Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 3502 (2012-11)

Accident Measuring Systems

(Störfallinstrumentierung)

The previous version of this safety standard was issued in 1984-11

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 2012	**Accident Measuring Systems**	KTA 3502

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1984-11 (BAnz.-No. 40a of February 27, 1985)
1999-06 (BAnz.-No. 243b of December. 23, 1999)

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 the Bundesanzeiger (BAnz) of January, 23th, 2013. Copies may be ordered through the Wolters Kluwer Deutschland GmbH, Postfach 2352, 56513 Neuwied, Germany (Telefax +49 (0) 2631 801-2223, E-Mail: info@wolterskluwer.de).

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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.

Basic Principles

(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 - AtG) in order to attain the protective goals specified in AtG and the Radiological Protection Ordinance (StrlSchV) and further detailed in the "Safety Criteria" and in the “Design Basis Accident Guidelines”.

(2) Criterion 5.2 “Incident Instrumentation” of the Safety Criteria specifies that equipment for measuring and recording must be provided which, during and after design basis accidents and in the case of unpredictable event sequences

a) will deliver sufficient data regarding the condition of the plant in order to make it possible to initiate the required protective measures with respect to personnel and plant,

b) will give indications on the sequence of events and will enable the documentation of this sequence, and

c) will enable estimating the effects on the environment.

(3) In accordance with Sec. 53 StrlSchV, the owner of a nuclear facility has the obligation to provide on the site of the plant all auxiliary means that are necessary to mitigate damages in the case of personal accidents and design basis accidents. This obligation extends to having to provide the necessary information to the competent authorities on the sequence of events of the design basis accident.

(4) The present safety standard specifies requirements for the design and construction of accident measuring systems for achieving the objectives in accordance with Sec. 53 StrlSchV. It is permissible, however, to fulfill these requirements by other measures.

(5) The present safety standard is based on the assumption that the conventional regulations and standards (e.g., German Accident Prevention Regulations, DIN standards and VDE regulations) are observed.

(6) Requirements regarding lightning protection are specified in safety standard KTA 2206.

(7) Requirements regarding cable penetrations through the containment vessel are specified in safety standard KTA 3403.

(8) Requirements regarding pipe penetrations through the containment vessel are contained in safety standard KTA 3407.

(9) Requirements regarding the arrangement of the display and recording equipment in the control room, the remote shutdown station and the local control stations are specified in safety standard KTA 3904.

(10) A number of measurement parameters specified in safety standard series KTA 1500 also pertain to the accident measuring system, thus supplementing the radiological and meteorological measurement parameters specified in Sections 3 and 5 of the present safety standard. With regard to these measurement parameters, the design and construction requirements for the accident measuring system are specified in safety standards series KTA 1500.

(11) With regard to the electrical power supply, safety standards KTA 3701 through KTA 3705 supplement the corresponding requirements specified in Sections 3.5, 5.4 and 6.3 of the present safety standard.

1 Scope

(1) This safety standard applies to the accident measuring system of stationary nuclear power plants with light water reactors.

This safety standard does not apply to:

a) the equipment of the reactor protection system including the hazard alarm system,

b) that part of the instrumentation necessary or provided solely for the performance of specified normal operating tasks,

c) the nuclear power plant remote surveillance system.

(2) Requirements that do not specifically concern the accident measuring system are not dealt with in the present safety standard.

N o t e :

Requirements that do not specifically concern the accident measuring system are, e.g., requirements that pertain to equipment for the clarification of operational malfunctions and are based of other engineering standards or requirements. This also includes accident management measures.

2 Definitions

N o t e :

The hierarchy of the terms defined in the present safety standard is shown in **Figure 2-1**.

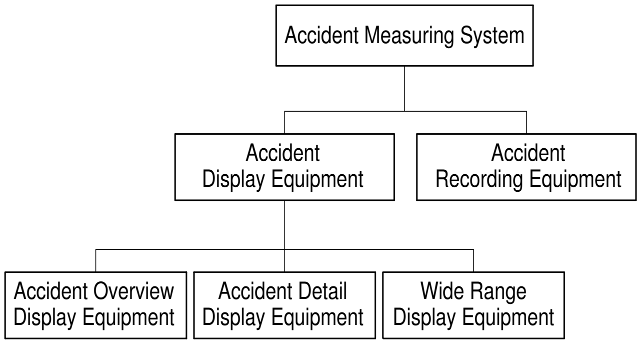

Figure 2-1: Hierarchy of terms of the accident measuring system

(1) Authorized expert

Authorized expert is a competent person or agency consulted in accordance with Sec. 20 AtG by the licensing or supervisory authority.

(2) Design basis accident

A design basis accident is a chain of events upon the occurrence of which the plant operation or the work activity 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 activity, precautionary protective measures shall be provided.

