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REVIEW OF ACCIDENT ANALYSES PERFORMED AT MOCHOVCE NPP

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Review of Accident analysis performed at Mochovce NPP.

1. Objective and scope of the safety measure

The accident analysis reports that were originally provided by the plant designer available at the plants did not correspond to the international practices from the point of view scope of analysed failures and used methodology. A systematic review of the safety analysis methods is needed which must be based on the common modern approach, namely regarding the acceptance criteria, the conservative assumption, initial and boundary conditions, the application of single failure criteria, the quality and completeness of the performed analysis and the used computer codes etc..

2. Safety measure scope and related improvements

Development of exhausting accident analysis of the Design Basis and Beyond Design Accidents is goal of safety improvements AA01 till AA12.

The scope of these safety measures was defined and development in the "TSSM for NPP Mochovce Nuclear Safety Improvements Report" issued in July 1995. The main objectives of these safety measures were the followings:

a) to establish the criteria for selection and classification of accidental events, as well as defining the list of initiating events to be analysed. Accident classification to the individual groups must be performed in accordance with RG 1.70 and IAEA recommendations "Guidelines for Accidental Analysis of WWER NPP (IAEA-EBR-WWER-01to select boundary cases to be calculated from the scope of initiating events.

b) to elaborate the accident analysis methodology that also includes acceptance criteria for their result evaluation, initial and boundary conditions, assumption related with the application of the single failure criteria, requirements on the analysis quality, used computer codes, as well as NPP models and input data for the accident analysis.

c) to perform the accident analysis for the Pre-operational Safety Report (POSAR)

d) to provide a synthetic report addressing the validity range of codes models and correlations, the assessment against relevant tests results, the evidence of the user qualification, the modelisation and nodding scheme for the plant and the justification of used computer codes.

3. Accident analyses

It has been decided to apply western practices, e.g. that following US practice as defined in the US NRC RG1.70, and to have the main part of these accident analyses performed by EUCOM and VÚJE companies using modern computer codes.

In 1995 IAEA has edited a guideline for the performance of accidental analysis of WWER plants "Guidelines for Accidental Analysis of WWER Nuclear Power Plants" (IAAEA-EBP-WWER-01). This guideline was the basis for all tasks resulting in the choice of initiating events, the settlement of acceptance criteria and the creation of the methodology for EMO unit 1 safety analysis.

3.1 Classification of events

The list of postulated initiating events of the Design Basis Accidents (DBA) in accordance with the IAEA recommendations according to their anticipated probability of occurrence, the listed events are assigned to three categories:

- Anticipated Transient (probability of occurrence > 10^{-2}) T
- Postulated accident (probability of occurrence < 10^{-2}) A
- Beyond Design Basis Accidents (probability of occurrence < 10^{-4}) BDBA

3.2 Methodology

The methodology reports for POSAR LOCA and Non-LOCA analyses is in accordance with IAEA Guidelines for Accident analysis for WWER NPP. These methodologies cover the spectrum of events to be analysed, their initial and boundary conditions and the acceptance criteria to be verified. Accordance to these methodologies , the following aspects are applied to define assumption for accident analysis:

- application of single failure criteria,
- account for control system if aggravating,
- account for failures occurring as a consequence of the event itself,
- analysis with and without offsite power,
- account for the first signal HO-1 for reactor trip with the assumption of stuck control rod'
- conservative values for initial and boundary conditions, setpoints, delay, etc.,
- appropriate time for operator intervention and assumption that the operator takes the correct action

Methodology reports concerning the radiological consequences. These documents identify dose acceptance criteria for the design of the plant regarding the radiological consequences of radioactive releases in normal operation, and propose dose target values for the safety analysis of postulated accidents. They also present the methodology to calculate the radiological consequences of radioactive releases after postulated accidents.

3.3 Initiation conditions

Procedures for the data base management and the collection of missing data, the collection of these missing information, the definition of this data base structure according to the requests from the users.

Quality Assurance rules to be applied, the elaboration and the validation of this data base. These data were used for the accident analyses in the frame of safety measures and POSAR.

Three versions of the data base were issued in February, April and October 1997 in order to comply with the degree of implementation of the modifications.

