Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 2201.6 (2015-11)

Design of Nuclear Power Plants Against Seismic Events;
Part 6: Post-Seismic Measures

(Auslegung von Kernkraftwerken gegen seismische Einwirkungen;

Teil 6: Maßnahmen nach Erdbeben)

The previous version of this safety standard was issued in 1992-06

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KTA SAFETY STANDARD

2015-11	**Design of Nuclear Power Plants against Seismic Events; Part 6: Post-Seismic Measures**	KTA 2201.6

Previous versions of the present safety standard: 1992-06 (BAnz No. 36a of February 23, 1993)

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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 the present 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 Requirements for Nuclear Power Plants (SiAnf) and the SiAnf-Interpretations.

(2) To achieve these goals, the present safety standard KTA 2201.6 deals with the measures that shall be taken whenever certain acceleration limit values have been exceeded during an earthquake. The safety standards series KTA 2201 “Design of Nuclear Power Plants against Seismic Events” comprises the following six parts:

KTA 2201.1: Principles

KTA 2201.2: Subsoil (soil and rock)

KTA 2201.3: Civil structures

KTA 2201.4: Components

KTA 2201.5: Seismic instrumentation

KTA 2201.6: Post-seismic measures
(*the present safety standard*).

(3) The requirements in the present safety standard are based on the verification concept ‘Design basis earthquake - Inspection level’ detailed in safety standard KTA 2201.1.

(4) Up to a point where the value of the inspection level is decisively exceeded, no earthquake-related deviations that could put specified normal conditions into question need to be expected in any areas designed against seismic events. Nevertheless, certain measures must be performed to verify specified normal conditions before the inspection level is decisively exceeded; those measures are specified in this safety standard.

(5) Whenever the value of the inspection level is decisively exceeded, earthquake-related deviations that could put the specified normal condition into question cannot anymore be ruled out in the areas designed against seismic events. Therefore, if the inspection level is decisively exceeded the nuclear power plant must be shut down and the measures be performed that are specified in this safety standard.

1 Scope

The present safety standard shall apply to nuclear power plants with light water reactors.

Note:
The present safety standard does not apply to earthquakes stronger than the design basis earthquake.

2 Definitions

(1) Condition, specified normal

The specified normal condition is that condition of the nuclear power plant that is associated with the specified normal operation.

(2) Operation, specified normal

In accordance with SiAnf:

The specified normal operation for which the nuclear power plant is technically intended, designed and suited comprises the operating conditions and operating procedures

a) during functioning condition of the facilities (undisturbed operational state, normal operation, Safety Level 1),

b) during abnormal operation (disturbed operation, malfunction, Safety Level 2),
as well as

c) during maintenance procedures (inspection, maintenance, repair).

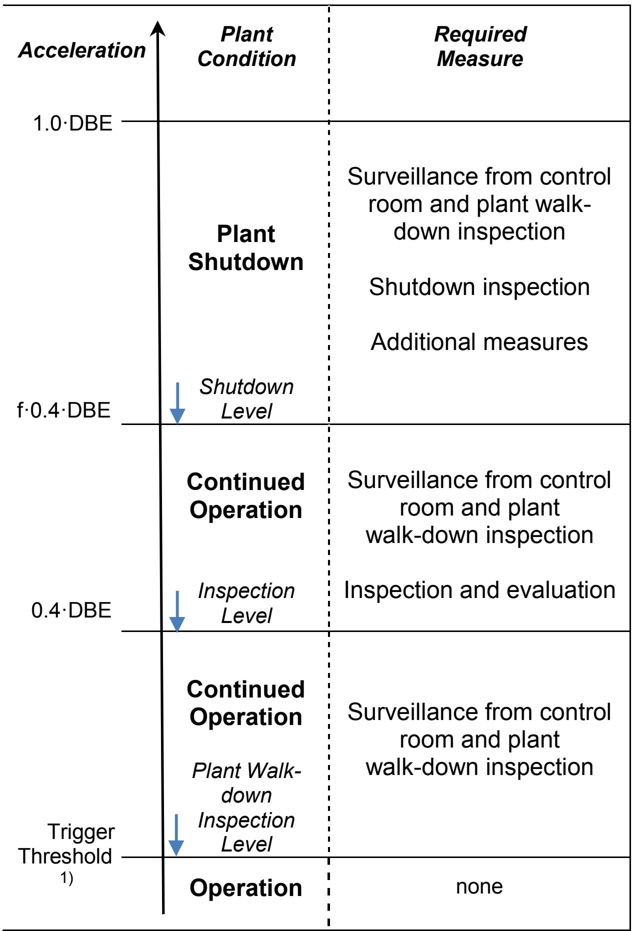
3 Procedure

3.1 General Requirements

(1) After occurrence of an earthquake and depending on the evaluation of the recorded acceleration time history a concept of graded measures shall be applied. This concept is shown in **Figure 3-1**.