(3) Accident display equipment

The accident display equipment is that part of the accident measuring system which displays the measurement parameters needed as information on the condition of the plant.

N o t e :

The accident display equipment includes all components required for this display, i.e., equipment of the measured value detection, transmission, processing and display.

(4) Accident recording equipment

The accident recording equipment is that part of the accident measuring system which records the measured values. The course of these measured values allows reconstructing the chronological sequence of events during the design basis accident, estimating the radiological effects after a design basis accident and identifying the causes in case of a plant internal design basis accidents. These data are needed for deciding whether or not important components may continue to be used.

Notes:

(1) The accident recording equipment includes all components for processing and recording and, as necessary, for the detection of the measured values. The equipment for the detection of measured values are subject to the requirements for the systems that are the source of the signals (e.g. reactor protection system, accident display equipment, radiological protection instrumentation).

(2) The exact determination of radiological effects in the environment will only be possible after additional information from the emission and the immission monitoring equipment has been evaluated.

(5) Accident detail display equipment

The accident detail display equipment is that part of the accident display equipment which displays the measurement parameters specific to the functional monitoring of the individual safety equipment and of the auxiliary systems required for their operation.

(6) Accident measuring system

The accident measuring system is the equipment which displays and records information specific to the condition of the plant before, during and after the occurrence of a design basis accident or of an event which could lead to an increased release of radioactive substances.

(7) Accident overview display equipment

Accident overview display equipment is that part of the accident display equipment which displays the essential measurement parameters specific to the condition of the plant in the case of design basis accidents.

(8) Wide range display equipment

The wide ranging display equipment is that part of the accident display equipment which displays the measurement parameters specific to the information about whether or not the plant parameters are approaching the design values of the activity barriers and which, in case the design values are exceeded, displays the further development of these plant parameters.

3 Accident Overview Display Equipment

3.1 Measurement Parameters

3.1.1 Selection criteria

(1) The measurement parameters for the accident overview display equipment shall be selected such that after a design basis accident it will be possible to assess the condition of the plant in light of the following criteria:

a) Effectiveness of reactor shutdown,

b) Effectiveness of residual heat removal,

c) Effectiveness of measures for the primary-side and secondary-side pressure limitation and pressure reduction,

d) Effectiveness of activity retention barriers,

e) Determination of external conditions,

In the case of pressurized water reactors and boiling water reactors these are those external conditions inside the containment vessel. In the case of boiling water reactors they, additionally, comprise the external conditions in the reactor building outside of the containment vessel and those inside the turbine building.

(2) Specific measurement parameters shall be selected such that an estimation of the radiological effects upon the environment of the nuclear power plant is possible after a design basis accident has occurred.

Note:

An exact determination of the radiological effects on the environment of the nuclear power plant requires the determination of additional measurement parameters. The requirements for determining these measurement parameters are specified in the safety standards series KTA 1500.

(3) The range of the accident overview display equipment based on these selection criteria is presented for nuclear power plants with pressurized water reactors in **Table 3-1** and for nuclear power plants with boiling water reactors in **Table 3-2**.

Note:

With regard to possible deviations of the expected plant conditions in the case of design basis accidents, the accident detail display equipment shall be consulted.

3.1.2 Location of the displays

The measurement values of the accident overview display equipment shall be displayed in the control room. **Tables 3-1** or **3-2** lists those measurement values of the accident surveillance equipment that shall normally be displayed in the remote shutdown station.

3.2 Display of Measured Values

(1) The measured value displays shall cover the measurement ranges specified in Section 3.3.

(2) The measured values shall normally be presented both as momentary value displays and as progressive value displays. The two types of displays may be combined as far as hardware is concerned. The progressive value displays shall have a time resolution that is suitable with respect to perceiving the chronology of the events.

(3) The displaying devices in the control room and in the remote shutdown station shall normally be correlated to the respective process-engineering related system displays. They shall be accentuated by a clear and unambiguous identification marking.

(4) The display of several measured values on a single device is permissible, provided,

a) the measured values concerned belong exclusively to the accident overview display equipment, and

b) no more than two different scales are required in the case of a mutual progressive value recording.

(5) The scales on the displaying devices shall normally be identification. marked with and unambiguously correlated to the physical unit of the measurement parameter

(6) The physical unit of the measurement parameter shall be clearly discernible on the device.

(7) If, in the case of progressive value displays, the resolution of the measured values is changed, the exact time at which the switchover occurs and the scales shall be well discernible.

(8) The measured values shall be easy to survey, shall be clearly displayed and such that it is possible to read them off unambiguously.

3.3 Requirements Regarding Measurement Range, Accuracy and Temporal Behavior

(1) The measurement range shall normally be selected such that it includes the values occurring during specified

normal operation and the extreme values determined in the design basis accident analysis with an added measurement range margin of 10 %.

