Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 1503.3 (2017-11)

Monitoring the Discharge of Radioactive Gases and Airborne Radioactive Particulates
Part 3: Monitoring the Non-Stack Discharge of Radioactive Matter

(Überwachung der Ableitung gasförmiger und an Schwebstoffen gebundener radioaktiver Stoffe
Teil 3: Überwachung der nicht mit der Kaminfortluft abgeleiteten radioaktiven Stoffe)

The previous version of this safety standard were issued in 1999-06 and 2013-11

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KTA SAFETY STANDARD

November 2017	Monitoring the Discharge of Radioactive Gases and Airborne Radioactive Particulates Part 3: Monitoring the Non-Stack Discharge of Radioactive Matter	KTA 1503.3

Previous versions of the present safety standard: 1999-06 (BAnz No. 243b of December 23, 1999)
2013-11 (BAnz. of January 17, 2014)

Contents

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, 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 the present safety standard.

Basic Principles

(1) The safety standards of the Nuclear Safety Standards Commission (KTA) have the objective to specify safety-related requirements, compliance of which provides the necessary precautions in accordance with the state of the art in science and technology against damage arising from the construction and operation of the facility (Sec. 7 para. 2 subpara. 3 Atomic Energy Act - AtG) in order to achieve the fundamental safety functions specified in the Atomic Energy Act and the Radiological Protection Ordinance (StrlSchV) and further detailed in the Safety Requirements for Nuclear Power Plants as well as in the Interpretations of the Safety Requirements for Nuclear Power Plants.

(2) The stationary and the mobile radiation protection instrumentation serves, among others, in protecting the persons inside and outside of the facility from ionizing radiation, and in ascertaining the specified normal functioning order of the equipment

a) for keeping solid, liquid and gaseous radioactive substances within the provided enclosures,

b) for handling and the controlled conducting of the radioactive substances within the facility, and

c) for monitoring the discharge of radioactive substances.

Concrete safety related requirements with regard to this instrumentation are specified in the safety standards of the series KTA 1500.

(3) The parts of safety standard KTA 1503 comprise the requirements regarding the technical equipment and additional organizational measures considered necessary with respect to monitoring emissions of gaseous radioactive substances and of radioactive substances bound to aerosols. These parts are, specifically:

Part 1: Monitoring the discharge of radioactive substances with the vent stack exhaust air during specified normal operation,

Part 2: Monitoring the discharge of radioactive substances with the vent stack exhaust air during design-basis accidents,

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

(4) Monitoring the discharge of radioactive substances contributes to fulfilling the requirements specified under Secs. 6, 47 and 48 StrlSchV, in accordance of which it is required

a) that any radiation exposure or contamination of man and environment must be minimized even to levels below the respective limit values specified in StrlSchV by taking into consideration the state of the art and by taking into account all circumstances of individual situations (Sec. 6, para. (2) StrlSchV),

b) that no radioactive substances are discharged uncontrolled into the environment (Sec. 47, para. (1), 2nd sentence, StrlSchV), and

c) that discharges are monitored and reported to the proper authority at least once a year specifying their kind and activity (Sec. 48, para. (1) StrlSchV).

The monitoring equipment must fulfill the requirements under Sec. 67 StrlSchV.

(5) In accordance with Sec. 51 StrlSchV, it is required that, in the event of design-basis accidents and emergencies, all necessary measures are initiated without delay in order to minimize the danger to life, health and material goods. In accordance with Sec. 53 StrlSchV, it is required that, in preparation for an effective damage control and mitigation, the necessary auxiliary aids are kept in readiness. Basis for the decision to take these measures as well as for their type, extent and duration is, among others, the monitoring of the discharged activity performed with the radiation protection instrumentation.

(6) The present safety standard (KTA 1503.3) specifies the measures to be taken with respect to the equipment required for fulfilling the tasks under paras. (4) and (5) insofar as a discharge of radioactive substances occurs other than with the vent stack exhaust air. Thus, the present safety standard contributes to meeting the requirements in accordance with Sec. 6 para. (2), Sec. 47 para. (1), 2nd sentence, Sec. 48 para. (1), Sec. 51, Sec. 58, and Sec. 67 para. (1) StrlSchV.

(7) By monitoring the activity of the discharged radioactive substances (emission monitoring) in connection, on the one hand, with dispersion calculations and, on the other, by direct measurements in the environment (immission monitoring), factual statements can be made about the radiological effects on the environment of the facility. Whereas, during specified normal operation, the essential data is acquired by emission monitoring – here the data derived from immission monitoring lie within the margin of fluctuation of the natural radiation –, it is the immission monitoring that becomes ever more important during design-basis accidents because it enables identifying the radiological effects.

(8) The equipment required for fulfilling these tasks are subdivided as

a) stationary measuring and sampling equipment, and

b) mobile measurement equipment deployed for determining the activity of cumulative samples.

1 Scope

(1) The present safety standard shall apply to the equipment for monitoring the non-stack discharge of gaseous radioactive substances and radioactive substances bound to aerosols from nuclear power plants with light water reactors during specified normal operation and during design-basis accidents.

(2) In order to achieve the monitoring objective, 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; the technical means and administrative measures to be applied in power plants with pressurized water reactors (PWR) and boiling water reactors (BWR) as specified under Sections 4, 5, 6 and 7 are 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 for determining the activity discharge in case of accidents that go beyond the design-basis accidents if this discharge would occur along the paths described in the present safety standard. If the monitoring of discharged radioactive substances cannot anymore be ensured with these technical means and administrative measures, the backup measurements carried out within the framework of environmental monitoring (immission monitoring) are used.

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

2 Definitions

(1) Discharge of radioactive substances

Discharge of radioactive substances is the intentional release of liquid radioactive substances bound to aerosols or of gaseous radioactive substances from the facility along paths provided for this purpose.

(2) Specified normal operation

Specified normal operation encompasses

a) Operating processes for which the plant, assuming the able function of all systems (fault free condition), is intended and suited (normal operation);

b) Operating processes which occur in the event of a plant component or system malfunction (fault condition) as far as safety related reasons do not oppose continued operation (abnormal operation);

c) Maintenance procedures (inspection, servicing, repair).

(3) Detailed assessment of radioactive substances

The detailed assessment is a special form of monitoring consisting of identifying, and determining the activity of, the radionuclides or radionuclide groups discharged over a given time span. For detailed assessment measured value is used. The measurement uncertainty is indicated separately.

