Reaffirmed: 6/97, 6/02

Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 2201.6 (6/92)

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

(Auslegung von Kernkraftwerken gegen seismische Einwirkungen;
Teil 6: Maßnahmen nach Erdbeben)

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

Editor:

KTA-Geschaeftsstelle c/o Bundesamt fuer Strahlenschutz (BfS)
Willy-Brandt-Strasse 5 • 38226 Salzgitter • Germany
Telephone +49-1888/333-(0) 1621 • Telefax +49-1888/333-1625

KTA SAFETY STANDARD

June 1992	Design of Nuclear Power Plants against Seismic Events Part 6: Post-Seismic Measures	KTA 2201.6

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PLEASE NOTE: Only the original German version of this safety standard represents the joint resolution of the 50-member Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA). The German version was made public in Bundesanzeiger No. 194a on October 14, 2000. Copies may be ordered through the Carl Heymanns Verlag KG, Luxemburger Str. 449, 50939 Koeln (Telefax +49-221-94373-603).

All questions regarding this English translation should please be directed to:

KTA-Geschaeftsstelle c/o BfS, Willy-Brandt-Strasse 5, 38226 Salzgitter, Germany

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, the 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 safety related requirements which shall be met with regard to precautions to be taken in accordance with the state of science and technology against the hazards arising front the construction and operation of tire facility (Sec. 7 para. 2 No. 3 Atomic Energy Act), in order to attain tire protective goals specified in the Atomic Energy Act and the Radiological Protection Ordinance. (StrlSchV) further detailed in the "Safety Criteria for Nuclear Power Plants" and in tire "Guidelines for tire Assessment of tire Design of Nuclear Power Plants with Pressurized Water Reactors against Incidents pursuant to Sec. 28 para. 3 of the Radiological Protection Ordinance - Incident Guidelines".

(2) hl order to attain these goals, this safety standard KTA 2201.6 deals with measures to be taken after air earthquake and to be provided for in the design of nuclear power plants against seismic impacts in case certain acceleration limits of tire inspection level have been exceeded. KTA 2201 includes additionally, the following safety standards:

KTA 2201.1:	Principles
KTA 2201.2:	Subsurface Materials (Soil and Rock)
KTA 2201.3:	Design of Structural Components
KTA 2201.4:	Requirements for Procedures for Verifying the Safety of Mechanical and Electrical Components against Earthquakes
KTA 2201.5:	Seismic Instrumentation

(3) This safety standard KTA 2201.6 is based on the verification concept design basis earthquake versus inspection level. The limits of tire inspection level are 0.4 times the design levels of tire design basis earthquake.

(4) If it was proved that air application of other limits is admissible with respect to tire measures to be taken after air earthquake, such other limits may be used as inspection level.

This applies, e.g., to nuclear power plants the design of which was based on a demonstration for the design basis earthquake on the basis of version 6/75 of KTA 2201.1 (safe shutdown earthquake versus design basis earthquake demonstration concept).

1 Scope

This safety standard applies to nuclear power plants. It applies to nuclear power plants at sites for which a seismic instrumentation in accordance with KTA 2201.5 is required to be installed.

2 Definitions

(1) Specified normal operation state of the plant

The specified normal operation state of the plant is the state of the plant which exists during specified normal operation.

(2) Inspection level

The inspection level is the level derived from the design levels of the design basis earthquake when applying a reduction factor of 0.4 (maximum acceleration $a_{IN,max}$) and from the history of the acceleration response spectra of tire design basis earthquake scaled with the reduction factor of 0.4.

3 Analysis and Measures

3.1 Analysis

3.1.1 Response to the 'earthquake' group alarm

Upon occurrence of the 'earthquake' group alarm in the control room, an analysis shall immediately be carried out in accordance with the analysis chart in **Figure 3-1** and the associated explanations in accordance with Section 3.1.2.

The measures specified in chart items 1 through 7 shall be carried out within a period of lime as shall be specified in the operating manual.

3.1.2 Performance of the analysis

(1) Determination of the prevailing circumstances

To determine the prevailing circumstances, tests and examinations shall be carried out in accordance with paras. (2) through (7) of this section.

(2) Group alarm triggered by air earthquake?
(Chart item 1)

It shall be checked whether an earthquake did occur. for example by establishing contact with organizations outside the nuclear power plant. If at least two seismic triggers have responded it shall be assumed that an earthquake has occurred.

(3) Binary seismic monitor has responded?
(Chart item 3)

It shall be checked whether at least one binary seismic monitor has responded.

(4) Plant is in the specified normal operation state?
(Chart item 5)

The state of the plant shall be checked from the control room (e.g. computer printouts, analog displays, incident alarms, any indications of leakages). A plant walk through shall subsequently be performed to visually inspect the entire plant (e.g., comparison of actual and as-planned conditions, inspection for integrity, checking of local measurement equipment and checking for any deformations). Kind and extent of these inspections and checks depend on the specific characteristics of the plant. For an evaluation of the effects of an earthquake, the walk-through inspection should also include those plant areas not designed to cope with earthquakes.

