to attain the protective goals specified in the Atomic Energy Act and the Radiological Protection Ordinance (StrlSchV) and which are further detailed in "Safety Criteria for Nuclear Power Plants" and in "Guidelines for the Assessment of the Design of Nuclear Power Plants with Pressurized Water Reactors against Incidents pursuant to Section 28 para. 3 of the Radiological Protection Ordinance (StrlSchV) - Incident Guidelines".

(2) The permanently installed and mobile radiation protection instrumentation, among other equipment, serves to protect the persons inside and outside of the facility from ionizing radiation, to ascertain the specified normal functioning of the equipment for keeping solid, liquid and gaseous radioactive substances within the intended enclosure, for the handling and controlled conduction of the radioactive substances within the facility and for Monitoring the discharge of radioactive substances. The safety standards in the KTA 1500 series formulate specific safety related requirements for this instrumentation.

(3) Safety standard KTA 1503 deals with requirements for the technical equipment and additional organizational measures considered necessary with respect to monitoring the emission of gaseous and aerosol-bound radioactive substances. It is comprised of the following parts:

Part 1: Monitoring the stack discharge of radioactive substances during specified normal operation,

Part 2: Monitoring the stack discharge of radioactive substances during incidents,

Part 3: Monitoring the non-stack discharge of radioactive substances.

(4) As specified under Sec. 46 para. 1 StrlSchV, it must be ascertained that any uncontrolled discharge is prevented, that the discharged radioactivity is as low as possible, that the discharge is monitored and, specified by type and radioactivity, is reported to the proper authority in no longer than yearly intervals. In accordance with these requirements, equipment for monitoring the discharge of radioactive substances shall be installed and operated. This monitoring equipment shall, in accordance with Sec. 72 StrlSchV, correspond to the state of the art in science and technology.

(5) In the case of design basis and other accidents Sec. 36 StrlSchV requires the immediate introduction of all necessary actions to minimize the dangers to life, health and material goods. In accordance with the requirements under Sec. 38 StrlSchV the necessary auxiliary means shall be kept available with regard to damage control in case of design basis and other accidents. The surveillance carried out with the instrumentation for measuring discharged radioactive substances serves, among others, as the basis for taking these measures and for their type, extent and duration.

(6) KTA 1503.3 specifies the measures to be taken with respect to equipment required for fulfilling the tasks under paras. 4 and 5 insofar as radioactive substances are discharged other than with the stack air. This safety standard thereby contributes toward meeting the requirements in accordance with Sec. 28 para.1 item 2, Secs. 36, 38, 46 para. 1 and Sec. 72 para. 1 StrlSchV.

(7) By monitoring the radioactivity of the discharged radioactive substances (emission monitoring) in connection with meteorological measurements and atmospheric dispersion calculations on the one hand and by direct measurements in

the environment (pollution monitoring) on the other, it is possible to identify the radiological effects of the facility on the environment. Whereas, during specified normal operation the essential data will result from emission monitoring – the pollution measurements values will stay within the tolerance limits of natural radiation –, these pollution measurements will increase in importance during incidents (design basis accidents) since they will enable establishing the radiological effects.

## 1 Scope

(1) This safety standard shall apply to the equipment and procedures for monitoring the gaseous and aerosolbound radioactive substances discharged from nuclear power plants with light water reactors other than with the stack air during specified normal operation and during incidents (design basis accidents)

(2) In order to achieve the surveillance goal, the basic requirements specified under Section 3 shall be fulfilled in every nuclear power plant with light water reactors. It is permissible to apply different technical means and administrative measures in order to meet these basic requirements; correspondingly, Sections 4, 5, 6 and 7 give specific examples for the technical means and administrative measures applied in PWR and BWR plants of particular designs that can be considered to be sufficient with respect to fulfilling the basic requirements specified under Section 3.

### Note:

The technical means and administrative measures required to meet the basic requirements specified under Section 3 are also applicable in determining the radioactivity discharge in case of accidents that go beyond the design basis accidents insofar as this discharge would occur along the paths described in this safety standard. If monitoring of the discharged radioactive substances cannot anymore be assured with these technical means and administrative measures, backup measurements will be used that are carried out within the framework of environmental surveillance (emission monitoring).

(3) Surveillance equipment and procedures for those events that would require plant-internal accident management measures are not within the scope of this safety standard.

## 2 Definitions

(1) Discharge of radioactive substances

Discharge of radioactive substances is the release of radioactive substances in liquid, aerosol or gaseous form from the facility along the paths provided for in the design.

(2) Specified normal operation

Specified normal operation encompasses

- Operating processes for which the plant, assuming the able function of all systems (fault free condition), is intended and suited (normal operation);
- Operating processes which occur in the event of plant component or system malfunction (fault condition) as far as safety related reasons do not oppose continued operation (abnormal operation);
- Maintenance procedures (inspection servicing, repair).

(3) Detailed assessment of radioactive substances

Detailed assessment of radioactive substances is a special form of monitoring and consists of the identification and radioactivity determination of the radionuclides or radionuclide groups discharged in a specified time period.

(4) Release of radioactive substances

Release of radioactive substances is the escape of radioactive substances from the prescribed containment into the plant or into the environment.

(5) Calibrating a measurement arrangement of the radiation surveillance

Calibrating a measurement arrangement of the radiation surveillance is the procedure of determining the relationship between the displayed value and the value of the measured quantity.

(6) Incident (design basis accident)

An incident (design basis accident) is a chain of events which interferes with the operation of the plant or with the activity in such a way that operation or the activity must be interrupted for reasons of safety, and for which the plant must be designed or for which protective measures must be provided with regard to the activities.

(7) Monitoring / Surveillance

Surveillance is a collective term for any kind of controlled measurement of physical values including their comparison with predetermined specified values.

Note:

Surveillance is carried out, e.g., by

- a) continuous measurement or
- b) discontinuous evaluation of samples (e.g. in the laboratory) or
- c) correlation of measured values

always in combination with a comparison to predetermined specific values of the physical value (e.g. licensing values, operational values).

### 3 Determining the Discharge Paths and Monitoring Concept

#### 3.1 Criteria for Determining the Discharge Paths to be Surveilled

(1) In order to be able to specify the required surveillance measures for the non-stack discharge of radioactive substances it is essential to know the further possible discharge paths. All of the further paths shall be taken into consideration by which, during specified normal operation or during incidents (design basis accidents), gaseous and aerosolbound radioactive substances could be discharged into the environment in such amounts that emission monitoring cannot be abstained from.

(2) No radioactivity discharge need be considered possible in those cases where the system under consideration and the environment are separated from each other by at least two physical barriers or one physical barrier and one pressure barrier and if even under incident conditions one physical barrier will remain intact.

(3) A detailed assessment of the radioactivity discharge during specified normal operation is not required for those discharge paths for which it is determined that the radioactivity discharge to the environment fulfills the criterion of insignificance. This criterion is considered fulfilled where – during specified normal operation – the radioactivity that would be discharged over a period of one week is less than one tenth of the radioactivity resulting from the product of the stack air volume discharged in this time period and the detection limit for the detailed assessment of the respective nuclide group specified in accordance with KTA 1503.1 Table 3-4.

#### 3.2 Surveillance Concept

(1) The paths that in accordance with the criteria under Section 3.1 require surveillance shall each be surveilled by determining both the specific radioactivity or radioactivity concentration in the medium of the corresponding system and the amount of the discharged medium and, then, by calculating the discharged radioactivity from these values.

(2) No procedure for determining the amount of the medium that could escape in case of a release is required, provided

the radioactivity concentration in the corresponding system is monitored continuously and a release from the system is prevented by isolating the system in case a limit value of the radioactivity concentration is exceeded. This limit value shall then be set such that the possible radioactivity discharge meets the criterion specified under Section 3.1 para. 3.

