KTA-2000

Another attempt is underway to modernise the existing rules and guidelines in Germany: the project “KTA-2000”. The objective of this project is the presentation of the reactor safety requirements in a comprehensive and hierarchic structure based on systematic approach. The pyramid shape of the new structure is based on the existing KTA safety standards. Existing gaps shall be closed according to the state of the art, new developments are included into plant assessment. The KTA safety standards determine and put the safety related technical requirements in concrete terms. It is understood that after fulfilment of these requirements the required precaution against damage is foreseen according to the state of science and technology and that the protection goals are achieved. “KTA 2000” consists of the following parts:

a) the KTA Safety Fundamentals 2000,
b) seven KTA basis rules, and the
c) KTA standards

Whereas the about 100 KTA standards are existing and under regular revision, the KTA fundamentals and the basis rules are presently in preparation.

According to the KTA fundamentals the integral holistic safety concept is basically preventive and follows closely the defence-in-depths concept which has to be applied for the three main areas: technology, man and organisation. The concept of the four safety protection goals is mainly based on deterministic principles, it can be supplemented by probabilistic elements in order to allow to state the safety related balance and the level of safety. This concept is applied for the safety evaluation.

The requirements necessary to achieve the protection goals are described in the KTA Basis rules. The first four rules contain design-independent requirements which can be directly assigned to the four protection goals:

1) Reactivity control
2) Cooling of fuel elements
3) Confinement of radioactive substances
4) Limitation of radiation exposure

The structure within these rules follows essentially the following order: protection goal, partial protection goals, operational or safety functions assigned to the partial protection goals, safety level. For example the structure of the second basis rule “cooling of fuel elements” contains the two partial protection goals “ensuring heat removal from fuel elements to ultimate heat sink” and “maintaining sufficient coolant inventory by minimising losses and by replenishment coolant from reservoirs”. The corresponding functions for the first partial safety goal are: heat removal from core, heat removal from secondary system (PWR only), heat removal from wet well (BWR only), heat removal from spent fuel elements in pool, heat removal from containment, heat transport in cooling chains (including ultimate heat sink); from the second partial safety goal: reactor coolant replenishment, minimisation of reactor coolant losses, steam generator feeding (PWR only), ensuring inventory of wet well (BWR only), minimisation of losses from wet well (BWR only), spent fuel pool water replenishment, minimisation of water losses from spent fuel pool. The requirements for each of these functions are subdivided according to the four safety levels (see Figure 3).

The remaining three basis rules contain additional design-independent requirements needed to achieve the protection goals. These rules are

5) General technical requirements
6) Methodology of safety demonstration
7) Personal-administrative measures

KTA basis rule 5 deals with requirements on civil engineering structures and plant equipment including the energy supply and the instrumentation and control systems, the protection of components and systems required for the fulfilment of operational and safety functions against internal and external impacts. In KTA basis rule 6 the requirements on the methodology of safety demonstration are dedicated to deterministic, probabilistic and engineering approaches. Regarding the scope of safety demonstration, it is distinguished between different causes: essential modifications of plants or of the plant operation, comprehensive safety examinations, assessments related to specific causes or events. Further requirements of this basis rule concern the reliability of safety functions and barrier functions as well as analytical methods, models and data. To the requirements with regard to the personal-administrative measures (basis rule 7) belongs the safety-oriented acting of all persons involved in the safe plant operation (regulatory body, assessors, vendors, utility, suppliers).

GRS Methodology Considering Operational Experience

The need for a methodology to be applied to assess the safety of operating German NPPs according to the defense-in-depth concept is strongly felt. Those nuclear regulations which represent the state of the art at the time of licensing of the KONVOI type NPPs - based on the Atomic Energy Act with its relevant ordinances and the BMI criteria, the RSK guidelines, the RSK and SSK recommendations as well as the KTA safety standards – are suitable to a limited extend only for this purpose. Although these rules and regulations are based on the defence-in-depth concept, their requirements do not make distinctions according to the individual safety levels of this concept. This is because these rules and regulations are related mainly to design, construction and commissioning of NPPs and were not primarily developed to consider requirements that can be derived from a long-term operational experience. These rules and regulations were created at a time when there was a broad willingness of consent to solve problems and find results among all participants and when the necessary expertise was still present to the required extent. It is therefore not surprising that the present rules and regulations have to be supplemented. Furthermore, they almost exclusively concentrate on a deterministic safety assessment, but in the meantime probabilistic safety assessments have become the state of the art.

At GRS such a development is underway (Rittig, 2000). To perform the safety assessments in a uniform and comprehensible way, GRS can make use of a methodology which had been developed early from practical work, which has been and is steadily improved, and which is now applied by GRS as a guidance for all safety assessments of existing reactors. This development is in line with the obligations within the framework of the Nuclear Safety Convention.