(2) The individually required measures are specified in Sections 3.4, 3.5 and 3.6.

Note:
Individual cases may require long-term measures. These may be performed even after the restart of the plant, however, are not subject of the present safety standard.



1) Trigger threshold for data registration in accordance with safety standard KTA 2201.5

Figure 3-1: Concept of graded measures

(3) In case an earthquake leads to an operational malfunction or design-basis accident, then the required measures to mitigate these events shall be performed with the highest priority.

3.2 Verification of the Earthquake

(1) Whenever the seismic recorder is activated (**Figure 3-2** Chart Item 1) it shall be investigated whether an earthquake has occurred. This requirement may be met, e.g., by contacting institutions outside of the nuclear power plant and evaluating the recorded time histories with respect to faulty signals.

(2) If the trigger thresholds for data recording of at least two installation locations of seismic instrumentations were exceeded (plant walk-down inspection level), it shall precautionarily be assumed that an earthquake has occurred.

(3) In case of a faulty signal, its cause shall be determined. Any faulty signal shall be documented.

3.3 Classification of the Earthquake

(1) When an earthquake has occurred, the response spectra generated from the recorded time histories shall be evaluated based on the following criteria.

(2) The earthquake shall be classified as specified in **Figure 3-2**, Chart Item 2. The factor f (cf. Appendix A) may be assumed as being equal to 1.5. Using a factor f larger than 1.5 requires an individual plant-specific verification.

(3) Should at least one frequency of the determined response spectrum (component or resultant) exceed a value of 0.4 times the design basis response spectrum, the inspection level shall basically be assumed as having been reached. A higher level is permissible in accordance with safety standard KTA 2201.1, provided, it was verified that specified normal operation of the plant is possible even after the occurrence of an earthquake of that size.

Note:

The last sentence applies, e.g., to nuclear power plants whose design was based on verifying the "operating basis earthquake" in accordance with safety standard KTA 2201.1, version 1975-06 (Verification concept: "Safety earthquake – Operating basis earthquake").

(4) If at least one frequency of a measured response spectrum (component or resultant) exceeds a value of $f \cdot 0.4$ times the design basis response spectrum, it shall be assumed that the inspection level has been decisively exceeded.

(5) If the inspection level is decisively exceeded only for frequencies above 16 Hz, these cases shall be evaluated by engineering-based considerations. For the length of these activities, a continued operation of the plant is permissible.

Notes:

1) The engineering-based evaluation may be based on, e.g., spectral intensity, spectral values, magnitudes or the cumulative absolute velocity (CAV) values.
2) In the case of safety-related buildings and components, the essential frequencies for the evaluation are the ones up to 16 Hz.

3.4 Initial Measures

(1) To obtain a quick overview of the effects that the earthquake had on the plant, the plant condition shall be determined by performing quickly executable measures.

(2) Independent of the earthquake classification, the plant condition shall be determined by a plant inspection. This requires performing plant check-ups from the control room and plant walk-down inspections.

(3) If the classification of the earthquake indicates that the inspection level was decisively exceeded then a plant shutdown inspection shall be performed and the plant shall be shut down.

3.4.1 Plant check-up from the control room and plant walk-down inspection (**Figure 3-2** Chart Item 3)

(1) The condition of the plant shall be checked from the control room (e.g., computer printouts, displays, failure and hazard alarms, indications of leakages).

(2) Within the framework of an immediately initiated plant walk-down inspection (see informative Appendix B) a visual inspection shall be performed to identify possible deviations caused by the earthquake. In this context, areas designed against, as well as areas not designed against seismic events shall be inspected. The type and extent of the plant walk-down inspection depend on the specific features of the plant and shall be specified in the operating regulations.

(3) The plant walk-down inspection shall be performed at least with the extent and quality of a regular inspection round.

(4) The results of the plant check-up and plant walk-down inspection shall be documented.