(4) Decision threshold

A decision threshold is a calculated value of a measurement parameter (e.g., activity, activity concentration, specific activity) for the comparison with a measured value in order to decide whether this measurement parameter has contributed to the measurement or has had a zero effect.

Note:

(1) Decision thresholds are determined in accordance with DIN ISO 11929.

(2) Application examples are given in report KTA-GS 82.

(5) Release of radioactive substances

Release of radioactive substances is the escape or leakage of radioactive substances from the intended enclosures into the facility or the environment.

(6) Calibration of a measurement equipment for radiation monitoring

The calibration of the measurement equipment for radiation monitoring is the determination of the relationship between the value defined by specific norms (e.g., activity of the calibration source) and the displayed value (e.g., count rate) of the measurement parameter.

(7) Mixture sample

A mixture sample is a mixture of individual samples or cumulative samples, or of parts of these samples, that were extracted within a specified time span.

(8) Detection limit

The detection limit is a calculated value for a measurement parameter (e.g., activity, activity concentration, specific activity) meant to be compared to a predetermined reference value to help decide whether or not the measurement procedure is suitable for a particular measurement task.

Note:

(1) Detection limits are determined in accordance with DIN ISO 11929.

(2) Application examples are given in report KTA-GS 82.

(9) Design-basis accident

A design-basis accident is a chain of events upon the occurrence of which the plant operation or the work task cannot be continued for safety-related reasons and which, with respect to the plant operation, was considered in the plant design or for which, with respect to the work task, precautionary protective measures shall be provided.

(10) Monitoring

Monitoring is a collective term for the various types of a controlled determination of physical parameters; monitoring includes comparing the results with previously specified values.

Notes:

(1) Monitoring is performed, e.g.,

a) by continuous measurements, or

b) by analyses of samples (e.g. in a laboratory), or

c) by a combination of selective measurement values,

and is always carried out in conjunction with a comparison of the results with previously specified values of the physical parameters (e.g., licensed limit values, operational values).

(2) To verify compliance with licensed limit values the upper limit of the confidence interval is applied.

(11) Confidence interval

The confidence interval is the interval that contains the true value of the measurement parameter with a specified probability.

Note:

The limits of the confidence interval are determined in accordance with DIN ISO 11929.

3 Determining the Discharge Paths, and Monitoring Concept

3.1 Criteria for Determining the Discharge Paths to be Monitored

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

(2) The possibility of a activity discharge does not need to be considered in those cases where the system under consideration is separated from the environment by at least two physical barriers or by one physical barrier and one pressure barrier, provided however, that even under design-basis accident conditions one physical barrier will remain intact.

(3) A detailed assessment of the activity discharged during specified normal operation is not required for those discharge paths for which it is determined that the activity discharge into the environment fulfills the criterion of insignificance. This criterion is considered fulfilled if the activity that can be discharged during specified normal operation over the time span of one week is less than one tenth of the activity resulting from the product of the vent stack exhaust air volume discharged in this time span and the detection limit specified in accordance with safety standard KTA 1503.1 Table 3-4 for the detailed assessment of the respective group of radionuclides.

3.2 Monitoring Concept

(1) For each of the paths that require monitoring in accordance with the criteria specified under Section 3.1, the specific activity or activity concentration in the medium of the respective system and the amount of the discharged medium shall be determined; the discharged activity shall then be calculated from these values.

(2) A procedure for determining the amount of the medium that could escape in case of a release is not required in those cases where the activity concentration in the associated system is continuously monitored and a release from the system is prevented by isolating this system whenever a threshold value of the activity concentration is exceeded. This threshold value shall be set such that a possible discharge of activity meets the criterion specified under Section 3.1 para. (3).

(3) Requirements regarding the monitoring concept for mobile measurement equipment are specified in safety standards KTA 1503.1 and KTA 1503.2.

4 Technical Equipment, Administrative Measures and Monitoring Procedures

4.1 Discharge Paths

In accordance with the criteria specified under Section 3, the multiple discharge paths (PWR) or the single discharge path (BWR) are listen in the first columns of **Table 4-1** and **Table 4-2**, respectively, for pressurized water reactors and for boiling water reactors.

Note:

The review carried out during preparation of the present safety standard with respect to which of the possible discharge paths would require emission monitoring according to the criteria specified under Section 3.1 has led to the result that the discharge paths listed in **Tables 4-1**and **4-2** for the analyzed PWR and BWR power plants are definitive with regard to implementing Sec. 47 para. (1), 2nd sentence, StrlSchV within the scope of the present safety standard (KTA 1503.3).

4.2 Monitoring Measures

The monitoring measures required for pressurized water reactors are presented in **Table 4-1** and for boiling water reactors in **Table 4-2**, in both cases, to be carried out with regard to the specific activity in the respective columns 2 and the media amounts in columns 3. The correspondingly required monitoring equipment for the two reactor types are shown in **Figures 4-1** and **4-2**, respectively.

4.3 Monitoring Procedures

4.3.1 Pressurized water reactors (PWR)

4.3.1.1 Discharged activity from the secondary system during specified normal operation caused by leakage into the turbine building atmosphere and, further, via roof vents, into the environment

(1) The activity of the blowdown water of each steam generator shall be monitored by a continuous integral measurement with gamma ray measurement equipment.

(2) In case of failure of the gamma ray measurement equipment in the steam generator blowdown system, a one-liter sample shall be extracted in daily intervals from the corresponding blowdown train and its Cesium-137 equivalency determined by an integral measurement of the gamma radiation in the energy range above 60 keV.

(3) If the Cesium-137 equivalency at one of the measurement equipment specified under para. (1) exceeds 4×10^5 Bq/m^3, a one-liter sample shall be extracted in daily intervals from the corresponding blowdown train (P 1, cf. **Figure 4-1**) and the discharge with the turbine building vent air shall be determined as specified under Appendix A.

(4) After each of the samplings under para. (3), the samples shall be evaluated as specified under Section C 1 Appendix C. The concentration of Strontium-89 and Strontium-90 as well as Tritium shall be determined for the time span of discharge. In this context, it is permissible to evaluate volume-proportional mixture sample (cf. Sections C 2 and C 3 Appendix C).

