Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 1501 (11/2004)

Stationary System for Monitoring the Local Dose Rate within Nuclear Power Plants

(Ortsfestes System zur Überwachung von Ortsdosisleistungen innerhalb von Kernkraftwerken)

Previous versions of this safety standard were issued in 10/1977 and 06/1991

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

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KTA SAFETY STANDARD

November 2004	**Stationary System for Monitoring the Local Dose Rate within Nuclear Power Plants**	KTA 1501

Preliminary remark: The safety standard KTA 1506, Measuring Local Dose Rates in Exclusion Areas of Nuclear Power Plants (06/1986), was withdrawn in November 2004. Thus, all requirements pertaining to exclusion areas not already contained in generally applicable radiological protection provisions were transferred to this safety standard.

Previous versions of this safety standard: *KTA 1501 10/77 (BAnz No. 234 of Dec. 15, 1977)*
KTA 1501 06/91 (BAnz No. 7a of Jan. 11, 1992)

Contents

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. 36a of February 19, 2005. Copies may be ordered through the Carl Heymanns Verlag KG, Luxemburger Str. 449, 50939 Koeln, Germany (Telefax +49-221-94373603).

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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 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 damage arising from the construction and operation of the plant (Sec. 7 para. 2 subpara. 3 Atomic Energy Act) in order to attain the protective goals specified in the Atomic Energy Act and the Radiological Protection Ordinance (StrlSchV) and further detailed in the "Safety Criteria for Nuclear Power Plants" and in the "Guidelines for the Assessment of the Design of PWR Nuclear Power Plants against Incidents pursuant to Sec. 28, para. 3 StrlSchV - Incident Guidelines" (the version released Oct. 18, 1983).

(2) The objective of monitoring the radiation and radioactivity is, among others, to protect the persons inside and outside the plant against ionizing radiation, to monitor the normal activity flow as specified of solid, liquid and gaseous radioactive substances inside the plant and of monitoring the discharge of radioactive substances.

(3) The stationary system for monitoring local dose rates (in short: LDR-System) serves to monitor the local dose rates in the controlled access areas of nuclear power plants during specified normal operation, and to actuate warning alarm thresholds are exceeded.

(4) During and after design basis accidents, the LDR-System shall normally give information regarding the accessibility of the monitored areas.

(5) The monitoring achieved with the LDR-System contributes towards fulfilling the requirements of Secs. 39 and 67 StrlSchV.

(6) Those measuring equipment required for measuring the local dose rates as specified in safety standard KTA 3502 for an evaluation of the plant condition during and after design basis accidents are not dealt with in the present safety standard. However, it may be useful to design particular measuring equipment in such a way that they meet the requirements of the present safety standard and of safety standard KTA 3502 and that can, thus, be employed to fulfill the tasks of both these safety standards.

(7) The LDR-System does not serve the tasks of Sec. 2 para. 1 Calibration Ordinance.

1 Scope

This safety standard applies to the stationary equipment for monitoring the local dose rate due to gamma radiation and neutron radiation in controlled access areas of nuclear power plants with light-water reactors during specified normal operation and in the case of design basis accidents.

2 Definitions

(1) Measurement location

The measurement location is the place at which the local dose rate is to be determined.

(2) Local dose

The local dose is the dose equivalent measured at a specific location in the measurands specified in Appendix VI Part A StrlSchV.

(3) Local dose rate

The local dose rate is equal to the local dose generated in a specified time interval, divided by the length of this time interval.

3 Measurand

The measurand addressed in this safety standard is the local dose rate of gamma radiation and, if applicable, the local dose rate of neutron radiation.

4 Measurement Locations

(1) Measurement locations for the LDR-System are:

a) Locations where, during specified normal operation, changes of the local dose rate are expected and persons would then have to be warned.

b) Locations where a determination of the local dose rate is required in the case of events which are not included in specified normal operation.

c) Exclusion areas, provided, the determination of the local dose rate in these areas requires stationary measuring equipment.