The measures for protecting the barriers, which enclose the radioactive substances concentrated in the reactor core - fuel rod cladding, the reactor coolant pressure boundary and the containment-, are assigned to the four levels of defence-in-depth concept. The fourth level is differentiated in the safety practice in Germany with a distinction being made according to the protection measures in

- measures to control ATWS and emerency conditions (comprehensive impacts owing to civilisation): level 4a,
- preventive accident management measures: level 4b,
- accident mitigation measures: level 4c.

The safety objectives are subdivided into technical and radiological criteria. The technical safety objectives are inter-linked with the radiological ones in the way that meeting the technical safety objectives represents a precondition for achieving the radiological safety objectives. The technical safety objectives are determined by the first three of the four protection goals of reactor safety: reactivity control, fuel element cooling, confinement of radioactive substances by the three barriers mentioned above. The integrity of the second barrier is supported strongly by the break preclusion concept.

There are strong basic deterministic requirements related to the break preclusion concept due to its importance within the barrier concept. The scope of application of the break preclusion concept up to the safety level 4b) comprises the reactor coolant pressure boundary (RCPB) including primary isolation, i.e. pipes > 200 mm (large RCPB pipes), vessel (RPV, PRZ), casing (MCP), primary head and tube sheet of steam generators, steam generator shell, components with a temporary or permanently high energy content inside the containment apart from the RCPB. The major pre-conditions for the application of the break preclusion concept are: advantageous material properties for processing, welding, expansion, a construction advantageous for stress conditions and inspection, qualified production and test procedures, and provisions to allow clear identification of manufacturing faults by in-service inspections. Essential conditions have to be fulfilled also during the plant operation (levels 1 and 2) besides the stress limitation, i.e. observation of the relevant system parameters, monitoring fatigue exploitation, water chemistry, changes of material properties, changes of movable component supports, main coolant pump vibrations, and leakage monitoring of wall-penetrating cracks and smallest boric leakage.

By this it must be reliably prevented that no inadmissible influence of stress limitation results from exceeding pressures, additional thermal loads, malfunctions of supports, corrosion, erosion processes, operational material changes. A redundant leak detection with specified sensitiveness as well as detection times in the range of hours, also an early detection of fault sizes in the range of registration limits with a high probability (typical: 90 %) must be ensured. These preconditions, together with additional fracture mechanical analyses and representative experiments concerning the crack stability and growth, result in the desired features of the concerned structures. These features are:

- no permanent growth of cracks in the range of the registration limits of test indications,
- stable crack behaviour for
  - a wall-penetrating crack of the large RCPB pipes clearly exceeding the sensitivity threshold of advanced leak detection systems,
  - a crack in the volume of RCPB vessels and SG shell measuring two times the dimensions of the registration limits of the test indications.

Provided the break preclusion concept can be applied the upper limit of the size of LOCA of level 3 can be restricted 0.1A. The LBLOCA can then be treated in level 4a in the safety assessment notwithstanding of the fact that the LBLOCA was the design basis for the emergency core cooling system and the containment.

The accident analysis needed mainly for the assessment in safety level 3 shall be preferably "best-estimate" analysis of the selected events.

The basic deterministic safety requirements alone are not sufficient for assessing safety, the safety level of a nuclear power plant is determined by the effectiveness of the foreseen equipment and the measures but also by the technical execution respectively the implementation of the administrative measures and by the positive operational experience of the barriers and the protection measures. For this reason the basic deterministic safety requirements are supplemented by probabilistic parameters. Three damage states are defined by considering the interconnections between the defence-in-depth concept and the PSA results:

- A <u>system damage state</u> is reached if the operational and safety measures for ensuring core cooling on the safety levels 1 - 3 are insufficiently available or not effective in the case of challenge. Core melting has not yet occurred in this state and can be prevented by further protection measures.
- A <u>core damage state</u> is reached if the protection measures of levels 4a and 4b are also not sufficiently available or not sufficiently effective in the case of challenge. The melting temperature of the nuclear fuel is reached with subsequent core melting.
- A <u>plant damage state</u> is reached if the safety measures of level 4c are also not sufficiently available or not sufficiently effective in the case of a challenge. The containment fails early as a consequence of the core damage.

Two parameters of these probabilistic values are internationally in use, namely the core damage state and the plant damage state. To better fulfil the requirements of safety practice in Germany, where there is a strict differentiation between the "classical failure mode design" of the first three safety levels and the measures reaching beyond that, GRS uses reference values for the system damage state and the plant damage states. However, these reference values, which represent the basic probabilistic safety requirements, are based on the international assessment criteria.

