Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 1301.1 (2022-11)

Radiation Protection Considerations for Plant Personnel in the Design and Operation of Nuclear Power Plants
Part 1: Design

(Berücksichtigung des Strahlenschutzes der Arbeitskräfte
bei Auslegung und Betrieb von Kernkraftwerken;
Teil 1: Auslegung)

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KTA SAFETY STANDARD

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2012-11 (BAnz. of January 23, 2013)
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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 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 the AtG, the Radiation Protection Act (StrlSchG) and the Radiation Protection Ordinance (StrlSchV) as well as further detailed in the Safety Requirements for Nuclear Power Plants (SiAnf) and the Interpretations of the SiAnf.

(2) The safety standard series KTA 1301 specifies protective-goal oriented requirements regarding the design of nuclear power plants and the design of the administrational and technical measures that are considered essential for the protection against exposure of the persons working in a nuclear power plant; they specifically address requirements for the design of the workplace, workflow and work environment. In this context, particular attention is paid to StrlSchG, Sec. 8.

The safety standard series KTA 1301 comprises

Part 1: Design (*the present standard)*, and

Part 2: Operation.

1 Scope

(1) This safety standard applies to the planning of buildings, systems and components within the permanent controlled area of nuclear power plants with light-water reactors and of that part of the sanitary tract bordering directly on the controlled area.

(2) The requirements concern both, the specified normal operation (Sections 3 through 8) as well as design basis accidents (Section 9).

2 Definitions

(1) Sanitary tract

The sanitary tract in a nuclear power plant comprises all those rooms of the entry and exit tracts leading to the permanent controlled area that contain the necessary monitoring equipment (e.g., controlled-area gate) as well as the sanitation facilities for the personnel and the rooms where the work wear and protective clothing is issued.

Inside the controlled area is that part of the sanitary tract that comprises the changing rooms, the rooms for issuing undergarments, the showers and washrooms for use by the persons leaving the controlled area.

Directly bordering on the controlled area (i.e., part of the supervised area) is that part of the sanitary tract that comprises recreation rooms, changing rooms, washrooms, showers and lavatories.

(2) Assembly opening

An assembly opening is an opening (e.g., hatch, walls of dry-stacked blocks, door) necessary for the assembly and disassembly of a component or component part.

(3) Controlled-area gate

The controlled-area gate comprises

a) the entry area to the controlled area with the equipment for checking the access authorization and issuing the direct-reading personal dose meters, and

b) the exit area from the controlled area with the equipment for checking person for contamination and for taking back the direct-reading dose meters and recording the respective dose.

The work place of the surveillance personnel in the controlled-area gate may lie within the controlled area or within the supervised area.

3 Rooms of the Controlled Area and Sanitary Tract

Note:
The controlled area in a nuclear power plant is an area with controlled access which includes all those areas in which the effective dose of the working personnel (as defined in StrlSchV) might exceed 6 mSv/a at individual locations.

3.1 General Requirements

3.1.1 Room classification

When planning the rooms, they shall be classified in accordance with DIN 25440.

3.1.2 Room arrangement

(1) It shall normally be possible to reach each room via traffic routes (in accordance with ArbStättV, Appendix 1.8).

(2) Rooms that cannot be reached via traffic routes shall basically be arranged such that they can be accessed only through rooms which are at most of the same class as the room to be accessed. Exceptions are permissible, provided, no considerable exposure is to be expected when passing through this room.

Note:
Requirements regarding the transportation of radioactive objects within nuclear power plants are specified in safety standard KTA 3604, and regarding rescue routes in safety standard KTA 2101.1.

3.1.3 Room design

3.1.3.1 Access openings

(1) Access openings to rooms (e.g., doors, narrow passage ways, ceiling and floor hatches, pressure hatches) shall basically be designed such that they do not determine the room classification of the preceding room from which the respective rooms are accessed. Exceptions are permissible where the purpose of the preceding room is mainly to provide this access.

Notes:

(1) Possible design measures are, e.g.,

a) suitable arrangement with respect to the radioactive components,

b) labyrinths (radiation traps),

c) radiation-protection doors, and

d) walls of dry-stacked blocks.

(2) Requirements for personnel and equipment airlocks are specified in safety standard KTA 3402 and KTA 3409, respectively.

(2) Access openings, furthermore, shall normally be arranged such that the work places in the room can be reached passing through regions with the lowest possible local dose rate.

(3) Moreover, access openings shall be designed to be large enough, both, for the auxiliary equipment assumed in the planning stage to be necessary (e.g., compressed-air respirators, tools, transportation means) as well as for the wear-and-tear parts expected to need replacing (cf. **Table 3-1**).

(4) Access openings shall not be walled up. The closing of access openings with mountable walls (e.g., of dry stacked blocks or slabs) is only permissible in those rooms which are not expected to be entered more than once a year.