## 4 Technical Means, Administrative Measures and Surveillance Procedures

### 4.1 Discharge Paths

Considering the criteria under Section 3, the multiple discharge paths (PWR) or the single discharge path (BWR) are listed in the first column of **Table 4-1** for plants of the KONVOI type, of **Table 4-2** for pressurized water reactors with straight-tube steam generators of the type built by BBR and of **Table 4-3** for boiling water reactors of the product line 72.

Note:

The review carried out when this safety standard was established concerning the possible discharge paths requiring emission monitoring under the criteria of Section 3.1 has led to the result that the discharge paths listed in Tables 4-1, 4-2 and 4-3 for the different PWR and BWR plants are definitive with regard to implementing Sec. 46 para. 1 StrlSchV within the scope of safety standard KTA 1503.3.

### 4.2 Surveillance Measures

Examples for realizing the surveillance measures are given below. The required surveillance measures are presented in **Table 4-1** for plants of the KONVOI type, in **Table 4-2** for pressurized water reactors with straight-tube steam generators of the type built by BBR and in **Table 4-3** for boiling water reactors of the product line 72, with the respective specific radioactivity in column 2 and media amounts in column 3. The required monitoring equipment are shown in **Figures 4-1, 4-2 and 4-3**, respectively.

### 4.3 Surveillance Procedures

#### 4.3.1 Plants of the KONVOI Type

##### 4.3.1.1 Radioactivity Discharged from the Secondary System during Specified Normal Operation due to Leakage into the Turbine Building Atmosphere and, via Roof Ventilators, into the Environment

(1) The radioactivity of the blow-down water of each steam generator shall be monitored by a continuous integral gamma ray measurement device.

(2) If the radioactivity concentration measured at one of the measurement devices specified under para.1 exceeds  $4 \times 10^5$  Bq/m<sup>3</sup> (cesium 137 equivalency) the discharge of iodine 131 with the turbine building vent air shall be determined as specified under Section A 1. It is permissible to conservatively assume that the measured overall radioactivity concentration is due entirely to iodine 131.

(3) If the radioactivity concentration measured at one of the measurement devices specified under para.1 exceeds  $1 \times 10^6$  Bq/m<sup>3</sup> (cesium 137 equivalency) a sample shall be extracted in weekly intervals from the corresponding blow-down pipe (P 1).

(4) If the radioactivity concentration measured at one of the measurement devices specified under para.1 exceeds  $1 \times 10^8$  Bq/m<sup>3</sup> (cesium 137 equivalency) a sample shall be extracted in daily intervals from the corresponding blow-down pipe (P 1).

(5) Each sample extracted as specified under para. 3 and 4 shall subsequently be evaluated as specified under Appendix C.

(6) The volume of demineralized water makeup in the secondary system,  $Q_N$ , and the monitored removal of fluids from the secondary system,  $Q_E$ , shall normally be determined as the time-weighted average of the measurements taken over the time period of, e.g., one week, thereby, evening out any irregular makeup volumes and removals in the detailed assessment.

**Note:**

Since, generally, not every removal of fluids from the secondary system is monitored, the result will be an overestimation of the vaporous leakage [discharged] with the turbine building vent air.

(7) The time-weighted average of the specific radioactivity of every nuclide and nuclide group identified in the evaluation specified under Appendix C shall be determined for the same time period specified under para. 6, and the emission of these nuclides and nuclide groups with the turbine building vent air shall be determined as specified under Section A 1.

**4.3.1.2 Radioactivity Discharged from the Secondary System via the Safety Valves and Blow-off Control Valves during Specified Normal Operation and Incidents (Design Basis Accidents)**

(1) In case radioactivity is discharged via the safety valves or blow-off control valves and the radioactivity concentration measurements at one of the measurement devices specified under Section 4.3.1.1 para. 1 that, simultaneously, exceed  $4 \times 10^5$  Bq/m<sup>3</sup> (cesium 137 equivalency), then without delay, a sample shall be extracted from the blow-down pipe of each steam generator (P 1) and this sample subjected to the nuclide specific evaluation as specified under Appendix C. After the initial sample extraction, further samples shall be extracted in 30 minute intervals and, likewise, subjected to the nuclide specific evaluation specified under Appendix C. Based on these measurement data and taking both the travel time of the medium from the steam generator to the sample extraction point and the transfer factor within the steam generator into account, the change in time of the specific main steam radioactivity - with the exception of noble gases - shall be determined. In the case of incidents (design basis accidents), the specific radioactivity of the noble gases in the main steam shall, additionally, be determined.

**Note:**

The specific radioactivity of noble gases in the main steam can be determined from the primary coolant radioactivity and the transfer rate.

(2) To determine the main steam discharge via the safety valves and the blow-off control valves it is necessary to establish and document the positions of the main steam safety valves (OPEN or CLOSED position), the main steam isolation valves (OPEN or CLOSED position) and the blow-off control valves (lift amplitude) as well as the positions of the isolation valves inline before the safety valves and blow-off control valves. The opening cross-section of the blow-off control valves shall be determined. It may be derived from the lift amplitude of the blow-off control valves.

(3) Either an exact analysis of the event sequence shall be performed or it shall be conservatively assumed that the entire radioactivity discharged stems from that steam generator for which the highest radioactivity concentration was determined.

(4) The pressure,  $P_E$ , in the main steam line between steam generator and blow-off station shall be continuously monitored and automatically recorded.

(5) The amount of main steam discharged via the safety valves and blow-off control valves shall be determined. This may be done on the basis of a reference point for which, during commissioning, the steam release rate is determined for a definite steam pressure and definite opening cross-section and by calculating the discharged volume from the change of main steam pressure with time, the opening cross-section determined under para. 2 and the value at the reference point.

(6) The radioactivity discharge shall be determined for each train of the main steam system on the basis of the data determined in accordance with paras. 1 and 5.

**4.3.1.3 Radioactivity Discharged during the Incident "Leakage of a Main Steam Line Downline from the Main Steam Isolation Valve"**

(1) Only minor amounts of radioactivity are expected to be discharged in case of a main steam line leakage that would lead to a closure of the main steam isolation valve. The discharged radioactivity shall be determined based on an analysis of the change of main steam pressure [in time], of the closing behavior of the main steam isolation valve and of the main steam radioactivity.

(2) The main steam radioactivity shall be determined from the measurement values of the gamma ray measurement device for continuous monitoring of the radioactivity in the corresponding steam generator blow-down line as well as from the results of analyzing in a gamma ray spectrometer one sample each from the corresponding steam generator blow-down line and from the primary coolant.

**4.3.2 Pressurized Water Reactors with Straight-tube Steam Generators of the Type Built by BBR**

**4.3.2.1 Radioactivity Discharged from the Secondary System during Specified Normal Operation due to Leakage into the Turbine Building Atmosphere and, via Roof Ventilators, into the Environment**

(1) The radioactivity in the main steam of each steam generator shall be monitored by a continuous measurement of the noble gas radioactivity concentration in the exhaust from the condenser air removal.

(2) If the noble gas radioactivity concentration in the exhaust from the condenser air removal exceeds  $8 \times 10^5$  Bq/m<sup>3</sup> (xenon 133 equivalency) a main steam sample shall be extracted in weekly intervals from each steam generator (P 1).

(3) After each sample extraction as specified under para. 2, the sample shall be analyzed as specified under Appendix C. If the iodine 131 radioactivity concentration exceeds  $8 \times 10^3$  Bq/m<sup>3</sup> a main steam sample shall be extracted in daily intervals from each steam generator and the sample analyzed as specified under Appendix C.

(4) The volume of demineralized water makeup in the secondary system,  $Q_N$ , and the monitored removal of fluids from the secondary system,  $Q_E$ , shall normally be determined as the time-weighted average of the measurements taken over the time period of, e.g., one week, thereby, evening out any irregular makeup volumes and removals in the detailed assessment.