INSAG-12, an reassessment of INSAG-3 of 1988, recommends the following probabilistic assessment criteria for existing plants: core damage states > 10^{-4}/ry, core damage states with an early re-lease < 10^{-5}/ry, and for future plants: core damage states < 10^{-5}/ry, core damage states with an early re-lease < 10^{-6}/ry.

For the basic probabilistic safety requirements, GRS applies reference summation values for the system damage states <10^{-4}/ry and for the plant damage states <10^{-6}/ry in the assessment of the existing power plants in Germany. This means that GRS applies assessment criteria to the nuclear power plants in Germany that are internationally applied to future plants. The factor 10 between the system damage state according to the GRS approach and the core damage states according to the IAEA or INSAG approach is based on GRS' findings from the PSAs carried out for the plants in Germany referring to the measures available beyond the operational and incident-related design to further prevent core damage.

ABBREVIATIONS

AM	Accident management
ATWS	Anticipated transients without scram
BMI	Bundesministerium des Innern (Federal ministry of the interior)
BWR	Boiling water reactor
DEGB	Double-ended guillotine break
FRM 2	Forschungsreaktor München 2 (Research reactor Munich 2)
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit mbH
IBC	Initial and boundary conditions
IAEA	International Atomic Energy Agency
INSAG	International nuclear safety advisory group (to the director general of IAEA)
KTA	Kerntechnischer Ausschuss (German nuclear standard committee)
LBLOCA	Large break LOCA
LOCA	Loss of coolant accident
MCP	Main coolant pump
NPP	Nuclear power plant
PRZ	Pressuriser
PSA	Probabilistic safety analysis
PSR	Periodic safety review
PWR	Pressurised water reactor
RCPB	Reactor coolant pressure boundary
RPV	Reactor pressure vessel
RSK	Reaktorsicherheitskommission (German reactor safety commission)
ry	per reactor and year of operation
SG	team generator
SSK	Strahlenschutzkommission (German radiation protection commission)

REFERENCES

Rittig, D., 2000, Analyses and Assessments of Reactor Safety, GRS, Annual Report 1999, Gesellschaft für Reaktorsicherheit (GRS) mbH, also found via the home-page of GRS: www.grs.de

APPLICANTS ACCIDENT ANALYSES IN SECOND PART LICENSE FOR KONVOI-PLANTS

Safety Analysis Report (S.A.R.) with 3 Handbooks:

- ECC Handbook (LOCA)
- Plant Dynamics Handbook (Transients incl. ATWS)
- Core Design Handbook

Content of ECC-Handbook:

0. Purpose, Scope, Handling, Abbreviations

Conceived as Living handbook, Basis for design, catalogue of transients, specifications and licensing. Handbook contains LOCA in primary system, it contains also core damage analysis, description of codes, description of essential plant data and code input data.

1. Initial and boundary conditions

1.1 Description of Essential Systems

Emergency core cooling and decay heat removal system, systems for secondary heat removal.

1.2 Reactor Protection System

Initiation criteria for scram, MCP shut-off, MFW shut-off, closure of MSIVs, for ECCS, emergency power diesel load sequence, accumulator shut-off, automatic secondary cooldown, secondary side safety valve isolation valve shut-off.

1.3 Reactor Core

Initial and boundary core conditions, kinetics data

1.4 Code-specific Boundary Conditions

Codes for LBLOCA (2A-0.25A): LECK-4/MOD2 (ANLASS = stationary conditions, BENN = leak mass flow rate, LECK = system code for instationary blowdown phases), HYDRANS = injection mass flow rate, REWAS = residual water in lower plenum, WAK-3 = refill and reflood, BETHY-AZ = maximum cladding temperatures, COCO = containment back pressure, CARO-D = initial conditions of fuel rods, Code for SBLOCA: RELAP5/MOD1

1.4.1 LECK-4/MOD2

boundary conditions, nodalisation

1.4.2 HYDRANS

boundary conditions, characteristics, nodalisation

1.4.3 WAK-3

boundary conditions

1.4.4 BETHY

boundary conditions

1.4.5 RELAP5/MOD1

initial and boundary conditions, nodalisation

1.1 Relevant Plant Differences between Pre-KONVOI (Generation 3: KBR, KWG, KKP 2) and PWR 1300 (KONVOI)

main differences: fuel elements, MCP, lower plenum internals

2. Summary of Results of LBLOCA ECC Calculations

7 LBLOCA calculations: 2A in CL, PCL, HL, 1A in CL, 0.5 A in CL, 0.25 A in CL, 2A in CL(best estimate)

3. Results of LBLOCA ECC Calculations

General description of LBLOCA sequence

3.1 Double-Ended Guillotine Breaks (2A)

3.1.1 2A in Cold Leg

Conservative initial and boundary conditions (e.g. SF and repair)

3.1.2 2A in Crossover Leg

3.1.3 2A in Hot Leg

3.2 Leak Size Less than 2A

3.2.1 1A in Cold Leg

3.2.2 0.5A in Cold Leg

3.2.3 0.25A in Cold Leg

3.1 Special Investigations

3.1.1 2A in Cold Leg with best estimate conditions

Purpose: Identification of safety margins, all systems available, no emergency power conditions,...