(2) Additional requirements apply to the following measurement parameters:

a) Neutron flux

   When the reactor is in the shutdown condition, any increase in the fission rate contributing noticeably to the heat generation shall normally be discernable.

b) Boric acid concentration of the water in the containment vessel sump (pressurized water reactor)

   The measurement range shall cover all possible operational boric acid concentrations and those during design basis accidents.

c) Departure from nucleate boiling (pressurized water reactor)

   The measurement range shall normally extend from the value of departure from nucleate boiling present between normal operation and the boiling state. In this context, the pressure change up to the response of the primary-side safety valves shall normally be taken into account.

d) Pressure in the reactor pressure vessel (boiling water reactor) or in the primary and secondary coolant systems (pressurized water reactor)

   The measurement range shall normally extend up to 1.3 times the design pressure of the primary and secondary coolant systems. In this context, the response values of the overpressure protection equipment shall also be covered.

e) Pressure in the annulus outside of the containment vessel (pressurized water reactor)

   The minimum value shall be determined based on the ventilation concept. The maximum value shall be determined from the design basis accident analysis.

f) Volume ratio of hydrogen

   In order to measure the volume ratio of hydrogen, measuring devices shall be employed with which the measurement parameter can be recorded and the chronological development of the measurement parameter can be determined.

g) Filling level of the reactor pressure vessel

   The measuring devices shall be designed such that the coolant cover of the core can be monitored. Full functionality of the measuring devices for the fluid level shall be ensured during normal operation as well as during design basis accidents.

(3) The accuracy and time behavior of the measuring devices shall remain within permissible tolerances during the course of the respective design basis accident. In order to meet these accuracy requirements, it is permissible to employ several measurement channels with mutually overlapping measurement ranges.

3.4 Requirements Regarding Monitoring and Processing of Measured Values

3.4.1 General requirements

(1) The monitoring and processing of the measured values shall normally be simple, clear and appropriate. If certain data can only be obtained from a combination of several measured values, a measured value processing shall normally be used to achieve the direct display of these data.

(2) The equipment provided shall be reliable, shall require little maintenance and shall be suited for the intended deployment. The equipment shall normally be designed such that a functional test is possible without changes to the wiring.

(3) Measurement signals from measurement locations of the reactor protection system may be used for the accident overview display equipment, provided, they meet the requirements of the present safety standard.

(4) If digital devices are used for the display in the accident overview display equipment, their suitability with regard to the requirements specified under para. (3) Section 3.3 shall be verified within the framework of the suitability check.

3.4.2 Redundancy and diversity

(1) A redundant measured value detection and processing is required unless it is verified

a) that the information contained in the respective measurement parameter is also obtained from measured values supplied by verifiably equally qualified instruments, or

b) that the non-availability of measured values for a measurement parameter is acceptable for a certain period of time and that, within this time period and under the then prevailing conditions, this failure can be repaired or an alternate solution can be found.

(2) Equipment diversity is not required.

3.4.3 Accident resistance

(1) The accident overview display equipment shall be able to withstand, and shall remain fully functional under the external conditions occurring at the respective place of installation in the case of design basis accidents and consequential events.

(2) The external conditions to be assumed to occur at each place of installation shall be determined from the design basis accident analyses.

(3) It is not required to design the measurement equipment of the radiological and meteorological instrumentation against external events. However, lightning protection measures may not be omitted.

(4) The accident overview display equipment inside the area protected against external events shall be decoupled from the equipment in the unprotected area and such that no feedback occurs. A decoupling within the protected area or within the unprotected area is not required.

(5) Equipment which is used both for the accident overview display as well as for other tasks shall be designed, constructed and operated to be in accordance with the requirements for the accident overview display equipment unless more stringent requirements result from the other tasks.

(6) The accident overview display equipment shall be designed such that, under accident conditions, it will remain fully functional at least for a period corresponding to the time between start of the design basis accident and the time when the equipment is either no longer required or can be repaired or replaced. This minimum required functioning period shall be determined from the accident analysis.

3.5 Power Supply

(1) The accident overview display equipment shall normally be supplied from an uninterruptible emergency power supply; the energy storage of this power supply shall comprise batteries in parallel and include rectifying equipment. Additional requirements for such an emergency power supply are specified in safety standard KTA 3703. An uninterruptible power supply is not required in those cases where, on account of its measurement tasks, the non-availability of the equipment is permissible for a short period of time.

