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NUCLEAR ENERGY AGENCY
COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

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VALIDATION MATRIX FOR THE ASSESSMENT OF THERMAL-HYDRAULIC CODES FOR VVER LOCA AND TRANSIENTS

A Report by the OECD Support Group on the VVER Thermal-Hydraulic Code Validation Matrix

English - Or. English

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The mission of the NEA is:

- to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes, as well as
- to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

Specific areas of competence of the NEA include safety and regulation of nuclear activities, radioactive waste management, radiological protection, nuclear science, economic and technical analyses of the nuclear fuel cycle, nuclear law and liability, and public information. The NEA Data Bank provides nuclear data and computer program services for participating countries.

In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS

The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of scientists and engineers. It was set up in 1973 to develop and co-ordinate the activities of the Nuclear Energy Agency concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations. The Committee's purpose is to foster international co-operation in nuclear safety amongst the OECD Member countries.

CSNI constitutes a forum for the exchange of technical information and for collaboration between organisations which can contribute, from their respective backgrounds in research, development, engineering or regulation, to these activities and to the definition of its programme of work. It also reviews the state of knowledge on selected topics of nuclear safety technology and safety assessment, including operating experience. It initiates and conducts programmes identified by these reviews and assessments in order to overcome discrepancies, develop improvements and reach international consensus in different projects and International Standard Problems, and assists in the feedback of the results to participating organisations. Full use is also made of traditional methods of co-operation, such as information exchanges, establishment of working groups and organisation of conferences and specialist meeting.

The greater part of CSNI's current programme of work is concerned with safety technology of water reactors. The principal areas covered are operating experience and the human factor, reactor coolant system behaviour, various aspects of reactor component integrity, the phenomenology of radioactive releases in reactor accidents and their confinement, containment performance, risk assessment and severe accidents. The Committee also studies the safety of the fuel cycle, conducts periodic surveys of reactor safety research programmes and operates an international mechanism for exchanging reports on nuclear power plant incidents.

In implementing its programme, CSNI establishes co-operative mechanisms with NEA's Committee on Nuclear Regulatory Activities (CNRA), responsible for the activities of the Agency concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with NEA's Committee on Radiation Protection and Public Health and NEA's Radioactive Waste Management Committee on matters of common interest.

VALIDATION MATRIX FOR THE ASSESSMENT OF THERMAL-HYDRAULIC CODES FOR VVER LOCA AND TRANSIENTS

A Report by the OECD Support Group
on the
VVER Thermal-Hydraulic Code Validation Matrix

July 2000

ABSTRACT

This report deals with an internationally agreed experimental test facility matrix for the validation of best estimate thermal-hydraulic computer codes applied for the analysis of VVER reactor primary systems in accident and transient conditions. Firstly, the main physical phenomena that occur during the considered accidents are identified, test types are specified, and test facilities that supplement the CSNI CCVMs and are suitable for reproducing these aspects are selected. Secondly, a list of selected experiments carried out in these facilities has been set down. The criteria to achieve the objectives are outlined.

The construction of VVER Thermal-Hydraulic Code Validation Matrix follows the logic of the CSNI Code Validation Matrices (CCVM). Similar to the CCVM it is an attempt to collect together in a systematic way the best sets of available test data for VVER specific code validation, assessment and improvement, including quantitative assessment of uncertainties in the modelling of phenomena by the codes. In addition to this objective, it is an attempt to record information which has been generated in countries operating VVER reactors over the last 20 years so that it is more accessible to present and future workers in that field than would otherwise be the case.

ACKNOWLEDGEMENT

This report represents the collective effort of the OECD Support Group on VVER Thermal-hydraulic Code Validation Matrix members (see Annex 1) all of whom provided valuable time and considerable knowledge towards its production. In offering its thanks to these experts, the NEA Secretariat wishes to express particular appreciation to Mr. Klaus Liesch, GRS Garching, Germany, who as the Group leader adeptly chaired the groups meetings and provided the overall co-ordination towards completing the report. Particular gratitude is expressed to Ms U. Grzesik, GRS Garching, Germany, for carrying out the large portion of word-processing and the implementation into the Internet.

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Annexes

List of Abbreviation

ACC	Accumulator
ADS	Automatic Depressurisation System
AFW	Auxiliary Feed Water
AM	Accident Management
APRM	Average Power Range Monitor
ARV	Atmospheric Relief Valves
ATWS	Anticipated Transient Without Scram
BBR	Brown Boveri Reactors
BD	Russian Test Facility
BDBA	Beyond Design Basis Accidents
BETHSY	Boucle d'Etudes Thermalhydrauliques Systèmes
BOC	Beginning of Cycle
BWR	Boiling Water Reactor
CCFL	Counter Current Flow Limitation
CCTF	Cylindrical Core Test Facility
CCVM	CSNI Code Validation Matrix
CEC	Commission of European Communities
CEEC	Central and East European Countries
CFD	Computational Fluid Dynamics
CHF	Critical Heat Flux
CL	Cold Leg
CLB	Cold Leg Break
CLG	Cold Leg Guillotine
CPT	Counter Part Test
CRM	Cross Reference Matrices
CSNI	Committee on the Safety of Nuclear Installations
DECLG	Double Ended Cold Leg Guillotine
DNB	Departure from Nucleate Boiling
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDO	Russian Test Facility
EO	Event Oriented
EOP	Emergency Operating System
EPRI	Electric Power Research Institute

EREC	Electrogorsk Research Center
ESF	Emergency Safety Features
F&B	Feed and Bleed
FCL	Free Circulation Loop
FIST	Fuel Integral Simulation Test
FIX	Swedish BWR - Related Test Facility
FLB	Feedwater line break
FTQM	Facility and Test Qualification Matrix
FWLB	Feed Water Line Break
GERDA	Geradrohr Dampferzeuger Anlage (a German straight rod pressuriser plant)
HL	Hot Leg
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HPI	High Pressure Injection
HPIS	High Pressure Injection System
HPL	High Pressure Loop
IB	Intermediate Break
INEL	Idaho National Engineering Laboratory
IPPE	Russian Research Institute
ISB-WWER	Russian Test Facility
ISP	International Standard Problem
ITF	Integral Test Facility
ITM	Integral Test Matrix
KKL	Kernkraftwerk Leibstadt (Power Plant in Switzerland)
KS	Russian Test Facility
KWU	Kraftwerk Union (German constructor)
LB	Large Break
LOAF	Loss of Auxiliary Feedwater
LOBI	Loop for Off-Normal Behaviour Investigation
LOCA	Loss of Coolant Accident
LOFT	Loss of Fluid Test
LOFW	Loss of Feed Water
LOMF	Loss of Main Feedwater
LONOP	Loss of Normal On-Site Power
LPCI	Lower Pressure Cooling Injection
LPCS	Low Pressure Core Spray
LPIS	Low Pressure Injection System
LPRM	Low Power Range Monitor

LSC	Loop Seal Clearing
LSTF	Large Scale Test Facility
LWL	Large Water Loop
LWR	Light Water Reactor
MCP	Main Coolant Pump
MIST	Multi-loop Integral System Test
MOC	Middle of Cycle
MPL	Medium Pressure Loop
MSIV	Main Steam Isolation Valve
NC	Natural Circulation
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OECD	Organisation for Economic Co-operation and Development
OTIS	Once Through Integral Systems
OTSG	Once-Through Steam Generators
PACTEL	Parallel Channel Test Loop
PCI	Pellet-Cladding Interaction
PCS	Primary Cooling System
PCT	Peak Clad Temperature
PIPER-ONE	Italian BWR - Related Test Facility
PKL	Primär Kreisläufe (Primary Coolant Loops)
PM-5	Russian Test Facility
PMK-2	Hungarian Test Facility
PORV	Power Operated Relief Valves
PS	Primary Side
PSB-WWER	Russian Test Facility
PWG	Principal Working Group
PWR	Pressurised Water Reactor
RCAs	Re-inforced Concerted Actions
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
REWETT	Finish Test Facility
RHR	Residual Heat Removal
RIA	Reactivity Initiated Accident
RNB	Return to Nucleate Boiling
ROSA	Rig of Safety Assessment (Test Facility)
RRC-KI	Russian research Center Kurchatov Institute

RWST	Refuelling Water Storage Tank
SB	Small Break; Russian Test Facility
SET	Separate Effects Tests
SETF	Separate Effects Test Facility
SETM	Separate Effects Test Matrix
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SLB	Steam Line Break
SM	Sub Matrix
SO	State Oriented
SOAR	State of the Art Report
SPES	Simulatore PWR per Esperienze di Sicurezza
SRV	Safety Relief Valves
SS	Secondary Side
SVD	Russian Test Facility
TBL	Two Bundle Loop
TC	Thermo Couple
TH	Thermal Hydraulic
TLTA	Two Loop Test Apparatus
TMI	Three Mile Island
TMLB'	This is not an acronym but it refers to Wash 1400 Accident Classification: Station blackout and small break
UCSP	Upper Core Support Plate
UMCP	University of Maryland College Park
UP	Upper Plenum
UPTF	Upper Plenum Test Facility
UPTF-TRAM	Upper Plenum Test Facility - Transient and Accident Management
UTSG	U-tube Steam Generators
VTT	Technical Research Centre of Finland
VVER, WWER	Russian Pressurised Water Type Reactor
W	Westinghouse

1 Introduction

1.1 Report Background

This is the Report of the OECD Support Group on VVER TH Code Validation Matrix. The Report is the result of the decision of the CSNI to continue with the elaboration of VVER specific validation matrices for thermal hydraulic computer codes on the basis of the well known

- Integral Test Facility CSNI Code Validation Matrix ITF-CCVM, which was initially published in 1987 and extensively updated in 1996 [1, 2],

- Separate Effect Test Facility CSNI Code Validation Matrix SETF-CCVM, which was issued in 1994 [3],

- Verification Matrix for Thermal Hydraulic System Codes Applied for VVER Analysis issued 1995 [4]. This supplement to the CSNI matrices is the result of an initiative of the Federal Minister for Research and Technology of the Federal Republic of Germany and the common work of experts from OECD, CEEC and NIS countries.

The Support Group which is comprised of experts from OECD and CEEC countries, Russia and Ukraine (Annex 1), combines the experience of the participating OECD countries with the specific experimental knowledge and data base of Russia primarily.

The Support Group, under the chairmanship of Mr Klaus Liesch met on five occasions:
In Moscow, June 1995; in Moscow March 1996; in Prague June 1997; in Budapest November 1997; and in Moscow October 1998. During additional meetings in Garching, January 1997, in Podolsk, May 1997 and in Paris, April 1998 the chairman, the task leaders and the NEA secretariat reviewed the frame and the actual technical inputs to the Report, discussed the contributions of experimental data for a planned implementation into the Russian and NEA databanks and developed a strategy how to reach the allowance. This draft report was compiled and reviewed by an editorial group composed of K. Liesch (Chairman), A. Suslov, Y. Bezrukov, S. Logvinov, I. Toth and M. Hrehor in Berlin, July 2000.

1.2 Report Objectives

Basically the mandate given to the Support Group was to review the level of validation of advanced thermal hydraulic codes applied for the analysis of VVER reactor primary systems in accident and transient conditions. Consequently the aim is to develop a supplement to the existing ITF and SETF CCVMs under consideration of the specific features of VVER reactor systems and their behaviour in normal and abnormal situations. This includes the necessary enlargement of the experimental data base for code assessment with data which were not taken into account in the previous CSNI CCVMs.
The report, in this version, is limited to the large and small break LOCAs and transients and therefore does not include shutdown transients and accident management scenarios.

The objectives of the OECD Support Group are:

- to identify the phenomena relevant in VVER reactor primary and secondary systems during LOCAs and transients.

- to compare the phenomena of VVER reactor systems with LWR reactor systems and to clarify similarities.

- to describe the phenomena involved in details as the basis for a common evaluation and an assessment by experimental data.

- to identify test facilities and experiments that supplement the CSNI CCVMs and are suitable for VVER specific code assessment.

- to establish criteria for the quality requirements and completeness of data finally to be used for the VVER specific code validation.

1.3 Report Process

The improved co-operation between Western countries and the CEEC and NIS brought the VVER topic to the attention of the CSNI scientific community. In this Report particular attention has been paid to the collection of all available technical information which supports the common understanding of the important differences that exist between VVERs and LWRs in design of systems and components, in materials of construction, in operating procedures, in both normal and emergency conditions of operation, and most important the modelling of systems and simulation of processes occurring during LOCAs and transients.

A major item namely the level of validation of advanced thermal hydraulic codes to VVER conditions was discussed, in specific VVER-440 and VVER-1000 reactor systems behaviour. For Western codes the question rose about the scope of simulation of VVER specific components, operational procedures, etc and the state of validation of the codes also including VVER specific experiments. Based on the extensive activities which have been completed by OECD/CSNI during the last decade the scientific and technical necessity resulted in the inclusion of VVER specific issues and finally led to the generation of a supplement to the existing CCVMs [4]. The general aim was to reach a common understanding in areas like relevant thermal hydraulic phenomena, current capabilities of simulations by codes, sufficiency and availability of qualified experimental data (both LWR and VVER specific) which constitute the basis for independent code assessment.

1.4 Report Structure

At the second meeting, the Support Group decided on the a structure for its review, and formed the following three Task Teams:

- VVER specific phenomena.
- VVER matrix optimisation.
- data storage.

Each Task Team prepared and presented its report to the Support Group as a whole for review and approval. Consequently, the Report represents a consensus of the Support Group that outlines the state of the art of the

- relevant VVER specific phenomena,
- three cross reference matrices,
- validation matrix.

Criteria for the selection and the selection descriptions of facilities and tests, as approved by the Support Group, are reflected in the structure of the Report.

Descriptions of VVER-440 and VVER-1000 systems, of phenomena, of selected test facilities and of requirements for VVER TH database are enclosed as appendices.

1.5 Related CSNI Activities

As a result of applied research over the last thirty years, several complex system thermalhydraulic codes exist for simulating the transient behaviour of water cooled reactors. These codes validated by a numerous experimental data obtained for accident and transient conditions relevant to PWR and BWR of Western design.

Extensive activities have been accomplished by OECD/CSNI in the Nineties to develop the basis for independent code assessment and to reach a common understanding in areas like relevant thermalhydraulic phenomena, available and needed data bases and current code capabilities. Milestones or reference reports, issued in the CSNI framework, are recalled hereafter together with related motivations or main findings.

- A State of the Art Report in modelling LOCA and non-LOCA transients was published in 1989 [5], broadly covering relevant thermalhydraulics, transients description, phenomena identification, code modelling capabilities, needs for experimental data, etc.

- The Integral Test Facility CSNI Code Validation Matrix was initially published in 1987 and extensively updated in 1996, [1] and [2]. Tests for code validation were selected on the basis of quality of the data, variation of scaling and geometry and suitability of the range of covered conditions. The decision was made in 1984 to limit the validation matrix to integral tests because it was assumed that sufficient validations against separate effects test data would be performed and documented by code developers. The last expectation has proved unrealistic. This stimulated the activity discussed in the next item.

- A group of scientists was formed at the end of the Eighties, to set-up the Separate Effect Test Facility CSNI Code Validation Matrix, that was issued in 1994, ref. [3]. The development of the SETF-CCVM required an extension of the methodology employed for the ITF-CCVM [2], both in the scope and the definition of the thermalhydraulic phenomena and in the categorisation and description of facilities. A significant result of the activity was the selection of sixty-seven phenomena assumed to cover all the thermalhydraulic situations of interest expected in PWR and BWR transients.

- The judgement of the currently available SETF data base, including parameter ranges, of the quality of measurements, of the scaling relevance and of the predictive capabilities of system codes was the objective of follow-up activities e.g. [6]. This resulted also in the identification of areas in which research efforts are still needed, concentrating on the experimental sector.

- The finalisation of code assessment includes the solution of different problems, that are strongly interrelated. In this context, CSNI reports have been issued dealing with the definition and the quantification of user or of compiler effects on code predictions, [7] and [8] respectively. Experiences have been gained from the conduction of International Standard Problems [9], [10] from the procedures currently adopted to achieve quality in results of code application [11] from the system codes use and needs in licensing [12].

- A special attention was given to the quantification of code uncertainties in predicting plant transients. The activity of UMS was started, that allowed the comparison of uncertainty results obtained from five different methodologies. Areas in which improvements are needed in code development and in the experiments have been identified from the study documented in [13].

CSNI workshops and seminars have been recently arranged in Annapolis (1996) and in Ankara (1998) and in Barcelona (2000) addressing the future needs in the area of code development/assessment/capabilities and the use of codes and uncertainty methodologies in the licensing process, respectively. The proceedings constitute the source of relevant information in the concerned areas [14], [15] and [16].

The improved co-operation between East and West, brought the VVER topic to the attention of the CSNI scientific community. The major question rose, to which extend the advanced Western system codes can be applied to VVER conditions, both VVER-440 and VVER-1000 systems respectively. Following a pioneering work in this area, [4], a group of experts from CSNI and Eastern countries was formed under the direct supervision of the CSNI Principal Working Group No. 2. Basically, the mandate given was to review the applicability of codes to VVER conditions and to enlarge the experimental data base for code assessment including data that were not taken into account in previous CSNI ITF or SETF CCVM (e.g. refs. [1] to [3], [17]). The description of the outcomings of the activities constitutes the objective of this report.

The work, that also implied the transfer of methodologies for matching the code assessment issue, was conducted following the steps identified below.

1. Characterisation of the main features of VVER reactor systems that are relevant to the thermalhydraulic design and the safety evaluation. Emphasis had to be given to the hardware and operational features that distinguish VVER from Western PWR-systems.

2. Description of postulated accident scenarios. Again an effort had to be made to characterise thermalhydraulic aspects and ranges of parameters distinguishing VVER from Western PWR-systems.

3. Identification of facilities and of experiments that supplement the PWR and the BWR SETF and ITF CCVM and are suitable for code assessment. A quality matrix or a set of conditions have been proposed for finalising the selection. Similar to the process that has been adopted for achieving PWR and BWR matrices.

