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Swiss Federal Nuclear Safety Inspectorate ENSI

Technical Safety Analysis for Existing Nuclear Installations: Scope, Methodology and Boundary Conditions

ENSI-A01/e

Guideline for Swiss Nuclear Installations

English is not an official language of the Swiss Confederation.
This translation is provided for information purposes only.

Edition September 2018 (Corrected version, 22 January 2020)

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Contents

1 Introduction

The Swiss Federal Nuclear Safety Inspectorate (ENSI) is the regulatory authority for nuclear safety and security of the nuclear installations in Switzerland. ENSI issues guidelines either in its capacity as regulatory authority or based on a mandate in an ordinance. Guidelines are implementation support documents that formalize the implementation of legal requirements, and facilitate uniformity of implementation practices. They further concretize the state-of-the-art in science and technology. ENSI may allow deviations from the guidelines in individual cases provided that the suggested solution ensures at least an equivalent level of nuclear safety or security.

2 Subject matter and scope

Safety analyses for existing nuclear installations comprise deterministic and a probabilistic evaluation of accident sequences. The requirements for the probabilistic analysis are regulated in the guideline ENSI-A05. The deterministic accident analysis consists of a technical and a radiological analysis. The requirements for the radiological accident analysis are regulated in the guidelines ENSI-A08 and ENSI-G14.

This guideline regulates the scope, methodology and the boundary conditions for the deterministic technical safety analysis for existing nuclear installations. It also specifies the methodology and boundary conditions for the review of criteria for provisional taking-out-of-service and backfitting of nuclear power plants (SR 732.114.5).

The specifications of boundary conditions for analysis and the determination of the extent of damage caused by aircraft crashes are classified and are not the subject of this guideline.

Objectives of the technical safety analysis are

a. to provide evidence that design basis accidents (SL3 accidents) comply with the fundamental safety objectives S1 to S3 as well as the criteria in accordance with Articles 8 to 11 of the DETEC (UVEK) Ordinance on the Hazard Assumptions and the Assessment of the Protection against Accidents in Nuclear Installations (SR 732.112.2);

b. to provide evidence of core coolability for SL4a accidents;

c. to specify boundary conditions for the radiological safety analysis;

d. to specify conditions and limitations for safe operation; and

e. to specify the contents of emergency operation procedures.

SL4b accidents which per definition lead to a severe core damage are analysed by PSA.

3 Legal Basis

This guideline implements the following legal provisions:

a. article 123 paragraph 5 of the Radiological Protection Ordinance RPO (SR 814.501 of 26 April 2017)

b. article 2 paragraph 4 of the DETEC Ordinance on the Hazard Assumptions and the Assessment of the Protection against Accidents in Nuclear Installations (SR 732.112.2 of 17 June 2009)

In addition, this guideline is based in particular on the following legal provisions:

c. article 7 letters c and d as well as article 8 of the Nuclear Energy Ordinance (SR 732.11 of 10 December 2004)

d. DETEC Ordinance on the Hazard Assumptions and the Assessment of the Protection against Accidents in Nuclear Installations (SR 732.112.2 of 17 June 2009

e. article 2 and 3 Ordinance on the Methodology and Boundary Conditions to Review of Criteria for Provisional Taking Out of Service of Nuclear Power Plants (SR 732.114.5 of 16 April 2008)

4 Design Basis Accidents in Nuclear Power Plants (SL3 Accidents)

4.1 Event Spectrum

a. An enveloping plant-specific spectrum of postulated initiating events based on deterministic and probabilistic methods as well as on national and international experience shall be considered. The event spectrum shall take into account specific plant states, as power operation, zero-power operation, start-up, shutdown and outage.

b. For the enveloping plant-specific event spectrum determined in letter a, the spectrum of representative events shall be defined and deterministically analysed. Appendixes A2.1 and A2.2 contain the minimum scope of the internal events for which technical safety analyses shall be carried out.

c. The spectrum of external initiating events shall be determined on the basis of site-specific hazards.

d. For the events from Appendixes A2.3, A3.1, A3.2, and A3.3 it shall be shown that with the available provisions the consequences are limited, so that these are covered by the technical accident analysis of other accidents. Otherwise, a technical safety analysis for the corresponding event shall be carried out.

e. For credible combinations of individual events it shall be shown that with the available provisions the consequences of such event combinations are limited, so that these are covered by the technical accident analysis of other accidents. Otherwise, a technical safety analysis for the corresponding event combinations has to be carried out.