3.4 Computer codes

- Lot of documents cover the codes assessment, the codes qualification, the users qualification and the plant model validation. The considered codes are the following:
  - RELAP 5/MOD2 used for system transient analyses,

- HEXTIME which is a three dimensional coupled neutronics/ thermal hydraulics analysis tool, used for safety analysis and all kind of transients in which the power distribution is significantly affected,
- CATHARE 2 which is a two fluid - thermal hydraulic code used in LOCA analyses,
- WAVCO codes which is used to analysis to analyse thermohydraulic loads within the compartments of the plant
- TYAGA which is used to analyse thermohydraulic forces on high energy lines following a postulated pipe break,
- TUS which is used to obtain the hydraulic forces on steam generator heat exchanging tubes in case of a feedwater line break,
- VKUS which is used to compute hydraulic forces on the steam generator internals in case of a steam line break,
- DPIPE which is devoted to dynamic and static analysis of pipeline systems,
- FEP that is devoted to the dynamic and static analysis of bidimentional plane or axisymetrical structures with complex configuration.
- ANSYS static and dynamic strength analysis of structures - linear and non-linear

3.5 Acceptance criteria

The acceptance criteria were used in full scope from IAEA methodology "Guidelines for Accidental Analysis of WWER Nuclear Power Plants" (IAAEA-EBP-WWER-01)

Acceptance Criteria for Accident Analyses (incorporating margins to design limits)

Non-LOCA

Safety Parameters	Transients	Accidents	
	'T'	'A'	'A' (ATWS)
Clad-to-Coolant Heat Transfer	No DNB	DNB admissible	DNB admissible
Fuel Cladding Temperature	Covered by ' No DNB '	$T_{clad} < 1480°C$	$T_{clad} < 1200°C$
Fuel Temperature/Enthalpy	$T_{max} < 2570°C$	E < 840 J/g, Fuel Melting < 10%	E < 840 J/g, Fuel Melting < 10%
RCS Pressure	p < 1.1 times Design Pressure Design pressure 13.73 MPa	p < 1.1 times Design Pressure	p < 1.35 times Design Pressure
Secondary-Side Pressure	p < 1.1 times Design Pressure Design pressure 5.59 MPa	p < 1.1 times Design Pressure	p < 1.35 times Design Pressure
Target Doses at Exclusion Area boundary for the Public	< 0.25 mSv	Effective dose ≤ 50 mSv Thyroid dose ≤ 500 mSv	

LOCA

Safety Parameters	Accidents 'A'
Fuel Cladding Temperature	< 1200°C
Fuel Cladding Local Oxidation	< 17 %
Fuel Cladding Core-wide Oxidation	1 %
Core Geometry	must remain amenable to cooling
Control Rods	must remain movable and no melting
Calculated Dosis	< limit for A

3.6 Accident analyses

The accidents analyses were grouped in the following classes:

1. - Reactivity and power distribution anomalies,
2. - Decrease in reactor coolant system flow rate,
3. - Loss of reactor coolant inventory (Leaks from primary to secondary side have been performed in the SM:AA07)
4. - Increase of reactor coolant inventory,
5. - Increase of heat removal through the secondary circuit (Spectrum of steam system break inside and outside the containment have been performed in the SM:AA05),
6. - Decrease of heat removal in the secondary circuit,
7. - Radioactive release from a component or a system (Drop of container with fresh and spent fuel has been performed in the SM:AA12),
8. - Fuel handling events,
9. - Anticipated Transients Without Scram (ATWS) (performed in the SM: AA13),
10. - Beyond Design Basis Accidents (performed in the SM: AA09),
11. - Containment (performed in the SM: CONT05)
12. - Loads on primary and secondary system components and internals (Loads on PRZ relief, safety valves and PRZ discharging pipes have been performed in the SM: S04),
13. - Pressurised thermal shocks (performed in the SM: AA06),
14. - RCP integrity at high speed.

For each analysis have been defined:

⇒ the identification of the event causes and acceptance criteria
⇒ the initial and the boundary conditions
⇒ the used codes and the modelling assumptions

They also contain the events description with table of chronological events, figures of the most important physical parameters to demonstrate that the acceptance criteria are met, operator actions (if needed) and summarising table of obtained results which are compared with the acceptance criteria and as well as the conclusions.