4. Set up of the VVER validation matrices that basically include range of conditions already covered by the previous available CSNI matrices, but also include VVER specific phenomena and features.

2. Identification of Relevant Phenomena

Objective of this chapter 2 is to provide an information on the thermalhydraulic phenomena relevant to safety of VVER reactors and to correlate these phenomena to experimental data sets available for code validation and development (refer to chapter 4). In short, VVERs are pressurised water reactors which differ from the Western PWRs in design details. VVER technology is quite similar to the PWR technology particularly considering the global design characteristics such as light water serving as primary coolant and moderator, operating pressure range and utilisation of secondary systems for steam generations, etc. To provide basis for understanding differences between Western PWRs and VVERs designs and for identification of VVER specific phenomena appendices A and B of this report contain descriptions of the VVER-440 and VVER-1000 systems.

Because of common characteristics of PWR and VVER technologies the phenomena identified in the CSNI thermalhydraulic code validation matrices [1, 2] are in general also relevant VVERs. Therefore, the OECD Support Group focused on the identification of unique and/or specific phenomena and processes during postulated VVER accidents.

2.1 Basis for Phenomena Selection

For the selection of the phenomena three principles have been applied.

The first principle is that the phenomena identified in the CSNI matrices are in general also relevant to VVERs because of common characteristics of PWR and VVER-systems. Therefore it is important to stress that code validation and assessment plans for thermalhydraulic codes to be used for safety assessments of VVERs should be made on the basis of both: the ITF CCVM and SETF CCVM as well as on the matrices presented in this report.

The second principle for selection of the phenomena for the VVER matrix is their relevance to safety. The selected phenomena have to be important to safety and furthermore their accurate modelling in computer codes is crucial to safety analyses. Section2.3 of this report provides a tabular overview of the selected phenomena and appendix C provides a detailed description of the phenomena and the discussion on their safety relevance.

The third principle for selection of phenomena relates to accident scenarios. Three separate accident scenario groups were considered, phenomena identified, and separate cross-reference matrices were developed. These are large break LOCA, small and intermediate break LOCA and transients. Other scenarios, in particular shutdown and accident management transients should be considered in a future revision of the report. For the selection of the phenomena three principles have been applied.

2.2 Accident Scenarios Description

Relevant hardware features and postulated accidents scenarios in PWR and BWR are described in detail in [1] to [2]. Hereafter an attempt is made to extend those descriptions to the VVER-440 and VVER-1000 reactor.

From the hardware view point the main differences between Western PWR-systems and VVER-systems are the following:

VVER-440

- six loops of primary circuit;
- loop seals in hot legs;
- horizontal steam generator with two headers;

- elevation of the top of steam generators tubes related to the top of the active (about 4 m, PWR about 10 m);
- shrouded fuel assemblies is with hexagonal fuel rod arrangement;
- injection points;
- secondary side water volume in steam generators compared with nominal thermal core power is larger;
- two isolation valves in each main loop;
- special pressure suppression system (bubble condenser);
- each control rod consists of two parts: lower fuel assembly and upper absorber;
- lower plenum volume larger and a different internal structures.

VVER-1000

- horizontal steam generators with 2 headers;
- ECCS injection points;
- secondary side water volume of the steam generators compared with the nominal thermal core power is larger;
- lower plenum internal structures;
- fuel assemblies with hexagonal fuel rod arrangements.

From the operational point of view differences are present in relation to:

- operational conditions and set points of actuation of ECCS;
- working conditions of secondary side of steam generators and set points for the operation of feedwater and steam line;

The considered differences may lead to different phenomena or may affect the course of the transients.

As an example, natural circulation between core and steam generators may be affected by the difference in the elevations. The presence of the hot leg loop seals may prevent the 'reflux condensing' natural circulation mode in VVER-440.

Additionally, different condensation rates can be expected inside the upper plenum when accumulators are actuated. Differences in secondary side water volume of steam generators may cause different transient evolution e. g. in a Loss of Feedwater transient.

2.2.1 Large break LOCAs

As experiments for large break LOCAs have not been performed to the moment in integral test facilities that simulate VVERs (with an exception for one test in SB test facility of EDO Gidropress) the LBLOCA scenario for PWRs is given in this section taken from [2].

The upper end of the large-break LOCA spectrum, the double-ended pipe rupture, is clearly defined. For the lower end, a rupture of 25% of the maximum pipe area is generally accepted as a reasonable estimation of the boundary. Large break scenarios involve a very rapid depressurisation ' with significant emptying of the primary system and core uncovery taking place within only tens of seconds. Reactor shutdown or scram occurs automatically and almost immediately following the rupture. The fuel cladding undergoes rapid temperature excursions with several characteristic peaks occurring at different stages of the accident.

When the primary system pressure falls below the injection pressure of the various ECC systems, borated coolant enters the primary system and flows through the available paths to refill the lower-plenum and then to reflood and finally recover the core. Available paths depend on the break and ECCS injection locations. At the end of the reflood period, which is typically of several minutes duration, the low pressure injection system continue to operate to dissipate residual heat. The large-break LOCA is therefore divided classically into separate phenomenological windows or periods: blowdown, refill and reflood.

A typical LOCA scenario for a PWR with U-tube steam generators and cold leg ECC injection is presented. The course of events and the phenomena involved are described for a guillotine rupture of a main coolant pipe between the pump and the reactor pressure vessel. This cold leg rupture is a design basis case which is commonly analysed, because it tends to give the highest cladding temperatures within the spectrum of large breaks.

Flows in the primary system develop somewhat differently for other break locations, such as a hot leg rupture, but nearly all of the phenomena involved remain the same. As break sizes get smaller, the sequence of events is delayed, but no new phenomena are encountered until phase separation begins to occur at intermediate to small break sizes.

2.2.1.1. Blowdown phase

The blowdown phase is initiated by break opening and terminated when accumulator injected water begins to flow down the downcomer and refill the lower plenum.

Immediately following break initiation, the pressure in the primary system falls to saturation pressure. Fluid in the primary system flashes and void propagation from flashing first extends to the hotter regions of the primary system (core, upper plenum, upper head and hot legs) and then to the colder regions. Stagnation conditions may arise in the core for a short period with downflow at the inlet and upflow at the outlet. Conditions for departure from nucleate boiling take place in the core within the first second and an abrupt cladding temperature increase occurs. The cladding temperature rise and its extent strongly depend on the energy stored in the fuel (i.e. its temperature profile prior to the accident).

Critical flow discharge rates during the initial depressurisation are extremely high due to subcooled water near the break. In the broken loop, the flow reverses from the vessel to the break. It is fed by coolant flow resulting from the draining of the pressure vessel and from the draining of the intact loops steam generators through their cold legs and the downcomer annulus. The downward core flow is fed by coolant flowing back through the hot legs from the intact loops and the steam generators. However, flow through the core in an upward direction may be briefly re-established when break flow becomes two-phase and coolant is still being supplied by the pumps coasting-down if some coolant in excess is directed into the lower plenum.

During the blowdown period which lasts about 20 s, both downward and upward flow of coolant through the core ensure efficient cooling of fuel rods. Cladding temperatures are turned over after reaching a maximum value (blowdown temperature peak) and a total or partial quenching may occur. This cooling continues until the core flow subsides then cladding temperatures, now driven by decay heat, rise again.

When the primary system pressure and temperature fall below those of the secondary side, reverse steam generator heat transfer occurs. At lower pressures, the accumulators start to inject into the cold legs. At first, emergency coolant accumulates in the cold legs. It is entrained and carried round the downcomer annulus and out to the break by the counter-current flow of steam from the lower plenum. Towards the end of the blowdown period, this so-called "downcomer bypass” process stops and the injected coolant "penetrates" the downcomer. Actually, the coolant bypass to the break is never complete due to the annulus configuration of the downcomer. Therefore the boundary between the blowdown phase and the refill phase is somewhat vague.

2.2.1.2 Refill phase

Since there is a period of overlap between the blowdown and refill phases, the coolant inventory in the pressure vessel heads a minimum before the blowdown has ended. This minimum does not necessarily correspond to the ECC injection rate exceeding the break discharge rate, which solely indicates an overall increase in primary system inventory.

The refilling of the lower plenum is largely governed by how quickly the accumulator injected ECC can penetrate the downcomer annulus and reach the lower plenum. Steam flows limiting this injection are generated not only due to the flashing of coolant during two overlapping blowdown window, but also due to the energy stored in the downcomer annulus walls, which is transferred to ECC coolant as annulus penetration begins. As the blowdown comes to an end, the counter-current flow of steam is reduced. By this time, massive quantities of ECC are being injected. Both high and low pressure injection systems are operational but their contribution to ECC are initially small compared with those from the accumulators. Multi-dimensional effects and steam condensation break down the remaining steam flow barrier allowing injected coolant to flow down the downcomer and refill the lower plenum progressively. The low quality mixture first cuts off the steam flow path around the lower edge of the core barrel. Shortly afterwards, it reaches the bottom of the core. This signifies the end of the refill period and the beginning of core reflood.

The refilling process duration is in the order of 20 s and during this period the core experiences nearly adiabatic heating. Some steam cooling is present and, if temperatures are high enough, zircaloy-steam reactions begin. Finally, a period of reverse break flow may occur as the ECC condenses steam in the primary system. This can cause the primary system pressure to fall below that of containment.

In plants with combined injection, water can penetrate the upper-plenum and quench the top of the core during the refill phase.

2.2.1.3 Reflood phase

The reflood phase begins as soon as, the ECC reaches the hot fuel rods at the bottom of the core. A quench front is formed on the fuel rods and large amounts of steam are generated by the energy released from the rods at a high temperature. This steam produces a back-pressure opposing the driving head of coolant in the annulus thereby slowing or even reversing the water level rise in the core. Thus, reflooding of the core proceeds with level oscillations (strong at the beginning, moderate later) occurring in both the core and downcomer.

Steam binding, as this back- pressure is called, is a complex phenomenon largely controlled by the pressure drop of the steam and strained droplet flows leaving the vessel and escaping to the break through the primary system loops. The pressure drop is augmented by additional steam generated outside the core by heat transfer from the secondary side or by steed energy in hot pipes or vessel walls. The source for this steam production is mainly due to liquid droplets entrained by steam at the vessel outlet

Emergency core coolant injected into the cold legs not only condenses steam, thereby mitigating the loop pressure-drop, but also increases the head of water in the downcomer, which is driving the reflood process.

Nitrogen is injected into the cold legs after the coolant has been driven from the accumulators for reactors where the discharge line cannot be isolated. This gas injection can completely change the flow and heat transfer characteristics of the loops. On the one ham), it can enhance the liquid flow into the core and thereby increase the core reflooding rate. On the other hand, it can interfere with heat transfer and steam condensation.

Once the accumulators are empty, or isolated, the low pressure injection system supply the coolant necessary to continue and complete the reflooding process. This process becomes somewhat complex in that a part of the water entering the core quenches the bottom of the fuel rods, bringing the cladding temperature down to saturation, whilst the rest is driven upwards through the core as a mixture of steam and entrained droplets. The mixture provides some cooling at upper core elevations, where the maximum cladding temperatures prior to the final quench are reached. De-entrainment of liquid may occur at the upper core tie-plate and on the forest of structures in the upper plenum. Liquid films on these structures form. Droplets may be entrained by the steam flow to the hot legs or may fall downwards and back into the core. These droplets and films can lead to the formation of a pool of water in the upper plenum and/or a quench front which propagates downwards into the core (top-down quenching).

During this reflooding process, the core flows and quench fronts are two-dimensional. The hottest core regions may experience upward flows and the cooler peripheral regions may experience liquid fall-back and early quenching. Fuel rod cooling differs between core regions, depending on the local flow situation. In the unquenched portions of the core, the rods experience inverted annular or dispersed flow film boiling heat transfer soon after the beginning of the reflood window. In the dispersed flow regime, the heat transfer effectiveness is largely governed by the density of entrained droplets. Sufficient heat transfer is provided by these regimes to turn around the cladding temperature excursion prior to quenching.

The end of the reflood period, the duration of which is about 150 s, is signified by complete quenching of the core. Thereafter, the water inventory in the primary system increases rapidly until the coolant lost through the break balances that injected. Continued injection of coolant into the primary system enables decay heat to be transported to the containment from where it is transferred to an ultimate heat sink for long term cooling purposes. This denotes the end of the large break LOCA condition.

2.2.2 Small break LOCAs

When the main circulating pumps are tripped the loop flow is dominated by gravitational effects. The efficiency of residual heat removal in natural circulation has been investigated in integral test facilities modelling VVER type nuclear reactors. Two type of tests are described here: a stationary test addressing energy transport on the primary side when primary inventory is reduced step by step and a test where behaviour of single-phase natural circulation heat transfer mode is observed when secondary side level is reduced. The experiments referred here has been conducted for VVER-440 but qualitatively very similar behaviour can be expected in VVER-1000 except for the hot leg loop seal effect.

2.2.2.1 Stationary test addressing energy transport on primary side

This type of experiments are performed to study natural circulation flow regimes. The tests are conducted by measuring a loop behaviour under steady state conditions over a range of primary side inventory levels. Constant secondary side conditions are maintained through out an experiment. The tests are terminated when the core heat-up is observed.

Three natural circulation modes occur when the primary inventory is changing:

- single-phase natural circulation,

- two-phase natural circulation and

- boiler-condenser natural circulation.

The heat transfer from the core to the steam generator is obtained through single-phase natural circulation when the primary inventory is >90%. There is a reverse flow in the lower part of the heat exchange tube cluster in the steam generator.

When the level in the upper plenum reaches the level of hot leg entrances the flow starts to change from single-phase to two-phase flow. The transition is not smooth but hot leg loop seal causes oscillatory flow behaviour. Liquid in the bottom part of the hot leg blocks the flow path of the two-phase mixture and the heat transfer starts to deteriorate which initiates the re-pressurisation of the primary side. Some of the primary coolant flows to the pressuriser which clears at least one of the hot leg loop seals. The flow resumes and the pressure starts to decrease. The loop flow is highly asymmetric during this period. Since the upper plenum pressure is decreasing, coolant from the pressuriser surges to loops and upper plenum and hot leg loop seals are blocked again. This initiates the next primary pressure peak and cyclic operation continues.

When the inventory is below 70% two-phase flow is established. The loop mass flow rate decreases along with the reducing inventory.

In boiler-condenser natural circulation mode the mass flow rate is very low due to the high energy transfer rate. The steam produced in the core flows through the hot leg to the steam generator and condenses and the condensate flows to the cold leg. The boiler-condenser heat transfer mode starts when the primary inventory is below 60%. The core uncovery occurs when the inventory is around 35%.

2.2.2.2 Stationary test addressing energy transport on secondary side

The purpose of this type of experiment is to determine how single-phase natural circulation in the primary side is affected by descending secondary side level. Large water inventory in the secondary side makes the dry out of the secondary side a very slow transient. The heat transfer is sufficient for core cooling even with low secondary side level. Secondary side steam becomes super-heated above swell level when there are uncovered heat exchange tubes in the top part of the tube cluster. The cold leg temperature increases when a tube layer uncovers which induces the hot leg temperature increase. The loop mass flow rate reduces momentarily when the temperature difference between the hot and cold leg is reducing, but returns close to the original value as the hot leg temperature increases. The differential pressure between the hot and cold collector changes because of temperature (density) change in the collectors. This results to descending flow reversal level in the heat exchange tube bundle. Primary coolant flows to the pressuriser when the primary temperature increases which finally yields to rising primary pressure.

2.2.2.3 Small break overfed with HPIS, secondary side necessary

This scenario for VVER-1000 plant takes place at leak flowrates up to 200 t/hour. Respective flowrate of HPIS pumps can be ensured only if primary pressure decreases to the values of about 80 to 90 bar. So in this scenario level in pressuriser drops and reactor trip occurs due to low primary pressure. Reactor trip causes closure of turbine stop valves and MCP trip. Primary pressure decrease leads to decrease of saturation margin for primary coolant. If saturation margin becomes lower than 10 K then step-wise start of safety systems occurs and HPIS pumps are actuated. Secondary pressure is maintained by operation of steam dump devices.

After actuation of HPIS pumps primary pressure begins to grow. Stabilisation of primary pressure takes place at pressures of 80 to 90 bar. If HPIS flowrate is larger than leak flowrate then pressuriser level grows and its overfilling is possible. So the operator should switch one or two HPIS pumps to recirculation and to keep balance of leak and HPIS flowrates with number of HPIS pumps operated to primary circuit. When required concentration of boric acid is established in primary circuit the operator may start the NSSS cooldown through secondary circuit.

2.2.2.4 Small break without HPIS overfeeding, secondary side necessary

In such a transient a high pressure injection does not compensate the loss of coolant through the break and the energy transported by the break flow is insufficient to depressurise the primary system. Secondary side cooling is necessary to reach the set-point of the low-pressure safety injection system.

Initial depressurisation leads to emptying of the pressuriser and saturated conditions in the hottest parts of the primary circuit. It is in this period, that high pressure pumps start to inject water to primary, the main circulating pumps are automatically switched off by low primary pressure and low pressuriser level and reactor scram occurs as well as turbine trip. If the break is in a cold leg it can be assumed that part of the injected water will flow directly to the break. Steam is produced and accumulated in the pressure vessel upper head and this results in a reduction of the depressurisation rate. When the level in the pressure vessel reaches the hot leg entrances, the heat transfer from primary to secondary starts to deteriorate in a VVER-440, because steam can not enter to the hot leg loop seal. Intense boiling occurs in the core and the primary pressure starts to increase.

Since the primary side is loosing coolant to the break, the hot leg loop seals are cleared and the loop flow resumes. There are significant differences between the loops during this period. Some of the loops may be in single-phase natural circulation mode and only very limited amount of energy is transferred to the secondary side in those loops, while the other loops take care most of the heat transfer. These loops are in boiler condenser mode for a short period after loop flow resumption. Such an efficient heat transfer mode induce a fast decrease in the primary pressure. The hydro accumulators start to inject during this period. The timing of injection depends on the available high pressure injection capacity, the break size and the break location. Because of increasing primary inventory, hot leg loop seal clearing and filling may produce oscillation in the primary parameters.