No.	Measurement Parameter	Unit or Reference Value	Measurement Range [1)]	Display in the		Selection Criteria specified in Section 3.1.1
				Control Room	Remote Shutdown Station	
1	Neutron flux	P_N	at least from 10^{-6} to 10^{-3}	x	x	(1) a)
2	Boric acid concentration of the water of the containment vessel sump [2)]	ppm	50 to 2600	x	–	(1) a)
3	Coolant temperature at the inlet and at the outlet of each loop	°C	50 to 400	x	x	(1) b)
4	Core outlet temperature	°C	100 to 1000	x	–	(1) b)
5	Fluid level in the pressurizer	m	cf. para. (1) Section 3.3	x	x	(1) b)
6	Fluid level in the reactor pressure vessel	m	cf. para. (2), item g) Section 3.3	x	x	(1) b)
7	Fluid level of secondary side in each steam generator	m	cf. para. (1) Section 3.3	x	x	(1) b)
8	Water temperature in the containment vessel sump	°C	10 to 150	x	–	(1) b)
9	Fluid level in the containment vessel sump	m	cf. para. (1) Section 3.3	x	–	(1) b)
10	Departure from nucleate boiling	K	50 to 0	x	–	(1) b)
11	Water temperature in the fuel pool	°C	10 to 150	x	x	(1) b)
12	Pressure in the reactor coolant system	bar	1 to 250	x	x	(1) c)
13	Pressure in the secondary side of each pressurizer	bar	1 to 150	x	x	(1) c)
14	Pressure inside containment vessel (differential pressure measurement) [3)]	bar	-0.5 to 5.5	x	x	(1) d)
15	Pressure inside annulus (differential pressure measurement) [3)]	bar	cf. para. (2), item e) Section 3.3	x	x	(1) d)
16	Hydrogen concentration in the containment vessel	vol. %	0 to 5	x	–	(1) e)
17	Air temperature in the upper region of the containment vessel	°C	20 to 160	x	x [4)]	(1) e)
18	Absorbed dose rate in the containment vessel	Gy/h	10^{-3} to 10^{4}	x	x [4)]	(1) e)
19	Discharge of radioactive substances with the exhaust stack air	Details are specified in safety standard KTA 1503.2				(2)
20	Volumetric flow of the exhaust stack air					(2)
21	Wind direction [5)]	Details are specified in safety standard KTA 1508				(2)
22	Wind velocity [5)]					(2)

1) The requirements for the measurement ranges in this table are specified on the basis of both Section 3.3 and the state of science and technology as well as on the system-engineering design of nuclear power plants.

2) Sampling at pre-installed sampling locations combined with a laboratory evaluation is permissible.

3) The measurement location for the reference pressure is dependent on the individual nuclear power plant.

4) A display outside of the remote shutdown station is permissible in well substantiated cases.

5) An estimation of the radiological effects on the environment is possible only after measuring or determining further meteorological parameters, e.g., diffusion category. Details are specified in safety standard KTA 1508

Table 3-1: Accident overview display equipment in nuclear power plants with pressurized water reactors

No.	***Measurement Parameter***	***Unit or Reference Value***	***Measurement Range*** [1]	***Display in the***		***Selection Criteria specified in Section 3.1.1***
				Control Room	***Remote Shutdown Station***	
1	Neutron flux	P_N	at least from 10^{-6} to 10^{-3}	x	x	(1) a)
2	Fluid level in the reactor pressure vessel	m	cf. para. (1) Section 3.3	x	x	(1) b)
3	Fluid level in the steam suppression pool	m	cf. para. (1) Section 3.3	x	x	(1) b)
4	Water temperature in the steam suppression pool	°C	10 to 110	x	x	(1) b)
5	Fluid level in the containment vessel	m	cf. para. (1) Section 3.3	x	–	(1) b)
6	Water temperature in the fuel pool	°C	20 to 110	x	x	(1) b)
7	Pressure in the reactor pressure vessel	bar	1 to 115	x	x	(1) c)
8	Pressure inside containment vessel (differential pressure measurement) [2]	bar	- 0.5 to 5.5	x	x	(1) d)
9	Pressure inside annulus gap (differential pressure measurement) [2]	mbar	- 30 to 30	x	–	(1) d)
10	Pressure inside reactor building (differential pressure measurement) [2]	mbar	cf. para. (1) Section 3.3	x	x	(1) d)
11	Pressure inside turbine building (differential pressure measurement) [2]	mbar	cf. para. (1) Section 3.3	x	–	(1) d)
12	Air temperature in the containment vessel	°C	20 to 150	x	x [3]	(1) e)
13	Hydrogen concentration in the containment vessel [4] [5]	vol. %	0 to 5	x	–	(1) e)
14	Temperature inside the reactor building	°C	20 to 110	x	x	(1) e)
15	Absorbed dose rate in the containment vessel	Gy/h	10^{-3} to 10^{4}	x	x [3]	(1) e)
16	Absorbed dose rate in the turbine building	Gy/h	10^{-6} to 10^{0}	x	x	(2)
17	Discharge of radioactive substances with exhaust stack air	Details are specified in safety standard KTA 1503.2				(2)
18	Volumetric flow of the exhaust stack air					(2)
19	Wind direction [6]	Details are specified in safety standard KTA 1508				(2)
20	Wind velocity [6]					(2)

1) The requirements for the measurement ranges in this table are specified on the basis of both Section 3.3 and the state of science and technology as well as the system-engineering design of the nuclear power plants.