Access Openings	*Clear Width*	*Clear Height*
Doors	in accordance with ASR A 1.7	
Narrow passage way	0.60 m	1.40 m
Ceiling or floor hatches	0.70 m	0.60 m (0.80 m) [1)]
Pressure hatches	0.57 m	1.40 m (lower edge at 0,25 m above ground)
[1)] Minimum dimension with regard to the use of self-contained respirators if it is not possible to supply air via flexible hoses for reasons of, e.g., their length.		

Table 3-1: Minimum dimensions of access openings

3.1.3.2 Assembly openings

The arrangement and design of assembly openings shall be such that the disassembly and re-assembly of radioactively contaminated or activated components is possible taking into account the necessary assembly equipment and radiation shielding (cf. Section 4.1.2, para. (2)). In this context, dose minimization shall also be taken into account with regard to the transportation route. The assembly openings shall normally be designed such that they do not determine the room classification of the adjoining rooms.

3.1.3.3 Wall penetrations

Wall penetrations shall normally be designed such that they do not determine the room classification of the adjoining rooms.

Note:

Possible measures are, e.g.,

a) slits and wall penetrations at an inclined angle,
b) covers or lead-wool packings for wall penetrations, and
c) locating them away from the generally accessible area (e.g., under the ceiling).

3.1.3.4 Walls, shielding

(1) Rooms shall be shielded such that the dose rate in a room caused by the radiation from adjacent rooms does not exceed 20 % of the upper limit of the respective room classification.

(2) The following locations shall be shielded such that the local dose rate does not exceed 3 µSv/h:

a) workplaces expected to be manned more than 1000 hours per year (e.g., control station of the liquid waste processing system, controlled-area gate),
b) sanitary tract (cf. Section 3.2), and
c) first-aid station (cf. Section 3.3).

(3) Certain parts of frequently used traffic routes (e.g., main passageway in the reactor auxiliary building) shall be shielded such that the local dose rate in the generally accessible area of the respective rooms does not exceed 10 µSv/h up to a height of 2 m above the accessible levels.

3.1.3.5 Load-bearing capacity of floors

(1) The traffic routes shall be designed with regard to their load-bearing capacity taking the expected transportation means into account, as well as the loads anticipated during assembly and operation together with the necessary shielding.

(2) If a local dose rate of more than 1 mSv/h is anticipated at a distance of 0.5 m from a component, the load-bearing capacity of the floors, platforms and ceilings shall basically be designed such that additional shielding can be installed. In this context, a line load of 10 kN/m shall be assumed for a maximum length of 3 m at the most unfavorable point.

(3) Exemptions from the load-bearing capacity specified in para. (2) are permissible, provided, the shielding effect is ensured by a sufficient spatial separation of the components.

(4) The design of floors and platforms for dynamic loadings does not, basically, have to take additional shielding into account. Additional shielding does, however, have to be taken into account where such shielding is installed on a long-term basis and a failure of the floors or platforms

a) during power operation could cause damage to safety-related components, or
b) while work tasks are performed during plant shut-down on one redundancy could damage a necessary redundancy other than the one being worked on.

(5) The floors of radioactive waste storage facilities not charged by remote control, as well as the floors of the hot workshop, of the decontamination room and of the stowage room for parts and components shall be designed for a line load of 10 kN/m over a maximum length of 5 m (cf. safety standard KTA 3604).

(6) The respective load-bearing capacity shall be locally marked on the floors and platforms.

3.1.3.6 Room dimensions

(1) Rooms shall normally be designed such that maintenance tasks can be speedily executed. In this context, the shielding of the components as well the necessary additional shielding (cf. Section 4.1.1 and Section 7) and other required pre-requisites for the work task (e.g., full protection clothing) shall be taken into account.

(2) Care shall be taken to ensure that sufficient space is available near the disassembly area for setting down temporarily disassembled components and component parts and for the related radiological protection measures (cf. Section 4.1.1, para. (5)).

(3) A stowage room serving as interim storage location for the disassembled components and parts in accordance with safety standard KTA 3604, Sec. 5.1, or equivalent set-down areas which meet the same requirements shall be provided.

3.1.4 Room furnishings and fittings

3.1.4.1 Surfaces

(1) The walls, ceilings and floors of rooms inside the containment vessel shall be provided with surfaces that are easy to decontaminate. Rooms of the controlled area outside of the containment vessel that house pressurized systems carrying radioactive fluids shall be treated in like manner. In rooms of the controlled area outside of the containment vessel in which unsealed radioactive substances are expected to be handled, an easily decontaminable surface up to a height of 2 meters is sufficient.

Note:

Requirements regarding the methods for testing and evaluating radioactively contaminated surfaces for their decontaminability are specified in DIN 25415-1 and DIN 55991-1.

(2) The floor of the sanitary tract, both inside and outside of the controlled area, as well as the base board region of the respective walls up to a height of 10 cm shall be easily decontaminable.