**Note:**

Since, generally, not every removal of fluids from the secondary system is monitored, the result will be an overestimation of the vaporous leakage [discharged] with the turbine building vent air.

(5) The time-weighted average of the specific radioactivity of every nuclide and nuclide group identified in the evaluation specified under Appendix C shall be determined for the same time period specified under para. 6, and the emission of these nuclides and nuclide groups with the turbine building vent air shall be determined as specified under Section A 2.

**4.3.2.2 Radioactivity Discharged from the Secondary Circuit via the Safety Valves during Specified Normal Operation and Incidents (design basis accidents)**

(1) In case of radioactivity discharged via the safety valves, the main steam radioactivity of the corresponding steam gen-

erator shall be determined without delay. This shall be based on measurement values as specified under Section 4.3.2.1 paras. 1 and 3 that are available at the time of beginning radioactivity discharge. Additionally, the transfer rate of primary coolant to the secondary side of the defective steam generator shall be determined from the primary coolant radioactivity.

(2) The position of the main steam safety valves (OPEN or CLOSED position) shall be determined and documented.

(3) The discharged radioactivity shall be determined for each main steam safety valve from the opening time, the respective main steam radioactivity and the flow rate per time unit specified for blow-down procedures.

**Note:**

The blow-down through the main steam safety valves will last only a few minutes due to the closed emergency feed-water system.

**4.3.2.3 Radioactivity Discharged During the Incident  
"Leakage of a Main Steam Line Downline from the Main Steam Isolation Valve"**

(1) A main steam line leakage will lead to a closure of the main steam isolation valve, therefore, only minor amounts of radioactivity are expected to be discharged. The discharged radioactivity shall be determined based on an analysis of the change in time of the main steam pressure, the closing behavior of the main steam isolation valve and the main steam radioactivity.

(2) To determine the main steam radioactivity, the transfer rate in the corresponding steam generator shall be calculated as specified under Section 4.3.2.2 para. 1.

**4.3.3 Boiling Water Reactors of the Product Line 72**

**4.3.3.1 Discharge of radioactivity from the Main Steam or Feed-water System Upon Opening the Turbine Building Roof Flaps due to a Leakage of the Main Steam or Feed-water Line Inside the Reactor Building or the Turbine Building**

(1) One sample shall be extracted in weekly intervals from each of the following systems:

- a) Main steam (P 1) or main condensate prior to condensate polishing (P 2),
- b) Reactor water (P 3),
- c) Feed water (P 4),
- d) Exhaust gas from the condenser air removal (P 5).

(2) The specific radioactivity of the samples extracted as specified under para. 1 shall be determined in the laboratory with a gamma ray spectrometer. The measured values or the detection limits achieved shall be documented. The list of nuclides to be surveilled shall be specified in the operating manual.

(3) The radioactivity concentration of the exhaust from the condenser air removal shall be continuously monitored by a gamma ray measurement point located before the delay line of the exhaust gas system. Based on the plant specific operating experience, an alarm threshold value shall be established for this measurement point such that, upon exceeding this value, a failure of the fuel assembly must be assumed. Whenever this alarm threshold value is exceeded and with every further doubling of this measurement value, additional sample extractions and measurements as specified under para. 1 and 2 shall be performed.

(4) All system fill levels, make-up volumes and internal pressures required to determine the mass of discharged media in case of a leak in the main steam or feed water pipe shall be continuously monitored. The necessary measurement

points and the procedures to determine the mass of discharged media shall be specified in an operating instruction.

(5) The proper interlock of the turbine building roof flaps shall be checked in regular intervals. The pressure difference between the outside atmosphere and the turbine building atmosphere shall be continuously monitored and automatically recorded.

(6) In case of a leak in the main steam or feed water pipe and a radioactivity discharge via the turbine building roof flaps, the following calculations shall be carried out:

- a) Calculation of the mass of the discharged medium on the basis of the system fill levels or of the size of the leak in connection with the internal system pressure specified under para. 4,
- b) Calculation of the amount of steam-air mixture released into the environment considering the opening cross section of the roof flaps and the pressure difference between the turbine building atmosphere and the environment,
- c) Calculation of the discharged radioactivity as specified under Appendix B.

**5 Design of Equipment for the Continuous Radioactivity Measurements and Sample Extractions**

**Note:**

Any equipment not covered in this section, e.g. isolation valves and water meters, shall, under consideration of the measurement tasks involved, be designed in accordance with the pertinent technical standards and guidelines.

**5.1 General Requirements**

**5.1.1 Design and Housing**

(1) Devices not located inside the laboratory shall be designed to meet the degree of protection Type IP 54 in accordance with DIN VDE 0470-1 (protection from foreign bodies and splash water).

(2) The measurement and sample extraction equipment shall be installed and located such that

- a) they are protected against any external influences that could impede proper operation,
- b) the nominal operating ranges in accordance with the corresponding equipment specifications are adhered to,
- c) inspection, servicing and repair are easily possible,
- d) they are at all times easily accessible during specified normal operation for authorized personnel.

**Note:**

A guideline with respect to the design of sample extraction systems is presented in DIN 25 477.

(3) Measuring containers and sample extraction containers shall be designed such that they can be easily removed and decontaminated.

**5.1.2 Protection Against Failure**

(1) Any continuously operating radioactivity measurement device shall be connected to an emergency power supply.

(2) Any continuously operating radioactivity measurement device shall be designed to be self-monitoring.

(3) All surveillance equipment shall be designed to restart automatically after a power interruption.

(4) When measurement devices are installed within or on a bypass then the bypass flow through shall be monitored. When measurement devices are installed on a system then the flow through of the measurement medium shall be monitored.

Section	Discharge Path	Determining the Specific Radioactivity	Determining the Amount of Discharged Media
4.3.1.1	Radioactivity discharged from the secondary system during specified normal operation due to leakage into the turbine building atmosphere and, via roof fans, into the environment.	Continuous surveillance of the gamma ray radioactivity of the blow-down water of each steam generator. Extraction of samples from the corresponding blowdown lines, and nuclide specific evaluation of the samples with respect to determining the specific radioactivity of the main steam.	Monitoring the demineralized water makeup fed into the secondary system and the removal of fluids from the secondary system.
4.3.1.2	Radioactivity discharged from the secondary system via the safety valves and blow-off control valves during specified normal operation and incidents (design basis accidents).	Extraction of samples from the steam generator blow-down pipes and, in case of incidents (design basis accidents), also from the primary coolant. Analysis of the samples in a gamma ray spectrometer with respect to determining the specific radioactivity of the main steam.	Determination of the discharged main steam volume by measuring the main steam pressure, establishing the positions of the safety and blow-off control valves, of the main steam isolation valves and of the isolation valves before the safety and blow-off control valves, and determination of the opening time and opening behavior of the valves.
4.3.1.3	Radioactivity discharged during the incident "Leakage of a main steam line downline from the main steam isolation valve".	Continuous monitoring of the gamma ray radioactivity in the blowdown water of each steam generator. Sample extraction from the corresponding steam generator blowdown line and from the primary coolant. Analysis of the samples in a gamma ray spectrometer with respect to determining the specific radioactivity of the main steam.	Determination of the released main steam volume by measuring the main steam pressure taking the closing behavior of the main steam isolation valves into account.