2) The measurement location for the reference pressure is dependent on the individual nuclear power plant.

3) A display outside of the remote shutdown station is permissible in well substantiated cases.

4) Sampling at pre-installed sampling locations combined with a laboratory evaluation is permissible.

5) In case of an inerted pressure reduction system, no measurement of the hydrogen concentration is required.

6) An estimation of the radiological effects on the environment is possible only after measuring or determining further meteorological parameters, e.g., diffusion category. Details are specified in safety standard KTA 1508

Table 3-2: Accident overview display equipment in nuclear power plants with boiling water reactors

(2) Devices of the accident overview display equipment located in the remote shutdown station shall basically be supplied from the same emergency power facility as the remote shutdown station.

(3) The power supply for the accident overview display equipment which, on account of their measuring location, are not protected against external events may be supplied by emergency power facilities, the protection against external events of which corresponds to that of these measuring devices.

3.6 Tests and Inspections

3.6.1 Suitability test

It shall be proven to the authorized expert that the plant-specific suitability of the accident overview display equipment (e.g., measuring transmitters and transducers, cables, wiring and connections) are in accordance with the present safety standard. It is permissible to supply this proof on the basis of a physical test of the accident overview display equipment (types) to be performed in the presence of an authorized expert.

3.6.2 Design review

It shall be demonstrated to the authorized expert on the basis of binding design documents (e.g., specifications, assembly and arrangement drawings, data sheets) that the planned accident overview display equipment meets all plant-specific requirements it is required to meet.

3.6.3 Factory test

It shall be demonstrated by means of factory tests that the accident overview display equipment has been manufactured as specified. These tests shall be performed by plant experts or under their responsibility. In well substantiated cases, these tests shall be performed in the presence of authorized experts.

3.6.4 Tests during pre-nuclear operation

In the course of pre-nuclear operation of the plant, tests (i.e., acceptance and functional tests) of the accident overview display equipment shall be performed by the manufacturer or operating utility in the presence of authorized experts. In this context, it shall be verified that mechanical design and function of the accident overview display equipment are as specified in the design reviewed documents.

3.6.5 Periodic inservice inspections

Periodic inservice inspections shall be performed by the operating utility. The test interval between inspections shall normally not exceed one year. Whether or not the presence of authorized experts is required shall be specified in the testing manual.

3.6.6 Tests after replacement or repair

Following the replacement or repair of parts of the accident overview display equipment, the operating utility shall perform tests to an extent corresponding to that part of the factory tests that would confirm that the repair has been carried out properly.

3.6.7 Test documentation

Test certificates or test records shall be considered as proof that the tests in accordance with Sections 3.6.1 and 3.6.3 through 3.6.6 have been performed. The documents shall contain all data in connection with the tests.

3.7 Maintenance

(1) In the case of a loss of informational data for which no redundancy or alternate solution is available, the corresponding measuring channel shall be repaired within 100 hours. If this is not possible, it shall be determined on the basis of an analysis to be made prior to plant commissioning whether or not the loss of data is acceptable for a duration longer than 100 hours. If this is not the case, the reactor shall be shut down after these 100 hours.

(2) The maintenance tasks shall be carried out by qualified personnel.

(3) Only such components are permitted to be used for repairs the suitability and proper manufacture of which have been verified by tests in accordance with Section 3.6.

(4) The failures detected, their causes and the type of repair shall be documented. The exact time the failure was detected and of the beginning and end of the repair task shall be documented.

(5) The accident overview display equipment shall normally be subjected to maintenance in regular intervals.

4 Accident Detail Display Equipment

4.1 Measurement Parameters

(1) The operating instrumentation shall also be employed as accident detail display equipment.

(2) The measurement parameters shall be specified during the design of the respective process-engineering system.

4.2 Display of Measured Values

The accident detail display equipment in the control room and in the remote shutdown station shall normally be arranged to correlate with the display of the respective process-engineering system. The display shall be sufficiently accurate and have a sufficient resolution.

4.3 Requirements Regarding the Measuring Devices

(1) The measuring devices of the accident detail display equipment shall be designed such that, with respect to the external conditions to be found at the place of installation, it meets the same requirements as the system which it is provided for to monitor.

(2) A single-channel design is permissible; a redundancy is not required.

(3) Failed measuring devices of the accident detail display equipment shall be repaired within the maximum down-time permissible for the system which it is provided for to monitor.

4.4 Tests and Inspections

The extent of the tests and inspections to be performed and the intervals between these tests shall be specified to be in accordance with the requirements for the process-engineering system to which the instrumentation is correlated.