4. Time schedule

Elaboration of TSSM	Jul 1995
Ratify of contract between SE a.s. and EUCOM, VÚJE for calculation accident analysis	March 1996
Delivery of accident analysis	Jul 1999

5. List of accidental analysis

AA 01 SCOPE AND METHODS OF ACCIDENTAL ANALYSES

BO	Original	Preklad
AA 01	Elements of validation of the Dose Calculation Models or Radiological Consequences of Postulated Accidents + Príloha 1 (obrázky)	
AA 01	Design fuel handling accident with fresh and spent fuel	Projekčný základ pre haváriu pri manipulácii s čerstvým a vyhoreným palivom
AA 01	Assessment of core boron accumulation and dilution risks in the long term phase of loss of coolant accidents	Zhodnotenie rizika hromadenia a riadenia koncentrácie kyseliny boritej v aktívnej zóne reaktora počas dlhodobej etapy havárií so stratou chladiva
AA 01	Completion and safety improvement - Accident analysis for design check - Primary overpressure in cold state	Doplnenie a bezpečnostné zlepšenie - bezpečnostné analýzy projektu kontrola pretlakovania PO v studenom stave
AA 01	Case 6.1. Malfunction of sec. circuit pressure control decreasing steam flow	Prípad č. 6.1. Porucha regulácie tlaku sekundárneho okruhu, ktorá spôsobuje pokles prietoku pary
AA 01	Case 3.2. Rupture of the line connecting pressurizer and a pressurizer safety valve	Roztrhnutie potrubnej línie medzi kompenzátorom objemu a poistným ventilom kompenzátora objemu
AA 01	Inadvertent opening of one check or isolation valve separating the reactor coolant boundary and the low pressure part of the system	Poruchové otvorenie jednej spätnej klapky alebo uzatváracej armatúry, ktoré odeľujú PO a nízkotlakú časť systému.
AA 01	Spectrum of postulated piping break within the reactor coolant boundary	Spektrum postulovaných zlomov potrubí v rámci tlakovej hranice chladiva reaktora
AA 01	Inadvertent opening of one pressurizer safety valve	Poruchové otvorenie PV KO
AA 01	Main coolant pump flywheel integrity during LOCA. Mechanical calculations.	Integrita zotrvačníka HCČ počas LOCA havárie
AA 01	Case 5.3. Malfunction of secondary circuit pressure control increasing steam flow.	Chybná regulácia tlaku v sekundárnom okruhu zvyšujúca rýchlosť prúdenia pary
AA 01	Case 6.5. Loss of condenser vacuum	Prípad 6.5. Strata vákua v kondenzátore
AA 01	Case 6.2. Loss of electrical load.	Prípad 6.2. Strata elektrického zaťaženia
AA 01	Turbine(s) stop valve closure. Case 6.3	Zavretie uzatváracej armatúry turbín(y). Prípad č. 6.3