**Table 4-1:** Discharge paths and procedures for determining the discharged radioactivity in the case of KONVOI-type nuclear power plants

Section	Discharge Path	Determining the Specific Radioactivity	Determining the Amount of Discharged Media
4.3.2.1	Radioactivity discharged from the secondary system during specified normal operation due to leakage into the turbine building atmosphere and, via roof fans, into the environment.	Continuous surveillance of the noble gas radioactivity concentration in the exhaust from the condenser air removal. Extraction of main steam samples from each steam generator and the nuclide specific analysis of the samples with respect to determining the specific radioactivity of the main steam.	Monitoring the demineralized water makeup fed into the secondary system and the removal of fluids from the secondary system.
4.3.2.2	Radioactivity discharged from the secondary system via the safety valves during specified normal operation and incidents (design basis accidents).	Continuous surveillance of the noble gas radioactivity concentration in the exhaust from the condenser air removal. Extraction of main steam samples from each steam generator and the nuclide specific analysis of the samples with respect to determining the specific radioactivity of the main steam. Determination of the transfer rate in the defect steam generator.	Determination of the volumetric release of main steam from the position of the main steam safety valves, their opening times and the flow rate per time unit specified for blowdown procedures.
4.3.2.3	Radioactivity discharged during the incident "Leakage of a main steam line downline from the main steam isolation valve".	Determination of the transfer rate in the defect steam generator. Calculation of the specific radioactivity on the basis of the transfer rate.	Determination of the volumetric release of main steam by measuring the main steam pressure and taking the closing behavior of the main steam isolation valves into account.

**Table 4-2:** Discharge paths and procedures for determining the discharged radioactivity in the case of pressurized water reactors with straight-tube steam generators of the type built by BBR

Section	Discharge Path	Determining the Specific Radioactivity	Determining the Amount of Discharged Media
4.3.3.1	Discharge of radioactivity from the main steam or feedwater system upon opening the turbine building roof flaps on account of a leakage of the main steam or feedwater line inside the reactor building or the turbine building	<p>Extraction of samples from the main steam or main condensate, reactor water and feed water and analysis of the samples with respect to determining the specific radioactivity.</p> <p>Continuous surveillance of the radioactivity concentration of the exhaust from the condenser air removal with the objective of determining changes of the specific radioactivity of the main steam and feed water between subsequent sample extractions.</p>	<p>In case of a large leak combined with an opening of the turbine building roof flaps the steam volume released shall be calculated from the system fill levels, makeup volumes and system pressures.</p> <p>Determination of the steam volume discharged to the environment as specified under Appendix B.</p>

**Table 4-3:** Discharge paths and procedures for determining the discharged radioactivity in the case of nuclear power plants with boiling water reactors of the product line 72

### 5.1.3 Factorials for the Statistical Confidence Levels

(1) The factorial  $k_E$  for the statistical confidence level at the identification limit shall be equal to 1.645 in all measurement devices for continuous radioactivity measurements and equal to 3.0 for the detailed assessment measurements.

(2) The factorial  $k_N$  for the statistical confidence level at the detection limit shall be equal to  $k_N = k_E + 1.645$  for all measurements specified under para. 1.

### 5.1.4 Limit Values

(1) If it is necessary to readjust devices during operation then the required adjustment possibilities shall be provided for as built-in fixtures. All adjustment possibilities on the electronic devices of surveillance equipment shall be arranged and secured in such a way that maladjustment by non-authorized personnel can, to a great extent, be precluded.

(2) An undershooting of the lower limit value for signaling equipment failure and the exceeding of an upper limit value shall be optically displayed acoustically annunciated in the control room. Collective alarms are permitted, provided the measurement point originating the alarm is displayed in the control room or control room annex. The acoustic alarms may be cancelled individually or collectively before repair of the causes.

(3) The optical alarms in the control room signaling a failure or that an upper limit value was exceeded shall indicate the status condition of the alarm.

### 5.1.5 Measurement Value Display and Automatic Recording

(1) The measurement devices shall basically have only one display range. If more than one display range is necessary it is required,

- a) that, in the case of multiple linear display ranges, the measurement ranges overlap each other by at least 10% and the measurement-range factor is not be larger than 10,
- b) that, in the case of multiple logarithmic display ranges, the measurement ranges overlap each other by at least one decade.

(2) All measured value shall normally be displayed on the measurement device and shall be displayed and automatically recorded in the control room.

(3) The automatically recorded data on the chart paper shall be directly visible and well legible over a time period of 3 hours.

(4) The measurement values may be presented in the control room on CRT monitors, provided that one monitor is dedicated primarily to the presentation of these values, that a copy of the screen display can be printed out at all times and that the values are stored. A second monitor shall be available as a redundancy. The presentation on the monitor shall meet the requirements specified under para. 1.

### 5.1.6 Testability

The equipment shall be designed and manufactured such that the perfect functioning order of the individual devices can be verified within the framework of the initial tests specified under Section 6.2.3 and of the inservice inspections specified under Section 6.2.4. It shall be possible to also perform functional tests during full power operation of the nuclear power plant.

## 5.2 Measurement Devices for Continuous Radioactivity Measurements

### 5.2.1 Detection Limits

The detection limits of the measurement device for continuous radioactivity measurements shall not be higher than the lower limits of the minimum required measurement ranges specified below.

### 5.2.2 Gamma Ray Measurement Devices in the Steam Generator Blow-down Trains (KONVOI-type Plants)

(1) The measurement range of the gamma ray measurement devices for the continuous measurements in the steam generator blow-down line shall cover at least  $1 \times 10^5$  through  $2 \times 10^8$  Bq/m<sup>3</sup> (cesium 137 equivalency).

(2) Both, the exceeding of a radioactivity concentration of  $4 \times 10^5$  Bq/m<sup>3</sup> (cesium 137 equivalency) and the failure of a measurement device shall be automatically annunciated and recorded in the control room.