4. Results of ECC Calculations for SBLOCA and MBLOCA

General description of LBLOCA sequence, definition of SBLOCA, MBLOCA

4.1 Leaks of Main Coolant Lines

All individual reports contain description of events, initial and boundary conditions, assumptions on availability of ECCS, calculated results, nodalisation scheme, about 40 plots

4.1.1 1105 cm^2-leak (0.25A) in cold leg

4.1.2 380 cm^2-leak in cold leg (injection line)

4.1.3 380 cm^2-leak in hot leg (injection line)

4.1.4 437 cm^2-leak in hot leg (surge line)

4.1.5 250 cm^2-leak in cold leg

4.1.6 160 cm^2-leak in cold leg

4.1.7 100 cm^2-leak in cold leg

4.1.8 70 cm^2-leak in cold leg

4.1.9 50 cm^2-leak in cold leg

4.2 Leaks at Reactor Pressure Vessel

Fulfilment of RSK guide line

4.2.1 20 cm^2-leak at bottom

4.1 Leaks at pressuriser

4.1.1 Inadvertent opening and stuck-open of relief valve (20.8 cm^2-leak)

5. Results of core damage analysis

5.1 Core damage analysis

5.1.1 Realistic and conservative core damage

5.1.1 Influence of break location and size on core damage

6. Description of codes

6.1 LBLOCA

6.1.1 Blowdown phase (LECK-4/MOD2 code system)

Users manual, input data, output data

6.1.2 Refill and reflood phase (HYDRANS, WAK-3, REWAS)

Users manual, input data, output data, verification based on PKL test K9

6.1.3 Core heat-up (BETHY-AZ)

Model description, input deck

6.2 Small and medium leaks

6.2.1 RELAP-5/MOD1

Code manual (system models, numerical methods, user guide, input requirements

6.2.2 BETHY

see 6.1.3

6.1 Core Damage Analysis

6.3.1 BETHY-AZ

7. INPUT DATA

Complete input decks for each case

7.1 LBLOCA

7.2 SBLOCA and MBLOCA

7.3 Core Damage Analysis

Content of Plant Dynamics Handbook:

0. Purpose, Scope, Handling

Basis for design, commissioning, operation, catalogue of transients, specifications and licensing
Specified operation, Disturbed operation, incidents, not: LOCA, SS-procedures
Code description

1. Analyses

1.1 Specified Operation

1.1.1 Normal Operation

(analysed for KKP-2, because of same controllers for coolant pressure and temperature, Code LOOP-7)

1.1.1.1 Slow load changes (+- 10%/min)

1.1.1.2 Fast load changes (+- 10%)

1.1.2 Abnormal Operation

1.1.2.1 Turbine shutdown

(Code: NLOOP, Calculation for BOC and EOC)

1.1.2.2 Load Rejection (covered by 1.1.2.1)

1.1.2.3 Fast reactor shut down (scram)

(Code: NLOOP, Calculation for EOC, FP, Scram initiates turbine shutdown, i.e. PCS coolant temperature falls only to secondary saturation temperature)

1.1.2.4 Failure of one MCP

(Code: NLOOP, Calculation for FP at BOC and EOC, no scram, new power level about 40%)

1.1.2.5 Failed function of auxiliary systems in PCS

(no code application, failed function of CVCS: coolant mass, coolant temperature, boron concentration)

1.2 Disturbed Operation and Incidents

1.2.1 Reactivity and Power Distribution Anomalies

(analysed for KKP-2 except for 1.2.1.4.2 because of same effectivity of control elements and because of conservatism of the analyses, but KKP-2 = 16x16, Convoy = 18x18)
(in common: Code LOOP-7 with the exception of 1.2.1.4.2, ignorance of first scram signal and of all controllers for control rod movement)

1.2.1.1 Incidents during plant start-up

(ZL, 225°C, disturbance 60 pcm/s, 20 pcm/s due to erroneous withdrawal of control elements with 1.25 cm/s)

1.2.1.2 Insertion of control elements

(covered by 1.2.1.3)

1.2.1.3 Drop of one control elements

($\Delta\rho$=–0.2%)

1.2.1.4 Withdrawal of control elements

1. dρ/dt□2.875pcm/s□ ignorance of PBCORR as first scram signal, second: PK
2 Withdrawal of D-Bank with 15 pcm/s: 4 calculations with code NLOOP (BOC and EOC from 50% initial power, BOC and EOC from FP)