4.2 Single Failure

a. Additional to the initiating event, an independent single failure shall be assumed.

b. The single failure shall be assumed where it mostly limits the provisions to cope with the accident. The findings from the plant-specific probabilistic safety analysis (PSA) shall be considered.

c. As single failure, either the loss of an active or passive technical component or the absence of an operator action required to cope with the accident shall be assumed. For this:

   1. For passive components, the assumption of a single failure can be waived if these are demonstrably of high quality.

   2. The failure of an operator action has not to be assumed if adequate alerting is available.

4.3 Accident Frequency

a. If applicable, the frequency of an initiating event shall be determined in accordance with the requirements of the guideline ENSI-A05.

b. The accident frequency shall be determined by multiplying the frequency of an initiating event by the probability of a single failure of 0.1. A smaller value down to a minimum of 0.01 can be applied if this can be proven by operating experience.

c. The operating state shall be considered when determining the accident frequency. The occurrence frequency is reduced according to the average yearly time share of such an operating state.

d. If technical specifications permit a temporary limited operation with increased coolant activity, compliance with the corresponding acceptance criteria shall be demonstrated. The annual occurrence frequency is reduced according to the fraction of the permissible temporarily limited operating time per year.

e. If technical specifications permit a temporary limited preventive maintenance of safety or special bunkered emergency systems compliance with the corresponding acceptance criteria shall be demonstrated by consideration of its unavailability. The annual occurrence frequency is reduced according to the fraction of the permissible maintenance time per year.

f. The accident shall be assigned to an accident category based on its occurrence frequency.

g. If an accident is assigned to a higher accident category due to the assumption of a single failure, compliance with the safety objectives and corresponding acceptance criteria shall also be demonstrated without a single failure.

h. Following accidents caused by natural hazards shall be analysed:

   1. accident category 2: accident with a mean exceedance frequency of the initiating event of 10^{-3} per year

   2. accident category 3: accident with a mean exceedance frequency of the initiating event of 10^{-4} per year

i. If the exceedance frequency of a natural hazard cannot be determined, the hazard impacts for accident categories 2 and 3 shall be defined in a conservative manner.

4.4 General Requirements for Technical Safety Analysis

a. The failure of structures, systems and components (SSC) as a consequence of the initiating event shall be assumed, if the corresponding SSCs are not qualified for the specific event, or if the conditional failure probability is higher than 0.01.

b. For the reactor and spent fuel pool analyses a core configuration and burn-up shall assumed covering the in-core fuel management strategy.

c. The total loss of off-site power shall be assumed to occur at the most unfavourable time during the accident sequence, provided that this negatively affects the accident sequence.

d. A stuck rod shall be considered for the most effective control rod.

e. The response delay until the respective protective action becomes active shall be considered.

f. In general, only safety systems and special bunkered emergency systems can be credited to carry out a safety function within the first 10 hours of the accident. The credibility of other equipment, which is qualified to cope with the accident of the corresponding event, shall be justified.

g. The operation of non-safety classified systems (operation systems) shall be considered if they can have unfavourable effects on the accident sequence.

h. Operator actions must not be credited within 30 minutes after the initiating event. Deviations from this regulation shall be justified.

i. Prescribed safety-relevant actions of the operating personnel shall be considered if sufficient time is available for the diagnosis and the execution of the corresponding action.

j. For sites with multiple units, potential mutual interaction shall be considered.

4.5 Calculation Programmes and Plant Models

a. Neutron-physical, thermal-hydraulics and structure-mechanical calculation programmes shall be used for technical safety analyses. The programmes shall be compatible as far as for the specific analysis required.

b. The important physical phenomena for the specific accident sequence shall be considered in the technical safety analysis.

c. The calculation programmes shall be verified and validated.

d. The development, application, and maintenance of the calculation programmes shall be carried out by qualified personnel and in the frame of a certified QMS. The requirements of the guideline ENSI-G07 shall be considered.

e. The plant models shall be verified and as far as possible validated. The plant models shall reflect the current plant state or planned plant modifications, if relevant for the safety analysis.