3.4.2 Deviations caused by the earthquake (**Figure 3-2** Chart Item 4)

(1) During the plant walk-down inspection particular attention shall be paid to obviously recognizable, earthquake-related deviations.

(2) Provided, the earthquake classification shows that the inspection level was not reached and no earthquake-related deviations were discovered, then no in-depth measures are required and continued operation of the plant is permissible. If, however, earthquake-related deviations were discovered it shall be checked whether the specified normal condition is upheld.

(3) If the earthquake classification shows that the inspection level was reached but not decisively exceeded and that no earthquake-related deviations were discovered, then in-depth measures shall be initiated. However, if earthquake-related deviations were discovered it shall first be checked whether the specified normal condition is upheld before the in-depth measures are initiated.

3.4.3 Specified normal condition in accordance with the operating manual (**Figure 3-2** Chart Item 5)

(1) The specified normal condition may be considered as being upheld if the respective prerequisites and conditions specified in the operating manual are met and no earthquake-related deviations were detected that would lead to restricting specified normal operation.

(2) If earthquake-related deviations are detected it shall be checked whether the specified normal condition is upheld.

(3) Provided, the specified normal condition is upheld, a continuation of plant operation is permissible for the time being and the inspections and analyses described in Section 3.5.1 shall be performed.

(4) If, however, the specified normal condition is not upheld, a shutdown inspection as specified in Section 3.5.3 shall be performed and the plant shall be shut down as specified in Section 3.6.2.

3.5 In-depth Measures

(1) In-depth measures shall be performed depending on the classification of the earthquake and on the results of the initial measures. In-depth measures shall normally either ascertain the specified normal condition of the plant or facilitate its safe shutdown.

(2) Provided, the earthquake classification indicates that the inspection level was not reached, then in-depth measures are only required to be performed if earthquake-related deviations were detected.

(3) If the classification of the earthquake indicates that the inspection level is exceeded then in-depth measures are required to be performed.

3.5.1 Inspections and analyses (**Figure 3-2** Chart Item 6)

(1) The inspection shall be performed by a special plant inspection team as a walk-down inspection of the entire plant.

Note:
Examples for possible indications of earthquake-related deviations are presented in the informative Appendix C.

(2) The limited accessibility of exclusion areas shall be taken into account depending on the actual plant conditions.

(3) The plant walk-down inspection team shall normally be made up of qualified persons and of personnel that is familiar with the condition of the plant before the earthquake. The composition of the plant walk-down inspection team and the extent of the random inspections shall be individually specified for the respective plant.

(4) A loading analysis shall be performed for those Seismic Class I components and civil structures for which an earthquake-related deviation was detected.

(5) In addition to the components and civil structures specified in para. 4, additional exemplary Seismic Class I components shall be chosen for which the earthquake is the decisive load case and that are highly stressed:

a) two pipelines,

b) two pipeline supports,

c) two vessel support structures,

d) two pump supports and

e) two valves with high super structures.

(6) The components specified in para. 5 either shall be subjected to an analysis of the loading due to the earthquake or their previously identified locations of highest loading shall be subjected to nondestructive examinations. It is permissible to base the loading analysis on the plant and system conditions that had existed during the earthquake.

(7) The number of load cycle that occurred during the earthquake shall be determined at the measuring points and shall be evaluated.

(8) Earthquake-related deviations of Seismic Class IIa components and civil structures shall be evaluated with respect to possible effects they might have on Seismic Class I components and civil structures.

(9) The functioning of the terminating elements of the reactor protection system, of the components of the emergency power supply and of the emergency system shall be inspected, as far as this is possible under the actual operating condition.

(10) The results of the inspections and analyses shall be documented.

3.5.2 Specified normal condition and permissible loads (**Figure 3-2** Chart Item 7)

(1) Provided, the inspections and analyses have not uncovered any earthquake-related deviations, then the specified normal condition shall be considered as ascertained and continued operation of the plant is permissible (**Figure 3-2** Chart Item 8).

(2) If, however, the inspections and analyses have uncovered earthquake-related deviations, a continuation of the plant operation is not permissible for the time being and the shutdown inspection specified in Section 3.5.3 shall be performed.

3.5.3 Shutdown inspection (**Figure 3-2** Chart Item 9)

The availability of the systems necessary for a safe shutdown (e.g., emergency power supply, residual heat removal and required auxiliary systems) shall be checked and, if required, made available.