AA 01	Large breaks LOCA accidents	Havárie typu "veľký únik"
AA 01	LOCA - Radiological consequences	LOCA - radiačné následky
AA 01	Rupture of I&C line or other lines from reactor coolant pressure boundary that penetrate containment (radiological consequences)	Roztrhnutie impulznej línie alebo iných potrubných línií od tlakovej hranice chladiva reaktora, ktorá prechádza cez hermetickú zónu (rádiologické následky)
AA 01	Radioactive release from subsystem or component	Rádioaktívne výpuste z podsystému alebo komponentu
AA 01	Case 6.4. Main Steam Isolation Valves Closure.	Prípad č. 6.4. Zatvorenie hlavného parného oddeľovacieho ventilu (ventilov)
AA 01	Case 1.2. Uncontrolled withdrawal of a control rod group during power operation	Prípad č. 1.2. Nekontrolované vytiahnutie skupiny ARK počas prevádzky na výkone
AA 01	Case 5.1. Feedwater system malfunction. Decrease of feedwater temperature.	Prípad č. 5.1. Porucha na systéme napájacej vody. Pokles teploty napájacej vody
AA 01	Case 5.2. Feedwater system malfunction. Increase of feedwater flow.	Prípad č. 5.2. Porucha na systéme napájacej vody. Zvýšenie prietoku napájacej vody
AA 01	Case 6.8. Feedwater piping break	Prípad č. 6.8. Pretrhnutie potrubia napájacej vody
AA 01	Case 4.1. Inadvertent actuation of ECCS during power operation	Prípad č. 4.1. Neúmyselné spustenie ECCS počas výkonovej prevádzky
AA 01	Case n° 1.3/b. Control rod maloperation: Withdrawal of One Control Rod	Prípad č. 1.3/b. Prevádzková porucha regulačnej tyče: Vytiahnutie jednej regulačnej tyče
AA 01	Case 1.5. Control rod ejection	Prípad č. 1.5. Vystrelenie havarijnej regulačnej kazety
AA 01	Case 1.7. Inadvertent loading and operation of Fuel assembly in an improper Position	Neúmyselné zaťaženie a prevádzka palivovej kazety v nesprávnej polohe
AA 01	Case 1.3/c. Control rod maloperation. Statical misalignment of one CR	Prípad č. 1.3/c. Poruchová prevádzka regulačných tyčí. Statické presadenie jednej regulačnej tyče.
AA 01	Case 1.4. Incorrect connection of an inactive loop	Prípad č. 1.4. Nesprávne pripojenie odstavenej slučky
AA 01	Case 1.6. Decrease of boron concentration in the RCS	Prípad č. 1.6. Pokles koncentrácie bóru v chladiacom systéme reaktora
AA 01	Case 1.3/a. Control rod maloperation: Drop of one control rod.	Prípad č. 1.3/a. Poruchová prevádzka regulačnej tyče: Pád jednej regulačnej tyče
AA 01	Fluid dynamic Loads on the Internals of the RPV after 2A-Break in the hot and cold leg of the Main Coolant Line	Hydrodynamické záťaže na vnútorné časti tlakovej nádoby reaktora po prasknutí 2A v horúcej a studenej vetve hlavného cirkulačného potrubia chladiacej vody
AA 01	Stresses on the RPV - internals due to a 2A break of the main coolant loop	Napätia na vnútorných komponentoch TNR vyvolané pretrhnutím 2A hlavného cirkulačného potrubia
AA 01	Rupture of surge line at connection of pressurizer - Mass and Energy Releases calculations	Roztrhnutie potrubia KO v mieste pripojenia ku KO. Výpočet uvoľnenej hmotnosti a energie

AA 01	Methodology and acceptance criteria (non LOCA)	Metodika a kritériá prijateľnosti -
AA 01	Case 4.2. CVCS malfunction that increases reactor coolant inventory	Prípad č. 4.2. Zlá funkcia systému CVCS, ktorá spôsobuje nárast inventáru chladiva reaktora
AA 01	Case 6.7. Loss of on-site and off-site power to the station	Úplná strata vnútorného a vonkajšieho elektrického napájanie elektrárne
AA 01	Case 5.5. Main steam piping break	Prípad č. 5.5. Roztrhnutie potrubia ostrej pary
AA 01	Case 5.4. Inadvertent opening of one steam valve	Prípad č. 5.4. Náhodné otvorenie jednej parnej armatúry
AA 01	Case 2.1. Inadvertent closure of one MGV in an RCS loop	Prípad č. 2.1. Náhodné zatvorenie hlavnej uzatváracej armatúry
AA 01	Case 6.6. Main Feedwater pump trip	
AA 01	Case 2.4. Single and multiple MCP trips	Prípad č. 2.4. Odstavenie jedného alebo viacerých HCČ
AA 01	Case 2.2. Seizure of one MCP	Prípad č. 2.2. Zadretie jedného HCČ
AA 01	Case 2.3. Break of the shaft of one MCP	Prípad č. 2.3. Zlomenie hriadeľa jedného hlavného cirkulačného čerpadla
AA 01	Case 1.1. Uncontrolled withdrawal of a control rod group at zero power	Prípad č. 1.1. Neregulované vytiahnutie regulačných tyčí pri nulovom výkone
AA 01	Initiating Events. Representative list	Celkový zoznam iniciačných udalostí
AA 01	Large Break LOCA accident. Reactor coolant pump overspeed	Havárie veľkého roztrhnutia potrubia s únikom chladiva. Prekročenie dovolených otáčok hlavného cirkulačného čerpadla
AA 01	Rupture of I&C line or other lines from reactor coolant pressure boundary that penetrate containment	Prasknutie meracej trasy alebo iného potrubia na tlakovej hranici chladiva reaktora, prechádzajúceho stenou hermetickej zóny
AA 01	Large Break LOCA accidents. Mass and Energy Releases Calculations	Havárie typu veľký únik LOCA. Výpočty uvoľnenej hmoty a energie
AA 01	Analysis of steam generator in accident conditions. Thermohydraulic loads	Analýza parogenerátora počas havarijných podmienok. Termohydraulické zaťaženie
AA 01	Methodology and criteria (Radiological consequences)	Metodológia a kritériá (Rádiologické následky)
AA 01	Acceptance criteria and methodology LOCA	Kritériá prijateľnosti a metodiky LOCA
AA 01	Analysis of Steam generator accident conditions - Mechanical consequences (LOCA, FWLB, SLB)	Analýza havarijných podmienok parogenerátora - Mechanické následky (LOCA havária, roztrhnutie potrubnej línie napájacej vody, roztrhnutie potrubnej línie ostrej pary)