f. One of the following three options can be used for performing technical safety analyses:

   1. best estimate calculation programmes in combination with realistic initial and boundary conditions and uncertainty analyses (BEPU) (see Chapter 4.5.1)

   2. best estimate calculation programmes in combination with conservative initial and boundary conditions (see Chapter 4.5.2)

3. Conservative calculation programmes in combination with conservative initial and boundary conditions (see Chapter 4.5.3)

4.5.1 Best Estimate Calculation Programmes and Realistic Initial and Boundary Conditions with Uncertainty Analyses (BEPU)

a. Realistic initial and boundary conditions as well as realistic assumptions for the effectiveness of the safety systems shall be used. Appropriate statistical distributions shall be assumed for the (key) input parameters.

b. If conservative assumptions are necessary for modelling of specific phenomena, or if conservative initial and boundary conditions are used, this shall not lead to results that are unrealistic or affect the outcome significantly.

c. Compliance with the technical acceptance criteria shall be demonstrated using an adequate confidence level.

4.5.2 Best Estimate Calculation Programmes and Conservative Initial and Boundary Conditions

a. Conservative initial and boundary conditions shall be assumed.

b. The required systems shall be assumed to perform with minimal efficiency according to their design unless a higher efficiency has unfavourable effects on the accident sequence.

4.5.3 Conservative Calculation Programmes and Conservative Initial and Boundary Conditions

The use of conservative calculation programmes shall be justified. For the application case, it shall be demonstrated that the calculation programme complies with the state of the art.

4.6 Safety Objectives and Technical Acceptance Criteria

a. Technical safety analysis shall provide evidence that the technical acceptance criteria required in Articles 8 to 11 of the DETEC Ordinance (SR 732.112.2) have been adhered to in accordance with their corresponding accident category.

b. Technical safety analysis shall provide evidence that the safe state can be achieved and ensured for at least 72 hours using on-site available equipment.

c. The technical safety analysis shall provide evidence that in the long term the cold shutdown state can be achieved.

4.7 Additional Requirements for Technical Safety Analysis of Internal Fires and Internal Flooding

4.7.1 Internal Fires

a. For the event spectrum of internal fires, the sum of occurrence frequencies of all fire scenarios with the same damage patterns shall be determined for each relevant fire compartment.

b. The identification and selection of relevant fire compartments shall be conducted in accordance with the guideline ENSI-A05. In addition to the PSA relevant equipment, components potentially containing a sufficient amount of radioactive materials shall be considered. A screening out of a fire compartment in the deterministic safety analysis only based on the PSA quantitative screening criteria (see guideline ENSI-A05) is not allowed.

c. The occurrence frequencies of internal fires shall be determined in accordance with the relevant section of guideline ENSI-A05.

d. The extent of damage of each fire scenario shall be determined and documented taking into account the fire detection and fire-fighting measures as well as resistance of specific fire barriers (walls, doors, hatches, and separations).

e. If there is no detailed determination of the extent of damage, the most unfavourable consequences for the components of the corresponding fire compartment shall be assumed in regard to the accident sequence.

f. The SSCs available to cope with the accident shall be identified taking into account a single failure (see Chapter 4.2). The failure of automatic fire detection and fire-fighting measures as well as components of fire compartments shall be included in single failure considerations.

4.7.2 Internal Floods

a. For the event spectrum of internal floods, the sum of occurrence frequencies of all flooding scenarios with the same damage shall be determined for each relevant flood area and system causing flooding.

b. The identification and selection of relevant flood areas shall be conducted in accordance with the guideline ENSI-A05. In addition to the PSA relevant equipment, components potentially containing a sufficient amount of radioactive materials shall be considered. A screening out of flood areas in the deterministic safety analysis only based on the PSA quantitative screening criteria is not allowed.

c. The occurrence frequencies of internal floods shall be determined in accordance with the guideline ENSI-A05.

d. The extent of damage of each flood scenario shall be determined and documented as a function of time taking into account the flood detection and protection measures.

e. If there is no detailed determination of the extent of damage, the most unfavourable consequences for the components of the corresponding flood area shall be assumed in regard to the accident sequence.

f. The SSCs available to cope with the accident shall be identified taking into account a single failure (see Chapter 4.2). The failure of technical equipment (i.e. flood detection, sump pumps, and isolation valves) shall be included in single failure considerations.