The primary pressure behaviour differs from the above description if the break is in a hot leg. In this case the hot leg loop seal has no significant effect since there is continuous flow through the core. In addition to this the total high pressure safety injection capacity can now be assumed to be available, since the break is not in the vicinity of injection location.

Operators decrease the primary pressure by cooling down the secondary side so that the pressure reaches the delivery head lift of the low pressure injection pumps. The sequence of events depends on how early the secondary side relief valves are opened. The secondary side is heating the primary side if the operators open the relief valves during the late phases of the transient, while early intervention maintains the direction of heat transfer from primary to secondary during the whole transient.

2.2.2.5 Intermediate break, secondary side not necessary

For larger break sizes the energy transported by the break flow is sufficient to depressurise the primary system, secondary side cooling is not necessary to reach the set-point of the low-pressure safety injection system. The phenomena encountered in VVERs for this type of transient are different from PWRs mainly due to

- the horizontal layout of the SGs,

- the hydro-accumulator (HA) set-point pressure, which is higher than the secondary pressure,

- the pressure, where HAs are empty being significantly higher than activation pressure of LPIS.

In VVER-440s there are additional effects from

- the hot leg loop seals,

- the relatively low core power that allows relatively small break sizes to belong to this type of transient category.

Initial depressurisation is rather fast leading to emptying of the pressuriser and saturated conditions in the hottest parts of the primary circuit. Steam is produced and accumulated in the vessel upper head, the primary SG collectors on the hot side and this results in a reduction of the depressurisation rate. It is in this period that the MCPs are automatically switched off by high containment pressure. On the secondary side the steam dump valves are actuated after turbine trip and regulate the pressure in the early phase of the transient.

The HAs start injecting in the same period, when the vessel level decreases to the elevation of the hot legs. In VVER-440s this leads to deterioration of the natural circulation, since the steam produced in the core cannot pass the hot leg loop seal and is trapped in the upper plenum. Injection is fairly smooth: the HAs maintain the primary pressure. Hot leg loop seal clearing may produce oscillation in the primary parameters (especially, with smaller breaks), due to periodic passage of steam across the loop seal, condensation in the SG and start-up of natural circulation. Once the hot leg loop seal is completely cleared, the primary and secondary pressures equalise, which lasts until the cold leg loop seal is cleared. The main difference in this period to PWRs is that in VVERs reflux of condensed coolant via the hot leg to the vessel cannot be expected due to the horizontal lay-out of SGs and (in 440s) to the hot leg loop seal.

With the break situated in the cold leg, as the primary mass inventory further decreases the manometric head difference in the two vertical sections of the pump seal makes the vessel level decrease faster than that of the downcomer: this is the well-known cold leg loop seal effect, which takes place in VVERs in a similar way to PWRs and can lead to core uncovery. When finally steam passes the loop seal, the break conditions change from single phase to two-phase, resulting in increased rate of depressurisation, followed by a vigorous injection from HAs. As a result, the vessel level increases and even hot and cold leg loop seals are partially filled up. The primary pressure falls well below the secondary side pressure.

After HA injection is terminated reverse heat transfer in the SG makes primary pressure to rise. The fact that loop seals have been partly refilled hinders the passage of steam produced in the core to the break, which results in a renewed loop seal effect and in a possible second core uncovery. The problem is aggravated by the fact that primary system pressure is only slowly decreasing and there could be a long period, before LPIS is active. Whether there is a danger for the core in this period depends on break size and the number of HPIS available. Operator actions (e.g. secondary side depressurisation) may be necessary to handle this problem.

2.2.2.6 Pressuriser leak

The initiating event can be the rupture of the pressuriser safety valve line or the inadvertent opening of the safety valve. The system quickly depressurises to the low-low pressuriser pressure signal, which scrams the reactor and starts up the HPIS. As a consequence, the turbines are tripped and the secondary pressure will be regulated by the steam dump system. The MCPs are tripped later in the transient - depending on the break size - by high containment pressure.

The pressuriser level first decreases then, due to HPIS and voiding of the hottest parts of the primary circuit, increases: this is the well-known “TMI effect”. Depending on the break size and ECCS injection, the pressuriser can be filled up completely, or to the level of the break. The hydro-accumulators start to inject, when the vessel level drops near the hot leg elevation. If the mass lost via the break exceeds ECCS flow, the hot leg can be emptied as well. This leads to different behaviour in VVER-440 and -1000 plants, due to the hot leg loop seal in the former. When the primary level decreases to the elevation of the surge line/hot leg connection, the pressuriser level decreases and the break flow changes again to single phase steam. This reduces the mass lost via the break and the primary system level stabilises well above the core exit plane. Overheating of the core is not expected, even if minimum safeguard systems are assumed.

2.2.2.7 Steam generator tube rupture

PSA analysis show that primary to secondary leaks are one of the main contributors to the core melt frequency in VVER reactors. In VVERs two cases are considered in this category: a steam generator tube rapture and an opening of the SG header cap. The management of a primary to secondary leak is originally based on isolation of the leaking steam generator (primary and secondary side). However, there is a considerable risk that the isolation fails. In addition to this secondary side safety valve may ultimately stick open and cause the loss of primary coolant.

The aim of operator measures is to minimise coolant mass which may be released to atmosphere through steam generator safety valve. This goal is obtained by decreasing the primary pressure below the set-point of the secondary side safety valve. After identifying that there is a primary to secondary leak, the operator should isolate the broken steam generator secondary side and start lowering the primary pressure. The pressure can be reduced by using the pressuriser spray and by cooling the loop using intact steam generators. In a one tube rupture event the stopping of safety injection is an useful action in reducing the primary pressure.

A steam generator tube rupture event is a slow transient in comparison to opening of a steam generator header cap case. So, the sequence of events in the tube rupture transient depends on timing of assumed operator intervention, unlike the early phase of collector cover opening where the operators have no time to intervene. In addition to this there are significant differences in VVER reactors for handling primary to secondary leaks: Operator instructions vary between the power plants and the power plant configurations are not the same. Hence no sequence of events is given here.

2.2.2.8 Steam generator header rupture

The opening of the SG header lid is considered in most countries operating VVERs as a safety analysis report case since this accident occurred in the Rovno plant. The maximum possible opening of the lid represents a primary to secondary break with an equivalent break size of about 100 mm, i.e. an intermediate break well above the hot leg elevation. The main concern in this accident is two-fold:

- primary liquid can be released to the atmosphere via the SG relief and safety valves,
- the break represents a containment bypass, that may lead without operator intervention to core degradation.

The accident starts with subcooled primary coolant flowing to the secondary side of the faulted SG, with corresponding increase of the SG level. The turbines are tripped by the high SG level signal and the reactor is scrammed. In the mean time the primary pressure quickly decreases to the setpoint of ECCS start-up and high pressure injection is started.

The secondary pressure temporarily rises to the setpoint pressure of the SG relief and safety valves. Since in the affected SG the flow through the safety valve may change to two-phase condition, there is a danger for the safety valve to stuck in the open position, leading to continuous radioactivity release to the atmosphere and to loss of primary fluid. Without any operator action the primary pressure remains higher than the secondary one. Soon the primary pressure sufficiently decreases so that the hydro-accumulators start to inject as well: this assures sufficient primary inventory and there is no danger for the core. However, without operator intervention ECCS water reserves will be exhausted in about 3-3.5 hours.
Original VVER emergency operating procedures prescribe isolation of the faulted steam generator on the primary side by closing the main gate valves in the affected loop both in the hot and in the cold legs. In several countries new procedures are being developed, that envisage cooldown of the primary circuit via the intact SGs and its depressurisation by injection of HPIS water to the pressuriser. The procedure also prescribes the isolation of the hydro-accumulators and the HPIS not necessary to keep sufficient primary side coolant inventory, the overall aim being to reduce the break flow rate to the secondary side.

2.2.3 Transients

2.23.1 ATWS

The switch-off of all MCPs with failure of actuation of emergency protection system can be considered as an example of ATWS scenario. It is assumed that this failure of actuation of emergency protection system occurs because of failure of mechanical parts of control rods. Consequently neither accelerated unit unloading nor operation of power limiter (ROM) are possible.

After switch-off of all MCPs their coastdown takes place and flow rate of coolant through the core decreases. So temperature drop between primary and secondary circuits increases and heat transfer from primary to secondary circuit is enhanced. Evaporation of SG water increases and secondary pressure grows.

Increase of coolant heat-up in the core leads to growth of fuel temperature and decrease of coolant density. Reactivity becomes negative and power decreases. Power decrease causes in turn coolant heat-up and decrease of fuel temperature and coolant density. Finally parameters of the NSSS are stabilised at certain power level (for VVER-1000 the power is about 30% of nominal value). Behaviour of main parameters of primary and secondary circuit may be oscillatory. This can depend on operation of some secondary circuit controllers (feedwater flow rate controller and turbine flow rate controller).

2.2.3.2 Loss of feedwater, non ATWS

Just after switch-off of main feedwater pumps the level in all steam generators begins to drop and stop valves of the turbine close. Auxiliary feedwater pumps begin to operate on the signal of loss of main feedwater. Closure of the turbine stop valves leads to sharp pressure increase in main steam header that causes opening of steam dump to turbine condenser (BRU-K).

On the signal of the turbine stop valve closure the accelerated unit unloading begins (selected group of control rods is inserted into the core). After that power limiter continues to decrease reactor power.

The auxiliary feedwater pumps are not able to compensate the loss of main feedwater when the reactor is operated at nominal power. Deficiency of feedwater leads to the SG level decrease and to stop of all MCPs. This causes actuation of the emergency protection system which in turn terminates action of the power limiter.

Soon after stop of the main feedwater pumps the primary pressure increases due to decrease of heat transfer to the secondary circuit but after that it begins to decrease due to the action of accelerated unloading and power limiter. The reactor trip leads to further decrease of primary pressure and temperature. Soon the parameters of primary circuit are stabilised. Heat removal through the secondary circuit is ensured with operation of the auxiliary feedwater pumps and BRU-K operated in the pressure maintenance mode.

After the reactor trip the pressuriser level first decreases and after that it stabilises according to the power level. This is ensured by operation of the pressuriser level controller which affects the makeup flow rate.

After parameter stabilisation the operator may cool down the unit. To perform this the operator switches the BRU-K which operated in the pressure maintenance mode into the cooldown mode with rate of 30 ^{o}C/hour.

In case of total loss of feedwater (beyond design basis accident) the SG water is being boiled off. The SG level decreases steadily and to certain moment the SG tube bundle is completely dried out.

The unbalance between generated and removed heat in the primary circuit causes pressure and temperature increase, this leads to the pressuriser safety valve opening. Due to loss of coolant through the safety valves the reactor level drops below the hot leg nozzle elevation. After that natural circulation in primary circuit is terminated. When the reactor level decreases to the elevation of top of core the core heat-up begins starting from the upper part of fuel rods. At this stage the loop seal clearing is possible that leads to re-establishment of weak natural circulation in primary circuit. This natural circulation can terminate temporarily the cladding temperature growth and even to decrease slightly the cladding temperatures but, in general, this natural circulation is not sufficient to remove the decay heat and soon the cladding temperatures continue to increase.

For the total loss of feedwater accident the operator actions are needed to implement the “feed and bleed” procedure in primary circuit.

2.2.3.3 Loss of heat sink, non ATWS

Accidents of this type are associated with loss of feedwater (see section 2.2.3.2) or with loss of possibility to take away steam from steam generators. Let us consider the scenario with closure of the turbine stop valves for VVER-1000 plant.

Due to closure of the turbine stop valves the reactor power decreases because of actuation of accelerated reactor unloading and power limiter operation. Power decrease leads to the pressuriser level drop. Short-term secondary pressure increase is possible. After that steam dump devices maintain secondary pressure at the initial level. Feedwater temperature decreases due to switch off of the high pressure heaters. The NSSS parameters are stabilised in 5 to 6 minutes after accident beginning.

2.2.3.4 Loss of off-site power

Loss of off-site power leads to the reactor trip, to the switch-off of the turbine, MCPs, makeup pumps and feedwater pumps. The transition to the emergency power supply from diesel-generators is performed automatically.

In the beginning of the transient the closure of the turbine stop valves leads to the growth of the secondary pressure and actuation of steam dump to environment (BRU-A). Cooling of primary circuit is ensured first with the MCP coastdown and after that with natural circulation. When diesel generators start to operate the emergency feedwater is supplied to steam generators and thus heat removal from primary circuit is ensured. The emergency feedwater is sufficient to remove the decay heat. The steam generated in secondary circuit is removed through BRU-As.

The AM strategy for this accident is to decrease primary pressure in order to ensure the HPIS operation to primary circuit. This can be done by means of opening of the pressuriser safety valves or the emergency gas removal system.

In case of station blackout (additional failure of diesel-generators) the accident is more serious as the emergency feedwater and ECCS water are lost. As there are no means to remove decay heat from primary circuit, the accident develops at high primary pressure and periodical opening of the pressuriser safety valves. The loss of primary coolant through the safety valves leads to the core dryout and heat-up and to transition of the accident into the severe stage.

2.2.3.5 Main steam line break

Two rather different types of accidents can be pointed out in this group of accidents. The first type is a main steam line break in isolated part of steam line (behind the steam line shut valve). In accidents of this type the fast isolation of respective steam generator from the break location occurs by means of closure of the shut valve. Until the break is isolated the loss of steam-water mixture through the break takes place that leads to the SG level drop. Nevertheless there is no a serious challenge to the unit in this accident because the reactor trip occurs and MCPs are switched off soon. So the primary temperature decrease is not large and there is no danger from view point of the core re-criticality.

The second type of accidents of this group is the steam line break in non-isolatable part of steam line. This class of accidents is more serious because the total loss of water in affected steam generator occurs and pressure in this steam generator decreases to low value. This leads to enhancement of heat transfer from primary circuit to the affected steam generator and, consequently, to the overcooling of primary circuit. At large primary overcooling the core re-criticality becomes possible.

For this class of accidents the non-monotonic dependency of primary overcooling on the size of break can be noted. The matter is that enhancement of heat transfer into the affected SG takes place until the SG tube bundle is dried out. So the worst case is the case of partial break of the steam line at which the secondary pressure decreases quickly enough (enhancement of heat transfer) but entrainment of steam-water mixture is relatively small (prolongation of time period when enhanced heat transfer into secondary circuit takes place).

For this accident the danger of the core re-criticality is less if the HPIS pumps are available to supply boric acid concentrate into the primary circuit.

2.2.3.6 Feedwater pipeline break

The progression of this accident depends on the break location. If the break occurred behind the check valve, then the loss of feedwater takes place in respective steam generator. The water level in affected SG decreases due to boil-off of the SG water. The level decrease causes the MCP trip on the respective loop and power decrease to the required value.

If the break takes place between the check valve and the feedwater pipeline inlet nozzle (i.e. in non-isolated part), then outflow occurs first of steam and after that of steam-water mixture from the affected steam generator. The pressure in this steam generator drops that causes the enhancement of heat transfer from primary circuit. The enhanced heat transfer mode takes place until the SG tube bundle is dried out. Thus, in this variant the feedwater pipeline break accident is similar to the main steam line break in non-isolated part.

2.2.3.7 Cooldown with primary "feed and bleed" procedure

In accidents with high primary pressure (for instance station blackout or very small break LOCAs) some measures are needed to decrease primary pressure and to reach cold shutdown conditions.

If decay heat can be removed through secondary circuit the NSSS cooldown can be performed using steam dump devices of secondary circuit (BRU-K or BRU-A). Primary pressure decrease is performed by means of opening of the emergency gas removal system or pressuriser safety valves.

In scenarios with loss of heat removal through secondary circuit (for instance total loss of feedwater) the only means to cool down the NSSS is primary “feed and bleed” procedure.

Decrease of primary pressure is ensured with opening of the pressuriser safety valves or emergency gas removal system that leads to leak of hot coolant from primary circuit ("bleed”). First, steam flows through the opened valve. After that transition to water leakage occurs.

When primary pressure decreases to a certain level the HPIS pumps are actuated to primary circuit. The HPIS pumps supply cold water to primary circuit (“feed”). During this procedure the balance should be kept between amount of water supplied and leak flowrate.

2.2.3.8 Reactivity disturbances

A typical scenario of this type is a control cluster ejection from the core of VVER-1000.
This accident is characterised with the following features:

- Sharp short-term decrease of the reactor period, increase of neutron flux and reactor thermal power,
- Ejection of a VVER-1000 control cluster forms the primary leak with equivalent diameter of 55 mm,
- Balance between leak flowrate and HPIS water supply is established at pressures of about 55 to 60 bar. It means that a short-term actuation of hydro-accumulators is possible.

During the first second the reactor neutron power increases to the value of 115 to 130 % of nominal power. The reactor trip occurs when the power exceeds 107%. As the cluster ejection leads to the primary leakage the primary pressure and saturation margin decrease. After that the process is similar to that of small break without HPIS overfeeding.

Other scenarios with insertion of positive reactivity are inadvertent upward movement of operating control group and decrease of boric acid concentration in primary circuit. In these accidents the primary leakage does not occur. Reactivity insertion is slower than in the accident with control cluster ejection.

2.2.3.9 Overcooling

The transients with overcooling are caused by the following reasons:

- Failures of the SG level controllers (overfeed of SGs with cold water),
- Opening with further failure to close the SG safety valves or steam dump to environment,
- Leaks and breaks of feedwater pipelines and steam lines.

A typical scenario of this type is a steam line break or feedwater pipeline break. The most serious consequences of these accidents are low primary coolant temperatures, insertion of positive reactivity, unsymmetrical processes in different loops.