AA 04 ACCESSIBILITY OF THE ACCIDENT ANALYSES FOR SUPPORT OF THE NPP OPERATION

BO	Original	Preklad
AA 04	Accident analyses for supporting EOPS - Scope of analyses	Havarijná analýza pre podporné havarijné predpisy. Rozsah analýz.
AA 04	Analysis of Inadvertent Actuation of High Pressure Safety Injection	Analýza neúmyselnej aktivácie vysokotlakového bezpečnostného vstrekov ania
AA 04	Use of the RCS Emergency Venting System During Emergency Operating Procedures	Použitie havarijného odplyňovacieho systému IO pri havarijných prevádzkových postupoch
AA 04	Possibility for Response of the Neutron Flux Signals to a Level Decrease in the RPV	Možnosť odozvy signálov neutrónového toku na pokles hladiny v TNR
AA 04	Small break LOCA without high pressure safety injection	Malý únik LOCA bez činnosti vysokotlakého havarijného doplňovania PO
AA 04	Assessment of the probability of inadvertent Depressurization of all Steam Generators	Ohodnotenie pravdepodobnosti náhodného odtlakovania všetkých parogenerátorov v NPP Mochovce
AA 04	Recording of results (transient analysis) (with 2 Compact Disc)	Zaznamenávanie výsledkov
AA 04	Availability of accident analysis results for supporting plant operation. Main loop isolation valves closure - LOCAs	Použiteľnosť výsledkov havarijnej analýzy pre podporu prevádzky elektrárne. Uzavretie hlavných uzatváracích armatúr - LOCA
AA 04	Cooldown under natural circulation EOPs ES-0.2 and ES-0.4	Vychladenie v podmienkach prirodzenej cirkulácie HP ES-0.2 a ES-0.4
AA 04	Loss of coolant accident - Reactor coolant pump trip criteria	Havária s únikom chladiva primárneho okruhu (LOCA). Kritériá pre odstavenie hlavných cirkulačných čerpadiel
AA 04	Primary to secondary leaks. Analysis to support EOPS	Analýza únikov z primárneho okruhu do sekundárneho okruhu pre podporu havarijných predpisov
AA 04	SM AA 04 - Group II. Steam generator tube rupture with best estimate assumptions	Roztrhnutie trubky parogenerátora s predpokladmi najlepšieho odhadu
AA 04	Availability of accident analysis results for supporting plant operation - Small break LOCA - best estimate analysis	
AA 04	Recording transient results (2 CD)	
AA 04	Pressure - temperature curves for faulted conditions	Tlakovo-teplotné krivky pre abnormálne stavy
AA 04	Group II - MLIV closure Collector head lift up event	Skupiny II - Uzavretie HUA. Udalosť nadvihnutia veka kolektora PG
AA 04	ATWS	ATWS
AA 04	Total loss of power supply (Station Blackout)	Úplná strata napájania

AA 04	Availability of accident analysis results for supporting plant operation - LOCA isolation	Použiteľnosť výsledkov havarijnej analýzy pre podporu prevádzky elektrárne - izolácia LOCA
AA 04	Natural Circulation Cool-Down Without Letdown EOP	Prirodzená cirkulácia chladiacej vody bez klesania, NOP
AA 04	Required Boron Concentration During Natural Circulation Cool-Down	Požadovaná koncentrácia bóru počas chladenia prirodzenou cirkuláciou, NOP
AA 04	Loss of secondary heat sink (feed and bleed)	Strata sekundárneho absorbéra tepla (Doplňovanie a odpúšťanie)
AA 04	Plant response to emergency feedwater operation	Odozva elektrárne na spustenie super-havarijného napájania