(2) Typical measurement locations for pressurized water reactors and boiling water reactors are specified in **Table 4-1**.

Note:
The measurement of the local dose rate of neutron radiation is limited to specific cases, e.g., measurements in close vicinity of the reactor pressure vessel and when handling neutron sources. A monitoring with stationary measuring devices is, generally, not required.

(3) The local dose rate shall always be monitored with the LDR-System at the entrances (personnel lock and equipment lock) inside the containment vessel of a nuclear power plant.

5 Requirements for the Measuring equipment and their Components

5.1 General requirements

(1) The measuring equipment specified in this safety standard shall be calibrated prior to commissioning and shall, thereafter, be subjected to inservice inspections as specified in **Table 8-1**.

(2) The measuring equipment shall be connected to the emergency power system. When the line voltage has been restored after power interruptions, the functional capability of the measuring equipment shall not have been impaired.

(3) The measurement values shall be recorded with recording devices.

(4) Where measuring equipment is provided with warning alarm thresholds for the actuation of warning alarms, the alarm thresholds shall be adjustable.

(5) In the case of measuring equipment with several linear display ranges, the successive measurement ranges shall, at least, adjoin each other. The scale end value of an individual measurement range shall not be larger than ten times the scale end value of the next more sensitive measurement range.

(6) In the case of measuring equipment with several logarithmic display ranges, the successive measurement ranges shall overlap each other by at least one decade (by one power of magnitude).

(7) If the dose rate exceeds the end value of the measurement range by up to fifty times the end value, however, up to no more than a dose rate of 250 Gy/h, this over-range condition shall be clearly displayed for a duration of at least 5 minutes.

(8) After no more than 10 minutes from the time the dose rate has exceeded the end value of the measurement range

as described in para. 7, the measurement location involved shall again meet the requirements specified in **Tables 5-1** through **5-3**.

(9) By varying one influence parameter (including a variation of the dose rate itself) within the nominal operating range specified in **Tables 5-2** or **5-3** with the other influence parameters (with the exception of pressure) left constant near the individual reference values, the resulting measurement value shall not deviate from the actual value by more than the corresponding factor f_{max} specified in the tables.

(10) In the case of digital display devices, means shall normally be provided for checking the functional capability of all display segments.

(11) The measuring equipment shall be protected against any unauthorized changes of the alarm thresholds.

(12) It is recommended that no measuring equipment are used that have adjustment elements for setting the zero point or the discrimination. However, in the case of measuring equipment that require re-adjustments during operation, adjustment elements shall be provided that are integrated into the equipment.

(13) If programmed electronic elements are contained in the signal path between the detector and the display unit, a test program for an internal functional check shall be installed.

(14) Above a local dose rate of 1/50 of the upper measurement range limit, the measuring equipment shall display 95 % of the sudden increase of the local dose rate within 10 seconds, provided, the increase amounts to 80 % of the end value of the specific display range,

(15) The measuring equipment shall be designed and constructed to meet the requirements of Protection Degree IP 54 in accordance with DIN EN 60529 (protection against foreign particles and water).

(16) The measuring equipment shall meet the specifications of DIN VDE 0412-1 (protection against ionizing radiation).

(17) With regard to the disturbance resistance of the measuring equipment against electromagnetic disturbance signals (e.g., electrostatic discharges, electromagnetic fields, interference voltages), the Electromagnetic Compatibility Act (EMVG) shall be taken into account.

(18) In accordance with DIN 6818-1, local dose rates above 3×10^{-3} Sv/h shall be displayed after a response time of no more than 5 seconds.

(19) In the case of local dose rate measuring equipment intended for use in areas of mixed radiation fields (e.g., neutron and gamma radiation), the manufacturer shall specify how this equipment reacts to the different types of radiation.

5.2 Additional requirements concerning measuring equipment for specified normal operation

(1) The measuring equipment shall meet the requirements specified in **Table 5-1**.