(3) The surface coatings, especially in the case of the floors of traffic routes, shall be seamless, waterproof, and sufficiently pressure and wear resistant.

3.1.4.2 Room drainage

(1) Rooms where components containing radioactive fluids are installed shall basically be provided with room drainage systems feeding into the building drainage tank (building sump). It shall be ensured by adequate floor gradients that any fluids released are conducted directly to the room drainage system.

(2) A room drainage system is not required from those vessel rooms where the entire room is constructed as a waterproof trough and this trough can hold the contents of the largest vessel in the room. In this context, adjoining vessel rooms may be constructed as one mutual trough. It shall be possible to conduct any fluid from the trough into tanks (e.g., by means of mobile submersible pumps or by means of a permanently laid pipe line to a collection tank).

3.1.4.3 Furnishings

(1) The rooms shall be equipped with the required number of supply connections for power, compressed-air and water.

(2) Lighting fixtures in the rooms shall be designed for low maintenance and shall be fitted with easily replaceable lamps.

(3) In the case of components which are expected to require maintenance but where no permanently installed lifting equipment is provided, provisions shall be made for temporary auxiliary disassembly equipment. This is not required if the conditions specified in **Table 3-2** are met.

Mass of Component or Part	***Boundary Condition for Handling***
up to 15 kg	none
up to 25 kg	easily accessible for one person and assembly location not above chest height
up to 50 kg	easily accessible for two persons and assembly location not above chest height up

Table 3-2: Handling without auxiliary disassembly equipment

3.2 Sanitary Tract

(1) The sanitary tract shall be shielded such that the local dose rate does not exceed 3 μSv/h.

Note:
A lower local dose rate may be required locally for the places of installation of the contamination monitors at the exit of the controlled area and for the storage places of the official dose meters.

(2) With regard to the number of plant personnel, contract personnel and visitors, the sanitary tract shall be dimensioned such that sufficient space is available for changing and washing and for a sufficient number of personnel contamination monitors. Provisions for a gender specific separation of the rooms shall be made.

(3) A reference value of 800 persons shall be assumed in planning the sanitary tract (i.e., plant personnel – 200 persons; contract personnel – 600 persons).

(4) A contamination checkpoint shall be provided for small objects (e.g., tools) which are intended to be brought out of the controlled area. A decontamination facility should be provided close by.

(5) To reduce the danger of inadvertent spreading of contamination, that part of the sanitary tract within the controlled area adjoining the entry and exit monitoring section shall be designed such that the persons entering and those exiting can be guided spatially separated from each other.

(6) Space shall be provided for a preliminary check of persons intending to leave the controlled area to determine whether or not any decontamination measures are required.

3.3 First-Aid Station

(1) A special room inside the controlled area shall be designated for first-aid measures and for the initial medical treatment of injured persons. This room shall be located in an easily accessible area with little danger of contamination. Care shall be taken to ensure that an easy and gentle evacuation of injured persons is possible.

(2) The first-aid station shall be designed to be easily accessible with a stretcher. It shall be furnished with all facilities and equipment necessary for first-aid and the initial medical treatment and shall be dimensioned accordingly (cf. ArbStättV, Appendix 4.3).

Notes:
a) A room with a floor area of 4 m x 5 m and a clear height of 2.80 m is suited to contain the necessary facilities and equipment.
b) In line with ASiG, Sec. 3, the equipment in the first-aid station is specified in joint consultation between the authorized medical doctor (cf. StrlSchV, Sec. 175) and the licensee.
c) The first-aid station in full compliance with ASR A 4.3 is located outside of the controlled area.

(3) A facility shall be provided for a simple decontamination of injured persons. This facility may be located outside of the first-aid station.

3.4 Spatial Needs Regarding Health Physics

The design shall take the following spatial needs regarding health physics into consideration:

a) rooms for preparing, performing and evaluating radioactivity measurements,
b) space for the inservice inspection of mobile radiation monitoring equipment,

   Note:
   The shielding of the rooms under items a) and b) is dependent on the necessary measurement tasks.

c) space for storing the radiation monitoring equipment and auxiliary health physics material, and
d) space for storing the radioactive samples and the test and calibration sources.

3.5 Spatial Requirements Regarding Processing and Storing of Contaminated Parts

3.5.1 Decontamination compartment

The necessary decontamination compartments and space (cf. safety standard KTA 3604) shall be provided for the decontamination of disassembled components and component parts as well as of devices and tools.

3.5.2 Hot Workshop

A hot workshop (cf. safety standard KTA 3604) shall be provided inside the controlled area for handling and repair of radioactive components and component parts.

3.5.3 Separate storage location

A separate storage location (cf. safety standard KTA 3604, Sec. 5.4) shall be provided, both, for the contaminated component parts which have been repaired for re-use as well as for the contaminated tools.