AA 05 ANALYSE OF ACCIDENT: MAIN STEAM LINE RUPTURE

BO	Original	Preklad
AA 05	Main steam line break. Accident analysis	Havarijná analýza pretrhnutia hlavného parného potrubia
AA 05	Main steam line break. Sensitivity Study	Roztrhnutie parovodu. Citlivostná štúdia
AA 05	Main steam line break. Analysis of available experience.	
AA 05	Main steam line break. Analysis Methodology	Metodika analýzy roztrhnutia parovodu
AA 05	MSLB upstream check valve with secondary failures in neighbouring lines (MER)	Roztrhnutie potrubia ostrej pary pred spätnou klapkou so sekundárnym zlyhaním v susedných parovodoch (MER)

AA 06 TRANSIENTS OF THE PRIMARY CIRCUIT UNDERCOOLING WITH THE PRESSURE THERMAL SHOCK

BO	Original	Preklad
AA 06	Pressurized thermal shocks. Thermal-hydraulic conditions for the RPV wall	Tlakový teplotný šok: Teplotno-hydraulické podmienky pre stenu tlakovej nádoby reaktora
AA 06	Pressurized thermal shocks. Mechanical analysis	Tepelný šok v natlakovanom stave
AA 06	Pressurized Thermal Shock. List of initiating events	Tepelný šok pod tlakom. Zoznam iniciačných udalostí
AA 06	Pressurized Thermal shock. Methodology and acceptance Criteria Report	Tepelný šok pod tlakom: Správa o metodike a kritériách prijateľnosti
AA 06	Pressurized thermal shock LOCA analyses using RELAP 5	Tlakový tepelný šok. Analýzy LOCA s použitím RELAP 5
AA 06	Pressurized thermal shock: Non-LOCA analyses using RELAP 5	Tlakový tepelný šok (PTS): Analyzovanie udalostí Non-LoCA použitím kódu RELAP 5

AA 07 RUPTURE OF THE STEAMGENERATOR PRIMARY HEADER

BO	Original	Preklad
AA 07	SG header lift - up - Acceptance criteria	Utrhnutie veka kolektora PG - Kritériá úspešnosti
AA 07	Primary to secondary leak analysis. Impact on hardware.	Dopady analýzy únikov z primárneho do sekundárneho okruhu na technické zariadenie
AA 07	Primary to secondary leak for EOPs	Únik z primáru do sekundáru - pre havarijné predpisy
AA 07	Primary to secondary leak. Identification of affected SG	Únik z primáru do sokundáru. Identifikácia postihnutého parogenerátora
AA 07	Primary to secondary leaks. Equipment behaviour	Únik z primáru do sekundáru. Správanie sa zariadení
AA 07	Water reserve for long term needs	Zásoba vody pre dlhodobú potrebu
AA 07	Primary to secondary leak analysis. Transient analysis	Analýza úniku z primárneho do sekundárneho okruhu. Analýza prechodového procesu
AA 07	Primary to secondary leak - Strategy for operator actions	Únik z primárneho do sekundárneho okruhu. Stratégia činnosti operátorov
AA 07	One steam generator tube rupture (radiological consequences)	Roztrhnutie jednej trubky PG (rádiologické dôsledky)
AA 07	Primary collector leaks up to cover lift-up (radiological consequences)	Únik z primárneho kolektora PG po nadvihnutie veka kolektora PG (radiačné následky)

AA 09 SEVERE ACCIDENTS

BO	Original	Preklad
AA 09	Beyond Design Basis Accidents - List of initiating events	Nadprojektové havarijné udalosti, zoznam prvopríčin
AA 09	Beyond Design Basis Accidents Methodology and criteria (LOCA)	Metodika a kritériá pre nadporjektové havárie
AA 09	Beyond design Basis Accident - Small break LOCA without high pressure safety injection	Havária malý únik z LOCA bez vysokotlakého havarijného systému chladenia AZ
AA 09	Beyond design basis. Accidents, methodology and acceptance criteria (non LOCA)	Nadprojektové havárie. Metodika a preberacie kritériá
AA 09	Beyond design Basis Accident - Interfacing system LOCA	Systém rozhrania pre LOCA haváriu
AA 09	Steam line break concomitant with a steam generator tube rupture	Roztrhnutie parného potrubia sprevádzané roztrhnutím rúry na parogenerátore