1.2.1.5 Ejection of one control element

($\Delta\rho$=+0.2%, ignorance of PBCORR as first scram signal, second: PK)

1.2.1.6 Change of boron concentration due to failed function of CVCS

($d\rho/dt$=0.7pcm/s, ignorance of PBCORR as first scram signal)

1.2.1.7 Detaching of boron containing sediments in core

($\Delta\rho$=+0.25%, ignorance of PBCORR as first scram signal)

1.2.2 Disturbed Heat Removal without Loss of Primary Coolant

1.2.2.1 Malfunction of turbine bypass valves

(Code: NLOOP, 3 cases: Opening of one bypass valve, malfunction of all bypass valves with and without RELEB)

1.2.2.2 Fast turbine shutdown with failures of various systems

(Code: NLOOP, several cases at FP: failure of condenser at BOC and EOC, same but ignorance of first scram signal at BOC, turbine shutdown without turbine by-pass at BOC and EOC, same but ignorance of first scram signal at BOC)

1.2.2.3 LONOP

(Code NLOOP; FP, consideration SS pumps, of auxiliary spray from CVCS and of automatic partial cooldown if p>86 bar, scram due to n_{MCP}<94% and PBCORR at same time)

1.2.2.4 Erroneous closure of MSIVs

(Code: NLOOP; FP, SF: MS-SV, LONOP at time of scram; 5 cases: closure of one MSIV with first scram signal (p_{sec} > 86 bar) with BOC and EOC, same two cases with second scram signal (SG level < min 1) but partial cooldown to 75 bar considered as in the cases 1 and 2, closure of all MSIVs with simultaneous scram)

1.2.2.5 Malfunction of FW supply

(Code: NLOOP; FP, 4 cases: failure of one MFW pump at BOC and EOC, failure of both MFW pumps at BOC and EOC,)

1.2.2.6 Blockage of one MCP

(Code: NLOOP, several cases with two different time constants: at BOC with and without consideration of RELEB and at EOC)

1.2.3 Incidents with Loss of Main Steam

1.2.3.1 Inadvertent opening of one safety valve

(Code: NLOOP, several cases at ZL(no decay heat) and at FP (FP: 900 ppm, EOC: 0 ppm)), at p <60 bar safety valve isolation valve closes)

1.2.3.2 Break of one main steam line (MSL)

see 1.2.3.4

1.2.3.3 Break of one main feed water line (MFWL)

(no analysis, but argumentation: break upstream of one-way valve is covered by total failure of main feed water, break between one-way valve and steam generator is covered by leak in MSL)

1.2.3.4 Leaks in MSL or MFWL

(Code : NLOOP, calculations for FP, ZL(no decay heat), EOC without boron, analyses of following cases: DEGB of MSL outside containment, 0.1A leak in MSL inside containment, to increase the re-criticality at leak inside containment for ZL, the additional failure of safety injection into defect loop was assumed as a further case)

1.2.4 Incidents with Loss of Primary Coolant

1.2.4.1 Steam generator tube rupture

(Code: NLOOP; FP, 2A SGTR, EOC, consideration of all automatic actions derived from N16-signal, no operator actions before 30 min.; 2 cases: with and without loss of power at the turbine shutdown after scram due to PCS pressure)

1.2.4.2 Inadvertent opening of pressuriser valves

(Code: NLOOP, two cases at FP, BOL: without and with ignorance of 1st scram signal (p<132 bar), second signal: level in pressuriser > 9.5m)

1.2.5 Incidents with Loss of Main Steam and SGTR

1.2.5.1 Break of MSL and SGTR

(Code: NLOOP; FP, EOC without boron, SGTR assumed as an additional, independent, random failure, considered as SF, no repair case because of small probability event, no operator actions before 30 min.)

1.1.2 Stuck open of MS Safety Valve and SGTR

(Code: NLOOP; FP,SGTR assumed as SF, no operator actions before 30 min.)

1.3 Anticipated Transient without Scram (ATWS)

Analysis of 8 ATWS cases, i.e. failure of the scram system during operational transients, according to RSK guideline 20. For each case two different calculation were performed: failure of the signalisation for reactor scram, mechanical blockage of all control elements. Generally BOC (960 ppm), for sub-cooling transients also EOC assumptions.

All analyses were performed originally for KKP-2. (Differences small, comparative calculations were performed.)

1.3.1 Failure of Main Heat Sink

Possible reasons: loss of condenser vacuum, closing of MSIV

1.3.2 Failure of Main Heat Sink + LONOP (station service power supply)

1.3.3 Maximum Increase in Steam Extraction

Possible reasons: opening the turbine bypass station, opening of main steam safety valves. Calculations for BOC and EOC.