4 Components

4.1 General Requirements

4.1.1 Component arrangement

(1) With regard to exposure during maintenance tasks, highly radioactive components shall normally be installed in their own separate rooms. If essential considerations stand against this approach (e.g., structural or process engineering considerations), other measures shall be considered already in the planning phase that would reduce exposure (e.g. distance, special tools, shielding).

(2) Components and their parts that are expected to require frequent maintenance shall normally be arranged in their rooms such that they can be reached with only minimal exposure.

(3) With regard to maintenance on components, the required space for performing these maintenance tasks shall be provided. In this context, particular attention shall be paid to special tools (e.g., lift trucks or manipulators) and to the necessary additional shielding.

(4) Components should be properly arranged such that work tasks can be ergonomically performed (cf. Section 7).

(5) Care shall be taken to ensure that sufficient space is available near the components to be able to set down and provide interim storage for the disassembled parts during maintenance tasks (cf. Section 3.1.3.6, para. (2)).

(6) Components should be arranged such that any necessary mobile shielding can be quickly brought in and set up with the least possible hindrances. In this context, preferably pre-assembled shielding shall be taken into consideration.

(7) Auxiliary and monitoring equipment should be arranged such that they are shielded from highly radioactive components.

(8) In the case of drives of highly radioactive components (e.g., agitators in concentrate storage tanks, or coolant pumps of pressurized water reactors), a shielding should be provided between drive and component unless this is inadvisable for structural or other design reasons.

(9) Connected pipes and equipment attached to pumps and vessels of systems containing radioactive substances shall be routed such that any required maintenance tasks on the respective components can be performed as far as possible without having to disassemble the connected pipes and attached equipment.

4.1.2 Design of components

(1) Components in regions of a high local dose rate shall be designed with particularly low maintenance requirements (e.g., long lifetimes of wear parts) and such that any maintenance tasks can be easily performed.

(2) If the maintenance of components requires their periodic disassembly and re-assembly, the assembly equipment required and, if necessary, shielded transportation equipment shall be considered already in the desing phase.

(3) With regard to reducing the exposure during inservice inspections, the mechanical design of components shall be such that the test set-up and testing times are as short as possible.

N o t e :
It is assumed that when preparing the testing schedule for inservice inspections and the testing instructions, the need for the inservice inspections and the resulting collective dose are carefully weighed against each other.

(4) With regard to short test set-up and testing times, the following points, among other, shall to be taken into account:

a) Welds subject to inservice inspections shall be few in number, conveniently arranged and requiring only a minimum of preparatory activities.

b) Remotely operated testing equipment shall be used for components of the reactor coolant pressure boundary.

c) Alternative tests in accordance with BetrSichV, Sec. 16, shall be taken into consideration if these can reduce exposure, provided, the test objective is achieved (e.g., pressure test and the use of remote cameras instead of an internal visual inspection).

(5) It shall be possible to purge components or system sections in which significant amounts of non-adhering deposits of radioactive substances are anticipated (e.g., vessels of the plant auxiliary system, system sections containing concentrates or ion-exchanger resins) in order to wash out these substances.

(6) Components carrying radioactive media shall be routed and their drainage points selected such that the components can be easily and completely drained. Suitable purging fittings shall be provided. The number of drainage points shall be kept to a minimum.

(7) With regard to the quality of the inner surface of components containing radioactive substances, particular attention shall be paid to a low surface roughness and to preventing weld seam protrusions or edge misalignments.

(8) In regions with a local dose rate higher than 100 µSv/h, the controls that have to be frequently operated (e.g., regularly on an inspection round) and the measuring devices that have to be frequently read off (e.g. thermometers, level and position indicators) should be designed such that they can be remotely operated or remotely read off from locations with a lower local dose rate.

(9) Manholes in vessels and vessel-like components shall be designed in accordance with AD 2000-Code A5, Sec. 2.2. If the work task requires full protection clothing, the manholes shall basically be designed with larger dimensions than the minimum measurements specified in AD 2000-Code A5. This does not apply to those vessels where larger dimensions are inadvisable for structural or other design reasons (e.g., steam generators).

4.1.3 Choice of materials

(1) The creation of activation products, such as cobalt-58, cobalt-60, silver-110m and antimony-124, shall be prevented by selecting structural materials with a low content of cobalt, nickel, silver and antimony.

N o t e :
The selection of the materials coming in contact with the reactor coolant has a significant effect due to the creation and deposition of activated corrosion products on the exposure of personnel (e.g. during maintenance tasks or waste treatment). In this context, the creation of Cr-51 from Cr-50, Co-60 from Co-59, Co-58 from Ni-58, Ag-110m from Ag-109 and Sb-124 from Sb-123 is of particular importance. The radionuclides Zr-95 and Nb-95, both activation products of Zr-94 (i.e., the structural material of the fuel-rod cladding) may also largely contribute to the nuclide mixture contaminating the systems. It is very difficult to describe in advance the behavior of the materials brought into the cooling circuit by erosion and corrosion and, in particular, the procedure of their deposition.
The selection of the materials can influence the operational mode of the plant and the water chemistry.