(5) The volume of demineralized water makeup in the secondary system, Q_N, and the measured amounts of watery fluids removed from the secondary system, Q_E, shall normally be determined as time-weighted averages of the measured values calculated for a time span of, e.g., one week; thereby, any irregular makeup volumes or removals will be evened out in the detailed assessment.

Note:

Because, generally, not all removals of watery fluids from the secondary system are metrologically registered, the result will be an overestimation of the vaporous leakage discharged with the turbine building vent air.

(6) The time-weighted average of the specific activity of all radionuclides and radionuclide groups identified by the evaluation specified under Appendix C shall be determined for the same time span specified under para. (5), and the emission of these radionuclides and radionuclide groups with the turbine building vent air shall be determined as specified under Appendix A.

4.3.1.2 Discharged activity from the secondary system via the safety and blow-off control valves during specified normal operation and design-basis accidents

(1) In case activity is discharged via the safety valves or the blow-off control valves and the measurement equipment specified under Section 4.3.1.1 para. (1) shows, simultaneously, that the Cesium-137 equivalency exceeds 4×10^5 Bq/m³, a sample shall be extracted without delay from the blowdown water of each steam generator (P 1, cf. **Figure 4-1**) and shall be subjected to the radionuclide-specific evaluation specified under Appendix C. After the initial sampling, further samples shall be extracted in 30 minute intervals until the end of activity discharge and shall, likewise, be subjected to the radionuclide-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 sampling point and the transfer factor within the steam generator into account, the temporal development of the specific activity in the main steam – with the exception of noble gases – shall be determined. In the case of design-basis accidents, the specific activity of the noble gases in the main steam shall, additionally, be determined.

Note:

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

(2) To determine the discharge of main steam via the safety valves or the blow-off control valves it is necessary to monitor and record the positional settings of the main steam safety valves (positions OPEN or CLOSED), the main steam isolation valves (positions OPEN or CLOSED) and the blow-off control valves (stroke position) as well as the positional settings of the isolation valves upstream of the safety valves and blow-off control valves. The opening cross-section of the blow-off control valves shall be determined. It is permissible that this opening cross-section is derived from the stroke position of the blow-off control valves.

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

(4) The pressure, P_E, in the main-steam line between steam generator and steam dump station shall be continuously measured and automatically recorded.

(5) The amount of main steam discharged via the safety valves or the blow-off control valves shall be determined. This may be based on a reference point for which, during commissioning, the waste-steam release rate at a defined steam pressure and a defined opening cross-section was determined, and

by calculating, for this reference point, the actual volume discharged from the temporal change of the main-steam pressure and from the opening cross-section determined under para. (2).

(6) The discharged activity shall be determined for each train of the main-steam system on the basis of the data determined as specified under paras. (1) and (5).

4.3.1.3 Discharged activity during the design-basis accident “Leakage in a main-steam line downstream of the main-steam isolation valves”

(1) In case of a leakage in a main-steam line that leads to a closure of the main-steam isolation valves, the discharged activity shall be determined based on an analysis of the temporal change of main-steam pressure, of the closing behavior of the main-steam isolation valves and of the activity in the main steam.

Note:

For this kind of leakage, only a small amount of discharged activity is expected.

(2) The main-steam activity shall be determined from the measurement values achieved with the gamma ray measurement equipment for the continuous monitoring of activity in the associated steam generator blowdown line as well as from the results of the gamma-spectrometric analysis of one sampling each from the associated steam generator blowdown line and from the primary coolant.

4.3.2 Boiling water reactors

4.3.2.1 Discharged activity from the main-steam or feedwater system upon opening of the turbine building roof flaps caused by a leakage of the main-steam or feedwater line inside the reactor building or inside 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) Feedwater (P 4),

d) Exhaust gas from the condenser air removal (P 5).

(2) The specific activity of the samplings specified under para. (1) shall be determined in the laboratory by a gamma-spectrometric analysis. The radionuclides listed in **Table 4-3** shall be taken into account. The measured values or the achieved decision thresholds and detection limits shall be documented.

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

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

Table 4-3: Radionuclides to be taken into account in determining the discharged activity: gamma radiating nuclides

(4) All those system filling levels, make-up volumes and system-internal pressures required to determine the mass of discharged media in case of a leakage in the main-steam or feed water line shall be continuously monitored. The necessary measurement equipment and the procedures for determining the mass of the discharged media shall be specified in an operating instruction.

(5) The proper functioning order of the 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 recorded.

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

a) Calculation of the mass of the released medium on the basis of the system filling levels or of the size of the leakage in conjunction with the internal pressures of the associated system determined as specified under para. (4),

b) Calculation of the amount of steam-air mixture discharged 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 activity as specified under Appendix B for the radionuclides listed in **Table 4-3**.

Section	*Discharge Path*	*Determining the Specific Activity*	*Determining the Amount of Discharged Media*
4.3.1.1	Discharged activity from the secondary system during specified normal operation caused by leakage into the turbine building atmosphere and, further, via roof vents into the environment.	Continuous monitoring of the gamma ray activity in the blowdown water of each steam generator. Sampling from the associated blowdown lines and radionuclide-specific evaluation of the samples with respect to determining the specific activity of the main steam.	Measuring the demineralized water makeup fed into the secondary system and the watery fluids removed from the secondary system.
4.3.1.2	Discharged activity from the secondary system via the safety valves and blow-off control valves during specified normal operation and design-basis accidents.	Sampling from the steam generator blowdown lines and, in case of design-basis accidents, also from the primary coolant. Gamma-spectrometric evaluation of the samples with respect to determining the specific activity of the main steam.	The amount of discharged main steam is determined from measuring the main-steam pressure, establishing the positional settings of the safety and blow-off control valves, of the main-steam isolation valves and of the isolation valves upstream of the safety valves and blow-off control valves, and by taking the point in time and opening behavior of the valves into account.
4.3.1.3	Discharged activity during the design-basis accident “Leakage in a main-steam line downstream of the main-steam isolation valves”.	Continuous monitoring of the gamma ray activity in the blowdown water of each steam generator. Sampling from the associated steam generator blowdown line and from the primary coolant. Gamma-spectrometric evaluation of the samples with respect to determining the specific activity of the main steam.	The released main-steam volume is determined by measuring the main-steam pressure and taking the closing behavior of the main-steam isolation valves and the main-steam activity into account.