5 Wide Range Display Equipment

5.1 Measurement Parameters

In order to obtain information on whether or not plant parameters are approaching the design values of the activity retention barriers and also whether or not these design values have been exceeded in the case of unforeseen sequences of events which cannot be taken into account in the design of the plant, the measurement parameters listed in **Table 5-1** shall be determined for pressurized water reactors and those listed in **Table 5-2** for boiling water reactors.

No.	Measurement Parameter	Unit	Measurement Range [1)]
1	Core outlet temperature	°C	100 to 1000
2	Pressure in the reactor coolant system	bar	1 to 400
3	Pressure in the containment vessel (measurement of differential pressure) [2)]	bar	-1 to 15
4	Fluid level in the fuel pool	m	empty to full
5	Fluid level in the containment vessel sump	m	Measurement range shall be specified depending on the plant [3)]
6	Discharge of radioactive substances with exhaust stack air	Details are specified in safety standard KTA 1503.2	
7	Volumetric flow of the exhaust stack air		
8	Absorbed dose rate in the containment vessel	Gy/h	10^{-3} to 10^{5}
9	Radioactive substances in the circulating water outfall culvert or circulating water outfall structure [4)] – activity concentration	Bq/m^3	10^{4} to 10^{8}

1) The requirements for the measurement ranges are specified on the basis of the state of science and technology as well as the system-engineering design of the nuclear power plants.

2) The measurement location for the reference pressure is dependent on the individual nuclear power plant.

3) The measurement range shall normally be chosen such that definite information is obtained on whether or not the design limits are exceeded.

4) One continuous gamma measurement facility in the circulating water outfall culvert or circulating water outfall structure is sufficient. Requirements regarding measurements of this measurement parameter are specified in safety standard KTA 1504.

Table 5-1: Wide range display equipment for nuclear power plants with pressurized water reactors

No.	Measurement Parameters	Unit	Measurement Range [1)]
1	Fluid level in the reactor pressure vessel	m	Bottom core edge to closure head flange
2	Pressure in the reactor pressure vessel	bar	1 to 250
3	Pressure in the containment vessel (measurement of differential pressure) [2)]	bar	-1 to 15
4	Fluid level in the fuel pool	m	Empty to full
5	Fluid level in the containment vessel	m	Measurement range shall be specified depending on the plant [3)]
6	Discharge of radioactive substances with exhaust stack air	Details are specified in safety standard KTA 1503.2	
7	Volumetric flow of the exhaust stack air		
8	Absorbed dose rate in the containment vessel	Gy/h	10^{-3} to 10^{5}
9	Radioactive substances in the circulating water outfall culvert or circulating water outfall structure [4)] – activity concentration	Bq/m^3	10^{4} to 10^{8}

1) The requirements for the measurement ranges are specified on the basis of the state of science and technology as well as the system-engineering design of the nuclear power plants.

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3) The measurement range shall normally be chosen such that definite information is obtained on whether or not the design limits are exceeded.

4) One continuous gamma measurement facility in the circulating water outfall culvert or circulating water outfall structure is sufficient. Requirements regarding measurements of this measurement parameter are specified in safety standard KTA 1504.

Table 5-2: Wide range display equipment for nuclear power plants with boiling water reactors

5.2 Requirements Regarding Measuring Devices

(1) As far as technically feasible, the measuring devices shall normally be installed outside of the containment vessel.

(2) The equipment shall normally be designed to withstand the external conditions occurring at its respective place of installation. At least those external conditions shall be taken into account that were determined in the design basis accident analysis.

Notes:

(1) Due to the available margins of safety in the design basis accident analyses and in the design of components, it is expected that the equipment will remain functional even under more severe environmental conditions than the conditions resulting from design basis accidents.

(2) The measurement range limits of the wide range display equipment may extend beyond the design specifications of the monitored components.

(3) The measurement parameters shall be displayed and recorded in the control room and shall normally also be displayed and recorded in the remote shutdown station. The measurement values shall be displayed in the same way as those from the accident overview display equipment.

(4) No redundancies are required for the wide range display equipment.

(5) The wide range display equipment may be a part of the accident overview display equipment, provided, it also meets the requirements of the accident overview display equipment.

5.3 Sampling

5.3.1 Sampling at the containment vessel

Sampling points shall be provided for monitoring the radioactivity concentration and hydrogen concentration in the atmosphere and the sump of the containment vessel of pressurized and boiling water reactors. These sampling points shall normally be designed for an activity concentration in the sample of 3.7×10^{14} Bq/m³.

5.3.2 Sampling at the exhaust air system

(1) The sampling points shall normally be selected such that sampling will be possible during a design basis accident. Length and mechanical design of the sampling lines shall be specified accordingly.

(2) Those locations that must be accessible for sampling shall be selected and shielded such that the local dose rates at these locations will not exceed the reference design value of 10 mSv/h for an assumed radioactivity concentration of 10^{13} Bq/m^3 radioactive noble gases in the vent air.