(2) If more than 20 % of the local dose rate will be caused by gamma radiation above 1.3 MeV, then the nominal operating range shall cover 80 keV to 3 MeV. If, in addition, more than 20 % of the local dose rate is caused by gamma rays in an energy range above 3 MeV, then the effect on the discrimination of gamma rays between 3 MeV and the highest occurring gamma ray energy shall be known.

(3) The measuring equipment for determining the dose rate of gamma radiation outside of exclusion areas shall have a nominal operating range covering at least 10^{-6} Sv/h to 10^{-2} Sv/h, and inside of exclusion areas a minimum nominal operating range covering 10^{-4} Sv/h to 1 Sv/h.

(4) The manufacturer shall specified the reference value of the local dose rate that must be adjusted in case one of the remaining influence parameters changes as specified under Section 5.1 para. 9. When the dose rate varies within the nominal operation range, then the displayed value of an optical display of more than four decades may not deviate from the actual value by more than ± 30 %, in all cases by not more than 20 %.

(5) The measurement value display of the measuring equipment in an exclusion area shall be readable outside of the exclusion area in an accessible area.

(6) In the case of measuring equipment for neutron radiation, the nominal operating ranges and reference values specified in **Table 5-1** shall be applied – with the exception of the values specified for gamma radiation. The nominal operating range for the neutron energy shall extend from 0.025 eV to 10 MeV. The factor f_{max} to be observed in the case of a variation of the neutron energy shall range from +400 % to -75 %. The other reference values (dose rate and neutron energy) shall be specified by the manufacturer. The manufacturer shall, likewise, specify the maximum permissible error of the dose rate (deviation from the actual value) within the nominal operating range and the maximum changes of measurement values, f_{max}, caused by a variation of the other influence parameters. In this instance, the requirements specified under Section 5.1 para. 9 shall be applied.

(7) Dose rate measuring equipment for neutron radiation shall normally have a measurement range covering at least 10^{-6} Sv/h to 10^{-1} Sv/h.

5.3 Additional requirements concerning measuring equipment for measurements during and directly after design basis accidents

(1) The measuring equipment for determining the dose rate of gamma radiation inside the containment vessel during and directly after design basis accidents shall have a nominal operating range covering at least 10^{-6} Sv/h to 10^{2} Sv/h. This requirement may be fulfilled by providing two measuring devices with each one covering a part of the overall measurement range. Otherwise, the requirements specified in **Table 5-2** shall be applied.

(2) The measuring equipment for determining the dose rate of gamma radiation inside the turbine building of a plant with boiling water reactor during and directly after design basis accidents shall have a nominal operating range covering at least 10^{-6} Sv/h to 1 Sv/h. Otherwise, the requirements specified in **Table 5-2** shall be applied.

(3) For the equipment specified under paras. 1 and 2, the manufacturer shall specify the dose rate reference value to be adjusted in the case of a variation of the other influence parameters as specified under Section 5.1 para. 9. The displayed measurement value for a dose rate within the measurement range may not deviate from the actual value by more than 40 %.

(4) The design of the components of the measuring equipment, including cables, shall take those conditions into account which these components could be subjected to during and directly after design basis accidents. This applies, in particular, to the radiation resistance of the electronic components.

(5) For the protection against effects from design basis accidents, the detector may also be installed away from the measurement location, provided, the ratio between dose rate at the installation location and dose rate at the measurement location is known and is considered in the calibration. This provisional condition shall be proven under consideration of possible changes of the nuclide spectrum. In this case, the requirements regarding the measuring device shall be adjusted accordingly.

(6) The measuring equipment shall be arranged such that it can not be made dysfunctional by debris, pipe whips and escaping coolant. If a corresponding design cannot be achieved with one measuring equipment, spatially separated redundant measuring equipment shall be provided.