3.6 Resulting Measures

3.6.1 Continued operation (**Figure 3-2** Chart Item 8)

Provided, the inspections under Sections 3.4.2 or 3.5.2 show that the specified normal condition is upheld, a continuation of the plant operation is permissible.

3.6.2 Shutting down the plant (**Figure 3-2** Chart Item 10)

If, however, the inspections under Sections 3.4.3 or 3.5.2 show that the specified normal condition is not upheld or that the inspection level is decisively exceeded, then the plant shall be shut down under consideration of the findings under Section 3.5.3.

3.6.3 Additional procedures (**Figure 3-2** Chart Item 11)

Additional measures required in individual cases shall be specified in close consultation with the supervisory authority.

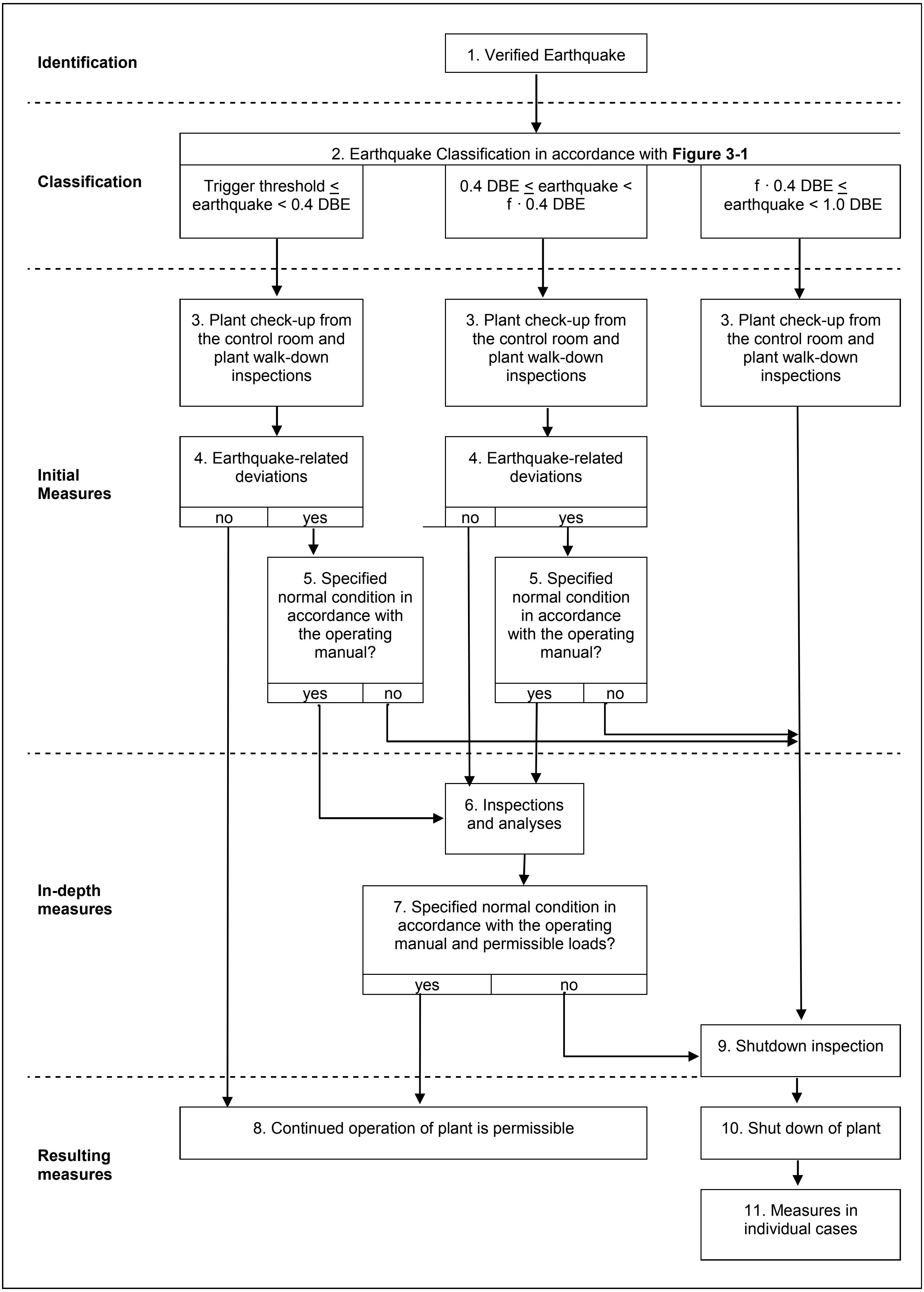


Figure 3-2: Post-seismic measures

Appendix A

Derivation of Factor f

The inspection level shall be considered to be decisively exceeded if the level of the actual earthquake (response spectrum) is f-times higher than the inspection level.