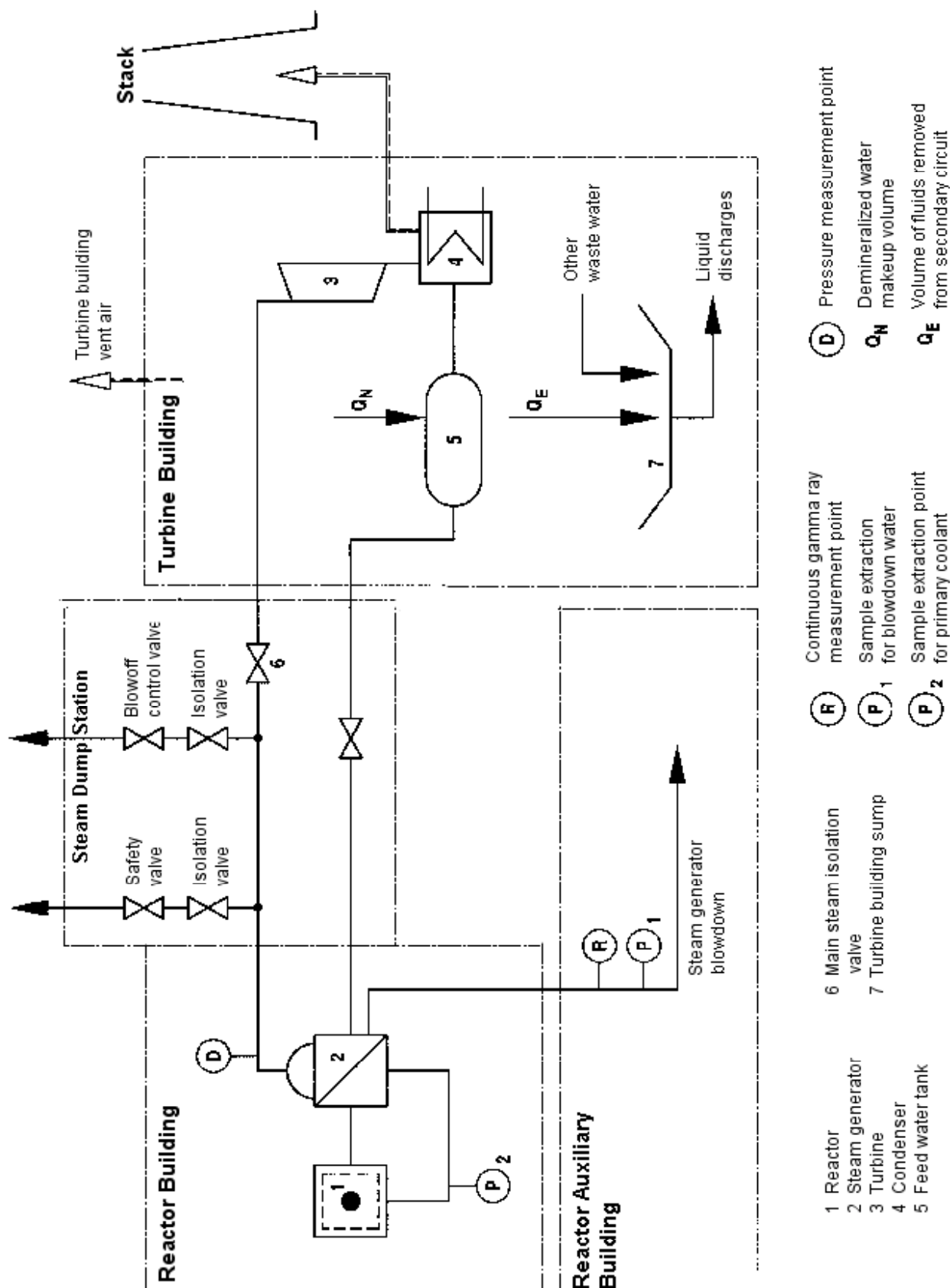
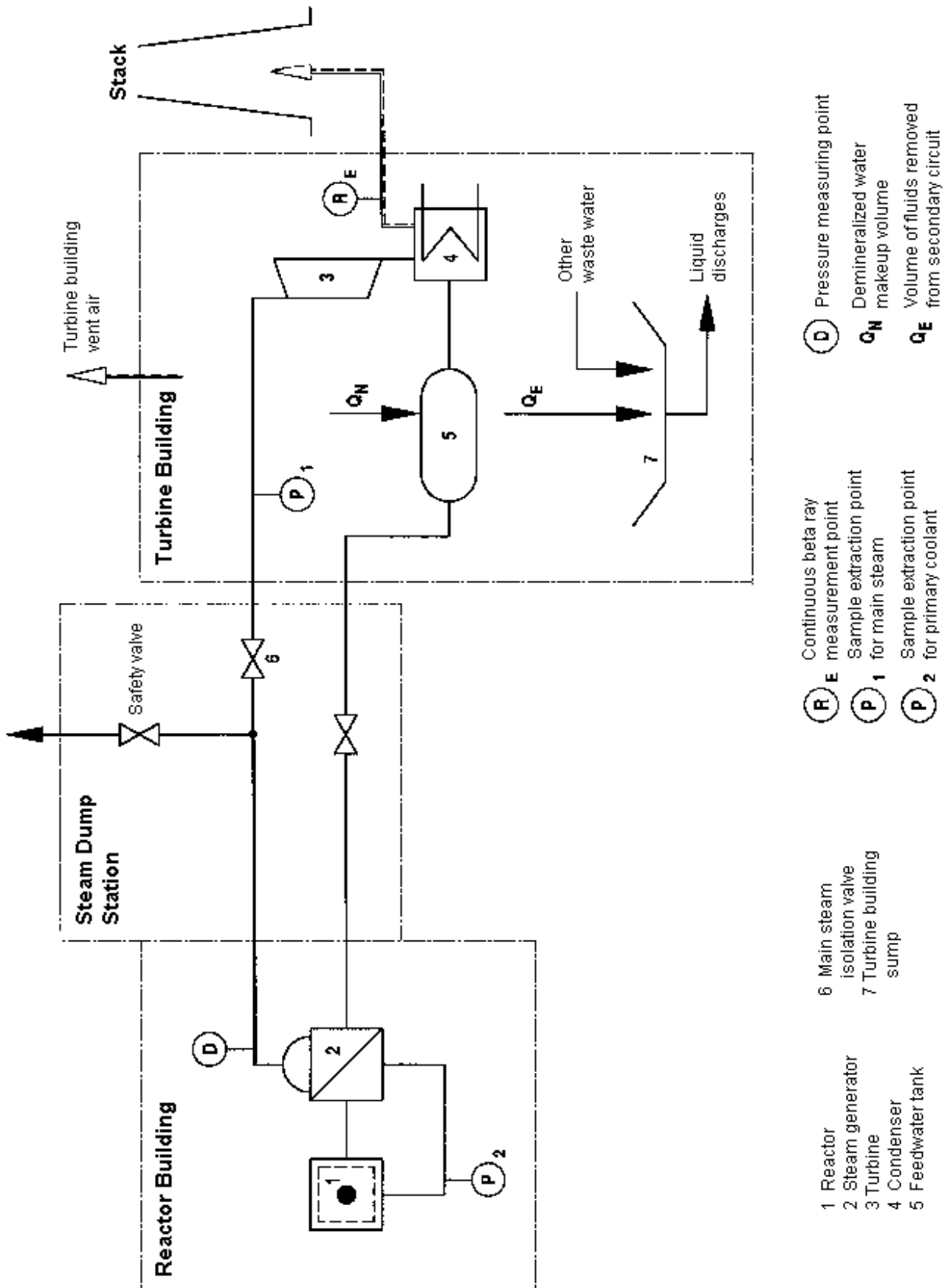
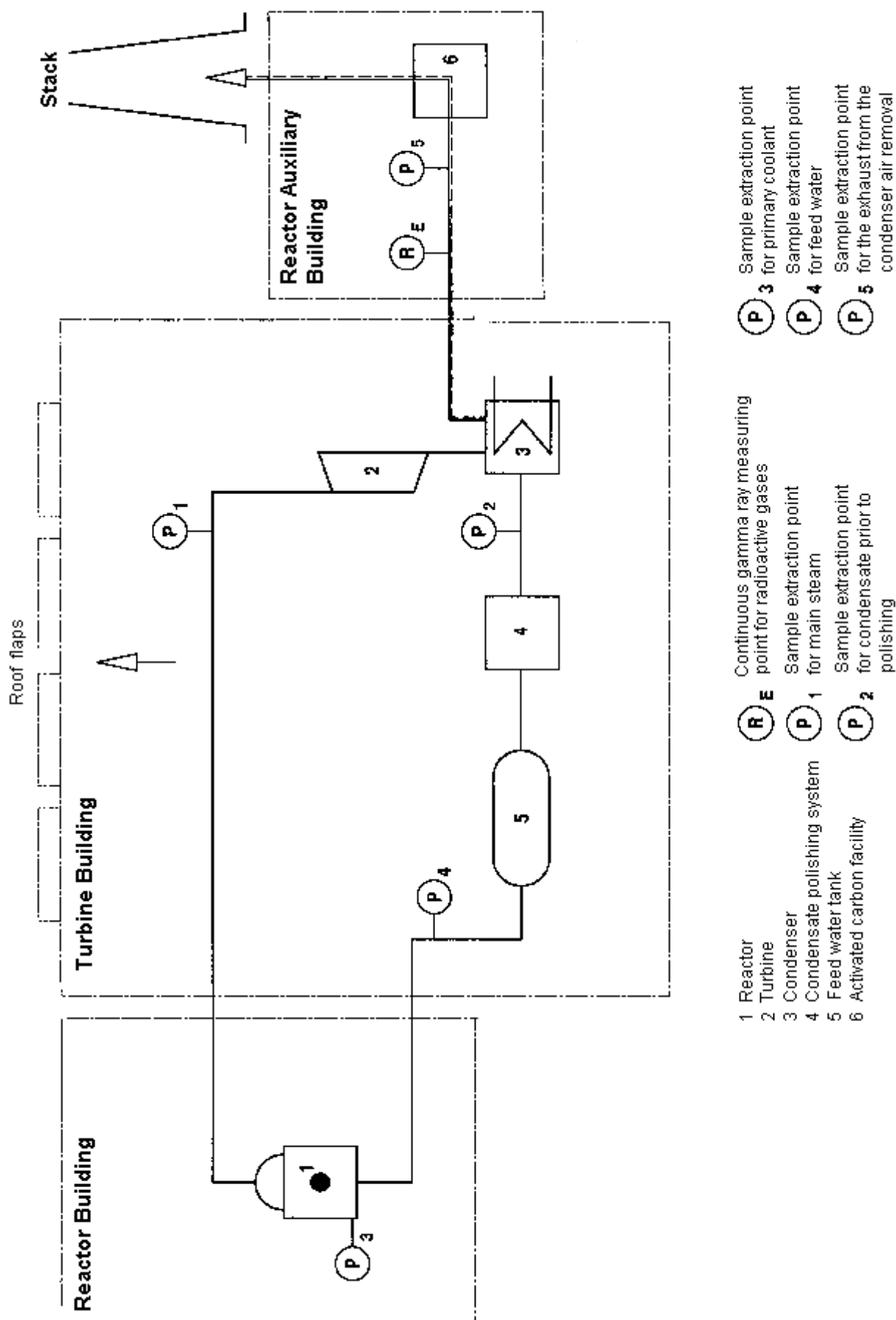