2.3 Tables of Phenomena Descriptions

Large Break LOCA

No.)[1,2]*	**Phenomena**	**Description**	**Safety significance**
1 (B1.1) [3.1.1]	Break flow	Single and two phase critical flow at the break.	Controls the availability of coolant mass for core cooling and determines the depressurisation rate of the system.
2 (B1.2) [3.2]	Phase separation	Major concern is the vapour liquid separation at the end of blowdown in the reactor vessel. The separation, is not a unique process to VVER, but because of the specific geometry of the vessel and internals, the applicability of the PWR data is not directly applicable and different, VVER specific, data will be necessary for code validation.	The phase separation in the upper plenum and the core will control the fuel cladding temperature at the blowdown end. The separation in the downcomer and formation of liquid level will influence the driving force for the reflood and therefore the cladding temperature.
3 (B1.3) [3.6]	Mixing and condensation during injection	As the ECC water is injected, it mixes with the fluid in the UP and downcomer causing rapid condensation of the vapour on the cold liquid. The efficiency of this process affects the depressurisation of the system and causes also system pressure oscillations. The VVER injection sites are different then those in PWRs and therefore the rate and magnitude of the process will be different. Also, this process has direct feedback on other processes such as phase separation CCFL and entrainment.	This process has an effect on core cooling because it affects the fluid flow through the core and therefore the cladding temperatures.
4	2-phase flow in SG primary and secondary side	Only 2-phase flow in SG primary side is relevant to large break LOCA in VVERs (blowdown, refill, reflood). (See item 9)	See item 9
5 (B1.4) [3.10.2]	Core wide void and flow distribution	Three-dimensional flow distribution in the core during blowdown and refill. The flow and void distribution is very specific to each core structures and geometry and in VVERs they will be different than in PWRs and will also vary between VVER 440 and VVER 1000. The boxed fuel assemblies in VVER440 will prevent stranded flows within the core.	The void distribution and coolant flow directly affects the fuel cladding temperatures.

[1] numbers in the () indicate section of the "CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients" [NEA/CSNI/R(1996)17] providing description of the phenomena; numbers in the [] brackets indicate sections in "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation" [NEA/CSNI/R(1993)14]report description the phenomena

[2] Last revision October 8, 1997

No.)[1,2]*	Phenomena	Description	Safety significance
6 (B1.5) [3.21]	ECC downcomer bypass and penetration	The ECC water bypass flow occurs when most of the coolant injected into the downcomer during blowdown flows into the broken cold leg instead down to the lower plenum. This phenomenon is similar to the ECC downcomer bypass in PWRs.	This phenomenon has direct influence on refill of the lower plenum and initiation of the reflood, and therefore on fuel cladding temperature.
7 [3.10.1]	UP Injection and penetration	In VVERs ECC water is also injected into the upper plenum. Condensation in upper plenum and CCFL, due to vapour flowing upwards, effects on UCSP and within the core will limit the penetration and efficiency of this injection. The extend of this phenomenon is very specific to VVERs. (see also 3 and 8)	The ECC injection penetration has direct influence on cladding temperatures.
8 (B1.6) [3.9]	CCFL	CCFL phenomena may occur at the outlet of the core and UCSP. The process and its effects are very similar to PWR, however very geometry dependent.	CCFL usually affects delivery of water from higher parts of the system to lower parts such as the core and therefore influences core cooling.
9 (B1.7)	Steam Binding (liquid carry over, etc.)	Evaporation of liquid droplets carried out of the core results in local pressure increase in hot legs and upper plenum slowing down the water level rise in the core. This phenomenon is counteracted by the upper plenum injection Steam binding denotes the generation of backpressure in the steam generators by the evaporation of droplets which were entrained by steam from the liquid in the core and UP. This phenomenon is similar to that in PWRs with ECC injection into hot legs. However, the magnitude of backpressure generated in the horizontal steam generator may be different than that is in vertical U-tubes steam generator.	Has some potential effects on the core cooling.
10 (B1.8) [3.10.1]	Pool Formation in UP	Water may be trapped in the upper plenum. Additionally to de-entrained water carried from the core in VVERs the upper plenum injection will significantly contribute to the pool formation. The pool formation is closely linked to UP Injection and Penetration under Item 7	Pool formation in the upper plenum relates to effectiveness of upper plenum ECC injection and top-down core cooling capability.
11 (B1.9) [3.11]	Core Heat Transfer incl. DNB, Dryout, RNB	Mechanism of heat removals from the fuel rods. On a local pin scale, these heat transfer mechanisms are similar to those in PWR. However the bundle or overall core heat transfer process may differ due to different geometries.	Control of the fuel clad temperature.
12 (B1.10) [3.12]	Quench Front Propagation - up and down	During reflooding, the water is forced into essentially hot and empty core from the lower plenum and quenching of the hot cladding surfaces occurs. As more water is added the quenching front moves upward through the core. The re-establishing of the contact between the coolant and hot cladding surfaces forms the quench front. The quench front phenomena are similar to those in PWR, however the front propagation may differ because of different VVER core geometries. Due to upper plenum injection also a quench front will form in the upper part of the core and will penetrate down into the core	The re-wetting characteristic are largely responsible for maximum fuel cladding temperatures.

No.)[1, 2]*	**Phenomena**	**Description**	**Safety significance**
13 (B1.11) [3.5]	Entrainment (core, UP)	Due to sufficiently high steam velocities, water can be entrained and carried to other parts of the system. This phenomenon is in VVERs similar as in PWRs within the core, however strongly affected by the core geometry, but quite different in the upper plenum because of the upper plenum injection.	This process will affect the amount of coolant remaining in the core during refill and reflood phases of the LBLOCA and potentially have an influence on fuel cladding temperature.
14 (B1.12) [3.5]	Deentrainment (core, UP)	Water removal from the steam-carrying droplets. This process is strongly dependent on local geometries and will be different in VVERs than in PWRs. Also it will be strongly affected in the upper plenum by the upper plenum injection.	Deentrainment in the core reduces cladding temperatures.
15 (B1.13) [3.15]	1 and 2-phase pump behaviour	Only two-phase behaviour of the pumps is relevant to large break LOCA since degradation or phase separation phenomena on the pump impellers may affect the timing and point of stagnation **of the core flow**. The pump behaviour is specific to the pump.	May have effects on the fuel cladding temperature excursion during blowdown.
17 (B1.14) [3.24]	Non-condensable gas effects	Nitrogen released from the accumulators has mechanical and thermal effects on the system behaviour. Through expansion in the downcomer and upper plenum it will have impact on downcomer water level. It will also affect condensation phenomena in various parts of the system.	It may have effects on general thermal-hydraulic behaviour of the system.

Small Break LOCA

No.)[3, 4]	Phenomena	Description	Safety significance
1 (B2.1)	Natural Circulation, 1 phase primary side	During early phases of some SB LOCA, after RCPs are stopped, natural single-phase circulation is established between the core and steam generators.	The single phase NC removes decay heat from the core.
2 (B2.2)	Natural Circulation, 2 phase, primary side	As the coolant inventory decreases voids produced in the core are condensed in the steam generators establishing a two-phase natural circulation. This is a very important phase in SB LOCA because it is the most efficient mechanism of removing decay heat. The process is depended very much on specific design of the reactor and the steam generators. Existence of hot leg loop seals in VVER-440 has an important impact on 2-phase natural circulation, with a possibility of oscillatory behaviour at mass inventory close to hot leg loop seals clearing, due to periodic condensation of steam in the SG.	Efficient heat removal process prevents core from overheating.
3 (B2.3)	Reflux condenser mode and CCFL	Any significant reflux condensation is not expected in VVER-440s due to the effect of hot leg loop seals, but it can be a very important heat removal mechanism in VVER-1000. The steam from the core is condensed in the steam generator tubes, and may flow to both tube ends. The reflux condensation process is very specific in VVERs because of horizontal SGs. This may be important in the plant, where SG tubes are slightly inclined.	Efficient heat removal process at low steam velocities.
4 (B2.4)	Asymmetric loop behaviour	Different heat removal capacity of loops, under similar boundary conditions, caused by asymmetric mass flow and distribution. Possible reasons for this behaviour are: clearing of the loop seals only in one of the loops, which reduces the possibility for other loop seals to clear, or presence of non-condensable gases in one of the loop.	Will reduce the overall heat removal capacity because some of the SGs may not take part in the heat removal process.
5 (B2.5)	Break flow	Single and two phase critical flow at the break.	Controls the availability of coolant mass fore core cooling.
6 (B2.6)	Phase separation without mixture level formation	Gravity separation at low steam velocities. It can be in reactor vessel in intermediate break LOCA at the end of blowdown phase or in various components	Separation in the core controls the cladding temperature. Separation in the downcomer

[3] numbers in the () indicate section of the "CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients" [NEA/CSNI/R(1996)17] providing description of the phenomena; numbers in the [] brackets indicate sections in "Separate Effects test Matrix for Thermal-Hydraulic Code Validation" [NEA/CSNI/R(1993)14] report description the phenomena

[4] Last revision October 8, 1997

No.)[3, 4]	Phenomena	Description	Safety significance
		including horizontal pipes during small break LOCA. It may have an important effect in the hot leg loop seals of VVER-440s and in the primary side SG collectors, where it leads to upper tubes being filled with steam, lower ones with water.	controls the driving force (static head) for the core flow. Separation within other primary components influences coolant distribution and break flow during the SB LOCA.
7 (B2.7)	Mixture level and entrainment in SG (SS +PS)	Mixture level in secondary side may decrease below top of the bundle. In this case, in horizontal SG complete rows of tubes are uncovered (not possible in vertical SGs). On primary side , two phase mixture and level in the vertical collectors can lead to different behaviour of tube layers (different quality at inlet to individual tubes or layers of tubes).	In some phases of SB LOCA, the mixture level will determine the heat transfer through the steam generator tubes and therefore decay heat removal from the core.
8 (B2.8)	Mixture level and entrainment in core	Due to sufficiently high steam velocities water can be entrained in form of droplets from the core region. Also, two phase mixture can be present in the core. Both, the entrainment and the mixture affect significantly core cooling. In VVERs these phenomena, despite being very similar to those in PWRs, are very VVER specific because of different grid spacers, shroud in the VVER 440.	Entrainment provides for effective cladding cooling, affects core level and also affects concentration of boron in the core.
9 (B2.9)	Stratification in horizontal pipes	As for 2-phase flow, it is covered under item 6. Stratification also may occur in single-phase conditions when flows of different temperatures (densities) enter the pipe. This phenomenon is similar to that in the PWR pipes	Covered under item 6 for 2-phase flow. At 1-phase conditions may lead to excessive thermal stresses on the pressure boundary
10 (B2.11)	ECC mixing and condensation	a. As the ECC water is injected it mixes with the fluid and steam in the upper plenum, downcomer and loops causing rapid condensation of the steam on the cold liquid. The VVER injection sites are different than those in PWRs therefore coarse of the process will be different. b. ECC mixing in 1-phase conditions arises at ECC injection into cold leg or downcomer. It mitigates temperature gradients in coolant. The condensation process may result in flow oscillation.	a. This process has an effect on core cooling because it affects the fluid flow through the core and ECC water delivery. b. 1-phase mixing mitigates temperature and boron concentration gradients
11 (B2.12)	Loop seal clearance (CL)	At cold and hot legs voided, liquid present in the loop seal piping between the steam generator outlet and pump inlet, which bottom is well below the elevation of the top of the core, can form an effective plug steam flow. To clear the loop seal a pressure differential across the loop seal greater than the hydrostatic head of the loop seal must be created. This pressure differential **may** cause core uncovery. Same issue as in PWRs.	Loop seal may block steam flow and therefore temporarily reduce the heat removal capacity. Core uncovery may occur.
12 (B2.13)	Pool formation in UP/CCFL (UCSP)	In VVER ECC water is being injected into the upper plenum. This water may accumulate on the upper core support plate forming a pool. Draining of this pool into the core depends on the vapour velocity which may be high enough preventing draining. See also item 10.	Pool formation and CCFL at the UCSP has direct effects on core coolant inventory and possibly core temperature.
13 (B2.14)	Core wide void and flow distribution	Three-dimensional flow behaviour in the core region. Less important than in large break LOCA, however the UP injection may cause some not typical for PWR flow	Affects cladding temperatures.

No.)[3, 4]	Phenomena	Description	Safety significance
		oscillations in the core.	
14 (B2.15)	Heat transfer in covered core	The heat transfer in covered core relates to pre accident conditions, initial phases of a LOCA and also to late phases. Heat transfer regimes are very similar to the PWR.	This heat transfer determines the initial conditions for core uncovery.
15 (B2.16)	Heat transfer in partially uncovered core	Mixture level is within the core. It involves single phase and two phase heat transfer. All associated phenomena are similar to those in PWRs.	Direct control over cladding temperatures.
16 (B2.17)	Heat transfer in SG primary side	Heat transfer under single and two phase conditions. In general similar to PWR however the tubes are horizontal. Relates to items 1, 2, and 3. Requires specific data for code validation.	Important mechanism for decay heat removal.
17 (B2.18)	Heat transfer in SG secondary side	Phenomena similar to PWR however with horizontal tubes. Potential for vapour blocking of tubes. Requires specific data for code validation. See also Item 7.	Important mechanism for decay heat removal.
18 (B2.19)	Pressurise thermal-hydraulics	Important for breaks in pressuriser (PORV opening). Includes mixture level development and rise. The same phenomena as in PWRs.	Affects mass loss through the break and primary coolant inventory.
19 B2.20	Surge line hydraulics	Affects the pressure control by the pressuriser during early stages of the accident. Potential for CCFL, and overall flow control while upflow during PORV discharge. Very similar to PWR behaviour, although connection of the surge line to the bottom of the hot e.g. loop seal in VVER-440s may create specific phenomena.	Affects pressure and coolant inventory.
20 (B2.21)	1- and 2-phase pump behaviour	Pump characteristics degradation during two phase flow related to phase separation on the impeller. The same process as in PWR pumps.	Important only if pumps left running during small break LOCA. It will affect coolant distribution and inventory.
21 (B2.22)	Structural heat and heat losses	Release of stored heat in the metal structures of the primary system will affect pressure in slower SB LOCAs. Also the heat losses to the environment will contribute to the pressure transient. Phenomena as in PWRs.	Has little significance to safety except affecting the overall energy balance.
22 B2.23	Non-condensable gas effects - Steam Generator	In very late phase non-condensable may affect heat transfer in general or block steam generators and or cause asymmetric system behaviour. It is very specific for VVER because in design differences between PWRs and VVERs.	Some effects on heat transfer.
23 (B2.10)	Phase separation in T-junctions and effects on break flow	Phase separation and flow partitioning at branches and T-connections. See also items 6 and 9.	Influences mass and energy loss from the system.
24	Natural Circulation core-gap-downcomer, dummy elements	Natural circulation within the reactor vessel while the core is covered. Very specific for VVER 440 and similar to PWRs for VVER 1000.	Support core cooling and contributes to energy distribution, however of little significance to safety.

25	Loop seal behaviour in HL	It may create the same problem as loop seal in cold leg (Item 11).	As Item 11.
26	Recirculation in SG primary side	At natural circulation in primary circuit with heat removal through steam generators a backward flow of coolant occurs in lower rows of SG tubes. This phenomenon decreases thermal efficiency of steam generator at decay heat removal but this efficiency remains sufficient due to large surface of heat transfer. Although the coolant temperature in these conditions increases to some degree.	This phenomenon can play some role in accidents with loss of feed water, when recirculation in SG tubes will decrease and the heat transfer is reduced leading to pressure rises on the primary side.
27	Boron mixing and transport	Mainly concerns two issues: 1. mixing of deborated slugs which may form in loops as a result of steam condensation in the SG tubes, primary-to secondary leak; 2. boron accumulation in the core region at long-term boiling conditions	Penetration of de-borated slugs into the core at circulation may lead to local criticality; excessive boron accumulation may lead to boron deposition on fuel rods and their overheating
28	Water accumulation in SG tubes	Water (condensate) accumulation in the SG tubes is possible, because of small tube inclination and bending. Influence of surface tension may play a role. Postulated phenomenon, no experimental evidence.	Influences coolant distribution in the primary circuit. The water accumulated in SGs tubes is not available for core cooling. May reduce circulation through SG and its heat removal capability.

Transients

No. [5, 6]	Phenomena	Description	Safety significance
1 (B4.1)	Natural circulation in 1-phase flow	After stop of the reactor coolant pumps, core heat is removed by natural circulation due to temperature difference between the loops. Single phase natural circulation is well understood, VVER specifics concerning code validation does not exist.	If heat removal from the secondary side is sufficient at transients, the single phase circulation is a reliable mechanism for removal the decay heat.
2 (B4.2)	Natural circulation in 2-phase flow	In some transients with reactor coolant pump out of operation coolant boils up. Steam condenses in slightly inclined U-shaped tubes providing high heat removal rate. In the vertical coolant collectors of the horizontal steam generators stratification of steam and water may occur. This is specific to the horizontal steam generators of VVERs.	Natural circulation of two phase flow during transients has itself no limiting features in the viewpoint of safety. The VVER-440 has loop seals also in hot legs, which may obstruct the two phase natural circulation.
3 (B4.3)	Core thermalhydraulics	At transients the conditions for heat transfer may be changed in a very wide range from forced or natural circulation with or without boiling to critical heat flux and post critical heat flux conditions. In different zones of the core different heat transfer modes can exist simultaneously. VVER has no specific features in comparison with PWRs, except of different fuel rod pitch and spacing grids.	Heat transfer of the core is of high safety importance, because it defines the compliance of the safety criteria.
4 (B4.4)	Thermalhydraulics on primary side of steam generator	During some transients as ATWS with loss of feedwater or station blackout primary coolant can boil up. In this case steam is condensed inside tubes. Condensation is a quite effective heat removal mechanism. It should be taken into account, that in horizontal steam generators recirculation arises between upper and lower tubes of the bundle. This effect is important at partial uncovery of the tube bundle, when heat transfer capability of the steam generator is decreased.	Thermalhydraulics on primary side of steam generators have no restricting conditions during transients except cases with lowered secondary level.
5 (B4.5)	Thermalhydraulics on secondary side of SG	During some transients, the steam generator tube bundle may be partially uncovered and dried, which leads to degradation of the steam generator efficiency. Tube bundle uncovery affects as partial loss of heat sink with primary pressure and temperature increase. That leads to actuation of pressuriser safety valves and partial loss of the primary coolant inventory. During such transients as steam line rupture, the loss of secondary side inventory reduces the overcooling in the concerned loop.	Thermalhydraulics of the steam generator secondary side is of high importance because it determines the behaviour of the parameters in the primary system and the core.