(7) Measuring equipment installed inside the containment vessel shall meet the requirements specified in **Table 5-2**. **Table 5-3** applies to the measuring equipment installed in the turbine building of boiling water reactors. The nominal operating ranges specified in **Table 5-2** or **Table 5-3** for ambient temperature, pressure, and relative humidity of the inner atmosphere need to be applied only to those parts of the measuring device which could be subjected to the design basis accident conditions and which are required to remain functional during the design basis accident. Section 5.1 para. 9 also applies in this case.

(8) Instead of the proof that the changes, f_{max}, of the measurement value display for the influence parameters ambient temperature, pressure of the inner atmosphere and relative humidity are as specified in **Tables 5-2** and **5-3**, a test in accordance with Sec. 5.8.3 KTA 3505 (design basis accident test curve) may be performed. However, for nuclear power plants other than those in accordance with Sec. 5.8.3 para. 2 KTA 3505, a load curve shall be chosen covering the conditions resulting from design basis accident analyses at the individual installation location. During the test, a local dose rate of 10^{-2} Sv/h, however, not below 100 times the lower measurement range limit, shall be generated at the location of the detector, and the measurement value shall be recorded. The test is considered successful if the measured value deviates by no more than 50 % from the set dose rate; exceptions are permissible within time intervals of no more than 10 minutes after rapid pressure or temperature changes.

(9) The measuring equipment for monitoring the local dose rate at or near the personnel lock inside the containment vessel shall be designed as redundant equipment. The detectors shall be installed spatially separated from each other.

6 Measurement Value Display, Recording and Documentation

(1) The measurement value shall be displayed on-location (cf. Section 5.2 para. 5) and shall be displayed and recorded in the control room. The on-location display unit may be installed spatially separated from the detector. The display and recording shall be in the units of Sv/h, mSv/h or µSv/h.

(2) The recording devices for the measurement values shall be located in the control room or in an annex to the control room. Multi-track plotters or printers may be used, however, with no more than 6 tracks. The recordings shall be in direct view and clearly legible on the recording strips for a period of at least three hours.

(3) The display and recording of the measurement values on CRT monitors is permissible, provided, that a monitor is available primarily for displaying these values, that a hardcopy of the display can be obtained at any time and that the measurement values are stored. A second monitor shall be available as redundancy.

(4) These recordings shall be regularly evaluated and shall be stored in accordance with the legal regulations or provisional conditions.

(5) All equipment failure alarms and warning alarms shall be displayed optically and acoustically in the control room.

(6) Group alarms may be used in the control room, provided, it is displayed in or near the control room, from which measurement location the alarm signal originates.

(7) The optical signals of the equipment failure alarms and of the warning alarms in the control room shall be such that the state of the alarms can be identified (e.g. registered, acknowledged).

(8) It shall be possible to reset the acoustic alarms in the control room. A renewed transition of the upper or lower limit values shall cause renewed acoustic alarms.

(9) The acoustic signals may be individually or collectively annulled prior to the removal of their causes.

(10) An equipment failure shall also be optically displayed on-location. A transition of the failure alarm threshold shall be optically and acoustically displayed on-location. An acoustic failure alarm on-location may be dispensed with, provided, a sufficient number of blinking or flashing lights are available in the hazard region. The cause leading to the transition of the threshold for the failure alarm shall be recorded.

(11) The alarm displays on-location shall be designed and constructed (e.g., brightness, loudness) such that the signals can be reliably perceived under all expected ambient conditions. It should be possible to distinguish between equipment failure alarms and the warning alarms.

(12) Equipment failure alarms and the warning alarms may not cancel themselves automatically while the transition of the upper or lower limit values exists. It shall be possible to cancel the acoustic warning alarms at all times, whereas the optical warning alarms may only be cancelled when the alarm condition has ended.

(13) The threshold values for the warning alarms may be altered only with approval of the radiological protection officer; the alteration shall be documented.

7 Servicing and Repair

(1) Regular servicing and the repairs of measuring equipment shall be carried out by competent personnel and in accordance with servicing schedules and repair instructions.