Table 4-1: Discharge paths and procedures for determining the discharged activity in the case of nuclear power plants with pressurized water reactors

Section	*Discharge Path*	*Determining the Specific Activity*	*Determining the Amount of Discharged Media*
4.3.2.1	Discharged activity from the main-steam or feedwater system upon opening of the turbine building roof flaps caused by a leakage of the main-steam or feedwater line inside the reactor building or inside the turbine building	Sampling from the main steam or main condensate, from the reactor water and feed water and evaluation of the samples with respect to determining the specific activity. Continuous monitoring of the activity concentration of the exhaust gas from the condenser air removal with the objective of determining changes of the specific activity of the main steam and feed water between successive samplings.	In case of a large leak combined with an opening of the turbine building roof flaps, the amount of steam released shall be calculated from the system fill levels, the makeup volumes and system pressures. The amount of steam discharged to the environment shall be determined as specified under Appendix B.

Table 4-2: Discharge paths and procedures for determining the discharged activity in the case of nuclear power plants with boiling water reactors

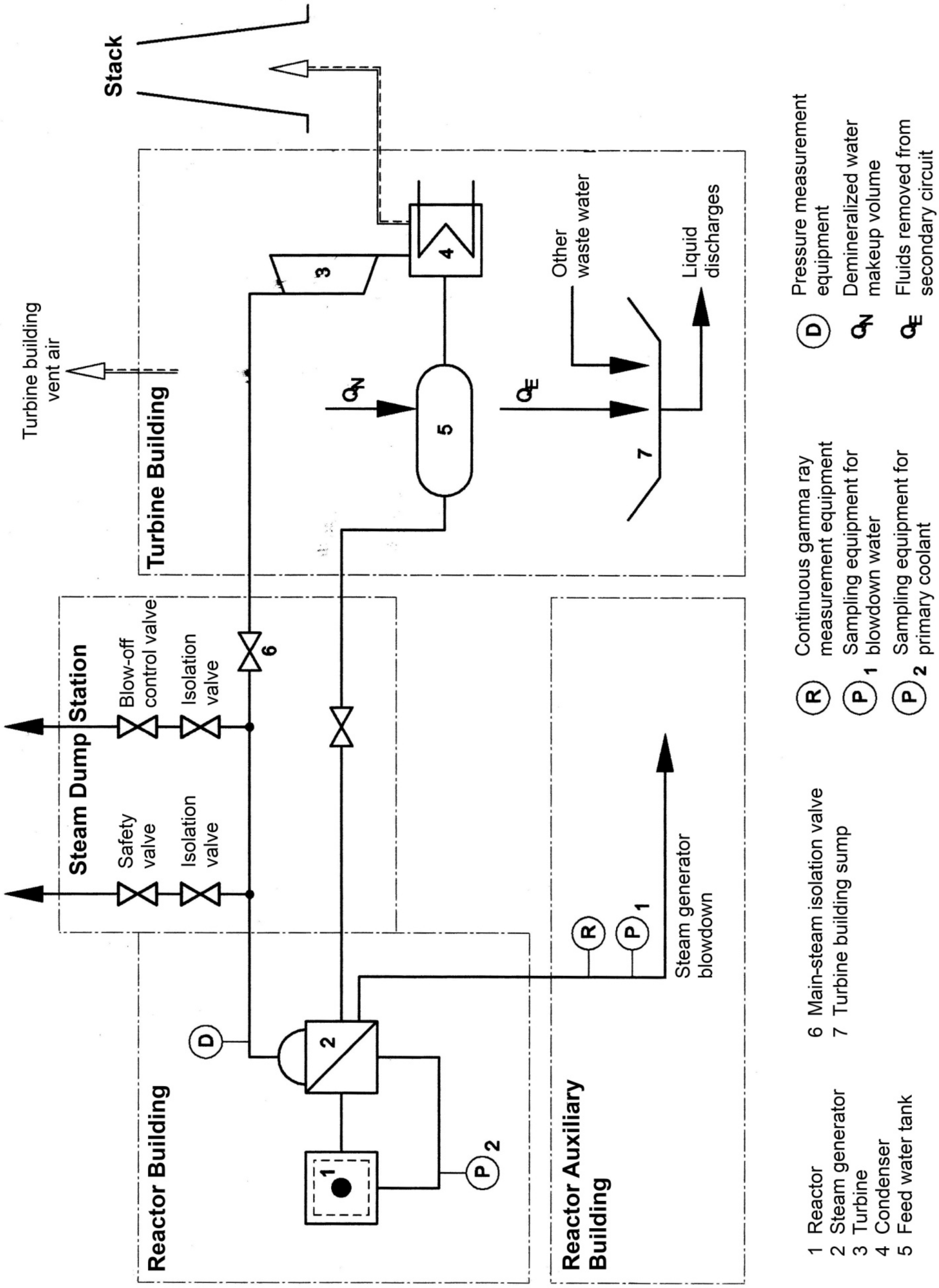

Figure 4-1: Schematic presentation of the monitored discharge paths and the measurement and sampling equipment nuclear power plants with pressurized water reactors (PWR)