[5] numbers in the () indicate section of the "CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients" [NEA/CSNI/R(1996)17] providing description of the phenomena

[6] Last revision August 26, 1999

No. [5, 6]	Phenomena	Description	Safety significance
6 (B4.6)	Pressuriser Thermal-hydraulics	Pressuriser thermalhydraulics is important when valves on pressuriser are open with steam water mixture discharge. At fast pressure increase in the primary system there are two competing phenomena as steam compression in the pressuriser and fountain effects of subcooled water should be taken into account. The thermalhydraulic behaviour of the pressuriser in VVERs is almost the same as in PWRs	Pressuriser behaviour affects on thermal-hydraulic parameters in the primary system and the core.
7 (B4.7)	Surge line hydraulics	During transients surge line may restrict flow between pressuriser and primary circuit due to CCFL, when steam is removed through pressuriser valves. If the hydraulic resistance of the surge line is too large (e.g. at test facilities), it cannot equalise pressure between primary circuit and the pressuriser at fast transients. The thermalhydraulic behaviour of the surge line in VVERs is almost the same as in PWRs	The flow restriction at CCFL may decrease coolant inventory in the primary circuit and can affect on thermalhydraulic parameters.
8 (B4.9)	1- and 2-phase pump behaviour	see. Tab. 2 SBLOCA. There are no principal differences in the thermalhydraulic behaviour of the reactor coolant pumps in VVERs and PWRs.	During transients reactor coolant pumps operate in single phase region. When the pumps are out of operation the natural circulation is important.
9 (B4.10)	Thermalhydraulic nuclear feedback	Increase in void fraction leads to a lower moderator density and reduces the nuclear power in the core	Important safety feature. Reactivity coefficient depends on the nuclear design.
10 (B4.11)	Structural heat and heat losses	Heat release or accumulation in the structures are not negligible in the overall energy balance and somewhat define the behaviour of time dependent parameters. Heat losses have no significance in this respect but they are very important for scaled test facilities as well as stored heat	In real plant structural heat and heat losses have small effect on the evolution of the thermalhydraulic parameters.
11 (B4.12)	Boron mixing and transport	When RCPs are out of operation steep boron concentration gradients are possible. Special consideration require deborated slugs which may form in loops due to inadvertent injection from external systems. After start-up of the 1-st RCP the slug is transported quickly to the core inlet. This event introduces positive reactivity into the core and may lead to its degradation, especially in fresh core at cold shut-down. Mixing before the core is of high importance to mitigate this danger. There are no principal differences between PWRs and VVERs, but the mixing degree depends on in-vessel geometry.	Formation of steep boron concentration gradients in primary system could lead to local criticality with hazard of reactivity initiated accident.

3. Cross Reference Matrices

The three matrices (large break, small and intermediate leaks and transient) accompanying this section were based on those developed originally by the CSNI for Western PWR systems [2]. The CSNI matrices were extended by Liesch and Réocreux [4] to include aspects relevant to VVER systems which were not already covered. They were formulated as a supplement to the CSNI matrices so the original structure was retained. Subsequently the VVER matrices have been revised, as a result of a more thorough description of VVER-specific phenomena and transients presented in the preceding chapter.

This chapter first describes the structure of the VVER matrices and their use in overall terms. An explanation is given of the symbols used in filling in the matrix. In the final sections of the chapter more detailed aspects of each of the three matrices are described as a further aid to their use.

3.1 Structure of the Matrices

Cross Reference Matrices related to LOCA and Transients have been drawn up with the objective of allowing a systematic selection of tests suitable for code assessment.

Each matrix is composed of six sub-matrices related to the following items:

- phenomena covered by CSNI matrix,
- phenomena versus plant types,
- phenomena versus test types,
- test facilities versus phenomena (both system and separate effects tests),
- test types versus test facilities (only system tests),
- plant type versus test facility.

In the term ‘phenomena’ all the important thermal-hydraulic processes expected to occur during an accident are included. In the column “CSNI” it is indicated whether a phenomenon has already been evaluated by experimental investigation within the CSNI frame; “plant type” gives a ranking according to the characteristics of VVER-systems; “type of test” relates to the definition of the experiment; the meaning of “test facilities” is self evident; both system test (integral facilities) and separate effects facilities are included.

Principles of the phenomena selection have been discussed in section 2.1. Test types and test facilities were selected essentially on the basis of personal experience of the participants of the Support Group, and, more specifically, on the basis of the knowledge of the national representative to which the test facility belonged. However, criteria for selection were commonly agreed upon, such as:

- general technical suitability,
- experimental coverage of phenomena,
- adequacy of instrumentation,
- adequacy of documentation.

Since the aim of the Support Group was to review all test facilities which fulfilled the above criteria, no pre-selection was made in [4] with respect to availability of the data.. So the list of test facilities given in

Appendix D can be considered as an exhaustive one, from which tests for code validation purposes can be selected.

As already mentioned, emphasis has been laid on integral systems, but a large number of separate effect test facilities has also been included. Only experiments already executed or planned to date are considered for filling the matrices.

The symbol X, introduced for these matrices (but not employed in the CSNI matrices), shows that the new facilities, are especially suitable for the simulation of a number of the identified issues not covered in previous facilities.

3.2 Use of the Matrices

c. The correlation between phenomena and CSNI-matrices is given in three levels:

- Covered (plus, [+]);
- Partially covered (open circle, [o]);
- Not covered (dash, [-]).

d. The phenomena also are correlated with plant type in three levels:

- Fully specific to VVER-440/213 and VVER-1000 respectively (plus, [+]);
- Partially specific (open circle, [o]);
- Not specific (dash, [-]).

e. The correlation between phenomena and test type is given in three levels:

- Occurring, which means that the particular phenomenon occurs in this kind of test (plus, [+]);
- Partially, occurring: only some aspects of the phenomenon are occurring (open circle [o]);
- Not in the list (dash, [-]).

f. The correlation between phenomena and test facilities is given in four levels:

- Suitable for code assessment, which means that a facility is designed in such a way to simulate the phenomenon assumed to occur in the plant and it is sufficiently instrumented to reveal it (plus [+]);
- Limited suitability: the same as above with problems due to imperfect scaling or insufficient instrumentation (open circle, [o]);
- Not suitable: obvious meaning, taking into account the two previous items (dash, [-]);
- Expected to be suitable: definition introduced in some cases to emphasise that new facility particularly addresses the simulation of this aspect; clearly a conclusive comment cannot be made at present (cross [x]).

g. The correlation between test facilities and type of tests is given in three levels:

- Already performed: the test type is useful for code assessment purposes (plus[+]);
- Performed but of limited use: this kind of test has been performed in the facility, but has limited usefulness for code assessment purposes, due to poor scaling or to lack of instrumentation (open circle, [o]);

- Not performed (dash, [-]).

h. The correlation between plant type and test facilities is given in three levels:

- Covered by (plus, [+]);
- Partially covered (open circle,[o]);
- Not covered (dash,[-]).

All spaces which have been left blank correspond to cases where experimental evidence is missing.

3.3 Matrix of Large Break in VVERs

The Large Break Matrix is given in Matrix I. Sixteen phenomena, two plant types, three test types, four system test facilities and twelve separate effects test facilities are included. The phenomena “2-phase-flow in SG primary and secondary side” has been added due to the importance in VVER systems.
According to the importance the phenomenon ECC bypass and penetration has been split into two phenomena, namely

- ECC downcomer bypass and penetration,
- UP injection and penetration.

Widely used nomenclature has been adopted in the identification of phenomena and test types. The explanation of the reasons for all the choices would be too long; nevertheless, for a better understanding of the matrices, it may be useful to describe one or two lines of each of the three main matrices:

- Test type,
- Test facilities and
- Test facility system test.

Test type
For VVERs de-entrainment from upper plenum to core is fully specific to a VVER-440/213 system and partially specific to VVER-1000 system. It is assumed to be of high importance for reflood tests, of limited interest in refill tests and long term cooling, and of no interest in blow-down tests.

Test facility
The de-entrainment phenomenon is suitable for code assessment when it is observed in the SB facility. When the data become available from the PSB-VVER and ISB-VVER facilities, they area also expected to be suitable for code assessment. There is limited assessment potential in the data from PM-5.

Test facility system tests
Blowdown, refill and reflood types of tests have been performed in the SB facility.

Matrix 1: Cross Reference Matrix for Large Breaks in VVERs

- **CSNI**
  + covered by
  o partially covered
  - not covered

- **Phenomenon vs plant type**
  + fully specific to WWER
  o partially specific
  - not specific

- **Phenomenon vs test type**
  + occuring
  o partially occuring
  - not in list

- **Test facility vs phenomenon**
  + suitable for code assessment
  o limited suitability
  - not suitable
  x expected to be suitable

- **Test type vs test facility**
  + already performed
  o performed but of limited use
  - not performed

- **Plant type vs test facility**
  + covered by
  o partially covered
  - not covered

		Plant type		Test type			TEST FACILITY *1															
							System Tests				Separate Effects Tests											
	CSNI	WWER-440/213	WWER-1000	Blowdown	Reflood	Refill	PSB-WWER	PM-5	SB	ISB-WWER	Data bank (EREC)	LWL	REWET-II	IVO-CCFL	SKN	SVD-1	SVD-2	TVC-440	EVTUS	KS	TOPAZ	SG-NPP
Phenomena																						
Break flow	+	-	-	+	+	+	x		+	x	+											
Phase separation	o	+	o	o	+	+	x			x				o								
Mixing and condensation during injection	o	+	+	o	+	+	x			x												
2--phase flow in SG primary and secondary side	-	o	o	o	o	o	x			x												
Core wide void + flow distribution	o	+	+	o	+	+	x	o		-												
ECC downcomer bypass and penetration	o	+	+	o	o	+	-			-												
UP injection and penetration	-	+	+	0	+	+	x		+	x			+				+	+				
CCFL (UCSP)	o	+	+	o	+	+	x	+		x				+								
Steam binding (liquid carry over, ect.)	o	o	o	-	+	o	x		+	x												
Pool formation in UP	o	+	+	-	+	+	x	+		x												
Core heat transfer incl. DNB, dryout, RNB	o	+	+	+	+	+	x	o	+	x		+			+	+	+	o	o	+		
Quench front propagation	o	+	+	o	+	o	x		+	x		+	+			+	x	+	o		+	
Entrainment (Core, UP)	-	+	o	o	+	o	x	o	+	x			+				x	o				
Deentrainment (Core, UP)	-	+	o	-	+	o	x	o	+	x			+				x	o				
1 - and 2-phase pump behaviour	-	+	+	+	0	0	x			-												
Noncondensible gas effects	-	o	o	-	+	+	x			x												x

Test Faciliy System Test	CSNI	WWER-440/213	WWER-1000	Blowdown	Reflood	Refill
ISB-WWER		o	+	-	-	-
PSB-WWER			+	-	-	-
SB		+	+	+	+	+
PM-5			+		+	

important test parameter
- pumps off/pumps on
- ECC injection mode
- ACC-pressure

*1 refer to description of test facilities - leak location/leak size

3.4 Matrix of Small and Intermediate Leaks in VVERs

The Small and Intermediate Leaks Matrix is given in Matrix II. Twenty eight phenomena, eight test types, six system test facilities and eleven separate effects test facilities are included.
Four VVER specific phenomena have been added:

- natural circulation core-gap-downcomer,
- loop seal behaviour in the hot leg,
- recirculation in the SG primary side,
- water accumulation in the SG tubes.

The same observations as in section 3.3 apply here. Also it should be noted that, among the phenomena, the ‘structural heat and heat losses’ has been considered in order to emphasise the noticeable distortions introduced by the heat release from structures in scaled-down facilities with respect to the plant behaviour. This is due to the large structural mass and structure-to-fluid heat exchange areas relative to the full-scale values.
A relatively large number of test types has been considered to emphasise that a number of different transients are possible under the general category of Small Break LOCA.

Matrix II: Cross Reference Matrix for Small and Intermediate Leaks

- **CSNI**
  - + covered by
  - o partially covered
  - - not covered
- **Phenomenon vs plant type**
  - + fully specific to WWER
  - o partially specific
  - - not specific
- **Phenomenon vs test type**
  - + occuring
  - o partially occuring
  - - not occuring
- **Test facility vs phenomenon**
  - + suitable for code assessment
  - o limited suitability
  - - not suitable
  - x expected to be suitable
- **Test type vs test facility**
  - + already performed
  - o performed but of limited use
  - - not performed
- **Plant type vs test facility**
  - + covered by
  - o partially covered

			Plant Type		Test Type								TEST FACILITY *1 System Tests						TEST FACILITY *1 Separate Effects Tests											
		CSNI	WWER-440/213	WWER-1000	Stationary Test addressing energy transport on primary	Stationary Test addressing energy transport on secondary side	Small leak overfeed by HPIS, secondary side necessary	Small leak without HPIS overfeeding, secondary side	Intermediate leak, secondary side not necessary	Pressurizer leak	SG-tube rupture	SG-header rupture	PACTEL	PMK-2	PSB-WWER	PM-5	SB	ISB-WWER	Data bank (EREC)	IVO-Loop Seal *2	IVO-CCFL *2	HORUS-II	Thermal Mixing	SVD-1	SVD-2	EVTUS	KS-1	SG-NPP-NV AEP	IF-NC
Phenomena	1-pha	+	-	-	+	+	+	o	-	+	+	+	+	+	+	o		+		-	-								+
	Natural circulation in 2-phase flow, primary side	o	o+	o+	+	-	o	+	+	o	-	+	+	+	+	o		+		-	-								+
	Reflux condenser mode and CCFL	o	-	+	+	-	-	+	+	-	-	+	-	-	-			-										x	
	Asymmetric loop behaviour	-*1	+	o	+	-	+	+	-	o	+	-	+	-	o			-		-	-								
	Leak flow	+	-	-	-	-	+	+	+	+	+	+	o	o	o			+	+										
	Phase separation without mixture level formation	o	+	o	+	-	o	+	+	+	o	+	o	o	x			o											
	Mixture level and entraiment in SG (SS+PS)	-	+	+	-	+	+	+	+	+	+	+	o	o	x	o		o		-	-	-							
	Mixture level and entraiment in the core	-	+	o	+	-	-	+	+	+	-	o	o	o	x			o		-	-			x					
	Stratification in horizontal pipes	+	-	-	+	-	-	+	+	-	-	o	o	o	x			-		o	-							x	
	ECC-mixing and condensation	o	+	+	-	-	o	+	+	+	+	+	o	o	o			o		-	-		+						
	Loop seal clearance (CL)	+	-	-	-	-	-	+	+	o	-	-	o	o	+	+	+	+		+	-								
	Pool formation in UP/CCFL (UCSP)	o	+	+	+	-	-	o	+	+	-	o	o	-	-	+		-		-	+			x					
	Core wide void and flow distribution	o	+	o	+	-	-	o	+	+	-	o	o	o	x	o		-		-	-				x				
	Heat transfer in covered core	+	o	-	+	+	+	+	+	+	+	+	+	+	x	+	+	+		-	-			x	x	o	+		
	Heat transfer in partly uncovered core	+	o	-	+	-	-	o	+	-	-	o	o	o	x	+	+	+		-	-			x	x		+		
	Heat transfer in SG primary side	-	+	+	+	o	o	+	+	o	o	+	o	o	x			o	o	-	-	+						o	
	Heat transfer in SG secondary side	-	+	+	o	+	+	+	+	+	+	+	o	o	o			-		-	-	-						+	
	Pressurizer thermohydraulics	-	+	+	o	-	o	o	+	+	+	o	o	o	x			o	0										
	Surgeline hydraulics	o	o	o	o	-	-	o	+	+	o	o	o	o	x			o		-	-								
	1- and 2-phase pump behaviour	-	+	+	-	-	-	o	+	-	-	o	x	-	x			-		-	-								
	Structural heat and heat losses	+	-	-	+	-	o	+	+	o	o	+	o	o	x			o		-	-								
	Noncondensible gas effects	o*3	+	+	+	-	-	-	-	-	-	-	x	+	x	x		x		-	-	x						x	
	Phase separation in T-junct. and effect on leakflow	+*3	-	-	-	-	-	+	+	-	-	o	-	-	-			-	0	-	-								
	Nat. circul., core-gap-downcomer, dummy elem.	-	+	+	o	+	+	-	-	-	+	o	-	-	-		+	-											
	Loop seal behaviour in HL	-	+	-	+	-	-	+	+	+	-	o	+	+	-		+	-											
	Recirculation in the SG primary side	-	+	+	+	-	-	+	o	o	-	-	+	x	x			x				-						x	
	Boron mixing and transport	-	+	+	-	-	-	+	+	+	o	o			o														
	Water accumulation in SG tubes	-	+	+	+	-	-	+	+	+	-	-	o	o	-	-	-	-				+						x	
Test Facility System Tests	WWER 1 : 1										+																		
	PMK-2		+		+	-	-	+	+	o	-	+																	
	REWET-III		+		+	+	+	-	-	-	o	-																	
	PACTEL		+		+	+	o	+	-	-	-	-																	
	ISB-WWER		o	+	-	-	-	o	-	-	-	-																	
	PSB-WWER			+	-	-	-	-	-	-	-	-																	
	SB		+	+			o	o	+																				
	PM-5			+	o		o																						

*1 refer todescription of test facilities

*2 included in the CSNI

*3 will be provided in the revised IT-matrix

3.5 Matrix for Transients in VVERs

The Transients Matrix is given in Matrix III. Eleven phenomena, nine test types, seven system test facilities and five separate effects test facilities are included.
Two phenomena have been removed:

- Valve leak flow,

- Separator behaviour.