Factors contributing to the formation of radioactive corrosion products include:

a) high cobalt content weld surfaces (e.g. Stellites®) which are required for their resistance to abrasion,
b) steam generator tubes containing nickel (in the case of pressurized water reactors),
c) components of the feedwater system components (in the case of boiling water reactors),
d) austenitic surfaces of pipes and reactor pressure vessel internals,
e) structural parts of fuel assemblies,
f) control assemblies, in-core instrumentation,
g) metal alloys containing antimony of the pump bearings and of fittings, and
h) gaskets containing silver or antimony.

(2) Reference values for limiting the mass fraction of cobalt are listed in **Table 4-1**.

Component / Material	***Maximum Mass Fraction of Cobalt [ppm]***
Reactor pressure vessel / Basic material	300
Reactor pressure vessel, reactor coolant line and reactor coolant pumps, steam generator, pressurizer / Cladding	800
Steam generator tubes	1000
Reactor pressure vessel internals / Austenitic materials	800
Reactor core internals	1000
Reactor pressure vessel thermal insulation / Block insulation material	500

Table 4-1: Reference values for limiting the mass fraction of cobalt

4.1.4 Thermal insulation of components

(1) Those parts of thermally insulated components of the reactor coolant pressure boundary for which it is planned or expected that they will be subject to maintenance or inservice inspections shall be provided with an insulating encasement which shall be easy to remove and reinstall (e.g., by means of quick-action fasteners).

(2) Components outside of the reactor coolant pressure boundary shall be treated same as under para. (1) if it is planned that they will be subject to maintenance or inservice inspections and it is expected that the local dose rate will exceed 1 mSv/h at a distance of 0.5 m.

4.1.5 Component identification marking

Components shall be labeled by a durable, unambiguous and clearly visible mark to enable a rapid orientation. If the components are provided with shielding or with removable thermal insulation, the identification mark shall be clearly recognizable both with and without the shielding or thermal insulation.

4.2 Reactor Pressure Vessel

Mechanized devices shall be provided for opening and closing the reactor pressure vessel of light water reactors in order to minimize the time spent in the radiation field.

4.3 Control Rod Drives

In the case of boiling water reactors, the devices securing the control rods against ejection shall be arranged such that these devices can either be easily removed or are not a major hindrance during maintenance tasks on the drives.

4.4 Steam Generators

In the case of a pressurized water reactor, each of the individual accessible compartments of a steam generator shall be equipped with an individual manhole accessible from the outside. Handling of the manhole covers shall be facilitated by means of auxiliary equipment. Tensioning devices or easy to handle closures shall normally be provided. Care shall be taken to ensure ease of handling of the gaskets.

4.5 Pumps and Compressors

(1) Special leak-tightness requirements apply to the pumps transporting highly radioactive media.

(2) Horizontally arranged pumps shall normally be designed such that maintenance tasks can be performed without having to disassemble or move the pump drive (e.g. by means of a removable part of the drive shaft).

4.6 Valves

(1) As far as possible, heavy valve tops shall be arranged vertically in order to speed up their removal and alignment. Valves installed at an angle shall be provided with suitable auxiliary equipment for their disassembly.

(2) Sufficient space shall normally be provided for setting down the machines needed for working on the valves (e.g. valve seat grinding machines).

(3) Safety valves shall normally be easily replaceable.

(4) Valves should be arranged such that the drive can be temporarily shielded from the body of the valve.

(5) Special leak-tightness requirements apply to valves handling highly radioactively contaminated media.

(6) It should be possible to replace and retighten compression seals without any larger effort.

4.7 Pipes

(1) Pipes carrying radioactive media should be physically separated from pipes carrying nonradioactive media (e.g. by routing them in pipe ducts or by spatially separating them from each other).

(2) Pipes carrying radioactive substances should be designed for an as high a hydrodynamic efficiency as possible (e.g., by avoiding dead ends, ensuring a sufficient flow velocity) in order to minimize deposition of radioactive substances.

(3) The free space around pipes carrying radioactive media which are planned to be subjected to inservice inspections or on which other maintenance tasks are expected to be performed should be designed such that the respective locations around and along the pipes can be freely approached with the testing equipment, welding tools or other equipment required for performing the work tasks.

(4) The draining of systems carrying radioactive substances, with the exception of the residual draining of components, shall normally be carried out through permanently installed closed pipe systems (e.g., no open funnels). Exceptions regarding a permanent installation are permissible in the case of

a) connecting hoses for draining procedures, and

b) transfer systems for the purpose of filling transport containers or mobile conditioning facilities.

The flexible hoses used shall normally be short and closable. It shall normally be possible to check the draining procedure.