Figure 4-1: Schematic presentation of the surveilled discharge paths and the measurement and sample extraction points in KONVOI-type nuclear power plants



**Figure 4-2:** Schematic presentation of the surveilled discharge paths and the measurement and sample extraction points in nuclear power plants with pressurized water reactors with straight-tube steam generators of the type built by BBR



**Figure 4-3:** Schematic presentation of the surveilled discharge paths and the measurement and sample extraction points in nuclear power plants with boiling water reactors of product line 72

**5.2.3** Beta Ray Measurement Device for Monitoring the Noble Gas Radioactivity Concentration in the Exhaust from the Condenser Air Removal (PWR with Straight-tube Steam Generators of the Type Built by BBR)

(1) The measurement range of the beta ray measurement device for monitoring the noble gas radioactivity concentration in the exhaust from the condenser air removal shall extend at least from  $1 \times 10^5$  to  $5 \times 10^9$  Bq/m<sup>3</sup> (xenon 133 equivalency).

(2) Both, the exceeding of a noble gas radioactivity concentration of  $8 \times 10^3$  Bq/m<sup>3</sup> (xenon 133 equivalency) and the failure of a measurement device shall be automatically announced and recorded in the control room.

**5.2.4** Gamma Ray Measurement device for Monitoring the Exhaust from the Condenser Air Removal Before the Delay Line of the Exhaust Gas System (BWR of Product Line 72)

(1) The measurement range for the continuous surveillance of the radioactivity concentration in the exhaust from the condenser air removal before the delay line of the exhaust gas system shall cover a range at least from the 0.5fold through 10fold alarm threshold value specified under Section 4.3.3.1 para. 3.

(2) Both, the exceeding of the alarm threshold value as specified under Section 4.3.3.1 para. 3 and the failure of a measurement device shall be automatically indicated and recorded in the control room.

## 6 Maintenance of the Equipment for Continuous Radioactivity Measurements and Sample Extraction

### 6.1 Servicing and Repair

#### 6.1.1 Execution

Servicing and repair of the surveillance equipment shall be carried out by qualified personnel and in accordance with the corresponding operating and maintenance instructions.

#### 6.1.2 Documentation

All servicing and repair tasks shall be documented. The documentation shall contain the following information:

- unambiguous identification of the surveillance equipment involved,
- type of the servicing or repair performed,
- type and number of parts exchanged,
- reasons for exchanging the parts,
- regarding exchanged parts: date and identification of the test certifications and of the test certificates required under this safety standard,
- information on the duration of the failure,
- date of performing servicing or repair,
- name and signature of the qualified personnel.

### 6.2 Tests and Inspections

The surveillance equipment shall be subjected to the following test and inspections:

- before their initial deployment of the device type in a nuclear power plant:
  - type test or presenting proof of service life,
  - calibration,

- before their initial deployment of each individual device in a nuclear power plant:

- verification of suitability,
- verification of the calibration with calibration sources,
- factory test,
- commissioning tests,

- in the course of their deployment in the nuclear power plant:

- regular inservice inspections,
- tests and inspections after servicing and repair.

#### 6.2.1 Test Schedule

Type, extent and intervals of the tests and examinations shall be specified in the documents in accordance with KTA 1202.

#### 6.2.2 Test Certification

All tests and Inspections shall be documented by test certificates in accordance with KTA 1202. The test certificates shall be kept in storage.

#### 6.2.3 Tests prior to Initial Deployment in a Nuclear Power Plant

##### 6.2.3.1 Certification of Suitability

(1) Before their initial deployment in a nuclear power plant it shall be demonstrated that the surveillance equipment fulfill their purpose and meet the specified requirements. The long term behavior of the surveillance equipment under cyclic variation within the specified nominal operating ranges shall also be checked within the framework of this test.

(2) The test shall be performed or evaluated by authorized experts. The suitability may be proven in part or entirely by presenting proof of service life under comparable operating conditions.

**Note:**

This subject matter is being treated in safety standard KTA 1505 "Suitability Certification of Radiation Measurement device" (in preparation).

##### 6.2.3.2 Calibration and Verifying the Calibration

(1) The measurement device specified under Section 5.2 shall be calibrated prior to their initial deployment in a nuclear power plant. This calibration may also be performed on a type identical device.

(2) The measurement device in the steam generator blow-down trains (KONVOI-type plants) shall be calibrated with cesium 137. The discrimination for gamma rays shall be known in the energy range from 100 through 1700 keV.

(3) The measurement devices for monitoring the noble gas radioactive concentration in the exhaust from the condenser air removal (PWR with straight-tube steam generators of the type built by BBR) shall be calibrated with xenon 133. The discrimination for beta rays shall be known in the energy range from 150 through 2500 keV.

(4) The calibration factor of the measurement device for monitoring the exhaust from the condenser air removal before the delay line of the gas exhaust system shall be determined analytically. The discrimination for gamma rays shall be known in the energy range from 80 keV through 3 MeV.

(5) In course of the initial calibration of the measurement points specified under paras. 2 and 3, a set of calibration sources shall be specified with which one display value in one of the two lower decades and one display value in one of the two upper decades can be verified.

(6) Following initial calibration of the measurement devices of the measurement device a calibration source in definite and reproducible geometry shall be used to determine a value which can later be used to verify the calibration and to connect type identical equipment.

### 6.2.3.3 Factory Test

(1) The factory test shall verify the correct manufacture and perfect functioning of the measurement device.

(2) In case the measurement device is comprised of components from different manufacturers then the correct manufacture and perfect functioning of these components shall be verified by factory tests at the individual manufacturer.

(3) The factory test shall be performed on individual pieces (production test) and shall comprise:

- a) visual inspection,
- b) test of the dependence of output value on the specified voltage fluctuation,
- c) test of the response characteristic with a pulse or current generator and at least one test value for each decade of the measurement display range.

(3) The factory test shall be performed by plant experts and in justified cases in the presence of experts authorized by the competent authority.

### 6.2.3.4 Commissioning Test

(1) The post-installation commissioning test shall demonstrate the proper design and function of the surveillance equipment. The following items shall be tested:

- a) design of the surveillance equipment,
- b) installation of the surveillance equipment,
- c) display (with a pulse or current generator and at least one test value for each decade of the measurement display range),
- d) verification of the calibration (with a calibration source),
- e) limit value settings,
- f) connection to the emergency power system,
- g) automatic restart after interruption of the power supply
- h) monitoring of the volumetric flows,
- i) measured value processing (alarms),
- k) supply of operating media.

(2) The commissioning test shall be carried out by the plant operator and, to an extent specified by the proper authority, by authorized experts or in their presence.

## 6.2.4 Inservice Inspection

### 6.2.4.1 General Requirements

(1) The monitoring equipment shall be subjected to inservice inspections during operation of the nuclear power plant.

**Note:**

These tests comprise regular inservice inspections and the tests and inspections after repairs.