1.3.4 Maximum Reduction of Feedwater Supply

Possible reasons: Failure of one MFW pump (reserve pump not available), Failure of both MFW pumps (also reserve pump not available).

1.3.5 Maximum Reduction of Coolant Flow Rate

Failure of one MCP

1.3.6 Maximum Reactivity Gain

Possible reasons: Withdrawal of control assemblies or control assembly banks

1.3.7 Primary System Depressurisation

Possible reason: Unintentional opening of pressuriser safety valve

1.1.1 Maximum reduction of reactor inlet temperature

Possible reason: Malfunction of an active component of feedwater supply system, e.g. HP-preheater

2. Description of Computer Codes

NLOOP combines thermal hydraulic and I&C simulation. Fixed zones in PCS, energy and mass exchange with other components (pressuriser, SG, RPV-upper head,...), integrated momentum balance, 1 - 4 loops, one-dimensional in all components, point kinetics, homogeneous coolant, in general thermal equilibrium (exceptions are in pressuriser, SG,...), mixing in lower plenum, change of flow direction possible, average fuel rod per loop, radial heat conduction, decay heat (DIN), reactivity due to void, boron, spectral, Doppler, control rod, pressuriser: heater, spray, relief, relief tank, SG: level, average tube, RPV upper head steam bubble homogeneous, MS-system: quasi-stationary compressible model, SGTR simulation, MSL leak, all limitation systems, control systems, reactor protection system.

Verification by means KKP-2 commissioning test, e.g. shut-off of one MFW-pump and switch on of one AFW-pump.

NLOOP is an extended version of LOOP-7. Main difference: extended simulation capability of NLOOP (1---4 loops) compared with LOOP-7 (only one loop).

3. Reference Material and Data

Content of core design handbook:

0. Introduction

1. General fuel assembly and core data

Description of reactor core geometry, the reactor core loading map and the Fuel assembly design

2. Nuclear Design

Monitoring of power density and power density distribution.

Reactivity balance and reactivity coefficients, efficiency of shutdown systems. Calculation of burn up cycle, power density distribution, critical boron concentration.

Codes used: SAV79A standard analysis methodology including FASER for nuclear data generation, MEDIUM and PANBOX for static and transient core calculations.

3. Thermal-hydraulic Design

Collection of relevant design data.

4. Description of first Core Loading

5. Accident analysis

- Startup accident
  Maximum reactivity insertion by withdrawal of control rods
- Withdrawal of a single control rod or a group of control rods
  Core conditions: BOC, EOC, hot zero power, full power
- Rod ejection accident
  Core conditions: BOC, EOC, hot zero power, full power

Code for core transients: PANBOX, 3D coarse mesh solution of neutron diffusion equations by nodal expansion method.

ASSESSOR ACCIDENT ANALYSES IN SECOND PART LICENSE FOR KONVOI-PLANTS

The three TÜV (Technical Inspection Agencies) responsible for the three individual plants of type KONVOI: TÜV Bayern for ISAR-2, TÜV-Hannover for KKE, TÜV-Stuttgart for GKN-2 and GRS performed the safety assessment. The work was distributed as follows:

- TÜV-Bayern for disturbance and failure of secondary heat sink without loss of coolant (failure of main heat sink, erroneous operation of valves in MS and in FW system, failure of MFW supply), long term LONOP, performance of selected SBLOCA analyses,
- TÜV Hannover for disturbances due to failure of MCPs, short term LONOP, damages of SG tubes incl. SGTR, performance of selected LOCA analyses (blowdown phase of LBLOCA),
- TÜV-Stuttgart for breaks and leaks in MS and FW system with and without leaks in SG tubes,
- GRS for ATWS, sub-cooling transients due to disturbances on secondary side, initial and boundary conditions for transients with opening of pressuriser valves with and without stuck-open, most of the LOCA analyses

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Analyses Performed in the Framework of Assessing the vendors Calculations in the ECC Handbook (LOCA)

1. LBLOCA

1.1 Analysis of 2A DEGB in Cold Leg with DRUFAN/FLUT

(analysed for KONVOI by TÜV-Hannover and by GRS)

1.2 Analysis of 2A DEGB in Cold Leg with TRAC-PF1

(analysed for KBR by GRS)

1.1 Analysis of Hot Rod Analysis and Core Damage for 2A DEGB in Cold Leg

(analysed with TESPA by GRS, TESPA = probabilistic investigations considering distributions of decay heat, gap conductance, internal fuel rod pressure, criterion for fuel rod burst, thermal hydraulic boundary conditions)

2. SBLOCA AND MBLOCA

2.1 Analyses with DRUFAN

2.1.1 Analyses for KONVOI

2.1.1.1 800 cm^2 in cold leg

(Δ-Analysis Convoy/KKP-2)