4.8 Additional Requirements for Technical Safety Analysis of Events Initiated by Natural Hazards

a. The technical safety analysis of events initiated by natural hazards shall be carried out on the basis of the current hazard assumptions accepted by ENSI.

b. For sites with multiple units, all units shall be assumed to be affected.

4.8.1 Earthquake

a. All SSCs required to cope with the accident shall be identified. Further, the component and structures shall be identified whose failure within the earthquake can jeopardize the accident management due to mechanical interactions or seismic-induced fires, floods, and explosions.

b. Specific evidence for functionality, integrity, or stability shall be provided for all SSCs identified in letter a. The loads resulting from power operation and earthquake shall be considered cumulatively for the identified SSCs.

c. For determination of the floor response spectra realistic dynamic properties of relevant structures as well as dynamic soil-structure-interactions shall be considered.

d. The seismic capacity of the reactor coolant pressure boundary shall be estimated on the basis of a coupled model consisting of the structure of reactor building and components of the reactor coolant pressure boundary.

e. The seismic capacity of other SSCs required for the accident management shall be estimated by means of fragility analyses, CDFM calculations (Conservative Deterministic Failure Margin) or deterministic methods. For this, following aspects shall be taken into account:

1. The estimation of the seismic capacity of building structures and anchorages shall conform to the concept of partial factors according to SIA- or equivalent EU-Norms. For this, the specific loads respectively requirements for the nuclear power plants shall be considered.
2. HCLPF values shall be estimated for all mechanical and electrical equipment required for accident management.
3. The seismic capacity for a representative selection of mechanical and electrical components shall be determined by deterministic methods. The representative selection shall include the weakest components with regard to seismic capacity.
4. The seismic capacity of hydraulic facilities whose failure might endanger the plant safety shall be determined in accordance with the methodology of the Swiss Federal Office of Energy guideline on “Safety of Water-Retaining Structures” assuming the seismic hazard accepted by ENSI.

4.8.2 External Floods

a. Water-intake plugging and damage due to debris and sediments as a consequence of the initiating event shall be considered if this cannot be ruled out.

b. The response of SSCs to hydrostatic and hydrodynamic loads (including erosion and flood/debris impact) shall be analysed.

4.8.3 Extreme Weather Conditions

a. The effects on the plant and the resulting damage caused by the hazards listed in Appendix A3.1 shall be determined.

b. Credible combinations of extreme weather conditions shall be considered.

5 SL4a Accidents in Nuclear Power Plants

a. A plant-specific spectrum of SL4a accidents based on deterministic and probabilistic methods shall be defined. Appendix 4 contains the minimum scope of the events to be considered.

b. A safety margin analysis shall be carried out on the basis of the requirements in Chapter 6 for natural hazards according to Appendix A3.1. The analysis shall demonstrate the margin to which extend a severe fuel damage in the reactor core and spent fuel pools can be avoided.

c. The requirements in Chapter 4.4 letters a and j shall be considered in the safety analysis.

d. All SSCs available on the plant site can be credited. The availability and functionality shall be shown.

e. Actions of the operating personnel can be taken into account if sufficient time is available for diagnosis and execution.

f. Realistic initial and boundary conditions can be assumed. Uncertainties in the initial and boundary conditions shall be estimated.

g. The applicability of calculation programmes shall be demonstrated.

h. The peak clad temperature shall be limited in order to avoid an excessive embrittlement and oxidation of the cladding.

i. The technical safety analysis shall provide evidence that the safe state can be achieved and ensured for at least 72 hours using on-site available equipment.

6 Safety Margin Analysis of Events Initiated by Natural Hazards in Nuclear Power Plants

a. The SSCs required to safely shut down the plant (shutdown paths) and for ensuring the integrity of the primary containment shall be determined.

b. The safety margin for a shutdown path or the primary containment integrity shall be determined as the capacity of the weakest SSC in relation to the specific loads during the accident which has the annual exceedance frequency of the initiating event of 10^{-4} per year.