The plant shall initially be considered as being in the state of specified normal operation if the conditions specified in the operating manual are fulfilled and there is no visible damage which may lead to a restriction of specified normal operation.

(5) Inspection level has been exceeded?
(Chart items 4 and 6)

Simultaneously with the measures in accordance with para. (4), the recorded maximum accelerations ($a_{ir,max}$) shall be determined on the basis of the time histories recorded at the installation locations of the acceleration sensors as follows.

a) For all acceleration time histories recorded the maximum values ($a_{xr,max}$, $a_{yr,max}$ and $a_{zr,max}$) shall be determined for each individual track.
b) The resultant ($a_{hrR,max}$) from the horizontal acceleration components recorded at the respective recording locations shall be determined.

For all installation locations of the acceleration sensors, a comparison shall be made between the maximum accelerations recorded ($a_{ir,max}$) and the maximum accelerations on which the inspection level is based ($a_{IN,max}$).

The inspection level shall be considered to have been exceeded, if the following condition applies to at least one acceleration:

$a_{ir,max} > a_{IN,max}$

(6) Inspection level has been exceeded significantly?
(Chart item 7)

The inspection level shall be considered to have been exceeded significantly, if the following condition applies to at least two recording locations

$a_{hrR,max} > f \cdot a_{IN,max}$

The factor f shall be substituted by a value of 1.5 as determined in accordance with **Appendix A,** unless it is demonstrated on a plant-specific basis that a greater value is admissible.

(7) Load level is admissible?
(Chart item 8)

A comparison shall be made between the earthquake response spectra determined on the basis of the earthquake time histories recorded and the response spectra based on the inspection level at the installation locations of the acceleration sensors:

a) The response spectra based on the inspection level are exceeded if, in the case of at least one frequency value, the amplitude was exceeded by more than 10%.

b) If the response spectra based on the inspection level have been exceeded at certain frequencies, these frequencies shall be determined at the recording locations in question.

c) With respect to Seismic Class I plant components whose eigenfrequencies do not coincide with the frequencies to be assigned in accordance with item a) above, it may be assumed without any further demonstration that the admissible stresses were not exceeded.

Concerning all Seismic Class I plant components whose eigenfrequencies do coincide with the frequencies to be allocated in accordance with item a) above, additional investigations shall be carried out regarding the stresses which occurred: the load level that actually occurred shall be evaluated as follows:

d) A stress analysis shall be performed with respect to these plant components. The acceleration response spectra determined on the basis of the earthquake time histories recorded - and if necessary the earthquake time histories of all three directions of initiation, converted to the respective locations installation - shall be taken into account. The plant and system states as they existed during the earthquake may be used.

e) The load level which actually occurred shall be compared with that on which the design was based (‘earthquake’ load combination in accordance with KTA 2201.4), and its admissibility shall be demonstrated (demonstration of the fulfillment of the following safety-related tasks: stability, integrity and functional capability).

Note:
For ‘earthquake’ load combination also see draft standard KTA 2201.3 (version 6 90).

f) Further specifications are admissible in individual cases, in particular also with respect to Seismic Class II plant components which may jeopardize Seismic Class I plant components for example in the case non-load-bearing structural parts of Seismic Class I structures.

3.2 Measures

3.2.1 General Measures

(1) If, as a result of an earthquake, a malfunction or an incident occurs in that area of the plant which is not designed against earthquakes, priority shall be given to the measures contained in the operating manual for this purpose. The pleasures contained in this safety standard shall then be considered as additional verification measures to the extent required in each case.

(2) The pleasures taken after an earthquake shall be documented.

(3) In order to maintain the stand-by recording condition of the seismic instrumentation, the data carrier media shall be replaced immediately upon their removal from the recording equipment.

3.2.2 Measures Depending on the Prevailing Conditions

(1) As a result of the analysis in accordance with **Figure 3-1**, any of the measures described in paras. (2), (3) or (4) shall be taken, depending on the prevailing conditions.

(2) Inspection of the Seismic Instrumentation
(Chart item 2)

The cause of an erroneous initiation shall be determined, and provisions against recurrence shall be taken. Initiations not caused by earthquakes shall be considered as erroneous initiations of the seismic instrumentation.

(3) Shutdown of the Plant
(Chart item 9)

Provided a reactor scram has not already been caused by the earthquake, the plant shall be shut down in accordance with the operating manual and as provided for the respective state of the plant. Shutdown via the ultimate heat sink should be preferred. In each individual case, further measures shall be specified in coordination with the supervisor authority.