The following observations can be added to those in the preceding two sections:

- Almost all the phenomena discussed in the SL and IL Matrix II are important for transient analysis but for brevity they are not listed in this table;

- Thermal-hydraulic nuclear feedback is the phenomenon which characterises some of the transients. For this reason, VVER-NPP was judged to be more suitable than the other experimental loops in operation;

- The experience gained from accidents occurring in real plants is potentially of great importance, as is that from the analysis of data from start up, shut down, and other operations.

Matrix III: Cross Reference Matrix for Transients

- CSNI
- + covered by
- o partially covered
- - not covered

- Phenomenon vs plant type
- + fully specific to WWER
- o partially specific
- - not specific

- Phenomenon vs test typ
- + occuring
- o partially occuring
- - not occuring

- Test facility vs phenomenon
- + suitable for code assessment
- o limited suitability
- - not suitable
- x expected to be suitable

- Test type vs test facility
- + already performed
- o performed but of limited use
- - not performed

- Plant type vs test facility
- + covered by
- o partially covered
- - not covered

			Plant Type		Test Type									TEST FACILITY *6 System Tests							TEST FACILITY *6 Separate Effects Tests				
		CSNI	WWER-440/213	WWER-1000	ATWS	Loss of feedwater, non	Loss of heat sink, non ATWS	Station blackout	Steam line break	Feed line break	Cooldown prim. feed and bleed	Reactivity disturbance	Over-cooling	WWER 1:1 *1	PACTEL	PMK-2	PSB-WWER	PM-5	ISB-WWER	BD	Data bank (EREC)	VEERA	REWET II	SVD-2	Mixing Model
Phenomena	Natural circulation in 1-phase flow	+	-	-	+	+	+	+	+	+	+	o	o	+	+	+	+	o	+			-	-		
	Natural circulation in 2-phase flow	o	o+	o+	+	+	+	+	-	-	o	o	-		+	+	+	o	+			-	-		
	Core thermohydraulics	o	+	o	+	+	+	+	o	o	o	+	o	+	+	+	x	o	+			+	+		
	Thermohydraulics on primary side of SG	-	+	+	+	o	o	+	o	o	+	o	+	+	o	o	x		o			-	-	+	
	Thermohydraulics on secondary side of SG	-	+	+	+	+	+	+	+	+	+	o	+	+	o	o	x		-			-	-		
	Pressurizer thermohydraulics *2	-	+	+	+	+	+	+	o	o	o	o	+	o	o	o	o		o		o	-	-		
	Surgeline hydraulics (CCFL, chocking) *2	o	o	o	+	+	+	+	o	o	o	o	o	o	o	o	x		o			-	-		
	1- and 2-phase pump behaviour	-	+	+	+	+	+	+	o	o	o	o	o		x	-	x		-			-	-		
	Thermohydraulic-nuclear feedback	o	+	+	+	-	-	-	+	-	-	+	- +	+	-	-	-		-			-	-		
	Structural heat and heat losses *3	+	-	-	o	o	o	o	o	o	o	o	o		o	o	o		+			-	-		
	Boron mixing and transport	o*5	+	+	-	-	-	-	o	-	-	-	o		x	-	x	-	-	+		+	+		+

		CSNI	WWER-440/213	WWER-1000	ATWS	Loss of feedwater, non	Loss of heat sink, non ATWS	Station blackout	Steam line break	Feed line break	Cooldown prim. feed and bleed	Reactivity disturbance	Over-cooling
Test Facility System Tests	WWER 1 : 1		+	+									
	PMK-2		+		-	o	-	o	-	-	-	-	-
	PACTEL		+		-	o	-	o	-	-	-	-	o
	ISB-WWER		o	+	-	-	-	-	-	-	-	-	-
	PSB-WWER			+	-	-	-	-	-	-	-	-	-
	PM-5			+									
	BD		+	+									

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*1 volumetric scaling

*2 for phenomena requiring separate effects test, e.g. pressurizer behaviour, refer to small leak cross reference matrix

*3 problem for scaled test facillities

*4 included in the CSNI SET matrix

*5 refer to descriptions of test facilities

4. Experimental Database

In Chapter 3 it has been mentioned that the test facilities listed in Appendix D were selected irrespectively of the fact, whether the facility owners are ready to supply test data to a data bank or not. The Support Group had the mandate to start with the physical collection of the data. Knowing the difficulties the NEA Data Bank is experiencing [22], when trying to collect the data included in the ITF and SETF CCVMs, it was decided to define a first set of test data, which

- were performed on well-known test facilities,
- cover a wide range of VVER-specific phenomena,
- are of high quality,
- were promised to be supplied to the data bank.

Detailed criteria for the selection of facilities and tests have been defined. Although the proposed steps of selection could not be fully followed, the selection processes and the resulting facilities and tests are described.

4.1 Criteria for Facility and Test selection

Criteria for facility and test selection were identified, including guidelines to qualify both facilities and tests. It is important to note that there is a close interconnection between facilities and tests. Therefore, the qualification of facilities and tests performed in that facility should be made parallel.

For the selection of tests more suitable for computer code assessment the following requirements should be considered: representativity of the test with regard to the reactor conditions including the range of parameters; quality of the data measured with adequate instrumentation and with acceptable uncertainties; quality and completeness of test documentation; scaling considerations and boundary conditions.

4.1.1 Facility and Test Qualification Matrix

An important tool for the facility and test selection is a matrix given in Table 4.1. The facility and test qualification matrix (FTQM) consists of three main parts addressing the overall code qualification process and, specifically the conditions leading to the selection of a suitable experiment.
The three parts are as follows:

- Representativity of the facility and test, i.e. the capability of a data base to represent phenomena or situations expected in a plant
- Quality of the data
- Challenge to the code.

The above three items may be addressed separately to the experiment, to the facility or to both. The main purposes of the FTQM are to make clear the reasons for selecting a test and a test facility.

Table 4.1 Facility and test qualification matrix

MAIN FIELDS OF QUALIFICATION	TERMS OF THE MATRIX				Qualification of	
					Facility	Test
		*	x			
				Design (including scaling)		
	Quality of			Construction		
	Facility	F		Operation		
				Use in International Framework		
			1	Personnel Qualification		
REPRESENTATIVITY				Test Design Including Consistency with Plant Phenomenon		
	Scaling	T	4	Parameter Range		
(Facility and Test)				Counterpart Test or Similar Test in ITF		
				Availability of SETF Test		
				Pump Characteristics		
	Boundary			Heat Losses		
	Conditions •	T	5	Pressure Distribution		
				Valve Operations		
				Fuel Pin Simulation		
				Facility		
	Description	T/F	2	Instrumentation		
				Data Acquisition System		
QUALITY OF DATA				Boundary Condition		
(Data Base)		T	2	Time Trends		
	Data processing			Archiving		
				Evaluation of Data		
	Documentation	T	3	Quantification of Errors		
				Representation of Physical Phenomena		
CHALLENGE		T/F	6	Numerics		
(Bridge to Code)	Modelling			Nodalisation		
				Other (e.g. sensitivity to user effects)		

+ Fully Satisfied - Not Satisfied * T: Relates to Test x Numbers refer to criteria from mandate
o Partially Satisfied / Not Applicable F: Relates to Facility • Examples only

4.1.2 Terms in the Matrix

4.1.2.1 Quality of the Facility

The evaluation of a facility is based on five items as detailed below:
Design:
A suitable set of thermalhydraulics scaling laws should be used to derive design values.

In the case of ITFs, two main approaches have been followed in the reality. One essentially aims at preserving the time scale of phenomena: full height of main components is necessary together with full pressure, temperature and the use of the H_2O as working fluid; power to mass and power to volume ratios are in principle preserved. The other allows for the use of reduced pressure (Ishii approach) largely reducing the cost of construction and of operation of the facilities. The first approach is mostly widely diffused and it is recommended if priorities have to be decided.

In the case of SETF, a much wider range of possibilities is available for the scaling laws and the design criteria, specifically in those cases where only steady state data constitute the objective of the research. In some situations, parts of a nuclear reactor may directly constitute a SETF. The geometric dimensions of the facility (as close as possible to the concerned reactor component), the uncertainties in boundary conditions (see also below) coming from design choices or compromises and the "vicinity" of the operating conditions to those of the reference reactor situations must be considered in the selection process.

Construction:
A well designed facility can be badly constructed. For instance, a horizontal pipe in the drawings might be substantially inclined in the reality; thermal insulation might not be as designed; the use of carbon steel might introduce corrosion, thus changing some characteristics in the long term operation, etc.

Operation:
A suitably scaled and constructed facility could be badly operated. Written procedures should be available for instrumentation calibration, facility start-up before any test, maintenance of the facility itself including the Data Acquisition System, etc. Essentially, the repeatability of any test should be assured.

Use in International Framework:
A very wide experience is now available at several research centers all over the world. A very large number of experiments have been performed; specifically, the number of failed attempts and of lessons learned is large, too.
Under this item it must be checked whether original experience was gained from the concerned facility, whether comparison of data has been made with similar facilities in other countries, and whether independent groups of researchers (with respect to the facility owners) have already used the data.

Personnel Qualification:
The best conditions in items described above, are not sufficient to ensure data quality if the personnel qualification is not adequate. Competencies and scientific background of facility manager(s) should be evaluated here.

4.1.2.2 Scaling of the Test

Again, the availability of an excellent facility does not ensure suitability of data in the present context, if tests are not properly designed, i.e. test boundary and initial conditions properly scaled. Four items have been distinguished to this aim.

Test design, including Consistency with Plant Phenomenon:
Scaling laws must be used leading to criteria that are consistent with design criteria; the discussion under item 4.1.1 "Design" is applicable here.

In addition, an activity may be important aiming at the comparison between plant and facilities expected behaviours during the phase of test planning. In this case, the same code to be eventually qualified on the basis of the concerned data, can be adopted: if a feedback from such calculation to the test design may reveal necessary, this must be carefully considered.

Parameter ranges:
The range of variations of the considered phenomena must be as wide as possible and close to what expected for the plant. For instance, if the dryout (or Critical Heat Flux) phenomenon is under investigation, a test where rod surface temperature varies between 500 and 1200 K may be more valuable than a test where the same quantity ranges between 500 and 700 K, though in both cases the same phenomenon is experienced.

Counterpart Test or Similar Test in ITF:
With main reference to ITF, an effort should be made checking whether the considered experiment has been performed in other ITF. A recent OECD report, ref. [2], deals with a wide list of ITF experiments including Similar and Counterpart Tests already available.

Availability of SETF Test:
In general, the same effort mentioned under item 3 for ITF, should be repeated for SETF. In addition, it may happen that phenomena expected to be relevant during an assigned test, cannot be simulated in the concerned ITF owing to scaling limitation; in this case it may be important to address the code qualification against a SETF experiment specifically concerned with that phenomenon. An example of this is constituted by the CCFL (Counter-Current Flow Limiting) at the core upper tie plate whose simulation is usually distorted in ITF (1-D behaviour of the upper plenum); in this case, properly designed SETF may exist and should be considered under this item.

4.1.2.3 Boundary Conditions of a Test

From the practical experience of the SETF and the ITF operation, it easily derives that many boundary and initial condition values cannot be controlled by scaling laws or criteria. In this case compromises are needed and specific resources should be used to reduce the impact of these upon the obtained data base. Only a few examples, [2], are considered hereafter.

Pump Characteristics:
Homologous curves in scaled pumps are generally different from prototype homologous curves. The experimentalists should be aware of this. Countermeasures can be taken in simple situations. This item consists in evaluating whether the mentioned scaling distortion may introduce large discrepancies in the simulation and whether proper countermeasures are undertaken.

Heat Losses:
Power lost to the environment may introduce large distortions specifically in very small scale facilities designed following the "time preserving" scaling laws. This is specifically true in long lasting transients. Heat losses spatial and time distributions may also play an important role. Again, countermeasures can be and should be undertaken. This item applies to both SETF and ITF.

Pressure Distribution:
Pressure distribution along the primary and secondary loops of ITF, may not be the same as in the reference plant. A detailed comparison between these two quantities (i.e. values in the model and in the prototype) is necessary and should be available to judge the realism of the considered experiment.

Valve Operation:
The same comment for pumps applies here. In this case, the valve pressure drop versus flowrate, the

critical flowrates and the quantities affecting the opening/closure cycles timing, are the concerned parameters.

Fuel Pin Simulation:
Electrically heated pins, as well known, may behave very differently from nuclear pins, specifically in reflood situations. Software material is available to control the electrical power reducing the impact of this unavoidable scaling distortion (expensive "gap fuel rods electric simulators" are also available but are not widely diffused). A check should be made in relation to the use of such software.

4.1.2.4 Quality of Data: The Database

Description of Facility:
All of the components of the facility must be described. Detailed drawings of all the components including dimensions should be provided; preferably such drawings are made to scale. Materials utilized for each component should be identified. Information on how each component was manufactured or assembled, should be included. Quantities of obvious interest to code validation such as volumes, flow areas, pressure drops, pump performance, etc. should be determined and provided. Information and drawings describing in detail how and where instrumentation is mounted, must be provided. The procedure followed in preparing for and carrying out the test must be described in detail. The sequence and chronology of test operations and control should be included for both the desired and actually achieved actions.

Description of Instrumentation:
A list of all instrumentation should be provided that includes the assigned sensor designation, type of physical variable measured, type of sensor, range or limit of measurement, general location in the facility, and precise location coordinates. The measurement errors must be quantified for each sensor. For each sensor, information should be provided identifying the manufacturer, design or configuration and working principle of the instrument if it is not a common type of sensor, as well as a detailed description and drawings of how it is installed or mounted in the facility.

Description of Data Acquisition System:
The description should include data acquisition hardware and the sampling rate for the various sets of channels. A list of all data channels should be provided identifying the assigned data channel as well as the input sensor/signal or calculated quantity derived from two or more sensor measurements. Errors associated with the data acquisition system should be quantified.

Description of Boundary Conditions:
All of the desired boundary conditions and the approaches followed in attempting to achieve or approximate them must be described. For some types of boundary conditions such as heat sources, this includes detailed information about the design/configuration of the relevant portion of the facility (e.g., fuel rod simulators) that might bear upon the conditions actually attained during the test. The degree to which the desired boundary conditions were achieved or not achieved should be discussed.

Data Processing:Time Trends:
The procedure used to convert raw data into the presented test results must be described. Information on how sensor output voltages were converted into physical variables must be provided. Any data filtering, smoothing, fitting, etc. carried out must be discussed in detail. The procedures employed should not introduce significant uncertainties into the results or cause the results to be misleading.

Data Processing: Archiving:
Raw data should be permanently maintained such a way that they remain available for a possible future reprocessing. Object of consideration is whether the archival medium or format is likely to become obsolete the way that data will cease to be accessible in a practical sense.

Documentation: Evaluation of Data:
The presentation and discussion of experiment results must include documentation of how the data were evaluated to arrive at those results. This involves not only conversion of sensor output voltages to physical

variables but also documentation of the basis and application of any models used to infer the values of variables that could not be directly measured or were not directly measured. Ambiguities in the results that arise due to the lack of additional instrumentation not incorporated in the test or significant uncertainties in measurement and evaluation should be identified.

Documentation: Quantification of Errors:
Following the test, a quantification of the overall error in each result presented should be performed and documented in detail. The evaluation should not be limited to more sensor and data acquisition statistical errors but also include errors associated with modelling of the response of the sensor in its mounting environment as well as errors associated with modelling relationships between any inferred variables and those subject to direct measurement.

4.1.2.5 Challenge to Codes: The Bridge to the Code

Where experiments have already been performed, they will be judged in terms of their usefulness in validating codes for the aspects considered below. In designing new experiments priority should be given, where possible, to widening the data base to those aspects not already covered. In particular, experiments should be chosen to test areas of the codes that are thought to be weak. Such areas could include CHF and rewet phenomena, low pressure transients, condensation (with and without non-condensable gas) stratification and natural convection. The qualification of the facility and test under consideration will be considered under the following headings:

Representation of Physical Phenomena:
For a facility to be rated **fully satisfied** physical phenomena should be represented well with a minimum of interference from other phenomena. Where a particular application is to be tested, the chosen facility should be properly scaled for the application and it should have the appropriate geometry. In a test the conditions must be representative.

Numerics:
The numerics of a code must be robust. Rapid changes in parameters, mass inventory, fluid constituents and rig components will all challenge the robustness of codes. Also in changing conditions the calculation of fluid properties can challenge the numerical stability of a code.

Nodalisation:
Although all phenomena are 3-dimensional in nature, in many instances they can be satisfactorily represented in fewer dimensions. The challenge to the code will be judged on how well the models included in a code will represent the full multi-dimensional nature of the principal phenomena involved. A further aspect of nodalisation concerns the selection of node size and positions as represented in code input decks. It may be difficult to choose appropriate node sizes because of numerical restrictions for some calculation schemes.

Others:
This is a general heading and it is hoped to develop a more specific category as experience in compiling the matrix grows. The aspects in mind at present are the sensitivity to user effects in code modelling, the sensitivity to the initial conditions for the calculation and the calculation time.

4.2 Selection of Facilities and Tests

From the facilities and tests listed in Appendix D a selection had to be made in order to provide data to the first round of qualification activities. The basic selection principles are the ones listed in the introduction to chapter 4. An important aspect was that data should come from a maximum number of countries. The present selection covers facilities and tests from Russia, Czech Republic, Finland and Hungary. Most of the VVER-specific experiments were performed in these countries. These experiments cover a wide range of important fields in thermalhydraulics as follows: SB and LB LOCA, DNB and dryout, natural circulation, loop seal behaviour and others.

Altogether 10 test facilities and 14 tests were selected for the validation matrix to cover the most important phenomena, these are listed in Table 4.2. The selection of these tests in no way implies that other test data included in Appendix D would be of inferior quality.