7 Review of Core Coolability for Design Basis Accidents with regard to Provisional Taking Out of Service

a. All available and permanently mounted SSCs as well as qualified mobile equipment can be credited to cope with the accident.

b. In accordance with the requirements of Chapter 4.2, a single failure shall be assumed.

c. Actions of the operating personnel can be credited. The effectiveness of the actions shall be assessed.

d. Best estimate calculation programmes in combination with realistic initial and boundary conditions can be used.

e. The peak clad temperature shall be limited in order to avoid an excessive embrittlement and oxidation of the cladding.

8 Design Basis Accidents in Other Nuclear Installations and Storages (SL3 Accidents)

a. A facility-specific spectrum of postulated initiating events shall be determined. As far as applicable, the requirements of Chapter 4.1 letters a to d shall be taken into account.

b. A single failure shall be assumed in accordance with the requirements of Chapter 4.2.

c. The accident frequency shall be determined based on the requirements in Chapter 4.3.

d. The requirements of Chapter 4.4 letters a, c and i shall be taken into account for the technical safety analysis.

e. Only qualified equipment shall be credited for the accident management.

f. The technical safety analysis shall be carried out on the basis of:

   1. detailed calculations considering the requirements of Chapter 4.5 and complying with the state of the art, or;

   2. conservative engineering judgment.

g. If applicable, the requirements in Chapters 4.7 and 4.8 shall be considered.

h. The technical safety analysis shall provide evidence that the requirements in Article 8 of the DETEC Ordinance (SR 732.112.2) are complied with.

9 Documentation

a. The complete verification process and the results of the technical safety analysis shall be comprehensively documented.

b. The calculation methods applied and plant models shall be described in a complete manner.

c. The technical plant situation after the accident (extent of damage) shall be described as boundary conditions for the radiological safety analysis. The requirements of the guideline ENSI-A08 shall be considered.

d. The documentation required for the review shall be submitted to ENSI in electronic form.

10 List of References

The guideline of Federal Office of Energy (Bundesamt für Energie BFE) on Safety of Water-Retaining Structures, Part C3, Earthquake Safety, Version 2.0, 1. February 2016

This guideline was approved by ENSI on 17 September 2018.

Director of ENSI: sig. H. Wanner

Appendix 1: Terms and Definitions (According to ENSI Glossary)

Calculation Programme

A calculation programme is as computer code which models and simulates neutron-physical, thermohydraulic and structure-mechanical phenomena in a nuclear facility. The calculation programmes are plant-independent.

Deterministic Safety Analysis

The deterministic safety analysis is a quantitative prediction or assessment of the plant behavior during an accident. The deterministic safety analysis shall provide evidence that the protection sequences remain effective for the spectrum of representative events in compliance with the fundamental safety objectives.

Plant Model

The plant model is a computer-based reproduction of systems and equipment of a nuclear installation/ power plant for a numerical simulation of accidents.

Representative Event

Representative event is a bounding event sequence in terms of compliance with safety objectives. Representative event represents a group of postulated initiating events which lead to a similar challenge to the safety functions and barriers.

Shutdown Path

The shutdown path is a combination of systems and measures which ensure the safe shutdown state of a nuclear power plant.

The Swiss nuclear power plants have three defined shutdown paths usually featuring diverse redundancies:

Shutdown Path 1: It consists of the safety systems.

Shutdown Path 2: It consists of the special protected bunkered safety systems. It is primarily used to manage extreme external events or malicious actions.

Shutdown Path 3: It consists of all permanently installed SSCs as well as mobile equipment for which the availability and functionality can be shown for the resulting loads from the corresponding accident.

SL3 Accidents (Safety Level 3)

SL3 accidents are design basis accidents (SE3 is the German abbreviation for the third level of defense with regard to the concept of defense in depth).

SL4a Accidents (Safety Level 4a Events)

SL4a accidents are beyond design base accidents which are to be managed on the fourth level of defense with regard to the concept of defense in depth without a severe core damage (SE4 is the German abbreviation for the fourth level of defense).

SL4b Accidents (Safety Level 4b)

SL4b accidents are beyond design accidents which lead to a severe core damage (SE4 is the German abbreviation for the fourth level of defense with regard to the concept of defense in depth).

Qualified Equipment

Qualified equipment comprises permanently installed SSCs as well as mobile equipment, whose availability and functionality for the resulting loads from the corresponding accident (for which the equipment is qualified) can be shown.