(2) A coordination with the proper authority shall be established in the case of functional tests that require the removal of safety related interlocks.

(3) The tests and inspections shall be possible without any change to the circuitry (e.g. soldering).

(4) The test and inspections shall be performed on the basis of test and inspection documents in accordance with KTA 1202.

### 6.2.4.2 Regular Inservice Inspections

(1) Regular inservice inspections shall be performed to assure a fault-free functioning of the surveillance equipment. The tests and the test frequencies shall be as specified in **Table 6-1**.

(2) Verification of the calibration as per running number 1 b) of **Table 6-1** shall be carried out with the calibration source and defined geometry as specified under Section 6.2.2.2 para. 6 during initial calibration of the measurement device.

(3) The tests and inspections shall be performed by the plant operator and by experts authorized by the proper authority.

### 6.2.4.3 Tests and Inspections after Repairs

After having performed a repair task the fault-free functioning shall be demonstrated by a commissioning test as specified under Section 6.2.3.4 with an extent corresponding to the repair task.

## 6.3 Removal of Defects

Any defects identified by the tests shall be removed without delay and within the time periods specified in the test records.

## 7 Documentation of Measurement Results

### 7.1 Flow Chart

(1) The installed sample extraction and surveillance equipment required for monitoring the discharged gaseous and aerosol-bound radioactive substances shall be clearly presented in a flow chart. The different types of sample extractions and surveillance measurements shall be identified by different symbols.

(2) The required objectives and means of measurement of each individual sample extraction and surveillance equipment shall be described ( e.g. in tabular form) with the descriptions correlated to the flow chart. With regard to sample extraction equipment, the objective, type, location and frequency as well as required measurements shall be specified. With regard to surveillance equipment, the measurement objectives and technical measurement requirements, especially, the type of measurement, the measurement arrangement including radiation shielding, calibration, measurement display ranges, detection limits and measurement tolerances shall be specified. With regard to the measurement laboratory, the measurement tasks and technical measurement requirements shall, likewise, be described.

### 7.2 Extent of Documentation

(1) In case it is required in accordance with Section 3 to determine the discharged radioactivity during specified normal operation then a report form as in **Figure 7-1** should be used. These forms shall be forwarded to the proper authority in quarter yearly intervals.

(2) The column "Discharged radioactivity" shall list only those values that are derived from measurement values exceeding the detection limit.

(3) The documentation of the radioactivity discharged during incidents (design basis accidents) shall be structured such that

- a) the discharge path is unambiguously shown,
- b) the chronological development of the radioactivity discharged due to the incident is described,

- c) the sample extraction,  
 ca) prior to the occurrence of,  
 cb) during  
 cc) after

the incident (design basis accident) including point in time, duration and sample extraction type (continuous, discon-

tinuous) and other boundary conditions during sample extraction are listed,

- d) the methods for determining the specific radioactivity and the discharged amounts are described in detail and the determination of the specific radioactivity and discharged amounts for the past incidents (design basis accidents) can be traced.

1	2	3	4	5	
Running No.	Test Object	Testing Procedure	Test Frequency <sup>1)</sup>		
			by the plant operator	by the authorised expert <sup>2)</sup>	
1	Measurement device for continuous radioactive measurements	a) visual inspection	during inspection rounds	annually	
		b) verification of calibration with a radiation source	quarter annually		
		c) in case of counter tubes: verification of the plateau	—		
2	Inspection and servicing records	Visual inspection	—		
3	Electronic modules	Insertion of standard signals into the transmitters (at least one value in each decade of the measurement range) <sup>3)</sup> Comparison of all displayed and automatically recorded display values	annually		
4	Alarm signals	Operational availability: visual inspection	during inspection rounds		
		Failure alarm: by interrupting the voltage supply or interrupting the signal connection between measurement transducer and detector	quarter annually		
		Hazard alarm: with calibration source or electronically	quarter annually		
5	Flow monitoring and supply of operating media without automatic function control	Visual inspection	during inspection rounds		
	with automatic function control	Comparison of actual and required values	quarter annually		

1) If a test is performed as specified under column 5 then a test specified for the same time period under column 4 need not be performed.  
 2) These requirements pertain to the expert authorised by the proper authority.  
 3) The testing procedure of inserting at least one standard signal into the transmitters for each decade of the measurement range is not required in the case of digital measurement devices, provided the software program is certified and itself of a self-monitoring nature. In this case it is sufficient to insert a single signal in the uppermost decade of the measurement range, provided the pre-processing electronics do not require any switching procedures over the entire measurement range. This too is not required if the verification of the calibration is carried out with one measurement value in the uppermost decade of the measurement range.

**Table 6-1:** Regular inservice inspections

Report Form (in accordance with KTA 1503.3)				
Nuclear Power Plant:		Time Period of Discharge:		
Annual Quarter:				
Discharge Path:				
Nuclide <sup>1)</sup>	Detection Limit of Evaluated Samples (Bq/Mg)		Discharged radioactivity (Bq)	Remarks
	min.	max.		
Cr 51				
Mn 54				
Co 57				
Co 58				
Fe 59				
Co 60				
Zn 65				
Zr 95				
Nb 95				
Ru 103				
Ru 106				
Ag 110m				
Te 123m				
Sb 124				
Sb 125				
I 131				
Cs 134				
Cs 137				
Ba 140				
La 140				
Ce 141				
Ce 144				
2)				
Sr 89				
Sr 90				
Sum total				
Tritium				

1) In case of the detailed assessment specified under Section 4.3.1.1 para. 2 only one value of radioactivity, the one for iodine 131, has to be documented.

2) If necessary, additional gamma ray emitters

**Figure 7-1:** Example for a report form on radioactive substances discharged other than with the stack air during specified normal operation

## Appendix A

### Procedures for the Analysis of PWR

#### A 1 Procedures for the Analysis of KONVOI-type Plants

##### A 1.1 Calculation of Radioactive Substances Discharged with the Turbine Building Vent Air (Roof Ventilators)

The radioactivity, A, discharged with the turbine building vent air in a particular time period on account of a defective steam generator shall be calculated from the equation

$$A = A_A \cdot \frac{Q_N - Q_E}{n} \cdot f \quad (\text{A-1})$$

Nomenclature:

- $A_A$  specific radioactivity or radioactivity concentration of a nuclide or a nuclide group in the steam generator blow-down water, determined as specified under Section 4.3.1.1
- $Q_N$  amount of demineralized water makeup into the secondary circuit during the particular time interval (determined as time-weighted average of, e.g., weekly measurements)
- $Q_E$  amount of fluids monitored as having been removed from the secondary circuit during the particular time period
- $n$  number of steam generators
- $f$  transfer factor of water/steam phase. A value of 0.001 shall be assumed for all fission and corrosion products (except for noble gases) unless another certified value can be provided.

In the case that a threshold value as specified under Section 4.3.1.1 para. 2 through 4 was exceeded in more than one steam generator, then the overall radioactivity discharge shall be determined as the sum of the corresponding steam generators.

#### A 2 Procedures for the Analysis of PWR with Straight-tube Steam Generators of the Type Built by BBR

##### A 2.1 Calculation of Radioactive Substances Discharged with the Turbine Building Vent Air (Roof Ventilators)

The radioactivity, A, discharged with the turbine building vent air in a particular time period on account of a defective steam generator shall be calculated from the equation

$$A = A_{FD} \cdot (Q_N - Q_E) \quad (\text{A-2})$$

Nomenclature:

- $A_{FD}$  specific radioactivity or radioactivity concentration of a nuclide or a nuclide group in the main steam of all steam generators, determined as specified under Section 4.3.2.1
- $Q_N$  amount of demineralized water makeup into the secondary circuit during the particular time period (determined as time-weighted average of, e.g., weekly measurements)
- $Q_E$  amount of fluids monitored as having been removed from the secondary circuit during the particular time period.