(2) The servicing and repair tasks shall be recorded in device-specific documents. These documents shall be stored and shall be presented to the authorized experts (under Sec. 20 Atomic Energy Act) during the inservice inspections.

(3) When measuring equipment fails then, until their repair, the individual measurement tasks shall be carried out using suitable alternate measures. The required repair shall be carried out without delay.

Note:
In the case of measuring equipment inside an exclusion area, an "immediate repair" is understood as being carried out during the next plant shut-down or during the next major inspection (refueling).

8 Tests and Examinations

8.1 Test and examination documents

Test instructions shall be written for all tests and examinations specified in Sections 8.3 to 8.6. In the case of inservice inspections, the test intervals, the responsibilities for tests and examinations and the applicable test instructions shall be listed in the test schedules in accordance with Sec. 3.3 KTA 1202.

8.2 Demonstration of suitability

It shall be demonstrated that the measuring equipment specified in this safety standard meet the requirements of the measurement objective.

Note:
Requirements for the demonstration of suitability are specified in safety standard KTA 1505 "Verification of Suitability of Radiation Measuring equipment".

8.3 Factory test and inspection

(1) A factory test and inspection shall be performed to demonstrate proper manufacture and functioning of the measuring equipment.

(2) The factory test and inspection shall be carried out as a production test and shall comprise, at least:

a) visual inspection,

b) test of the change of display when varying the operating voltage within its nominal operating range,

c) calibration with radioactive test sources with at least two measurement points which shall be apart from each other by no less than one decade (one order of magnitude),

d) test of the measurement characteristic using, alternatively, radioactive test sources or an impulse or current generator with at least one test value per decade of the measurement range.

(3) The factory test and inspection shall be carried out by plant experts and, in accordance with a specification of the proper authority, in the presence of an authorized expert (under Sec. 20 Atomic Energy Act).

8.4 Commissioning test

After installation and before first operation, a commissioning test shall be carried out by the licensee and, in accordance with a specification of the proper authority, in the presence of an authorized expert (under Sec. 20 Atomic Energy Act). This test shall comprise, at least:

a) inspection of the installation,

b) test with an impulse or current generator with at least one measurement point per decade of the measurement range (display on-location and in the control room; recording),

c) specification of recalibration values using radioactive test sources in a defined geometry for the verification of the calibration within the framework of inservice inspections (cf. Section 8.5 para. 2),

d) tests of the alarms regarding equipment failure,

e) tests of the alarms regarding transition of limit values,

f) test of the limit value generators,

g) test of the connection to the power supply, and

h) function of the alarm equipment, if applicable.

8.5 Inservice inspections

(1) The measuring equipment shall be subjected to the inservice inspections specified in **Table 8-1**. The inspection intervals also specified in **Table 8-1**, shall be considered as orientation values that may be varied with the objective of dose reductions (cf. Sec. 6 para. 2 StrlSchV).

Note:
The radiation exposure caused by tests and inspections can be reduced if the tests and inspections are performed during the major inspection (refueling).

(2) The calibration test with the same geometry and similar radioactive test sources (taking radioactive decay into account) may deviate from the recalibration values specified in Section 8.4 item c by no more than ±30 % of these recalibration values.

(3) When Geiger counters are used as detectors, the functional capability and the residual life shall be verified by checking the plateau gradients at the individual operating voltages. The radiation source used in this test shall be chosen such that, at the detector operating voltage and at 100 V below and above this operating voltage, the measurement values lie approximately in the central decade of the nominal operating range. From these measurement values the plateau gradients relative to the measurement value at the operating voltage shall be determined, and these gradients shall, then, be compared with the specifications of the manufacturer. The detector shall be replaced the latest when the plateau gradient has reached a value of twice that specified by the manufacturer.

8.6 Tests and inspections after repairs

After completion of a repair task, the proper functioning shall be demonstrated by performing a commissioning test as specified in Section 8.4 to an extent corresponding to the extent of the repair task.