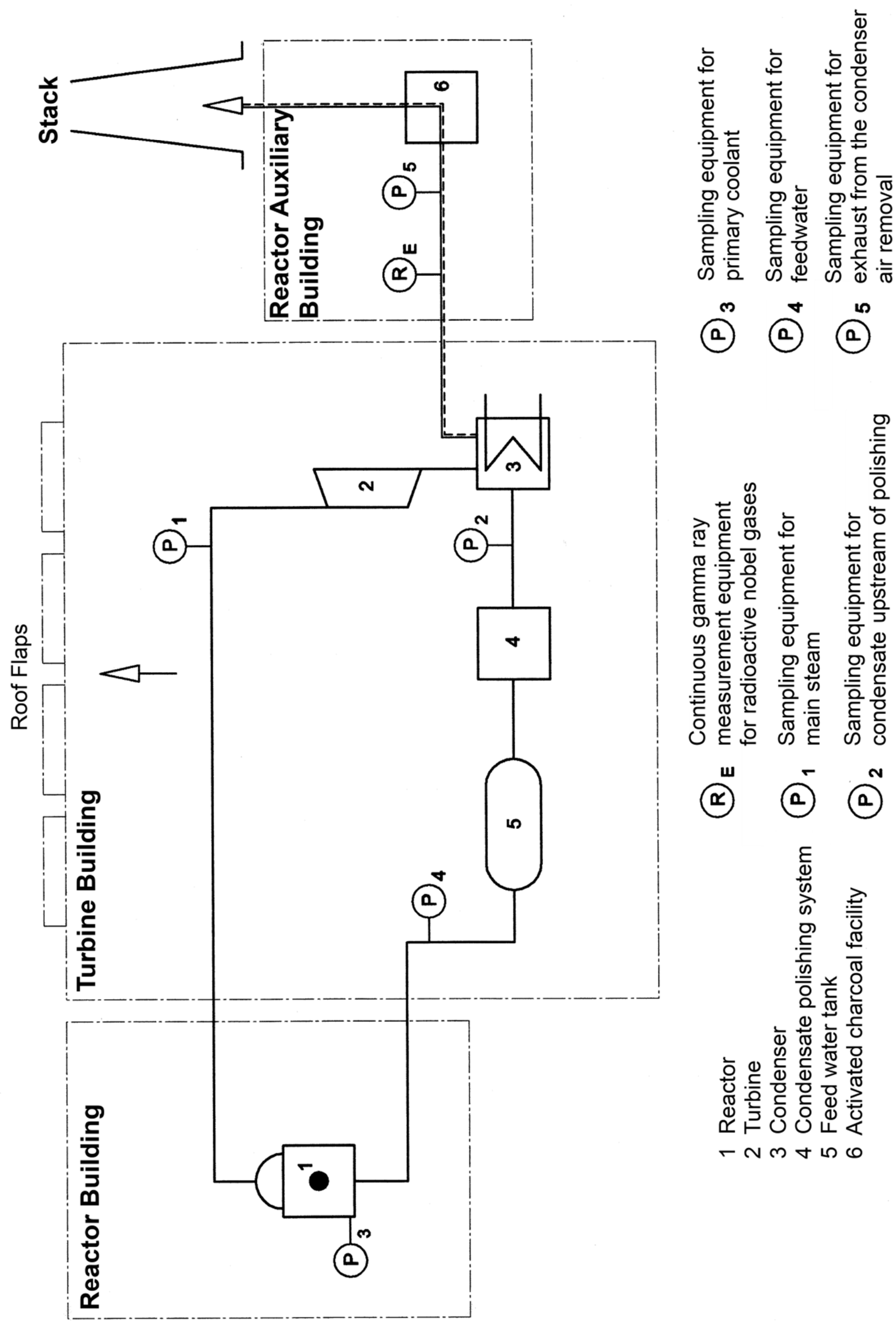

Figure 4-2: Schematic of the monitored discharge paths and the measurement and sampling equipment nuclear power plants with boiling water reactors (BWR)

5 Design of the Equipment for Continuous Activity Measurements and Samplings

Note:

Any equipment not covered in this section (e.g., isolation valves and water meters) shall be designed in accordance with the pertinent technical standards and guidelines taking the measurement tasks into account.

5.1 General Requirements

5.1.1 Design and housing

(1) Devices not located inside the laboratory shall be designed to meet the requirements of Protection Type IP 54 in accordance with DIN EN 60529 (i.e., protection from foreign matter and from water).

(2) The measurement and sampling equipment shall be installed and housed such that

a) the nominal operating ranges specified in the individual equipment specifications are maintained, and

b) tests and inspections, maintenance and repair can be easily performed.

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

(4) With regard to the immunity of the measurement equipment to electromagnetic influences (e.g., electrostatic discharges, electromagnetic fields, interference voltages) the Act on the Electromagnetic Compatibility of Devices (EMVG) shall be observed.

5.1.2 Protection against failure

(1) Continuously operating activity measurement equipment shall be connected to an emergency power supply.

(2) Continuously operating activity measurement equipment shall be designed to be self-monitoring.

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

(4) In case measurement equipment are installed on or inside a bypass, the volumetric flow in the respective bypass shall be monitored. In case the measurement equipment are installed directly on a system, the volumetric flow of the measurement medium shall be monitored.

(5) Possible pulse rate losses of the measurement equipment within the measurement range (e.g., due to delay times) shall be known as a function of the pulse rate and shall be taken into account. No decrease of the display with the increase of a measurement parameter (overloading) is permissible.

5.1.3 Statistical certainty

(1) The value of the factor $k_{1-\alpha}$ in accordance with DIN ISO 11929 shall be set equal to 1.645.

(2) The value of the factor $k_{1-\beta}$ in accordance with DIN ISO 11929 shall be set equal to 1.645.

(3) The value of the factor $k_{1-\gamma}$ in accordance with DIN ISO 11929 shall be set equal to 1.645.

5.1.4 Threshold values

(1) If it is necessary to readjust devices during operation, built-in adjustment controls shall be provided. All adjustment controls on electronic devices of the monitoring equipment shall be arranged or secured in such a way that a readjustment by non-authorized personnel can, to a large extent, be precluded. A misadjustment by the equipment itself shall be impossible.

(2) Equipment failure and the exceeding of threshold values shall be optically displayed and acoustically annunciated and recorded in the control room. Collective alarms are permissible, provided, the measurement location originating the alarm is displayed in the control room or in a control room annex. The acoustic alarms may be cancelled individually or collectively before remedying the cause of the alarm.

(3) The optical alarms in the control room indicating that a failure has occurred or that an upper threshold value has been exceeded shall also indicate the alarm condition (e.g., registered, or acknowledged).

5.1.5 Displaying and recording of the measured values

(1) The measurement equipment shall normally 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 sequential measurement ranges overlap each other by at least 10 % and the full-scale values do not differ by more than the factor of 10; and

b) that, in the case of multiple logarithmic display ranges, the sequential measurement ranges overlap each other by at least one decade.

(2) All measured values shall normally be displayed on the associated measurement equipment itself and shall be displayed and automatically recorded in the control room.

(3) The recorded data shall remain directly visible and well legible for a time span of at least 3 hours.

5.1.6 Testability

The monitoring equipment shall be designed and constructed such that a verification of the perfect functioning order of individual devices is possible within the framework of the initial tests specified under Section 6.2.1 and of the inservice inspections specified under Section 6.2.2. It shall be possible to perform functional tests even during full power operation of the nuclear power plant.

5.2 Measurement Equipment for Continuous Activity Measurements

5.2.1 Detection limits

The detection limits of the measurement equipment for continuous activity measurements shall not exceed the lower limits of the minimum required measurement ranges specified below.