(4) Continued Operation of the Plant
(Chart item 10)

The plant need not be shut down if it is in the specified normal state and was not subjected to any inadmissible stresses as a result of the earthquake.

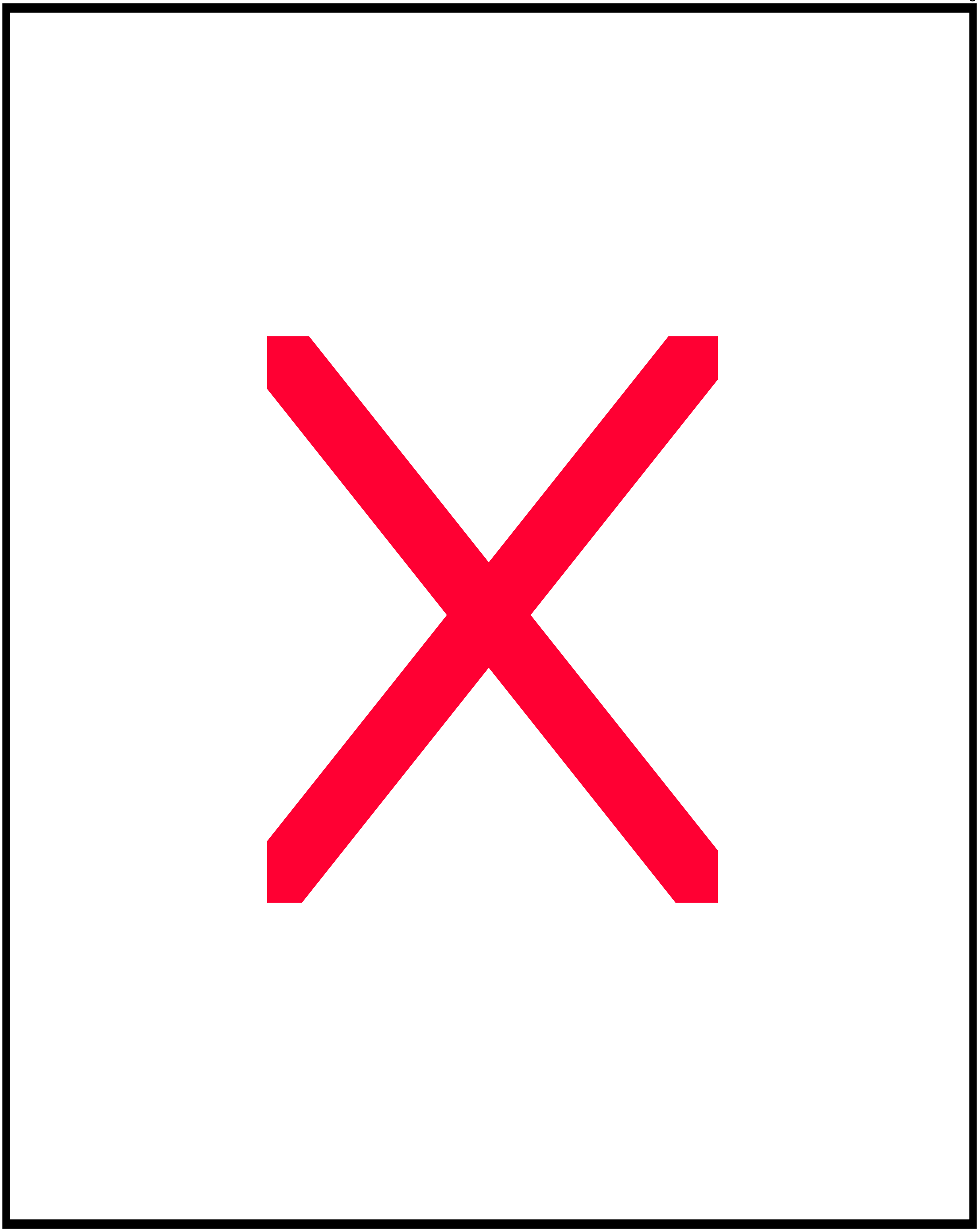

Figure 3-1: Analysis chart for the specification of the measures to be taken after an earthquake

Appendix A

Derivation of Factor f

The inspection level shall be considered to have been exceeded significantly if the level of the earthquake that occurred (a_{hrR}, response spectra) is above the inspection level by a factor f (in accordance with Section 3.1.2 (6)):

$$\text{shutdown level} = f \cdot 0.4 \cdot \text{DBE}$$

For the derivation of a conservative factor f it is assumed that continued operation of the plant is acceptable as long as the stresses of the earthquake in question remain below the elastic limits and plastic deformations remain restricted to the range of geometric discontinuities. The latter is the case, provided the stresses of Level C are not exceeded. In the case of a design against the design basis earthquake Level D, the Level C stresses are attained at $\alpha \cdot$ DBE where

Because of the following assumptions made, sufficient conservativities are still available:

- careful assumptions concerning vorh σ_A and vorh σ_D,
- partly more favorable operating conditions during this special earthquake,
- narrow bands of the spectra of this special earthquake,
- experience shows that actual component behavior is far more favorable than can be demonstrated mathematically on the basis of KTA 2201.