2.1.1.2 160 cm^2 in cold leg

(largest cladding temperature in GRS analysis)

2.1.1.3 20 cm^2 at pressuriser

(entrainment in pressuriser)

2.1.2 Analyses for NPP KKP-2

2.1.2.1 380 cm^2 in cold leg

(rupture of largest connecting pipe at cold leg)

2.1.2.2 800 cm^2 in cold leg

(Δ-Analysis Convoy/KKP-2)

2.1.2.3 70 cm^2 in cold leg

(check of vendor's statement that no core uncovery takes place for leaks $<100cm^2$)

2.1.2.4 380 cm^2 in hot leg

(relevant case with regard to systems failure assumptions)

2.1.3 Analyses for NPP KWG

2.1.3.1 380 cm^2 in cold leg

(rupture of largest connecting pipe at cold leg, Δ-Analysis KONVOI/Grohnde)

2.1.3.2 70 cm^2 in cold leg

(check of vendor's statement that no core uncovery takes place for leaks <100cm^2, Δ-Analysis KONVOI/Grohnde)

2.3 Analyses with RELAP5/MOD1

2.2.1 Analyses for KONVOI

2.2.1.1 380 cm^2 in cold leg

(rupture of largest connecting pipe at cold leg)

2.2.2 Analyses for KKP-2

2.2.2.1 70 cm^2 in cold leg

Analyses Performed in the Framework of Assessing the Vendors Calculations in the Plant Dynamics Handbook

1. Abnormal Operation

2.3 Failure of Main Heat Sink

(Code: RELAP4/2STD)

2.3 Failure and Restart of a MCP

(Code: RELAP4/2STD)

2. Disturbed Operation and Incidents

2.3 Ejection of one Control Element

(Code: RELAP4/2STD)

2.3 LONOP

(Code: RELAP4/2STD)

2.3 Closure of all MSIVs with Add. Failure of Depressurisation in one Loop

(Code: RELAP4/2STD)

2.4 Failure of MFW Supply

(Code: RELAP4/2STD)

2.5 Non-Isolable 2A MFW Line Break

(Code: RELAP4/2STD)

2.1 SGTR and erroneous opening of Relief Valve in Defect Loop

(Code: RELAP4/2STD)

3. ATWS

(Code: ALMOD 3.5, Selection of 3 worst cases from vendors calculations with the assumption of mechanical blockage of control elements, BOC, each 2 calculations: with KONVOI-specific reactivity feedback and with KKP-2-specific reactivity feedback)

3.1 Emergency Power Case

3.2 Failure of Main Feedwater Supply

(parametric study on the influence of steam generator heat transfer)

3.3 Failure of Main Heat Sink

Analyses Performed in the Framework of Assessing the Vendors Calculations in the Core Design Handbook

1. Nuclear Design

Calculation of fuel assembly characteristics like local pin power distribution and reactivity dependence of fuel assembly during burn-up cycle. Determination of homogenised two group cross-sections. Code: CASMO

Calculation of power density distribution in the reactor core and shutdown reactivity. Reactivity efficiency of boron and moderator temperature reactivity coefficient for a low-leakage core loading in GKN-2. Code: QUABOX/CUBBOX-HYCA

2. Reactivity Initiated Transients

Check of initial conditions and assumptions on reactivity insertion in vendor calculations. Additionally, calculations by a model like ALMOD.

3. Steam-line break analysis

Calculation of reactivity balance and power density distribution during cooldown in shutdown condition including a stuck-rod by static calculations with QUABOX/CUBBOX-HYCA.

REFERENCE LIST OF DBA TO BE CONSIDERED IN THE SAFETY STATUS ANALYSIS OF A PSR

Level 3, accidents, events to be considered for transients

1. PWR-specific

– Primary-side, due to
  - withdrawal of most effective control element or most effective group during start-up,

– secondary-side, due to
  - loss of main heat sink caused by the failure to open of the main-steam turbine bypass system after turbine trip,
  - loss of main feedwater supply,
  - loss of off-site power supply,
  - leaks of main-steam lines up to $0.1A_D$ designed in a break preclusion quality, otherwise $2A_D$ (A_D: open cross-sectional area of the tube).

2. BWR-specific

– Reactivity accidents
  - limited failure of most effective control rod,
  - uncontrolled withdrawal of control rods during start-up,

– loss of the main heat sink caused by spurious closure of the main-steam penetration valves,
  - loss of main heat sink caused by the failure to open of the main-steam turbine bypass system after turbine trip,
  - loss of main feedwater supply,
  - loss of off-site power supply.