5.2.2 Gamma ray measurement equipment in the steam generator blowdown lines (PWR)

(1) The measurement range of the gamma ray measurement equipment for the continuous measurements of the Cesium-137 equivalency in the steam generator blowdown lines shall cover a range from at least 1×10^5 Bq/m^3 through 2×10^8 Bq/m^3.

(2) An exceeding of the Cesium-137 equivalency of 4×10^5 Bq/m^3 as well as a failure of a measurement equipment shall be automatically annunciated and recorded in the control room.

5.2.3 Gamma ray measurement equipment for monitoring the exhaust from the condenser air removal upstream of the gas exhaust system (BWR)

(1) The measurement range of the measurement equipment for the continuous monitoring the activity concentration in the exhaust from the condenser air removal upstream of the gas exhaust system shall cover a range at least from a value of 0.5 times to 10 times the threshold value under Section 4.3.2.1 para. (3).

(2) An exceeding of the alarm threshold value specified under Section 4.3.2.1 para. (3) as well as a failure of a measurement equipment shall be automatically annunciated and recorded in the control room.

6 Maintenance of the Equipment for Continuous Activity Measurements and for Sampling

6.1 Servicing and Repair

6.1.1 Execution

Servicing and repair of the monitoring equipment shall be performed in accordance with the respective operating and repair instructions and by qualified persons.

6.1.2 Documentation

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

a) Unambiguous identification of the monitoring equipment involved,

b) Type of the servicing or repair task performed,

c) Type and number of exchanged parts,

d) Reasons for exchanging the parts,

e) Regarding the newly installed parts:
date and detailed identification of the test certificates and of the verifications required under the present safety standard,

f) Information regarding the outage times,

g) Date of the servicing or repair task, and

h) Name and signatures of the qualified persons.

6.2 Tests and Inspections

The monitoring equipment shall be subjected to the following tests and inspections:

a) Prior to their deployment in a nuclear power plant:
   - aa) Certification of suitability, and
   - ab) Calibration,

b) Prior to their first deployment in a particular nuclear power plant:
   - ba) Check of suitability,
   - bb) Check of calibration with solid calibration sources,
   - bc) Factory test, and
   - bd) Commissioning tests and inspections,

c) During deployment in the nuclear power plant
   - ca) regular inservice inspections, and
   - cb) tests and inspections after servicing and repair tasks.

6.2.1 Initial tests and inspections

6.2.1.1 Verification of suitability

(1) Prior to their initial deployment in a nuclear power plant, it shall be verified that the monitoring equipment can fulfill their tasks and meet the specified requirements.

Note:
Requirements regarding verifying the suitability of stationary measurement equipment for monitoring activity are dealt with in safety standard KTA 1505.

(2) The verification of suitability comprises a (plant independent) verification of the equipment characteristics and a plant-dependent suitability check.

(3) The plant-dependent suitability check shall be performed by the proper authority or an authorized expert appointed by the proper authority.

6.2.1.2 Calibration and calibration check

(1) The measurement equipment 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 measurement equipment.

(2) The measurement equipment in the steam generator blowdown lines (PWR) shall be calibrated with Cesium-137. The discrimination for gamma rays shall be known in the energy range from 100 keV through 1700 keV.

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

(4) During the initial calibration of the measurement equipment specified under para. (2), a set of solid calibration sources shall be specified with which one display value in one of the two lower decades of the measurement range and one display value in one of the two upper decades can be verified. For this purpose, the following solid calibration sources shall be provided:

a) Cesium-137 for monitoring the water, and one solid calibration source for checking the lower energy threshold;

b) Cesium-137 for the gamma ray measurement equipment in the condenser air removal system specified under 4.3.2.1.

(5) Directly after initial calibration of the monitoring equipment, a solid radiation source shall be applied in a defined and reproducible geometry to determine a display value which will later make it possible to check the calibration and to connect further type-identical equipment.

6.2.1.3 Factory test

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

(2) In case the monitoring equipment is comprised of components from different manufacturers, the correct manufacture and perfect functioning order of these components shall be verified by factory tests performed at the individual manufacturer.

(3) The factory test shall be performed as a production test and shall comprise:

a) Visual inspection,

b) Test of the output value as a function of the specified fluctuation of the operating voltage,

c) Test of the response characteristic with a pulse or current generator and at least one test value for each decade of the measurement range,

d) Test of the overload resistance (by using electronic means or a solid radiation source), and

e) Function test (using a solid radiation source),

(4) The factory test shall be performed by plant experts and, in justified cases, in the presence of the proper authority or an authorized expert appointed by the proper authority.

6.2.1.4 Commissioning Test

(1) The post-installation commissioning tests shall verify the proper design and function of the monitoring equipment. The following items shall be tested:

a) Design of the monitoring equipment,

b) Installation of the monitoring equipment,

c) Display (with at least one test value for each decade of the measurement range),

d) Check of the calibration (with a solid calibration source),

e) Connection to the emergency power system,

f) Volumetric flow monitoring,

g) Measurement value processing,

h) Supply of operating media,

i) Equipment failure alarms,

j) Threshold value settings and alarm signals,

k) Automatic restart after interruption of the power supply.

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

6.2.2 Inservice inspections

6.2.2.1 General requirements

(1) The testing schedule, the test instructions and test certificates shall be in accordance with safety standard KTA 1202.

(2) The inservice inspections shall be possible without manual changes of the circuitry (e.g., soldering).

6.2.2.2 Regular inservice inspections

(1) Regular inservice inspections shall be performed to verify the perfect functioning order of the monitoring equipment. The testing procedures and test frequencies shall be as specified in **Table 6-1**.

(2) The verification of the calibration listed under Running Number 1 of **Table 6-1** shall be carried out in the defined geometry and with the solid calibration source specified during initial calibration of the measurement equipment (cf. Section 6.2.1.2 para. (5)). The required value of the display shall be achieved with an accuracy that must be specified in the testing manual.

(3) The regular inservice inspections shall be performed by the plant operator or by the proper authority or an authorized expert appointed by the proper authority.

6.2.2.3 Post-repair testing

After completion of a repair task, the perfect functioning order shall be verified by a commissioning test as specified under Section 6.2.1.4 to an extent corresponding to the repair task.

6.3 Removal of defects

The time limits and, possibly, alternative measures for the removal of defects shall be specified in the operating manual. The defects including the measures taken for their removal shall be documented.