4.8 Electrical Equipment and Instrumentation and Control Equipment

(1) Electrical drives shall normally be installed at locations with a low local dose rate. If this is not possible, they shall normally be designed with low maintenance requirements and such that a time-saving replacement and adjustment is possible.

(2) Cables that have to pass through shielding walls shall be installed such that, in generally accessible areas, the shielding effect of the wall is not detrimentally affected.

(3) If it is unavoidable to install electrical measuring and control equipment at locations with higher local dose rates, provision should be made for their quick replacement (e.g., plug-in modules).

5 Ventilation Systems

(1) The ventilation systems (e.g., recirculated air filtration, air exchange rate) shall be designed taking the potential for radioactive release in the room groups in accordance with safety standard KTA 3601 into account.

(2) The equipment for monitoring the concentration of airborne radioactive substances in the room air shall be designed in accordance with safety standard KTA 1502.

(3) Facilities shall be provided that will prevent the influx of aerosol bound radioactive substances and radioactive iodine into rooms that must be continuously manned in the case of serious events (e.g., control room, central security station, incident response center, remote shutdown station). Provisions shall be made for an air recirculation operation.

(4) Care shall be taken in planning the treatment facilities for solid and liquid wastes as well as of the equipment for the decontamination compartment and the hot workshop to ensure that local exhaust equipment in accordance with safety standard KTA 3604 are provided.

(5) If it must be assumed that an increased release of radioactive substances into the air of rooms or room regions will occur during work tasks, technical protection measures such as local exhaust equipment shall be given preference over personal protection measures such as the wearing of respirators.

6 Communication Equipment

N o t e :
The requirements for communication equipment in nuclear power plants are specified in safety standard KTA 3901.

When selecting the locations for the sockets of communication and data access equipment (e.g., computer network and video surveillance) a low local dose rate and a close proximity to the workplace shall be the guiding factors.

7 Ergonomics

When specifying the radiological protection measures under Sections 3 through 6, all those ergonomic aspects shall be taken into account that, with particular regard to reducing exposure during maintenance tasks, will ensure a sufficiently short working time and an assessment of the prevailing condition that is as exact as possible.

N o t e :
From the point of view of ergonomics, e.g., with regard to the freedom of movement and working posture, the significant principles are the following:

a) Larger forces can be exerted with greater ease if applied in the vertical downward direction.

b) When a force must be exerted in the upward direction, the point on the component where the force must be applied should be as far as possible either above or below shoulder height.

c) When a force must be exerted in the horizontal direction, the point on the component where the force must be applied should be just below shoulder height. In this context, forces to be exerted in a forward or backward direction are more easily applied from the front of the body whereas forces to be exerted to the right or the left are more easily applied from the side of the body.

8 Documents Regarding Radiation Protection

With regard to possibly required large-scale or difficult maintenance tasks

a) at locations with a high local dose rate (e.g., 3 mSv/h), and

b) in areas difficult to reach within the controlled area

the documents of the technical documentation shall normally be supplemented as necessary with respect to a plant-specific radiation protection by

- 3-D graphics of the rooms and systems,
- photographs, photo-documentation (including benchmark scales),
- assembly instructions in the form of audiovisual recordings,
- exploded-view drawings, or
- models.

N o t e :
These documents can be used for the preparation of the work tasks and for training purposes.

9 Special Aspects Regarding Design Basis Accidents

9.1 Basics

The measures to be taken for the radiological protection of personnel regarding their work tasks in connection with the mitigation of design basis accidents are determined by two design basis accidents: loss of coolant in the containment vessel and differential pressure line rupture in the annulus.

9.2 Loss-of-Coolant Accident in the Containment Vessel

9.2.1 General requirements

(1) In the case of a loss-of-coolant accident in the containment vessel, it shall be ensured that the residual heat can be removed from the reactor core and from the fuel pool, that the necessary switching actions can be performed and that the data required for determining the chronological sequence of the accident can be obtained.

(2) The activity concentrations and dose rates for a pressurized water reactor shall be calculated in accordance with the accident calculation principles with due consideration of their respective scope of application.

9.2.2 Residual heat removal from the reactor core

(1) Components required for the long-term removal of residual heat from the reactor core shall be designed such that they can be operated for at least one year without maintenance.

(2) Repair measures do not need to be considered, provided, it is verified on the basis of probabilistic analyses and taking the redundancies into account, that a sufficient availability is ensured.

Note:
This is the case, for example, if the per-annum probability of failure of the residual heat removal is in the same order of magnitude as the non-availability of the residual heat removal at the time the design basis accident occurs.

(3) With regard to a repair of residual heat removal pumps the following conditions and reference values shall be applied:

a) Timing

From the viewpoint of radiology, the residual heat removal pumps shall be designed such that no repairs will be required within the first 30 days after the occurrence of the design basis accident.

b) Accessibility

Access (entire route) to and from the room of the pump to be repaired shall normally not lead to a personal exposure exceeding a total of 1 mSv from direct radiation. The local dose rate along this route shall normally not exceed 10 mSv/h.