(3) The ventilating ducts inside regions protected against external events shall be provided with connection lines that lead out of these regions and can be used for sampling.

5.4 Power Supply

(1) The wide range display equipment shall normally be supplied from an uninterruptible emergency power supply; the energy storage of this power supply shall comprise batteries in parallel and include rectifying equipment. Additional requirements for such an emergency power supply are specified in safety standard KTA 3703. An uninterruptible power supply is not required in those cases where, on account of its measurement tasks, the non-availability of the equipment is permissible for a short period of time.

(2) Devices of the wide range display equipment located in the remote shutdown station shall normally be supplied from the emergency power facility allocated to the remote shutdown station.

(3) The power supply for the measuring devices of the wide range display equipment which, on account of their measuring location, are not protected against external events may be supplied by emergency power facilities, the protection against external events of which corresponds to that of these measuring devices.

5.5 Maintenance

(1) The maintenance tasks shall be carried out by qualified personnel.

(2) Only such components are permitted to be used for repairs, the suitability and proper manufacture of which have been demonstrated.

(3) The failures detected, their causes and the type of repair required shall be documented. The exact time the failure was detected and of the beginning and end of the repair task shall be recorded.

(4) The wide range display equipment shall normally be subjected to maintenance in regular intervals.

6 Accident Recording Equipment

6.1 Measurement Parameters

6.1.1 Selection criteria regarding control room area

(1) At least the following analog and binary data shall be recorded in the control room area:

a) the measurement values of the accident overview display equipment as listed in **Table 3-1** or **Table 3-2**,

b) the measurement values of the accident detail display equipment,

c) the measurement values of the wide range display equipment as listed in **Table 5-1** or **Table 5-2**,

d) the Class S alarms in accordance with safety standard KTA 3501,

e) the Class I alarms in accordance with safety standard KTA 3501,

f) the Class II alarms in accordance with safety standard KTA 3501 in so far as they can be correlated to the definition of the accident recording equipment (cf. para. (3) Section 2).

g) selected binary data from the reactor protection system.

(2) The recording of these data shall be such that the following information is discernible:

a) the point in time and the type of component and system malfunction which triggered the design basis accident,

b) the point in time, the reason for, and the effectiveness of automatically or manually initiated countermeasures,

c) the amplitude and duration of accident-induced external conditions affecting safety related components and equipment.

(3) The data to be recorded even during a shut-down of the nuclear power plant shall be specified.

6.1.2 Selection criteria regarding remote shutdown station area

(1) At least those data shall be recorded in the remote shutdown station which make it possible to subsequently identify

a) whether or not the design limits of the pressure boundary and of the containment vessel were exceeded,

b) the integral effectiveness of automatically or manually initiated countermeasures, and

c) whether or not important design limits were approached or exceeded in the case of unpredictable sequences of events.

(2) The measured values of the accident recording in the remote shutdown station shall comprise:

a) the measured values of the accident overview display equipment as listed in **Table 3-1** or **Table 3-2**,

b) the measured values of the wide range display equipment as listed in **Table 5-1** or **Table 5-2**.

6.2 Requirements

6.2.1 Accident recording equipment for the control room area

(1) The measurement parameters selected for display in the control room area shall basically be recorded automatically. A manual recording is permissible in well substantiated cases, provided, the sufficiently exact recording and chronological resolution of the measurement values are ensured.

(2) If a central data recording system is used to record all the measurement parameters specified under Section 6.1.1, it shall be ensured that sufficient data is retained in the event of the occurrence of a random failure.

Note:

It is not required to design the central data recording systems against external events.

(3) If this central data recording system is also used for other tasks, it shall be demonstrated that no loss of data will occur within the accident documentation as a result of these other tasks.

(4) If the progressive displays specified under para. (2) Section 3.2 are designed as recording devices, it is permissible to use them as part of the accident recording system, provided, they meet the requirements of this section.

(5) The accident recording equipment shall be designed such that the selected measurement parameters are recorded clearly and with the required resolution regarding time, amplitude and accuracy, and in the chronologically correct sequence.

(6) For each measured value, it shall be possible to identify the corresponding date and point in time from the corresponding recording with such exactness that a chronological sequence can be established.

(7) It shall be possible to clearly correlate measurement unit, scale and measurement parameter to each other.

(8) The devices of the accident recording equipment in the control room area are only required to remain functional under conditions of specified normal operation.

6.2.2 Accident recording equipment for the remote shutdown station area

(1) The devices of the accident recording equipment in the remote shutdown station shall be designed to withstand the external conditions resulting from external events. In the course of the actual mechanical effects, it is permissible that the recording shows distorted measured values.