In the first step of the evaluation procedure, owners of the facilities and tests were asked to make an evaluation using the qualification matrix given in section 4.1.

The owners' evaluation is presented in Table 4.3.

Facility	Test	Brief Description
SB, EDO, Russia	SB/1	100% Double break in cold leg
SB, EDO, Russia	SB/2	1% Cold leg break
BD, EDO, Russia	BD-1	Boron dilution
SVD-2, IPPE, Russia	2	Dryout at low pressure
KS, RRC-KI, Russia	KS/19R/TF84	DNB, dryout
KS, RRC-KI, Russia	KS-1/05-91/No 34	Heat transfer in covered and partially covered core
PM-5, IPPE, Russia	6	Loop seal clearance
ISB-WWER, EREC, Russia	UPB-2.4	2.4% Upper plenum break
ISB-WWER, EREC, Russia	NC	Natural circulation
LWL, Skoda, Czech Rep.	DNB-D19	DNB in 19-rod bundle
PACTEL, VTT Energy, Finland	LOF-1	Loss of feedwater
PACTEL, VTT Energy, Finland	ITE-6	Natural circulation
REWET-II, VTT Energy, Finland	SGI/7	Reflood
PMK-2, KFKI-AEKI, Hungary	IAEA-SPE-4	CLB with secondary bleed and feed

Table 4.2: Facilities and tests selected for the VVER validation matrix

Table 4.3 Owners' facility and test evaluation

TERMS OF THE MATRIX				SB SB/1/2		BD BD-1		SVD-2 2		KS KS/19R/TF84	
				F	**T**	**F**	**T**	**F**	**T**	**F**	**T**
	*	X									
Quality of Facility			Design (including scaling)	+		+		+		o	
			Construction	+		+		+		+	
	F		Operation	o		+		+		+	
			Use in International Framework	/		/		-		-	
		1	Personnel Qualification	+	o	+		+		+	
Scaling			Test Design Including Consistency with Plant Phenomenon		o		+		o		o
	T	4	Parameter Range		+		+		o		+
			Counterpart Test or Similar Test in ITF		/		/		/		o
			Availability of SETF Test		o		-		/		o
Boundary Conditions •			Pump Characteristics		o		+		/		o
			Heat Losses		o		o		+		o
	T	5	Pressure Distribution		+		+		o		+
			Valve Operations		o		/				o
			Fuel Pin Simulation		+		/		o		o
			Facility		o		+	+		+	
Description	T/F	2	Instrumentation		o		o	+		o	
			Data Acquisition System		o		o	+		+	
			Boundary Condition		o		+		+		o
Data processing	T	2	Time Trends		+		+		o		o
			Archiving		o		+		+		+
Documentation			Evaluation of Data		o		+		+		+
			Quantification of Errors		+		+		+		o
	T	3	Quantification of Uncertainty		/		+		-		-
Modelling			Representation of Physical Phenomena						+		+
	T/F	6	Numerics						-		
			Nodalisation						-		
			Other (e.g. sensitivity to user effects)						-		

+ Fully Satisfied
o Partially Satisfied
- Not Satisfied
/ Not Applicable
* T: Relates to Test
F: Relates to Facility
x Numbers refer to criteria from mandate
• Examples only

Table 4.3 Owners' facility and test evaluation (continue)

TERMS OF THE MATRIX				KS KS-1/05-91/No. 34		PM-5, 6		ISB-VVER UPB-2.4		ISB-VVER NC		LWL DDNB-D19	
				F	**T**	**F**	**T**	**F**	**T**	**F**	**T**	F	**T**
	*	x											
Quality of Facility			Design (including scaling)	+		o		+		+		+	
			Construction	+		+		+		+		+	
	F		Operation	+		+		+		+		+	
			Use in International Framework	+		-		+		+		+	
		1	Personnel Qualification	+		+		+		+		+	
Scaling			Test Design Including Consistency with Plant Phenomenon		+		o		+		+		+
	T	4	Parameter Range		+		o		+		+		+
			Counterpart Test or Similar Test in ITF		o		/		/		/		/
			Availability of SETF Test		o		o		o		o		o
Boundary Conditions •			Pump Characteristics		o		/		/		o		/
			Heat Losses		+		o		+		+		+
	T	5	Pressure Distribution		o		o		+		+		+
			Valve Operations		o		/		o		o		o
			Fuel Pin Simulation		o		o		o		o		o
Description	T/F	2	Facility	+		+			+		+		+
			Instrumentation	+		o			+		+		+
			Data Acquisition System	o		+			+		+		+
			Boundary Condition		+		o		+		+		+
Data processing	T	2	Time Trends		o		o		+		+		+
			Archiving		+		+		+		+		+
Documentation			Evaluation of Data		+		+		+		+		+
			Quantification of Errors		+		-		+		+		+
	T	3	Quantification of Uncertainty		-		-		/		/		/
Modelling	T/F	6	Representation of Physical Phenomena		+		+		+		+		+
			Numerics		o		-		+		+		+
			Nodalisation		o		-		+		+		+
			Other (e.g. sensitivity to user effects)		o		-		o		o		o

+ Fully Satisfied   - Not Satisfied   * T: Relates to Test   x Numbers refer to criteria from mandate
o Partially Satisfied   / Not Applicable   F: Relates to Facility   • Examples only

Table 4.3 Owners' facility and test evaluation (continue)

TERMS OF THE MATRIX				**PACTEL LOF-1**		**PACTEL ITE-6**		**REWTT-II SGI/7**		**IAEA-SPE-4 PMK-2**	
				F	**T**	**F**	**T**	**F**	**T**	**F**	**T**
	*	x									
Quality of Facility	F		Design (including scaling)	o		o		+		+	
			Construction	+		+		+		o	
			Operation	+		+		+		+	
			Use in International Framework	+		+		o		+	
		1	Personnel Qualification	+		+		+		+	
Scaling	T	4	Test Design Including Consistency with Plant Phenomenon		+		+		+		o
			Parameter Range		o		o		o		+
			Counterpart Test or Similar Test in ITF		o		o		-		-
			Availability of SETF Test		/		/		/		/
Boundary Conditions •	T	5	Pump Characteristics		-		-		/		o
			Heat Losses		+		+		+		o
			Pressure Distribution		+		+		+		+
			Valve Operations		+		+		/		+
			Fuel Pin Simulation		/		/		+		o
Description	T/F	2	Facility	+		+		+		+	
			Instrumentation	+		+		+		+	+
			Data Acquisition System	o		o		+		+	+
			Boundary Condition		+		+		+		o
Data processing	T	2	Time Trends		+		+		+		+
			Archiving		+		+		o		+
Documentation	T	3	Evaluation of Data		+		+		o		+
			Quantification of Errors		+		+		+		+
			Quantification of Uncertainty		-		-		-		-
Modelling	T/F	6	Representation of Physical Phenomena		o		o				
			Numerics		+		o				
			Nodalisation		o		o				
			Other (e.g. sensitivity to user effects)		-		-				

+ Fully Satisfied - Not Satisfied * T: Relates to Test x Numbers refer to criteria from mandate
o Partially Satisfied / Not Applicable F: Relates to Facility • Examples only

4.3 Short Description of Selected Test Facilities and Tests

A short description of the facilities and tests listed in Table 4.2 is given in this chapter. More complete description of facilities is included in appendix D.

4.3.1 SB Facility

The facility was designed for investigation of the core heat removal at loss of coolant accidents. The test facility models the reactor vessel, two separate loops, the pressuriser, safety injection equipment and has some auxiliary systems. Steam generators are modelled by coolers in each loop. From the two loops, one models the loop with break, the other the intact loops. The intact loop contains main circulation pump, heater and cooler. Also the pressuriser is connected to the intact loop. The broken loop contains a break device. The reactor vessel model consist of core, lower and upper plenum and of an included downcomer. All elevations of the modelled components are kept 1:1.

The measurement system contains transducer for fast changing pressure in the reactor model, difference pressure in the core simulator and fluid temperatures in the whole circuit. Three rods of the core simulator are equipped with thermocouples at different heights for measuring the rod surface temperature. Also the flow rates through the break and the emergency injection are determined by measuring flow velocity and density.

4.3.1.1 Description of the Experiments

SB1: 2F-break in cold leg

In the SB1 test the core contained 7 rod simulators with an outer diameter of 9.10 to 9.15 mm and an inner diameter of 7.7 to 7.8 mm. The grid pitch amounts to 12.75 mm. The spacer grids have a height of 20 mm and a distance of 255 mm. The simulators have a Zr+1%Nb cladding material and He filled gaps between heater and clad. The rods have over the length a middle zone of increased power generation of 1990 mm, were the heater have a diameter of only 5.10 to 5.15 mm, whereas the diameter in the zone of normal power generation amounts to 6.20 to 6.35 mm. The electrically heated active length of the rod simulators amounts to 3450 mm. The simulators have passive ends of 730 mm. The intact loop had a length of about 45 m and a diameter 60.5 mm. The broken loop with a diameter 38.4 mm had a length of about 7 m.

Initial conditions:

pressure in the circuit	15.2 MPa
core inlet temperature	296 ^{o}C
core outlet temperature	328 ^{o}C
mass flow through the core simulator	20 m^3/h
initial power of core simulator	0.568 MW
remaining power of core simulator	0.040 MW
initial rated power in the core simulator	1.25 MW/m^2
remaining rated power	0.089 MW/ m^2
break diameter	2*0.03 m
pressure in the accumulators	5.88 MPa

The experiment is prepared by heating the circuit by the pre-heater and the pumps. In the accumulator and in the pressuriser a gas pressure is established. Reaching the operating temperature the temperature difference over the core is controlled by the mass flow. When all initial conditions are calibrated, the break is opened, the pumps are switched off and the core power is decreased to decay heat level.

SB2: Small break

The test was carried out at SB test facility in EDO “Gidropress”. The facility was reconstructed for investigation of the reactor model behaviour during Small Break accidents. A new bundle of fuel rod simulators was mounted into the reactor model. The rods were heated directly by electric current.

During this experiment the core simulator was equipped with 19 rods with a heated length of 3460 mm. The grid pitch amounts to 12.75 mm. The distance between the spacers was 255 mm, the height 20 mm. The diameter of the whole assembly amounted to 58 mm. The heat generation in the rods with a diameter of 9.1 mm was equally distributed over the rod length.

The length of the intact loop amounted to 32 m, the diameter 60.5 mm. The loop had a U-shaped loop seal with a sinking part of 4.5 m and a rising part of 2.5 m. The broken loop had a length of 16 m and a diameter of 38.4 mm. Also the broken loop was equipped with the same U-shaped loop seal. During the experiment the levels also in the loop seals were determined. Using the control valves of the broken loop, the simulation of a small break in the hot or in the cold leg was possible.

Main initial and boundary conditions:

- pressure of coolant: 14.6 MPa
- inlet temperature of the reactor model: 288°C
- outlet temperature of the reactor model: 296°C
- number of rods in the bundle: 19
- heated length of rods: 3460 mm
- power of bundle: 68.5 kW
- natural circulation in the reactor model before test
- ECCS didn´t work.

4.3.2 BD Facility

The facility was designed, to investigate boron mixing and transport phenomena in the cold leg loop seal and in the downcomer and lower plenum of reactor vessel. At special abnormal operation conditions of the nuclear reactor, in the pump seal deborated coolant may be accumulated. Turning on the main coolant pump or by natural circulation effects, the deborated coolant may enter the core and initiate a reactivity caused accident.

In the test facility coolant at different boron concentration is modelled by water at different temperatures. Reference reactor is the VVER-1000. The volume scale amounts to 1:5. A closed primary circuit is represented. The facility models downcomer and lower plenum of the reactor vessel and the pump seal of one cold leg. The other three loops are not modelled explicitly. In the real reactor facility, switching on only one from four main circulation pumps, a flow reversal in the other three cold legs occurs. In the test facility BD the flow reversal into the connections from pressure vessel to the cold legs may be simulated by operation of valves and auxiliary pumps. The core geometry is not modelled in detail.

The flow rate in the modelled loop may amount 220...1100 m^3/h, which corresponds to the scaled down value of the reference reactor of 27500 m^3/h. Possible temperature differences, which represent different boron concentrations, are 15...40 degree.

The measuring system contains 118 transducers for temperature, which are distributed over the facility as follows: 8 at the modelled cold leg, each 2 at the reactor inlet of each of the four loops, 12 in the downcomer and 90 at the core inlet. The measuring system consists of 96 channels with a sampling frequency of 10 Hz.

Preparing the experiment, the coolant is heated by running the main circulation pump. When the operation temperature is reached, the pump is switched off, the pump loop seal is switched out by valves and the coolant of the pump loop seal is replaced by cold water. The experiment is started by switching on again the pump.

4.3.2.1 Description of the Experiment

Test BD-1

In recent years, safety analyses for PWRs (VVERs) deal with a possibility of reactivity accident which is connected with penetration of slug with low boron concentration to the core inlet. The slug may be formed in reactor loops by different ways when forced or natural circulation has been stopped. When circulation is restored, the slug enters into the reactor and that is dangerous from the view point of criticality.

The most dangerous event is an inadvertent start-up of the first RCP when transient time has order of several seconds. Any corrective measures are impossible during the transient. Coolant mixing at the inlet part of reactor flow path (before core) may weaken this danger or exclude it entirely. In this part a complex three-dimensional flow arises that promotes to mixing.

Main initial and boundary conditions:

- temperature of water in reactor model: 54.5°C
- temperature of water in model of the loop seal: 25°C
- volume of the loop seal: 0.074 m^3
- volume of pipelines from the loop seal up to entrance in reactor model: 0.046 m^3
- volume of water in downcomer: 0.157 m^3
- volume of the lower part of the reactor model: 0.091 m^3

Behaviour of the boron concentration in the core entrance was determined on the relative change of temperature. Relative temperature was calculated under the formula:

$$\delta t = \frac{t_{hot} - t_i}{t_{hot} - t_{cold}} * 100\%$$

where

t_{hot} - temperature of the "hot" coolant in reactor model up to start-up of the pump, °C
t_{cold} - temperature of the "cold" coolant in the loop seal up to start-up of the pump, °C
t_i - current value of temperature of the coolant on the entrance of core model, °C.

Value of δt characterises the dilution level of boron coolant. Value δt=0 shows that reduction of the boron concentration was not observed. Value δt=1 shows that the pure distillate flows in this point.

4.3.3 SVD-2 Facility

The SVD-2 test facility is designated to carry out experimental studies on heat transfer, critical heat flux (CHF) and fluid dynamics using full height models of water cooled reactor fuel subassemblies and other analogous heat exchange devices. The primary circuit of the test facility incorporates three loops: a high pressure loop (HPL), a medium pressure loop (MPL) and a free circulation loop (FCL). The operating performance of loops is shown in Table 4.3.1. The test section is a bundle consisting of a 19 fuel rod simulator.

Parameter	Loop		
	HPL	MPL	FCL
Capacity of test section, MW Water pressure, MPa Water temperature at the inlet of test section, °C	11 25,5 450	11 19,6 350	11 6,8 240
Water temperature at the outlet of test section, °C Water flow rate, m^3/h	500 35	450 35	280 8

Table 4.3.1 The Operating Performance of SVD-2 Test Facility Loops

For the data collection and recording a so-called measurement complex is used. It consists of the necessary hardware and software to perform test and to collect test data.

4.3.3.1 Description of the Experiment

The experiments are conducted in the following range of parameters:

- water pressure at bundle: 0.2 - 3.0 MPa
- water mass velocity through channel: 50-300 kg/m^2s
- water temperature at test section inlet: 60-180 °C
- pressure throttling at test section inlet:up to 2.0 MPa.

Test section

- number of fuel rods: 19
- heated length: 3 m
- pitch: 12.75 mm
- spacer pitch: 300 mm

Measured parameters

- pressure at inlet and outlet of test section
- pressure drop in the channel
- coolant temperature at inlet and outlet of the test section
- coolant flow rate
- power given to the test section
- temperature of internal walls of heater rods.

Measurement of critical powers was taken as follows. The predetermined water pressure at the test section outlet, water flow rate through the channel and water temperature at the test section inlet were set. Further, the rod bundle power was increased in steps up to the critical value. With a power close to the critical value, the steps of power increase did not exceed 1% of the critical power. The power was decreased, when one of thermocouples measuring the heater rod wall temperature indicated a temperature rise improportional to the rod bundle power increase.

4.3.4 KS Facility

The aim of the experiments on this facility was the investigation of heat removal phenomena at partially core dryout and interruption of natural circulation. This event may happen at small break accidents, which are not completely compensated by emergency injection. The experiments were performed at decay heat level (1% of nominal power) and at a low pressure.

In the test facility a separate downcomer, the lower plenum, the reactor core equipped with electrical heated rod simulators corresponding to VVER-1000 fuel elements, the upper plenum and one loop are modelled.

The core simulator has the following characteristics:

heated length of rod simulator	3.53 m
overheated length of rod simulator	2.505 m
number of rod simulators	19
outer rod diameter	9.0 mm
pitch of the grid	12.75 mm
number of spacers in the region of overheating	11
spacer height 20 mm	
spacer distance	250 mm
diameter of the hexagonal channel	59 mm

The heat production is equally distributed over the length and radius of the rods. The downcomer has a 2.78-fold cross section compared with the core simulator. The loop is modelled as an open circuit with water in- and outlet at the lowest point of the loop.

The facility is equipped with devices for measurement of pressure (2), difference pressure (3), fluid temperatures (2) and heater rod cladding temperature measurements (24) in different heights and at different rods of the core simulator.

4.3.4.1 Description of the Experiment

DNB tests KS/19R/TF84

Objectives of the test series:

- to investigate CHF in VVER-type test bundle with large radial non-uniformity,

- to obtain data base for verification of subchannel thermal hydraulic codes.

Test bundle:

- number of fuel rods: 19

- bundle layout: hexagonal

- distance between centres of fuel rods: 12.2 mm

- heat length: 2.5 m

- number of spacer grids: 11

- radial non-uniformity factor: 1.828

- axial non-uniformity factor: 1.0

Test procedure:

Axial heat generation uniformity leads to CHF on the upper end of fuel rods. CHF was fixed with sharp rod temperature increase at smooth power growth. Number of experimental CHF points: 62.