The residual heat removal pumps shall each be installed in separate rooms.

There shall be an area in the vicinity of the room in which the pump is installed where preparatory work can be performed; the maximum local dose rate in this area shall normally not exceed 0.3 mSv/h.

c) Necessary spare parts

It shall be ensured that the spare parts necessary for the repair measures can be transported to the repair site.

d) Water filling and draining

It shall be possible to fill and drain or to flush the pumps from locations with a low local dose rate (cf. Section 4.1.2, para. (6)).

e) Work location

At the location of the pump to be repaired, the local dose rate from adjacent rooms and from adjacent components shall normally not exceed 5 mSv/h during the repair tasks.

(4) If instead of repair work on the residual heat removal system, an onsite manual switching to other systems is planned (e.g., to the cooling system of the fuel pool), the physical arrangement and shielding shall normally be such that the exposure of a person performing the switching task including access (entire route) to and from the designated location will not exceed the reference value of 25 mSv.

9.2.3 Cooling system of the fuel pool

If, in analogy to Section 9.2.2, para. (2), it cannot be ruled out that repairs of the cooling pumps of the fuel pool will be necessary, the possibility for a repair of at least one pump shall be ensured. In this context, the following conditions and reference values apply:

a) Timing

From the viewpoint of radiology, the pumps shall be designed such that no repairs will be required within the first 30 days after the occurrence of the design basis accident.

b) Accessibility

Access (entire route) to and from the room of the pump to be repaired shall normally not lead to a personal exposure exceeding a total of 1 mSv from direct radiation. The local dose rate along this route shall normally not exceed 10 mSv/h.

c) Necessary spare parts

It shall be ensured that the spare parts necessary for the repair measures can be transported to the repair site.

d) Work location

At the location of the pump to be repaired, the local dose rate from adjacent rooms and from adjacent components shall normally not exceed 5 mSv/h during the repair tasks.

9.2.4 System for the monitoring and limitation of hydrogen

(1) If the system for the monitoring and limitation of hydrogen is located inside the containment vessel, it shall be ensured that a maintenance-free operation of this system is possible over a long period of time.

(2) If this system is located outside of the containment vessel, the following precautionary measures shall be taken:

a) Well in advance of this system being put into operation, the designated rooms shall be shielded with regard to the design basis accident conditions and to the anticipated activity inventory.

b) When operation of this system is required, the air from the rooms where the system is installed and in which a leakage may occur, shall be conducted through high-efficiency particulate air filters and iodine filters.

c) The pipe lines leading to the hydrogen recombination system should be as short as possible.

9.2.5 Change-over switching of accident filters

In the case of a manual change-over switching of redundant accident filters, the personal exposure to be assumed for the entire switching procedure including access (entire route to and from the switching location) shall normally not exceed a reference value of 5 mSv at the point in time of maximum filter loading.

9.2.6 Major plant control room

The major plant control room shall be shielded such that it can be continuously manned.

9.2.7 Sampling

(1) In order to be able to assess the radiological conditions inside the containment vessel after a design basis accident, a possibility for taking samples shall be provided in accordance with safety standard KTA 3502, Sec. 5.3.1.

Note:
Requirements for the equipment for measuring the local dose rate during and after design basis accidents are specified in safety standard KTA 1501, Sec. 5.3.
Depending on the requirements of the post-accident situation it may be necessary to take samples of the primary water or of the containment vessel atmosphere only a few hours after the occurrence of the design basis accident.

(2) The following conditions and reference values shall be applied to the equipment used to take water samples (e.g., sump water or water from the pressure suppression chamber) or air samples:

a) Timing

From a viewpoint of radiology, a sampling is not required within the first three hours after occurrence of a design basis accident.

b) Accessibility

If onsite access is necessary, the access (entire route) to and from the sampling points shall normally not lead to a personal exposure exceeding a total of 1 mSv from direct radiation. The local dose rate along this route shall normally not exceed 10 mSv/h.

c) Radiation shielding

The shielding shall be designed and installed such that the personal exposure during sampling does not exceed 25 mSv.

d) Size of samples

Sample volumes shall normally be in accordance with the amounts required for the measuring procedure.

Note:

A few cubic centimeters of the sampled medium will be sufficient due to the expected relatively high activity concentration.

9.2.8 Vent air monitoring

Note:

Additional requirements regarding the accident measuring system for vent air monitoring are specified in safety standards KTA 1503.2 and KTA 3502.

(1) In order to determine the discharge of radioactive substances through the vent stack, the measuring filter at the exhaust air monitoring point shall be changed within a few hours after the occurrence of a design basis accident.