## Appendix B

### Procedure for the Analysis of BWR of Product Line 72

#### Calculation of the Radioactivity of the Steam or Steam-water Mixture Discharged into the Environment upon Opening the Turbine Building Roof Flaps on Account of a Leakage in the Main Steam or Feedwater Line Inside the Reactor Building or the Turbine Building

**Note:**

In order to simplify application and verification of the calculation procedures, a consistent set of dimensions is specified for the different measured quantities. The addendum "e.g." should indicate that the corresponding measured quantities may also be measured in other dimensions.

(1) The radioactivity,  $A$ , e.g. in Bq, discharged through the turbine building roof flaps on account of a leakage in a main steam or a feedwater line shall be calculated from the equation

$$A = \frac{A_A \cdot Q_A \cdot F_A \cdot \ddot{U}_A}{V_A} \cdot V_U \quad (\text{B-1})$$

**Nomenclature:**

$A_A$  specific radioactivity in the main steam or feedwater, e.g. in Bq/kg

(the specific radioactivity of radioactive noble gases in the feedwater shall be calculated as follows:

$$A_{AE} = \frac{A_{KE} \cdot \dot{V}_G}{\dot{R}_{FD}} \quad (\text{B-2})$$

where the nomenclature is as follows:

$A_{KE}$  activity concentration of radioactive noble gases in the exhaust of the condenser air removal (determined as specified under Section 4.3.3.1 paras. 1 and 2), e.g. in Bq/m<sup>3</sup>

$\dot{V}_G$  volumetric flow to the exhaust gas facility, e.g. in m<sup>3</sup>/h

$\dot{R}_{FD}$  rate of main steam creation, e.g. in kg/h)

$Q_A$  mass of the medium released into the turbine building (determined as specified under Section 4.3.3.1 para. 4), e.g. in kg

$V_A$  free air volume in the turbine building, e.g. in m<sup>3</sup>

$F_A$  portion of leakage released as steam

$F_A = 1$  in the case "Leakage in the main steam line"

$F_A = 0.2$  in the case "Leakage in the feed-water line"

$\ddot{U}_A$  transfer factor of radionuclides into the released steam

$\ddot{U}_A = 1$  in the case "Leakage in the main steam line"

$\ddot{U}_A = 0.2$  in the case "Leakage in the feed-water line"

$V_U$  volume of steam and air released to the environment via the turbine building roof flaps (calculated from equation B-3), e.g. in m<sup>3</sup>

(2) The volume of steam and air released to the environment upon opening the turbine building roof flaps,  $V_U$ , shall be calculated from the equation

$$V_U = n \cdot F_e \cdot \sum_{i=1}^m \sqrt{\frac{2 \cdot \Delta p_i(t)}{(1 + \xi) \cdot \rho}} \cdot \Delta t_i \quad (\text{B-3})$$

**Nomenclature:**

$n$  number of unlocked roof flaps  
(shall be determined onsite by visual inspection after the incident)

$F_e$  aperture of the roof flaps, e.g. in m<sup>2</sup>  
(it shall be conservatively estimated that the roof flaps are 100% open for the entire opening time)

$\xi$  loss coefficient for the individual roof flap  
(shall be conservatively estimated as equal to 0)

$\rho$  density of the steam-air mixture  
(approximately 1 kg/m<sup>3</sup>)

$\Delta p(t)$  pressure difference between the turbine building atmosphere and the environment (as a function of time), e.g. kg/(m · s<sup>2</sup>)

$\sum_{i=1}^m \Delta t_i$  opening time of the roof flaps, e.g. in s

(The values for  $\Delta p(t)$  and  $\Delta t_i$  shall be determined from the chart recordings of the pressure difference between the turbine building atmosphere and the environment.)

## Appendix C

### Nuclide Specific Evaluation during Specified Normal Operation of Samples from the Blowdown Water (KONVOI-type Plants) or Main Steam (PWR with Straight-tube Steam Generators of the Type Built by BBR)

#### C 1 Gamma Ray Emitters

In order to determine the discharged radioactivity the samples shall be analyzed in a gamma ray spectrometer. The nuclides listed in **Table C-1** shall be taken into consideration. The detection limit of the measurement arrangement for determining the radioactivity concentrations in demineralized-water samples shall not be higher than  $1 \times 10^3$  Bq/m<sup>3</sup> for cobalt 60. The measurement time for the sample shall be at least as long as it takes to achieve the required detection limit for cobalt 60. A check shall be carried out within the framework of the gamma ray spectrometer analyses whether additional radionuclides to the ones listed in **Table C-1** are present due to plant specific conditions. If other nuclides are detected they shall additionally be considered in determining the discharged radioactivity.

**Note:**

The detection limits specified in this appendix with respect to meeting the described measurement tasks are not required for the samples from the blowdown water. As specified in Sections 4.3.1.1 and 4.3.1.2, the determination of gamma ray emitters, of radioactive strontium and of tritium is required when the radioactivity concentration at one of the continuous integral gamma ray measurement devices for monitoring the radioactivity in the blowdown water exceeds  $4 \times 10^5$  Bq/m<sup>3</sup>. However, this case is already covered by KTA 1504 "Monitoring the Discharge of Radioactive Substances with Water", which requires that the above mentioned nuclides are determined in the blowdown water with the herein specified detection limits as part of the surveillance of the discharge of radioactive substances in the turbine building waste water; in other words, the determination of the data as specified in this appendix is already carried out on account of other requirements.

Nuclide			
Chromium	51	Silver	110m
Manganese	54	Tellurium	123m
Cobalt	57	Antimony	124
Cobalt	58	Antimony	125
Cobalt	60	Iodine	131
Iron	59	Cesium	134
Zinc	65	Cesium	137
Zirconium	95	Barium	140
Niobium	95	Lanthanum	140
Ruthenium	103	Cerium	141
Ruthenium	106	Cerium	144

**Table C-1:** Nuclides to be considered in determining the discharged radioactivity: Gamma ray emitters

#### C 2 Radioactive Strontium

The samples extracted in the discharge time period shall be evaluated with respect to their content of strontium 89 and strontium 90. It is permissible to carry this out on a volume-proportional mixture sample. The detection limit of the procedure for determining the radioactivity concentration shall not be higher than  $5 \times 10^2$  Bq/m<sup>3</sup>.

#### C 3 Tritium

The samples extracted in the discharge time period shall be evaluated with respect to their content of tritium. It is permissible to carry this out on a volume-proportional mixture sample. The detection limit of the procedure for determining the radioactivity concentrations shall not be larger than  $4 \times 10^4$  Bq/m<sup>3</sup>.

## Appendix D

### Regulations Referred to in this Safety Standard

(Regulations referred to in this safety standard are only valid in the version cited below. Regulations which are referred to within these regulations are valid only in the version that was valid when the later regulations were established or issued.)

Atomic Energy Act		Act on the peaceful utilization of atomic energy and the protection against its hazards (Atomic Energy Act) of December 23, 1959 (BGBl. I, p. 814) as amended on July 15, 1985 (BGBl. I, p. 1565), most recently changed by law on April 6, 1998 (BGBl. I, p. 694)
StrlSchV		Ordinance on the protection from damage by ionizing radiation (Radiological Protection Ordinance - StrlSchV) of October 13, 1976 (BGBl. I, p. 2905, 1977 p. 184, 269) in the version published on June 30, 1989 (BGBl. I, p. 1321, corrected p. 1926) most recently changed by ordinance on August 18, 1997 (BGBl. I, p. 2113)
KTA 1202	(6/84)	Requirements for the testing manual
KTA 1503.1	(6/93)	Monitoring the discharge of gaseous and aerosolbound radioactive substances; Part 1: Monitoring the stack discharge of radioactive substances during specified normal operation
DIN VDE 0470-1	(11/92)	Degrees of protection provided by enclosures (IP Code); (IEC 529 (1989), 2. Edition); German version EN 60529:1991