7 Documentation of Measurement Results

7.1 Flow chart

(1) The sampling and monitoring equipment provided with respect to meeting the specified requirements shall be clearly presented in a flow chart. Different symbols shall be used to identify the type of sampling and monitoring.

(2) In a description correlated to the flow chart (e.g., in the form of a table), the required measurement task and the measurement procedure shall be specified for each sampling and monitoring equipment. In case of sampling equipment, the task, type, location and frequency as well as the measurements to be performed shall be listed. In case of monitoring equipment, the measurement tasks and technical measurement requirements, in particular, the type of measurement, the arrangement of the measurement equipment including radiation shielding, the calibration, the measurement ranges, detection limits and measurement uncertainties shall be listed. Likewise, the measurement tasks of the measurement laboratory shall be described.

7.2 Extent of Documentation

(1) In case it is required, as specified under Section 3, to determine the discharged activity during specified normal operation, the report form shown in **Figure 7-1** shall normally be used. These forms shall be forwarded to the proper authority at the end of each yearly quarter.

(2) The column "Discharged Activity" shall list only those values that result from measured values of activity concentrations exceeding the decision thresholds. The measurement uncertainties shall be noted. If no values exceeding the decision thresholds are detected, the corresponding fields in the report form shall be marked as “smaller DT” or “unknown”.

(3) In the column „Discharged Activity and its uncertainty“ the summation of measurement uncertainties according to Gaussian error propagation shall be applied and list in the in the appropriate line.

(4) The documentation of the discharged activity during design-basis accidents shall be structured such that

a) the discharge path is unambiguously shown,

b) the chronological development of the discharged activity caused by the design-basis accident is described,

c) the sampling,

    ca) prior to the occurrence of,

    cb) during, and

    cc) after

    the design-basis accident including the point in time, duration and sampling type (continuous, non-continuous) and other boundary conditions occurring during sampling are listed, and such that

d) the methods for determining the specific activity and the discharged amounts are described in detail and the determination of the specific activity and discharged amounts can be traced back for the duration of the respective design-basis accident.

Running No.	***Test Object***	***Testing Procedure***	***Test Frequency*** [1)] ***by the plant operator***	***Test Frequency*** [1)] ***by the proper authority or an authorized expert appointed by the proper authority***
1	Monitoring equipment for continuous measurements	a) Visual inspection	during inspection rounds	annually
		b) Calibration check with a solid radiation source	quarter annually	
2	Inspection and maintenance records	Visual check	—	
3	Electronic modules	Insertion of suitable signals at inputs provided or simulation of signals directly into the measurement transducer input with at least one value in each decade of the measurement range [2)] for the integral test of the measurement transducer. For testing the measurement transducer output as well as registering devices such as displays, recorders, monitoring computers, at least one value shall be simulated for each decade of the measurement range; in the case of computer based measurement equipment, these values may be created by the software. Comparison of all displays and recordings.	annually	
4	Alarm signals	a) Operational availability: visual inspection	during inspection rounds	
		b) Failure alarm signal: by interrupting the voltage supply or interrupting the signal connection between measurement transducer and detector, or by inserting a value below the failure threshold	quarter annually	
		c) Hazard alarm: by radiation source or electrically	quarter annually	
		d) Volumetric flow monitoring: by varying the volumetric flow to values outside of the alarm thresholds	annually	
5	Flow monitoring and supply of operating media			
	without automatic function control	Visual inspection	during inspection rounds	
	with automatic function control	Comparison of the required value with the actual value	quarter annually	

1) If a test or inspection is performed according to Column 5, the test or inspection to be performed according to Column 4 at this point in time may be waived.

2) The test procedure of simulating of detector signals at the measurement transducer inputs for the integral test of measurement transducers and measurement circuits – with at least one value in each decade of the measurement range – is not required in the case of computer-based measurement equipment, provided, the software program is certified. In this case it is sufficient to insert one signal in the uppermost decade of the measurement range, provided, the pre-processing electronics do not perform any switching procedures throughout 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: Regularly recurring inservice inspections

Report Form (in accordance with safety standard KTA 1503.3)

Nuclear Power Plant:	**Time Period of Discharge:**	**Annual Quarter:**
Discharge Path:		

Radionuclides	**Decision Treshold and Detection Limit 1) of Evaluated Samples (Bq/m^3)**		**Discharged Activity 2) (Bq) and its uncertainty (Bq)**	**Remarks**
	DT min.	**DL 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				
Cs-134				
Cs-137				
Ba-140				
La-140				
Ce-141				
Ce-144				
I-131				
3)				
Sr-89				
Sr-90				
Sum Total				
Tritium				

1) DT and LT are the maximum decision thresholds and detection limits achieved with a single measurement in the detailed assessment period.

2) unk. = unknown

3) Additional gamma ray emitters, if applicable.

Figure 7-1: Example for a report form for the non-stack discharge of radioactive substances during specified normal operation

Appendix A

Analysis Procedure for PWR

Calculation of Radioactive Substances Discharged with the Turbine Building Vent Air (Roof Vents)

The activity, A, caused by a defective steam generator and discharged with the turbine building vent air in a particular time span shall be calculated by equation (A-1):

$$A = A_A \times \frac{Q_N - Q_E}{n} \times f \tag{A-1}$$

Nomenclature:

A_A specific activity or activity concentration of a radionuclide or a radionuclide group in the steam generator blowdown 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 span (determined as the time-weighted average of, e.g., weekly measurements)

Q_E amount of measured watery fluids removed from the secondary circuit during the particular time span

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 and Tritium), provided, no other value can be verified.

If a threshold value as specified under Section 4.3.1.1 paras. (3), (4) and (5) is exceeded in more than one steam generator, the overall activity discharge shall be determined as the sum of the amounts contributed by the steam generators concerned.

Appendix B

Analysis Procedure for BWR

Calculation of the Activity Discharged by Steam or a Steam-water Mixture 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:

To simplify the application and verification of the calculation procedures, a consistent set of dimensions is specified for the different measurement parameters. The introductory "e.g." should indicate that the associated measurement parameters may also be measured in other dimensions.