$$\text{Shutdown level} = f \cdot 0.4 \cdot \text{DBE} \qquad \text{(A-1)}$$

(DBE – design basis earthquake)

The conservativeness of the factor f is founded in the understanding that operation may harmlessly be continued, provided, the earthquake-related loads lie within elastic limits or plastic deformations are restricted to regions of geometric discontinuities.

For pressurized or radioactivity-containing components this is the case if Service Limit Level C is not exceeded. The service limit levels are specified in safety standard KTA 3201.2.

Provided, the design basis earthquake is accounted for by a design for Service Limit Level D, then Service Limit Level C is reached at $\alpha \cdot$ DBE where α shall be calculated by equation (A-2).

$$\alpha = \left(\frac{\text{perm } \sigma^{C}}{\text{act } \sigma^{A}} - 1\right) / \left(\frac{\text{act } \sigma^{D}}{\text{act } \sigma^{A}} - 1\right) \qquad \text{(A-2)}$$

(perm – permissible; act – actual)

The calculation of the factor α as well as the entire calculatory chain for the design against earthquakes is based on the following conservative assumptions:

- Cautious assumptions regarding act σ^{A} and act σ^{D},
- In part more favorable operating conditions during the respective earthquake,
- Narrow-banded spectra of the respective earthquake,
- Based on experience, a more advantageous component behavior than verified based on the analyses specified in safety standard series KTA 2201.

Table A-1 lists the safety-related components grouped according to their corresponding α-values. The engineering-based assessment of these results and of the remaining conservativeness leads to a factor α = 0.6. Depending of plant related investigations (i.e., calculatory check of the safety-related vessels with austenitic support lugs or beam supports), this value can possibly be increased up to α = 0.7.

This corresponds to a factor f = 1.5, possibly increased up to f = 1.75.

Row	***Component Group***		***Maximum Stress Limit for DBE***	***Shutdown Level = α · DBE***
1	Pipe lines	welded	$3\ S_m\ (= R_{mT})$	0.7
2		flanged	Flange: $R_{p0.2T}$	1
3	Supports, mounting brackets - steel structures		$R_{p0.2T}$	1
4	Active mechanical components		$R_{p0.2T}$ or deformation analysis	1
5	Vessels, heat exchangers		min. $(3.6\ S_m, R_{mT})$	> 0.5 (up to 0.7)
6	Electrotechnical components and controls		Experimental verification	1
7	Seismic Class IIa components and pipe lines		Same as Seismic Class I components and pipe lines	> 0.5 (up to 0.7)
8	Containment vessel		$0.94 \cdot R_{p0.2T}$	1

Explanatory Notes:

Row 1: On account of Design Level 0 and the required deflection limitation, a value for act $\sigma^{A} < 0.75\ S_m$ may be assumed. With the otherwise conservative assumptions this leads to
→ α = 0.7.

Rows 2, 3, 8: Since the design for the DBE is such that the yield strength will not be exceeded, the loads up to 100 % DBE will remain in the elastic range. Connecting elements, however, are rated higher anyway and therefore:
→ α = 1.0

Row 4: Since the design for the DBE requires either a verification of Design Level B or a deformation verification, no impermissible plastic deformations will occur up to 100 % DBE and therefore:
→ α = 1.0

Row 6: Provided, the active functioning is experimentally verified up to 100 % DBE:
→ α = 1.0

Rows 5, 7: Since the sum of the primary and secondary stresses ($P_L + Q \le 3\ S_m$) is significant for Service Limit Levels A and B, a value of act $\sigma_L^{A} \le S_m$ can be assumed for the austenitic materials in all situations occurring in the plant. With an otherwise conservative approach, this leads to
→ α = 0.5 for austenitic materials and a corresponding value
→ α = 0.9 up to 1.0 for ferritic materials
Since α = 0.5 only applies to vessels with brackets or supports, an intensive investigation of these (generally few) vessels opens the possibility to raise the value of α maybe even to the most advantageous value α = 0.7, the value for welded pipe lines.