8.7 Test reports

The results of the tests and inspections specified in Section 8 of this safety standard shall be documented in test reports. The test reports shall contain, at least, the following information:

a) test object,

b) test instruction,

c) performance of tests and inspections and the test results,

d) defects,

e) date of testing,

f) signature of tester.

9 Documentation

(1) The documents specified under Sections 7 para. 2, 8.1 and 8.7 shall be stored for every device and shall be available at all times.

(2) Data sheets shall be available for every measuring equipment. In addition, the following data shall be specified:

a) description of the measuring equipment,

b) technical documents regarding installation and set-up,

c) operating instruction,

d) calibration instruction,

e) calibration means,

f) servicing and test instructions.

Note:
The certifications on which the documents under items d, e and f are based are, usually, stored at the location of manufacture of the measuring devices.

Reactor Type	Measurement Location	Design Requirements [2)]
PWR	In front of the storage location for radioactive wastes	I
	Material lock (inside the containment vessel) [1)]	I, II
	Personnel lock (inside the containment vessel) [1)]	I, II
	Refueling platform (refueling machine)	I
	Aeroball measurement room	I, III
BWR	In front of storage location for radioactive wastes	I
	Material lock (inside the containment vessel) [1)]	I, II
	Personnel lock (inside the containment vessel) [1)]	I, II
	Refueling platform (refueling machine)	I
	Floor level of turbine building	I, II
	Compartment of control rod drive mechanisms	I, II
	Compartment of traversing in-core probe system	I, III

[1)] Display outside the containment vessel at entrance to lock

[2)] I: Designed for specified normal operation (Section 5.2)
II: Designed for design basis accidents (Section 5.3)
III: Temporary exclusion area

Table 4-1: Typical measurement locations

Influence Parameter	Nominal Range of Use	Reference Value of Influence Parameter	f_{max} [1)]
Gamma ray energy			
Measuring equipment for gamma radiation with a maximum energy of up to 1.3 MeV	80 keV to 1.3 MeV	662 keV	± 40 %
Measuring equipment for gamma radiation with a maximum energy of up to 3 MeV	80 keV to 3 MeV	662 keV	± 40 %
Measuring equipment for gamma radiation with a maximum energy over 3 MeV	80 keV to 3 MeV	662 keV	± 40 % within the range 80 keV to 3 MeV; additionally, the function of the discrimination with changing energy must be known
Incidence angle of radiation	preferred direction ± 45°	preferred direction	± 30 %
Operating voltage			
DC voltage supply	18 V to 30 V	24 V	± 5 %
AC voltage supply	187 V to 253 V	230 V	± 5 %
Ambient temperature	10 °C to 50 °C	20 °C	± 20 % [2)]
Pressure of inner atmosphere	900 hPa to 1300 hPa	1013 hPa	± 5 % [2)]
Relative humidity	30 % to 95 %	60 %	± 20 %

1) Largest permissible variation of the displayed measurement value with the respective influence parameter being varied within its nominal operating range relative to the measurement value at the reference value of the influence parameter.

2) To meet the requirements in the case of detectors that are not leak tight, the air density correction may be taken into account.

Table 5-1: Permissible variation of measured value display for local dose rate measuring equipment during specified normal operation

Influence Parameter	Nominal Range of Use	Reference Value of Influence Parameter	f_{max} [1)]
Gamma ray energy	80 keV to 3 MeV	662 keV	± 40 %
Incidence angle of radiation	preferred direction ± 45°	preferred direction	± 30 %
Operating voltage			
DC voltage supply	18 V to 30 V	24 V	± 5 %
AC voltage supply	187 V to 253 V	230 V	± 5 %
Ambient temperature	10 °C to 165 °C	20 °C	± 20 % [2)]
Pressure of inner atmosphere	900 hPa to 6300 hPa	1013 hPa	± 5 % [2)]
Relative humidity	30 % to 100 %, condensed steam	60 %	± 20 %

1) Largest permissible variation of the displayed measurement value with the respective influence parameter being varied within its nominal operating range relative to the measurement value at the reference value of the influence parameter.