(2) The following conditions and reference values shall be applied to the measuring point of the exhaust air monitoring:

a) Timing

It shall be ensured that the filter-changing procedure is initiated one hour after the occurrence of the design basis accident.

b) Accessibility

If onsite access is necessary, the access (entire route) to and from the measuring point of the exhaust air monitoring shall normally not lead to a personal exposure exceeding a total of 1 mSv from direct radiation. The local dose rate along this route shall normally not exceed 10 mSv/h.

c) Radiation shielding

The shielding shall be designed and installed such that the personal exposure during exchange of the measuring filter does not exceed 1 mSv.

9.3 Differential Pressure Line Rupture Accident Outside of the Containment Vessel

(1) In accordance with safety standard KTA 3501, Sec. 5.1.5.3, para. (3), the differential pressure lines of the reactor protection system leading out of the containment vessel shall normally not contain automatic isolation valves. Therefore, it must be possible to isolate these lines manually in case of a rupture outside of the containment vessel, i.e., these lines shall be accessible and capable of being closed off.

(2) The access (entire route) to and from the manually operated valves together with the manual closing procedure shall normally not lead to a personal exposure exceeding a total of 5 mSv from direct radiation. The local dose rate along this route shall normally not exceed 10 mSv/h.

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) Atomic Energy Act in the version promulgated on July 15, 1985 (BGBl. I, p. 1565), most recently changed by article 1 of the act dated December 4, 2022 (BGBl. I, p. 2153)
StrlSchG		Act on the Protection against the Harmful Effect of Ionising Radiation (Radiation Protection Act - StrlSchG) Radiation Protection Act of June 27, 2017 (BGBl. I, p. 1966), most recently changed by the promulgation of January 3, 2022 (BGBl. I, p. 15)
StrlSchV		Ordinance on the Protection against the Harmful Effects of Ionising Radiation (Radiation Protection Ordinance - StrlSchV) Radiation Protection Ordinance of November 29, 2018 (BGBl. I, p. 2034, 2036), most recently changed by article 1 of the ordinance dated October, 2021 (BGBl. I p. 4645)
SiAnf	(2015-03)	Safety Requirements for Nuclear Power Plants (SiAnf) of November 22, 2012, amended version of March 3, 2015 (BAnz AT 30.03.2015 B2), most recently changed as promulgated by BMUV on February 25, 2022 (BAnz AT 15.03.2022 B3)
Interpretations to SiAnf	(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)
ASiG		Act on company doctors, safety engineers and other experts for occupational health and safety of December 12,1973 (BGBl. I, p. 1885), most recently changed by Article 3, Sec. 5. of the Act of April 20, 2013 (BGBl. I, p. 868)
ArbStättV		Ordinance on workplaces (Arbeitsstättenverordnung – ArbStättV) of August 12, 2004 (BGBl. I p. 2179), most recently changed by Article 4 of the Ordinance of December 22, 2020 (BGBl. I p. 3334)
BetrSichV		Ordinance on industrial safety (Betriebssicherheitsverordnung – BetrSichV) of February 3, 2015 (BGBl. I, p. 49), most recently changed by Article 7 of the Act of July, 2021 (BGBl. I, p. 3146)
ASR A 1.7	(2009-11)	Technical rules for workplaces - Doors and gates, most recently changed GMBI 2022, p. 244
ASR A 4.3	(2010-12)	Technical rules for workplaces - First-aid stations; Equipment and facilities for first-aid, most recently changed GMBI 2022, p. 252
KTA 1501	(2022-11)	Stationary system for monitoring the local dose rate within nuclear power plants
KTA 1502	(2022-11)	Monitoring radioactivity in the inner atmosphere of nuclear power plants
KTA 1503.2	(2022-11)	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 2101.1	(2015-12)	Fire protection in nuclear power plants; Part 1: Basic requirements
KTA 3402	(2022-11)	Airlocks on the reactor containment of nuclear power plants - personnel airlocks
KTA 3409	(2022-11)	Airlocks on the reactor containment of nuclear power plants - equipment airlocks
KTA 3501	(2015-11)	Reactor protection system and monitoring equipment of the safety system
KTA 3502	(2012-11)	Accident measuring system
KTA 3601	(2022-11)	Ventilation systems in nuclear power plants
KTA 3604	(2020-12)	Storage, handling, and plant-internal transport of radioactive substances in nuclear power plants (with the exception of fuel assemblies)
KTA 3901	(2017-11)	Communication means for nuclear power plants
AD 2000-Code A5	(2020-01)	Technical rule – Openings, closures and closure elements
DIN ISO 8690	(2022-03)	Measurement of radioactivity - Gamma ray and beta emitting radionuclides - Test method to assess the ease of decontamination of surface materials (ISO 8690:2020)
DIN 55991-1	(2016-03)	Coating materials - Coatings for nuclear facilities - Part 1: Requirements and test methods
DIN 25440	(2021-06)	Classification of rooms in the controlled area of nuclear facilities and facilities according to local dose rates