Level 3, accidents, events to be considered for losses of coolant (LOCA)

1. PWR-specific

Leak cross-sections to be assumed for the reactor coolant boundary for plant-specific positions:

– leak cross-section < 120 cm²
  - overpressure protection systems stuck-open
  - break of connecting lines,
  - leaks at pipe branches, penetrations and seals,
  - leaks through cracks,
  - double-ended break of a steam generator tube in a steam generator,

– leak cross-section

- about 0.1A (A: open cross-sectional area of a main coolant line designed in a break preclusion quality,
- otherwise up to 2A.

2. BWR-specific

Leak cross-sections to be assumed for the coolant boundary for plant-specific positions:

– leak cross-section
  - about $0.1A_R$ (A_R : open cross-sectional area of tubes) when designed in a break preclusion quality, otherwise up to $2A_R$,

– leak cross-section smaller than 80 cm²
  - leaks through cracks in the area between the control rod drives of the reactor pressure vessel (RPV) bottom.

<u>Level 3, accidents, radiologically representative events</u>

1. PWR-specific

– Losses of coolant with
  - leak cross-section A_{RL} (A_{RL}: open cross-sectional area of a measuring line in the annulus) caused by break of a measuring line in the annulus not isolated for 30 minutes,
  - leak cross-section $2A_{DE}$(A_{DE}: open cross-sectional area of a steam generator tube) and leak in the main-steam line behind the isolating valve considering the closing times of the isolating valve,
  - leak cross-section 0.1A designed in a break preclusion quality, otherwise up to 2A.

– fuel element handling accidents
  - damage of all perimeter fuel rods of a fuel element,

– failure of auxiliary systems
  - break of a line in the off-gas purification system,
  - failure of the waste water evaporator unit in the coolant treatment.

2. BWR-specific

– Losses of coolant with
  - leak cross-section $2A_{AL}$ (A_{AL}: open cross-sectional area of a measuring line in the reactor building) caused by break of a measuring lance in the reactor building carrying reactor water not isolated for a period of 30 minutes,
  - leak cross-section $0.1A_{NL}$ (A_{NL}: open cross-sectional area of a residual-heat removal line in the reactor building) designed in a break preclusion quality,
  - otherwise up to $1A_{NL}$ by break of a residual-heat removal line in the reactor building during the shutdown phase taking into account closure times of the isolating valve,
  - leak cross-section $0.1A_R$ designed in a break preclusion quality, otherwise up to $2A_R$,

– leak cross-section smaller than 80 cm²
  - leaks through cracks in the area between the control rod drives of the reactor pressure vessel bottom,

– fuel element handling accidents

    - like PWR,
- failure of auxiliary systems
    - like PWR.

<u>Level 3, accidents, PWR and BWR-specific spreading impacts to be considered as possible initiating events for transients and SBLOCA</u>

Plant-internal impacts

- Flooding through leaks in pipes outside the coolant circuit of up to $0.1A_R$ upon designed in a break preclusion quality, otherwise up to $2A_R$,
- other plant-internal flooding (e.g. through leaks in service water lines),
- plant-internal fires,
- fragments with high kinetic energy as a consequence of component failure (e.g. turbine missiles due to turbine blade failure).

Plant-external impacts

- Site-specific, external impacts caused by nature (earthquakes, lightning, flooding and weather conditions like wind, ice, snow).

REFERENCE LIST OF SPECIAL VERY RARE EVENTS AND BDB PLANT CONDITIONS TO BE CONSIDERED IN THE SAFETY STATUS ANALYSIS OF A PSE

Level 4, PWR- and BWR-specific special, very rare events

- ATWS,
- site-specific external civil impacts (certain emergencies).

Level 4, beyond-design-basis plant conditions

1. PWR-specific

- Plant conditions owing to non-availability of engineered safety features challenged
  - Total loss of feedwater with the tendency towards a complete evaporation of the secondary sides,
  - loss of coolant with a small leak cross-section with the tendency to an increase of coolant pressure beyond the delivery pressure of the high-pressure (HP) injection pumps,
  - double-ended break of a SG-tube in a steam generator and increase of main-steam pressure with the tendency towards actuating the main-steam safety valve,
  - loss of the entire three-phase current supply, if not battery-supplied for a period of up to 2 hours (station black-out),
  - global, long-term increase of pressure inside the containment vessel with the tendency towards an increase beyond design pressure.

2. BWR-specific

- Plant conditions owing to non-availability of engineered safety features challenged
  - loss of coolant with subsequent overfeeding of one main-steam line and the possibility of water hammer outside the penetration isolation valve,
  - transients with the tendency of a decreasing of the RPV level to the lower edge of the core,
  - failure of the entire three-phase current supply, if not battery-supplied for a period of up to 2 hours (station black-out),
  - global, long-term increase of pressure in the containment vessel with the tendency towards an increase beyond design pressure.