Table A-1: Shutdown level in the case of safety related components

Appendix B (informative)

Directives for Plant Walk-Down Inspections

The following list contains examples (based on IAEA Report Series No. 66) for possible earthquake-related deviations that need special attention during plant walk-down investigations. Objective of the visual inspection in the course of a plant walk-down inspection is to identify obviously apparent damages.

- Leakages in pipe line systems, particularly at flanges, threaded nozzles and branching-off pipe lines
- Damage to low-pressure tank vessels, especially flat-bottom tanks
- Damage to switchyard components
- Increased vibrations, increased bearing temperatures and unusual sounds of rotating components
- Displaced, fallen-over or fallen-down objects
- Damages to, and loosening of anchorings
- Damages to pipes, electric cables and cable ways
- Indications of an excessive displacement of cable and component supports
- Indications that containment penetrations are possibly adversely affected
- Indications that components have impacted each other

Appendix C (informative)

Inspection

The following observations can, among others, be indicative of earthquake-related deviations.

Platforms and mountings

- Newly flaked off paint
- Visible cracks in weld seams
- Concrete dust, visible cracks in the wall near anchors
- Visible deformations or displacements

Civil structures

- Concrete spallings
- In-seepage of water
- Cracks in the concrete
- Visible damages to the containment isolation components (preventing a release of radioactivity)
- Damages to doors
- Damages to suspended ceilings
- Damages to lighting elements
- Uplift and lowering of the ground

Ventilation ducts

- Visible deformations
- Leakages

Cable way constructions

- Visible deformations
- Visible damages to cables

Pipe lines

- Restrained expansion, restrained vibration
- Damages to wall penetrations
- Visible deformations
- Leakages
- Damage to subsoil-embedded pipe lines and other distribution systems (pipe breaks and ground anomalies)

Fittings

- Leakages from connecting flanges, spindle seals

Pumps, ventilators

- Damages to foundations, connection bolts and linchpins
- Level of operating noise
- Leakages from mechanical seals (slip ring seals)
- Oil leakages

Vessels, heat exchangers, tanks

- Damages to foundations, connection bolts and linchpins
- Dents
- Displacements
- Damages to supporting structures (visible cracks of weld seams)
- Leakages

Electrical and control components

- Damages to panels and doors of control cabinets
- Visible deformations, displacements
- Visible damages to mounting constructions
- Visible damages to the energy supply (emergency power diesel generator, battery rooms, converters, etc.)

Fire protection equipment and facilities

- Visible damage to building and component related fire protection equipment and facilities
- Constriction of escape and rescue routes
- Constriction of fire brigade access routes and engagement areas

Plant security facilities

- Visible damages to the plant security facilities

Cranes

- Was hook loaded during earthquake? – yes/no
- Correct positioning in the tracks
- Visible damages to crane runway

Appendix D

Regulations Referred to in the Present Safety Standard

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

AtG		Act on the peaceful utilization of atomic energy and the protection against its hazards (Atomic Energy Act – AtG) of December 23, 1959, revised version of July 15, 1985 (BGBl. I, p. 1565), most recently changed by Article 307 of the Act of August 31, 2015 (BGBl. I 2015, No. 35, p. 1474)
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 5 of the Act of December 11, 2014 (BGBl. I, p. 2010)
SiAnf	(2015-03)	Safety requirements for nuclear power plants of November 22, 2012, revised version of March 3, 2015 (BAnz AT of March 30, 2015 B3)
Interpretations to SiAnf	(2015-03)	Interpretations regarding the safety requirements for nuclear power plants of November 22, 2012, most recently changed on March 3, 2015 (Banz AT of March 30, 2015)
KTA 2201.1	(2011-11)	Design of nuclear power plants against seismic events; Part 1: Principles
KTA 2201.2	(2012-11)	Design of nuclear power plants against seismic events; Part 2: Subsoil
KTA 2201.3	(2013-11)	Design of nuclear power plants against seismic events; Part 3: Civil structures
KTA 2201.4	(2012-11)	Design of nuclear power plants against seismic events; Part 4: Components
KTA 2201.5	(2015-11)	Design of nuclear power plants against seismic events; Part 5: Seismic instrumentation
KTA 3201.2	(2013-11)	Components of the reactor coolant pressure boundary of light water reactors; Part 2: Design and Analysis
IAEA Safety Report Series No. 66	(2011)	Earthquake Preparedness and Response for Nuclear Power Plants Safety Reports Series No. 66; ISBN:978-92-0-108810-9