Table A-1 specifies the groups of safety-related components and the associated values of α. An engineering assessment of these results and of the existing conservativities shows that α = 0.6 with a possibility of raising this factor up to α = 0.7 by means of plant-specific investigations (mathematical verification of all safety-related vessels having lugs or supports made of austenitic steels).

This corresponds to a factor f = 1.5 which may be increased to f = 1.75.

<table>
<tr><th>Line</th><th colspan="2">Groups of Components</th><th>Stress Limit in the Case of a DBE</th><th>Shutdown Level = α x DBE</th></tr>
<tr><td>1</td><td rowspan="2">Pipes</td><td>welded</td><td>3 S_m (= R_{mT})</td><td>0.7</td></tr>
<tr><td>2</td><td>flanged</td><td>Flange: $R_{p0.2T}$</td><td>1</td></tr>
<tr><td>3</td><td colspan="2">Supports, hangers; steel construction</td><td>$R_{p0.2T}$</td><td>1</td></tr>
<tr><td>4</td><td colspan="2">Active mechanical components</td><td>$R_{p0.2T}$ or verification of deformation</td><td>1</td></tr>
<tr><td>5</td><td colspan="2">Vessels, heat exchangers</td><td>minimum (3.6 S_m, R_{mT})</td><td>> 0.5 (up to 0.7)</td></tr>
<tr><td>6</td><td colspan="2">Electrical components, instrumentation and control components</td><td>experimental verification</td><td>1</td></tr>
<tr><td>7</td><td colspan="2">Earthquake Class IIa components and pipes</td><td>like Earthquake Class I components and pipes</td><td>1</td></tr>
<tr><td>8</td><td colspan="2">Containment</td><td>0.94 x $R_{p0.2T}$</td><td>1</td></tr>
</table>

Explanations:

Line 1

On the basis of Design Level () and the condition of deflection restriction the following can be assumed vorh σ_A < 0.75 S_m. Under conservative assumptions a = 0.7 will result.

Lines 2, 3 and 8

In as far as the yield point is basically adhered to within the scope of the deign against the design basis earthquake, the stresses remain in the elastic range rip to 100 %, design basis earthquake. Anyway, connecting elements (joints and fasteners) are assessed on a more stringent basis.

$\rightarrow \alpha$ =1.0

Line 4

Since the demonstration within the scope of the design against the design basis earthquake is basically carried out in compliance with Level B or as a deformation demonstration, no inadmissible plastic deformations occur up to 100 % design basis earthquake.

$\rightarrow \alpha$ =1.0

Line 6

In case the active functionality is demonstrated experimentally up to 100 %, design basis earthquake, the following applies:

$\rightarrow \alpha$ =1.0

Lines 5 and 7

Since the stun of the primary and secondary stresses ($P_L + Q \leq 3\ S_m$) is generally relevant for Levels A and B, the following can be assumed to apply for austenitic materials in the cases occurring in nuclear power plants: $\sigma_L^A < S_m$. Under conservative assumptions α = 0.5 will result. In the case of ferritic materials the corresponding values are α = 0.9 through 1.0. Since α =11.5 can only occur in the case of vessels with lugs or supports, there is the possibility when taking these (generally, few) vessels into exact consideration to raise the value in the most favorable case up to α = 0.7, the value for welded pipes.

Table A-1: Shutdown levels for safety-related plant components

Appendix B

List of Symbols and Abbreviations Used

$a_{ir,max}$ = recorded maximum acceleration in direction i

$a_{hrR,max}$ = resultant acceleration generated from the two recorded horizontal components $a_{ir,max}$

$a_{IN,max}$ = maximum acceleration, in relation to the design data of the design basis earthquake, using a reduction factor of 0.4

Note:
In each case, the definitions refer to the installation locations of the acceleration sensors (= locations of recording) of the seismic instrumentation.

Indexes:

r = recorded acceleration

R = resultant acceleration

IN = inspection level

i = direction in the system of coordinates (i = x, y, z)

Appendix C

Regulations Referred to in this Safety Standard

(Regulations referred to in this safety standard are valid only in the versions cited below.)

KTA 2201.4	(6/90)	Design of nuclear power plants against seismic events: Part 4: Requirements for procedures for verifying the safety of mechanical and electrical plant components against earthquakes
KTA 2201.5	(6/90)	Design of nuclear power plants against seismic events: Part 5: Seismic instrumentation