2) To meet the requirements in the case of detectors that are not leak tight, the air density correction may be taken into account.

Table 5-2: Permissible variation of measured value display for local dose rate measuring equipment inside the containment vessel during and after design basis accidents

Influence Parameter	Nominal Range of Use	Reference Value of Influence Parameter	f_{max} [1]
Gamma ray energy	80 keV to 3 MeV	662 keV	± 40 %
Incidence angle of radiation	preferred direction ± 45°	preferred direction	± 30 %
Operating voltage			
DC voltage supply	18 V to 30 V	24 V	± 5 %
AC voltage supply	187 V to 253 V	230 V	± 5 %
Ambient temperature	10 °C to 80 °C	20 °C	± 20 % [2]
Pressure of inner atmosphere	900 hPa to 1300 hPa	1013 hPa	± 5 % [2]
Relative humidity	30 % to 100 %, condensed steam	60 %	± 20 %

[1] Largest permissible variation of the displayed measurement value with the respective influence parameter being varied within its nominal operating range relative to the measurement value at the reference value of the influence parameter.

[2] To meet the requirements in the case of detectors that are not leak tight, the air density correction may be taken into account.

Table 5-3: Permissible variation of measured value display for local dose rate measuring equipment inside the turbine building of boiling water reactors during and after design basis accidents

Item No.	Test or Examination	Method of Testing	Testing Interval	
			licensee	expert authorized by the proper authority
1	Electronic tests	Input of standard signals into the transmitter [1] (at least one value per decade of the measurement range). Comparison of all displays and recorded data	yearly	yearly
2	Testing of the detectors	As specified in Section 8.5 para. 3 – (only if Geiger counters are used)	half-yearly	yearly
3	Testing of the calibration	Comparison of the specified and actual values using radiation sources ranging over two decades, with at least one value per decade of the measurement range	- half-yearly (if Geiger counters are used) - yearly (other measuring equipment)	yearly
4	Testing the signals (signaling)	Failure alarm: e.g. by interrupting the voltage supply or by breaking the signal connection between transducer and detector	half-yearly	yearly
		Hazard alarm: with radiation source or electrically	half-yearly	yearly
5	Visual inspection of the equipment outside of exclusion areas	–	quarter-yearly	yearly
6	Inspection of the records	–	–	yearly

[1] The testing method of inputting standard signals into the transmitter with at least one value per decade of the measurement range is not required for digital equipment, provided, the program is tested and it controls itself (self-surveillance). In this case, inputting a signal into the uppermost decade of the measurement range is sufficient, provided, no switchovers are carried out by the signal conditioning logic in the entire measurement range. This may, also, be dispensed with, if, during the calibration test, one measurement value lies in the uppermost decade.

Table 8-1: Inservice inspections

Appendix

Regulations Referred to in this Safety Standard

Regulations referred to in this 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.

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) in the version of July 15, 1985 (BGBl. I, p. 1565), most recently changed by Act of August 12, 2005 (BGBl. I, p. 2365)
StrlSchV		Ordinance on the protection from damage by ionizing radiation (Radiological Protection Ordinance - StrlSchV) of July 20, 2001 (BGBl. I, p. 1714), most recently changed by Act of September 1, 2005 (BGBl. I, p. 2618)
KTA 1202	(06/1984)	Requirements for the testing manual
KTA 3502	(06/1999)	Accident overview measuring systems
KTA 3505	(11/1984)	Type testing of measuring transmitters and transducers of the safety related instrumentation and control
DIN VDE 0412-1	(09/1998)	Nuclear instrumentation - Electrical measuring systems and instruments utilizing ionizing radiation sources - Part 1: Technological safety requirements
DIN 6818-1	(08/2004)	Radiation protection dosimeters - Part 1: General
DIN EN 60529	(09/2000)	Degrees of protection provided by enclosures (IP code)