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

$$A = \frac{A_A \times Q_A \times F_A \times Ü_A}{V_A} \times V_U \qquad \text{(B-1)}$$

Nomenclature:

A_A specific activity (e.g., in Bq/kg) in the main steam or feedwater; in this context, the specific activity of radioactive noble gases in the feedwater, A_{AE}, shall be calculated by equation (B-2):

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

where

A_{KE} activity concentration (e.g., in Bq/m^3) of radioactive noble gases in the exhaust of the condenser air removal system (to be determined as specified under Section 4.3.2.1 paras. (1) and (2)),

$\dot{V}_G$ volumetric flow to the gas exhaust facility (e.g., in m³/h)

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

Q_A mass (e.g., in kg) of the medium released into the turbine building (to be determined as specified under Section 4.3.2.1 para. (4))

V_A free air volume (e.g., in m^3) in the turbine building

F_A portion of leakage released as steam

$F_A = 1$ in case of "Leakage in a main-steam line"

$F_A = 0.2$ in case of "Leakage in a feedwater line"

$Ü_A$ transfer factor of radionuclides into the released steam

$Ü_A = 1$ in case of "Leakage in a main-steam line"

$Ü_A = 0.01$ in case of "Leakage in a feedwater line"

V_U volume (e.g., in m^3) of the steam and air released to the environment via the turbine building roof flaps (to be calculated by equation (B-3)),

(2) The volume, V_U, of steam and air released to the environment upon opening the turbine building roof flaps shall be calculated by equation (B-3):

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

Nomenclature:

n number of unlocked roof flaps (to be determined onsite by visual inspection after the design-basis accident)

F_e aperture of the roof flaps (e.g., in m^2) (to be conservatively estimated by assuming that the roof flaps are open 100 % for the entire opening time)

ξ loss coefficient for the individual roof flaps (to be conservatively estimated by setting them equal to 0)

ρ density of the steam-air mixture (approximately 1 kg/m³)

$\Delta p(t)$ time dependent pressure difference (e.g. in kg/(m × sec^2) between the turbine building atmosphere and the environment,

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

(The values for $\Delta p_i(t)$ and Δt_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 the Blowdown Water Samplings (PWR)

C 1 Gamma Ray Emitters

In order to determine the discharged activity, the samples shall be analyzed by a gamma spectrometric measurement. In this context, at least the radionuclides listed in **Table C-1** shall be taken into consideration. The detection limit of the measurement equipment for determining the activity concentrations of a demineralized-water sampling shall not exceed 1×10^3 Bq/m³ for Cobalt-60. The measurement time for the individual samplings shall be at least as long as it takes to achieve the required detection limit for Cobalt-60. Within the framework of the gamma spectrometric analyses, it shall be checked whether any radionuclides in addition to those listed in **Table C-1** are present on account of plant specific conditions. Other radionuclides that are detected shall be included in determining the discharged activity.

C 2 Radioactive Strontium

The samplings extracted in the discharge time span shall be evaluated with respect to their content of Strontium-89 and Strontium-90. It is permissible to determine this from a volume-proportional mixture sample. The detection limit of the procedure for determining this activity concentration shall not exceed 5×10^2 Bq/m^3.

C 3 Tritium

(1) The samplings extracted in the discharge time span shall be evaluated with respect to their content of Tritium. It is permissible to determine this from a volume-proportional mixture sample. The detection limit of the procedure for determining this activity concentrations shall not exceed 4×10^4 Bq/m^3.

(2) If the activity concentration of Tritium exceeds 1×10^6 Bq/m^3 in the steam generator blowdown water, a detailed assessment of the discharged Tritium is required.

Radionuclides	
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: Radionuclides to be considered in determining the discharged activity: Gamma ray emitters

Appendix D

Regulations Referred to in the Present Safety Standard

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

AtG		Act on the peaceful utilization of atomic energy and the protection against its hazards (Atomic Energy Act – AtG) of December 23, 1959, revised version of July 15, 1985 (BGBl. I, p. 1565), most recently changed by Article 2, Sec. 2. of the Act of July 20, 2017 (BGBl. I, p. 2808)
StrlSchV		Ordinance on the protection from damage by ionizing radiation (Radiological Protection Ordinance – StrlSchV) of July 20, 2001 (BGBl. I, p. 1714; 2002 I, p. 1459), most recently in accordance with Article 10 changed by Article 6 of the Act of January 27, 2017 (BGBl. I, p. 114, 1222)
Safety Requirements for Nuclear Power Plants (SiAnf)	(2015-03)	Safety Requirements for Nuclear Power Plants (SiAnf) of 22 November 2012 (BAnz AT 24.01.2013 B3), revised version of 3 March 2015 (BAnz AT 30.03.2015 B2).
Interpretations of the "Safety Requirements for Nuclear Power Plants" (Interpretations)	(2015-03)	Interpretations of the "Safety Requirements for Nuclear Power Plants of 22 November 2012" (BAnz AT 24.01.2013 B3), revised version of 3 March 2015 (BAnz AT 30.03.2015 B2)
EMVG		Act on the electromagnetic compatibility of operating media of February 26, 2008 ((BGBl. I, p. 220), most recently changed by Article 3 Sec. 1 of the Act of June 27, 2017 (BGBl. I, p. 1947)
KTA 1202	(2009-11)	Requirements for the testing manual
KTA 1503.1	(2016-11)	Monitoring the Discharge of Radioactive Gases and Airborne Radioactive Particulates Part 1: Monitoring the Discharge of Radioactive Matter with the Stack Exhaust Air During Specified Normal Operation
KTA 1503.2	(2017-11)	Monitoring the Discharge of Radioactive Gases and Airborne Radioactive Particulates Part 2: Monitoring the Discharge of Radioactive Matter with the Vent Stack Exhaust Air During Design-Basis Accidents
KTA 1505	(2017-11)	Suitability verification of the stationary measurement equipment for radiation monitoring
DIN ISO 11929	(2011-01)	Determination of the characteristic limits (decision threshold, detection limit and limits of the confidence interval) for measurements of ionizing radiation - Fundamentals and application (ISO 11929:2010)
DIN EN 60529	(2014-09)	Degrees of protection provided by enclosures (IP Code) (IEC 60529:1989 + A1:1999 + A2:2013); German version EN 60529:1991 + A1:2000 + A2:2013
KTA-GS 82	(2016-11)	Determination of the characteristic limits (decision threshold, detection limit and limits of the confidence interval) for nuclear radiation measurements according to DIN ISO 11929 - Application examples for the KTA safety standard series 1500, Revision 1