Appendix 2: Internal Initiating Events

A2.1 Events in Reactor Coolant Circuit

Event	PWR	BWR
Inadvertent Opening of or Erroneous Failure to Close Main Steam Relief or Main Steam Safety Valves	●	●
Inadvertent Closing of All Main Steam Isolation Valves	●	●
Small, Intermediate, and Large Breaks in Reactor Coolant Circuit or in a Reactor Coolant Line[1] (including Double-Ended Break of a Main Reactor Coolant Line)	●	●
Leaks and Breaks of a Main Steam Line Inside and Outside the Primary Containment[1] (including Double-Ended Break)	●	●
Leaks and Breaks of a Feedwater Line Inside and Outside the Primary Containment[1] (including Double-Ended Break)	●	●
Leaks and Breaks of Reactor-Coolant-Containing Extraction and Instrument Lines Outside the Primary Containment	●	●
Leaks in Heat Exchangers and Breaks of Piping Connected to the Reactor Coolant Circuit which are located completely or partially outside the Primary Containment (Interfacing System LOCA)	●	●
Total Loss of Offsite Power (SBO)	●	●
Inadvertent Withdrawal of a Control Assembly or Control Rod	●	●
Control Rod Ejection	●	
Control Rod Drop		●
Failure of Volume Control System	●	
Failure of a Reactor Coolant Pump or Reactor Recirculation Pump	●	●
Inadvertent Opening of or Erroneous Failure to Close a Pressurizer Safety Valve	●	
Double-Ended Break of a Steam Generator Tube (SGTR)	●	

[1] If a leak-before-break behavior has been demonstrated, an analysis of the mechanical consequences of double-ended pipe breaks can be omitted

A2.2 Events in Spent Fuel Pools

Event	PWR	BWR
Loss of Spent Fuel Pool Cooling	●	●
Leaks and Breaks of Piping connected to Spent Fuel Pool	●	●
Events according to A2.3, if applicable	●	●
Inadvertent Boron Dilution	●	

A2.3 Specific Internal Hazards

Event	PWR	BWR
Fire	●	●
Flood	●	●
Failure of Large Components (e.g. Turbine Damage)	●	●
Fuel Handling Accident	●	●
Heavy Load Drops	●	●
Pipe Whipping	●	●
Missile	●	●
Explosions	●	●

Appendix 3: External Initiating Events

A3.1 Natural Hazards

Event	PWR	BWR
Earthquake	●	●
Flood	●	●
Extreme Weather Conditions: • Wind • Tornado • Minimal and Maximal Atmospheric Temperature • Minimal and Maximal River Temperature • Forest Fire • Snow Load • Heavy Rain on the Site • Drought	●	●
Lightning Stroke	●	●

A3.2 Man-made External Events

Event	PWR	BWR
Explosions	●	●
Gas Clouds	●	●
Fire	●	●
Deterioration or Failure of External Cooling Water Supply	●	●

A3.3 Events in Spent Fuel Pools

The events in A3.1 and A3.2 shall be also applied for spent fuel pools.

Appendix 4: SL4a Accidents

A4.1 SL4a Accidents in Reactor Coolant Circuit

Event	PWR	BWR
Anticipated Transient without Scram (ATWS) for		
• Total Loss of Offsite Power (SBO)	●	●
• Main Isolation Valve Closure (MSIVC)		●
• Turbine Trip Without Bypass	●	
• Subcooling Transient	●	●
Loss of Main Control Room	●	●
Long-Term Loss of Primary Ultimate Heat Sink	●	●
Total Station Blackout (TSBO)	●	●
SL3 accidents (Chapter 4.1 letter b) with accident frequency lower than 10^{-6} per year resulted due to assumptions of • single failure or • temporarily limited operation conditions (Chapter 4.3 letters c and e[2])	●	●
SL3 accidents (Chapter 4.1 letter b) on the assumption of the second, time delayed criterion for reactor scram	●	●
Natural Hazards (Chapter 5 letter b)	●	●
Loss of the Secondary Residual Heat Removal Systems	●	
Multiple Steam Generator Tube Ruptures	●	

A4.2 SL4a Accidents in Spent Fuel Pools

Event	PWR	BWR
Long-Term Loss of Active Spent Fuel Pool Cooling	●	●

[2] Corrected on 22 January 2020

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