AA 09	Total Loss of Electrical Power Supply	Úplná strata elektrického napájania elektrárne
AA 09	Total Loss of Feedwater system	Úplný výpadok systému napájacej vody

AA 11 ACCIDENTS WITH THE PRIMARY CIRCUIT COOLANT DILUTION

BO	Original	Preklad
AA 11	Boron dilution in the pressure vessel after pump start in the boron - free loop. 3D CFD calculation with PHOENICS	Zníženie koncenrácie H3BO3 v tlakovej nádobe reaktora po nábehu HCČ v slučke s čistým kondenzátorom. Trojrozmerné CFD výpočty pomocou kódu PHOENICS
AA 11	Methodology for Boron dilution analysis	Metodika pre analýzu riedenia bóru
AA 11	Plant and Core Response on Boron Dilution at Start-up of First Loop	Reakcia elektrárne a aktívnej zóny reaktora na riedenie kyseliny boritej pri spustení prvej slučky

AA 12 SPENT FUEL CONTAINER DROP

BO	Original	Preklad
AA 12	Spent fuel container drop	Pád kontajnera na vyhorené palivo
AA 12	Spent fuel container drop, POSAR EMO, chapt. 15.8.2	Pád kontajnera na vyhorené palivo. PpBS AEMO, kap. 15.8.2

AA 13 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

BO	Original	Preklad
AA 13	Methodology for ATWS analysis	Metodika analýzy ATWS
AA 13	Case 9.3. Loss of on-site and off-site power to the station without SCRAM (ATWS)	Prípad č. 9.3. Strata vnútorných a vonkajších zdrojov napájania elektrárne bez havarijného odstavenia reaktora (ATWS)
AA 13	Case 9.1. Uncontrolled withdrawal of a control rod group (ATWS)	Prípad č. 9.1. Nekontrolované vysúvanie skupiny havarijných a regulačných kaziet (ATWS)
AA 13	Case 9.4. Loss of condenser vacuum ATWS	Prípad č. 9.4. Strata vákua v kondenzátore (ATWS)
AA 13	Case 9.2. Loss of main feedwater flow (ATWS)	Prípad č. 9.2. Strata prietoku hlavnej napájacej vody
AA 13	Case 9.8. Inadvertent opening of one steam valve without scram (ATWS)	Neúmyselné otvorenie jedného parného ventilu bez havarijného odstavenia reaktora (ATWS)
AA 13	Case 9.7. Main steam isolation valve closure ATWS	Prípad č. 9.7. Zatvorenie hlavného parného oddeľovacieho ventilu (ATWS)

AA 13	Case 9.6. Loss of electrical load without scram (ATWS)	Strata elektrickej záťaže bez havarijného odstavenia reaktora (ATWS)
AA 13	Case 9.5. Trip of both turbines without scram (ATWS)	Výpadok oboch turbín bez havarijného odstavenia reaktora (ATWS)

6. Results

Summary of accident analyses by pressure in primary circuit (Accidents + Transients)

Acceptance criteria for the pressure in the reactor coolant and main steam systems :

AT3. The pressure in the reactor coolant and main steam systems shall be maintained below 110% of the design values.

This acceptance criterion (15.10 MPa) was met, the most adverse is Case 4.2 "CVCS Malfunction that Increases Reactor Inventory", maximal pressure is 14,59 MPa.

Summary of accident analyses by the fuel temperature (Transients)

Acceptance criteria for melting of fuel pellets :

AT2. There shall be no melting of the fuel pellets, not even locally (melting point 2840°C for fresh fuel, 2570°C for burned fuel).

This acceptance criterion (2570 °C) was met, the most adverse is Case 1.3.1.1 "Control rod maloperation: Drop of one control rod, Case 1.3.1 " maximal temperature is 1985 °C.

Summary of accident analyses by the fuel rod cladding temperature (Accidents)

Acceptance criteria for melting of fuel pellets :

PA2. a, The fuel rod cladding temperature does not exceed 1200°C.

This acceptance criterion (1200°C) was met, the most adverse is Case 3.1.1 "Large LOCA, Case 3.1.1 " maximal temperature is 1022 °C.

Analyses results shoved that all acceptance criteria were met with satisfactory margin and design of the NPP Mochovce is accurate.