(2) In those nuclear power plants where it is assumed that the control room can fail, the measurement parameters to be detected by the accident overview display equipment and by the wide range display equipment shall continue to be recorded automatically in the remote shutdown station for at least 10 hours after the occurrence of a design basis accident. After this period, manual recording is permissible, provided, the sufficiently exact recording and chronological resolution of the measurement values are ensured.

(3) If the progressive displays specified under para. (2) Section 3.2 are designed as recording devices, it is permissible to use them as part of the accident recording system, provided they meet the requirements of this section.

(4) The accident recording equipment shall be designed such that the selected measurement parameters are recorded clearly and with the required resolution regarding time, amplitude and accuracy, and in the chronologically correct sequence.

(5) For each measured value, it shall be possible to identify the corresponding date and point in time from the corresponding recording with such exactness that a chronological sequence can be established.

(6) It shall be possible to clearly correlate the measurement units, the scale and the measurement parameter to each other.

6.3 Power Supply

(1) The accident recording equipment shall normally be supplied from an uninterruptible emergency power supply; the energy storage of this power supply shall comprise batteries in parallel and include rectifying equipment. An uninterruptible power supply is not required in those cases where, on account of its measurement tasks, the non-availability of the equipment is permissible for a short period of time. Additional requirements for such an emergency power supply are specified in safety standard KTA 3703.

(2) Devices of the accident recording equipment located in the remote shutdown station shall basically be supplied from the same emergency power facility as the remote shutdown station.

(3) The power supply for the measuring devices of the accident recording equipment which, on account of their measuring location, are not protected against external events may be supplied by emergency power facilities, the protection against external events of which corresponds to that of these measuring devices.

6.4 Tests and Inspections

(1) The concept of the data flow to individual equipment as well as the compliance of the equipment with the requirements specified in this safety standard shall be tested and inspected by authorized experts. Tests and inspections shall be performed in the course of plant commissioning. These shall demonstrate that construction and function of the accident recording equipment correspond to the design reviewed documents.

(2) Those devices of the accident recording equipment which will be activated only in the event of a design basis accident shall normally be tested in annual intervals.

6.5 Maintenance

6.5.1 Equipment of the control room area

(1) Any failures leading to a numerical reduction of the measurement parameters that are recorded shall normally be repaired as quickly as possible. The reactor plant shall be brought into a safe condition if a situation arises where only the minimum of required data listed in **Tables 3-1** and **5-1** or in **Tables 3-2** and **5-2** is recorded,.

N o t e :

The reactor plant can be brought into a safe condition, e.g., by carrying out the repair without delay.

(2) Any repairs and the duration in which the equipment was unavailable shall be documented.

6.5.2 Equipment of the remote shutdown station area

(1) In case the equipment for one or more of the measurement parameters specified under Section 6.1.2 fails, the repairs shall be carried out within a maximum of 100 hours from the time the failure was detected. If this is not possible it shall be determined on the basis of an analysis made prior to commissioning whether or not the loss of data is acceptable for a duration longer than 100 hours. If this is not the case, the reactor shall be shut down after these 100 hours.

(2) Any repairs and the duration in which the equipment was unavailable shall be documented.

Appendix A

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.)

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 1 of the Act of July 31, 2011 (BGBl. I, p. 556)
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 changed by Article 1 of the Act of October 4, 2011 (BGBl. I, p. 2000)
Safety Criteria	(1977-10)	Safety criteria for nuclear power plants of 21 October 1977 (BAnz. No. 206 of November 3, 1977)
Design Basis Accident Guidelines	(1983-10)	Guidelines for the assessment of the design of nuclear power plants with pressurized water reactors against design basis accidents as defined in Sec. 28, para. 3 StrlSchV (Design Basis Accident Guidelines) of October 18, 1983 (Addendum to BAnz. No. 245 of December 31, 1983)
KTA 1503.2	(1999-06)	Monitoring the discharge of gaseous and aerosol-bound radioactive substances; Part 2: Monitoring the stack discharge of radioactive substances during design basis accidents
KTA 1504	(2007-11)	Monitoring and assessing of the discharge of radioactive substances in liquid effluents
KTA 1508	(2006-11)	Instrumentation for determining the dispersion of radioactive substances in the atmosphere
KTA 2206	(2009-11)	Design of nuclear power plants against damaging effects from lightning
KTA 3403	(2010-11)	Cable penetrations through the reactor containment vessel of nuclear power plants
KTA 3407	(2010-11)	Pipe penetrations through the reactor containment vessel
KTA 3501	(1985-06)	Reactor protection system and monitoring equipment of the safety system
KTA 3701	(1999-06)	General requirements for the electrical power supply
KTA 3703	(1999-06)	Emergency power facilities with batteries and ac/dc converters in nuclear power plants
KTA 3705	(2006-11)	Switchgear, transformers and distribution networks for the electrical power supply of the safety system in nuclear power plants
KTA 3904	(2007-11)	Control room, remote shutdown station and local control stations in nuclear power plants