Initial and boundary conditions:

Pressure:	6.37 - 8.91 MPa
Mass flux:	1095 - 3234 kg/m^{2}s
Inlet temperature:	84 - 245 °C
Average heat flux:	0.6455 - 1.1160 MW/m^2

Test KS-1/05-91/No 34

In the framework of the test series KS-1/05-91/ No. 34 eight experiments at different pressure were performed. The experiment aimed at the modelling of a non compensated small break with interrupted natural circulation and partly uncovered core. At core inlet/outlet temperature of 155/265 ^{o}C, the integral heat loss of the facility amount to 50 kW. The following boundary conditions were established:

core inlet temperature	230 - 260 ^{o}C
core power	10 - 11 kW
rated heat flux at rod surface	8 kW/m^2
pressure in the facility	4.6 - 3.2 MPa

The natural circulation is established by heating the water respective steam generation in the core simulator and subsequent cooling of the water in the downcomer by the downcomer heat losses. To perform the experiment, the natural circulation is interrupted by steam outlet in the highest point of the facility or by water draining in the lowest point of the loop. The investigation of heat removal from the partially uncovered core is the performed at different decreasing primary pressure and at different sinking collapsed coolant levels in core and upper plenum.

4.3.5 PM-5 Facility

The PM-5 facility of IPPE is an integral model of the VVER-1000 reactor, with one primary loop. The test section in the loop consists of 5 parallel rod bundles. Each of them comprises 14 heated rods. The operating pressure is 1 to 3 bar, the fuel rods can be heated up to 800°C. The loop can be used for measurement of counter current flow limitation, for loop seal behaviour and heat transfer in partially covered cores.

4.3.5.1 Description of the Experiment

PM-5 Test No. 6: Loop seal clearance

Experimental study of loop seal influence on the processes in primary circuit of VVER-1000 was carried out on PM-5 model.

Received experimental data refer to three qualitatively different modes:

- a stable mode, with the water plug occupying practically the whole length of seal and partial uncovering of heater elements. Here we can observe their overheating;

- an unstable mode with periodic expulsion of liquid from seal and periodic core uncovering. In this mode sometimes the periodic temperature fluctuations of heater rods are possible,

- a stable mode with small fluctuations of pressure drop and coolant level above the core. This mode is characterised practically by the complete expulsion of liquid from seal and coolant level above the upper heated level of the core.

Main initial and boundary conditions:
Range of studied parameters:
- pressure: near atmospheric,
- power of core model: 10-50 kW,
- feed water flowrate to the seal: 30-120 kg/h,
- feed water flowrate to the core: 30-120 kg/h,
- feed water temperature: ~ 30°C,
- secondary side flowrate through the steam generator model: zero,
- characteristic period of unstable modes: 0.5 - 7.0 minutes,
- nonstability region in core power (under the fixed feeding): ~ 2-5 kW.

4.3.6 ISB-WWER Facility

The ISB facility was built to investigate the transient behaviour of WWER-1000 reactors. The reference plant is the fifth unit of Zaporojskaya NPP. The volume and power scaling ratio are 1:3000. The plant is modelled by 2 loops, one intact and one broken loop. The core model consists of 19 electrically heated rods.

The tests are conducted from nominal operating parameters of the plant. The range of parameters covers the expected parameter range of the plant.

The number of measurement channels is 208. A data processing system is used for the processing of pressures, pressure drops, void fractions and collapsed levels.

4.3.6.1 Description of the Experiments

Test UPB-2.4

The test is performed in the ISB facility to provide experiment for the first Russian Standard Problem.

The experiment is characterised as follows:
A ∅ 2.4 mm leak in the outlet chamber of the reactor vessel

- starting from full power
- simulating of pumps seizure

secondary side is isolated after transient initiation

The initial conditions are as follows:

- pressure in upper plenum	16.5 MPa
- coolant inlet/outlet temperature	297/327 °C
- core power	784 kW
- flow rate in intact loop	3.32 kg/s
- flow rate in broken loop	1.1 kg/s
- collapsed level in pressuriser	7 m
- pressure in SG secondary side	6.3 MPa
- collapsed	

Test NC Natural circulation test

The experiment was particularly aimed at the investigation of phenomena which play a role during the later phases of small break loss of coolant accidents (SBLOCA) when natural circulation is the only mechanism for decay heat removal. Test scenario provides safety relevant phenomena like single- and two-phase natural circulation, boiler condenser mode, natural circulation instabilities and cold leg loop seal clearing. The investigation of boundary conditions which induce the natural circulation instabilities were of particular interest. Similar experiments were carried out in 1989 at the Hungarian PMK-facility and in 1992 at PACTEL in Finland (ISP-33). Both facilities simulate VVER-440 thermalhydraulics.

Defined amounts of coolant were drained from the lowest point of the facility at defined times. During the test, the coolant mass is reduced in steps of 4.5 kg. This represents about 5% of the coolant inventory. In the calculation the drainage itself takes a time of 50 s. After each drainage the boundary conditions are kept constantly for 1000 s. This procedure is repeated, until the cladding temperatures start to increase.

Starting from nominal conditions, with full power of the rod simulators, the power was decreased to decay heat level (100 kW) and the pumps were switched off by opening the pump bypass valves and closing the pump valves. The primary pressure was kept constant at nominal value by switching on/off the pressuriser heater. After the depletion of the pressuriser, the pressuriser was isolated to avoid pressuriser refilling. The secondary steam generator level was kept constant by controlling the feedwater mass flow. The steam generator secondary pressure was controlled by throttling the steam outlet valve. The test was stopped when the cladding temperature started to increase.

Initial conditions of the ISB-facility at nominal conditions and for the natural circulation experiment

	Nominal conditions	**Natural circulation experiment**
Primary side:		
Power of rod simulator, kW	960.0	96.0
Power of core bypass, kW	5.0	0.55
Pressure in upper plenum, Mpa	15.6	15.4
Temperature in upper plenum, °C	320	320
Temperature in downcomer, °C	280	280
Mass flow triple loop, kg/s	4.24	0.26
Mass flow single loop, kg/s	1.44	0.10
Pressuriser level, m	5.0	5.0
Secondary side:		
Pressure in steam generator, MPa	6.0	6.5
Feedwater temperature, C	220	220
Steam generator level, m	2.3	2.1

4.3.7 LWL Facility

The LARGE WATER LOOP has been built at the NUCLEAR MACHINERY PLANT, ŠKODA, Plzen Ltd. The loop is a non-active pressurised-water equipment with technological and thermal parameters corresponding to those of PWR (VVER). The possible parameters ranges are suitable for all types of pressurised water reactors.

The CHF experimental facility (a part of the Large Water Loop) has been designed for research of CHF in water flow through a bundle of electrically heated rods.

- Flow in the closed loop was driven by a main circulating pump.

- The main and bypass branches with regulating valves were connected to the pump outlet.
- Water was pre-heated in the electrically heated vessel.
- The power input was continuously controllable from 0 to 155 kW.
- Water flow rate was measured by orifice gauges.
- The test section was heated by one D.C. 3.75 MW source whose voltage could be varied between 0 and 150 V.
- The pressurizer was electrically heated (50 kW).
- The steam condensers and the steam separators were vessels used to separate the steam from the water and also for steam condensation in experiments with vapour generation.
- The main cooler was a shell-and-tube type water-water heat exchanger with an output of 4 MW.
- All components of the experimental loop coming into contact with the working medium were made of stainless steel.
- Water was treated to correspond to the requirements of the PWRs primary circuits conditions.
- The test sections were formed by 7 or 19 parallel electrically heated rods with external diameters of 9 mm. Axially and radially uniform or non-uniform heat flux distribution and water upflow were used in the tests.
- The rods were modelled by hollow tubes with direct heating of the wall.
- The rods were specially manufactured with axially varying wall thickness while maintaining a constant outside diameter to achieve non-uniform axial heat flux.
- The rods were placed in regular hexagonal geometry with a pitch of 12.5-13 mm.
- The heated length of the rods was 3500 mm.

Critical conditions were determined under constant pressure, inlet water temperature and mass flux and for quasi steady-state by gradually increasing heat input.

Testing equipment

The experiments were performed on Large Water Loop (LWL) in ŠKODA NUCLEAR MACHINERY PLANT LTD. LWL is a pressure testing equipment operating with high technological and heat parameters, making possible to perform experiments in area of light water reactor economy and safety, and functional testing of reactor components.

LWL consists of a pump, pressure vessels, tubing, auxiliary systems and many miscellaneous sensors for measuring of various parameters. Control rooms are equipped with systems for operational and special measuring and with respective check systems. All LWL parts coming into contact with working medium are made of stainless steel, which makes possible to keep a high degree of working medium cleanliness. Demineralized water is used as working medium (water of reactor quality parameters can be also used, i.e. containing respective boric acid (H_3B0_3) concentration with pH amount between values 4-9 and with minimal content of chlorides and solid particles.

Equipment description - Pressurised water circuit

The loop is assembled of circulation pump, testing channels , pressurizer , condenser and separator , electrical heater, cooler , working medium continual filtration , air cooling tower and cooling circuit pump.

Parameters of the LWL Primary circuit :

Maximum water pressure	20 MPa
Maximum water flow rate	250 m^3/hour
Maximum pump head	220 m of water
Maximum water temperature	350^{O}C
Maximum D.C. power supply	3.75 MW, 150 V
Maximum flow rate	200 m^3/hour
Maximum temperature gradient	2 ^{o}C/min
Delivery head	2.2 MPa
Volume of the water loop	5.0 m^3
Maximum A.C. power supply	155 kW

Secondary circuit

Maximal overpressure of 2.5 MPa maximal temperature of 220 ^{o}C, delivery head of 0.5 MPa, cooling output of 4.5 m^3, max. flow rate of 60 -120 m^3/hour.

Main circulation pump - consists of an asynchronous motor which drives a hermetic pump, and a semiconductor frequency converter (3l-54 Hz). The pump working point, i.e. delivered quantity, can be continuously changed in designated range using a frequency variation, which becomes the cause of impeller wheel speed change.

Pressurizer - is created by pressure vessel equipped with electrical heating with input of 50 kW. Pressurizer serves as compensator of working medium volume variations and keeps constant pressure level in the loop.

Condenser and separator - are pressure vessels. Their purpose is to separate and subsequently to condense the vapour phase content of the steam-water mixture created during experiment in testing channel.

Heater - an electrical heated pressure vessel being used as working medium heater in the loop. Heating input of 155 kW can be continuously controlled in its whole range. Cooler - is a tube heat exchanger with cooling output of 4 MW. Secondary side of an exchanger is connected with a pressurised liquid cooling circuit.

Working medium continual filtration system - makes possible a non-stop filtration of working medium in high-pressure ion-exchanger filters. The equipment is furnished with a cooler and a cooling exchanger for preheating and cooling of filtered medium.

Air cooling tower and cooling circuit pump - are used for heat removal from the main circuit of LWL. The air cooling tower cooling capacity is about 4.5 MW. One pump discharge is 80 m3/h of cooling solution. The circuit is equipped with a pressurizer.

Measuring and control system - basic measurement including the technology parameters control system correspond to fossil type power plant operation control system supplemented with a computer system (PC) data acquisition and with mentioned loop parameters control from computer keyboard.

Operational sensors and loop technology instrumentation are simultaneously used for back-up parameters measuring. J-type thermocouples o.d. 2 mm with an insulated thermocouple end and with compensation to 20”C are used for temperature measurement. EB-type transducers were used for pressure measurement. ROSEMOUNT 1151-type sensor was used for measuring with ISA nozzle.

4.3.7.1 Description of the experiments

Test DNB-D19
DNB tests were performed in the LWL facility in a 19-rod bundle in the range of operating parameters of the VVER-1000 reactor. In the test section called “Bundle-D” the number of measured points was 166. Heated length was 3480 mm. Axial power distribution was uniform.

Operational limits of the LWL ŠKODA test facility

Maximum water pressure	20 MPa
Maximum water flow rate	250 m^3/hour
Maximum pump head	220 m of water
Maximum water temperature	350^{O}C
Maximum D.C. power supply	3.75 MW, 150 V
Maximal flow rate	200 m^3/hour
Maximal temperature gradient	2 ^{o}C/min
Delivery head	2.2 MPa
Volume of the water loop	5.0 m^3
Maximum A.C. power supply	155 kW

4.3.8 PACTEL facility

PACTEL is a volumetrically scaled (1:305) out of pile model of the VVER-440 type nuclear power plants located at Loviisa and operated by Imatran Voima Oy. All the main components of the reference reactor primary loop are included in PACTEL: a pressure vessel, main circulation loops, steam generators and a pressuriser. In addition to these emergency core cooling systems are simulated in PACTEL. The original elevations have been kept in order to maintain the natural circulation pressure heads.

The maximum primary side pressure is 8.0 MPa and on the secondary side the peak pressure is 4.6 MPa. The reactor pressure vessel is simulated with a U-tube construction including separate downcomer and core sections. The core consists of 144 electrically heated fuel rod simulators and the maximum heating power in the core is 1000 kW which is roughly 22% of the scaled down nominal power. The fuel rod pith (12.2 mm) and diameter (9.1 mm) are identical to those of the reference reactor. The rods are divided into three roughly triangular-shaped channels representing the intersection of the corners of three hexagonal VVER rod bundles.

The PACTEL test facility consists of three primary loops while the reference reactor has six primary loops. The hot and cold leg elevations of the reference reactor had been reproduced including the loop seal. The cold leg geometry differs from the other VVER-440 type reactors because of a unique main circulating pump construction.

The steam generators of the PACTEL facility contain 38 horizontal heat exchange U-tubes in nine layers. The average length of the tubes is 8.8 m. The diameter of the tubes and the space between them are the same as in the reference steam generator (vertical pitch 24 mm, horizontal pitch 30 mm). The heat transfer area of the tube bundles and the volume of each steam generator is scaled down so that one steam generator in PACTEL corresponds to two steam generators in the reference reactor (only three loops). The PACTEL steam generator is not a full height component, but the tube bundle height has been significantly reduced. The center of the PACTEL steam generator is at the same elevation as the center of steam generator in the VVER-440.

The measurement system of the PACTEL consists of about 390 transducers. The majority of the transducers are thermocouples (324 TCs) which measure fluid and structure temperatures in various locations in the primary and secondary side. The number of differential pressure transducers is 43 and they form a closed chain over the loop. There are ten flow rate measurement points. In addition to aforementioned measurements, primary pressure, secondary pressure and heating power are measured. Hewlett-Packard HP 3852A data acquisition unit is used to collect the data. The data acquisition unit is connected to a workstation HP 9000/360.

4.3.8.1 Description of the Experiments

LOF-1 experiment

Experiment LOF-1 (Loss of Feed water) was performed to study behaviour of a VVER reactor geometry during a loss of feed water transient. LOF-1 was a one loop test where low core power was used through out the experiment. So, the test did not simulate what is normally called a loss of feed water transient since nominal scaled down core power was not used in the beginning of the test. However, the test data is useful for verification of the steam generator models used in thermal hydraulic computer codes.

LOF-1 was started from steady-state conditions where the water level in the steam generator secondary side was above the heat exchange tubes. The test was performed with only one primary loop of the facility in operation (Loop 3), which was realized by closing the primary loop isolation valves in the other two loops. In the hot legs, these valves are located in the loop seals. Hence, the pressurizer was operable in this test even though the loop isolation valves were closed in the loop to which the pressurizer is connected. In LOF-1 feed water injection to the steam generator secondary side was closed at the beginning of the test. The experiment was continued until the heat transfer to the secondary side ceased and the primary system pressure started to rise. The heating power in this test was 75 kW, corresponding to 1.7 % power in the reference reactor.

In the experiment, the main primary system parameters remained unchanged until the uppermost layer of heat exchange tubes in the steam generator secondary side were no longer covered by water, i.e. the swell level in the steam generator secondary side dropped below these tubes. This happened when the collapsed liquid level in the secondary side was about 20 centimeters from the bottom of the steam generator. When the uppermost layer of tubes in the steam generator secondary side was no longer covered by water, steam in the secondary side started to superheat. At the end of the experiment, steam in the steam generator secondary side was almost 30 °C superheated.

The fact that the heat exchange tubes in the steam generator secondary side were not covered by water affected the flow rate in the primary side. The mass flow rate decreased when one tube layer in the secondary side became free of water. The flow rate changed since the tubes being no longer covered by water leads to increasing steam generator outlet temperature and, hence, to a smaller temperature difference between heat source (core simulator) and sink (steam generator). After a while the loop flow rate returned to near the original level. This happened, since the increase in heat sink temperature affected, after some time, also the heat source temperature. LOF-1 was terminated after the primary pressure exceeded 76 bar.

ITE-6 experiment

The ITE-6 experiment was performed to observe a natural circulation flow behaviour under quasi-steady state conditions over a range of primary side inventory levels. Core power was selected as 155 kW which is 3.4% of the scaled down nominal power of the reference power plant. The core power was constant through out the test. In the experiment, the facility was heated up until it approached the selected temperature and pressure and steady state was established near these conditions. The primary coolant was drained from the lower plenum in several steps. There was a 900 seconds period between each step in order to let the system to stabilize. The drained mass was about 60 kg in each depletion. For the duration of the test, the secondary side conditions were maintained near the nominal full power operating conditions of the reference power plant.

Three natural circulation heat transfer modes were observed in the course of the test. In the beginning a single phase flow was established and after the second depletion the heat transfer mode started to change from single phase to two phase natural circulation mode. The transition was not smooth, but several flow stagnations were measured. The oscillatory behaviour was caused by the hot leg loop seals. The flow was not distributed evenly between the primary loops, but the loop flow was highly asymmetric. After the oscillatory period two phase natural circulation flow was observed and asymmetric flow behaviour continued in this mode as well. When the inventory was dropped further the heat transfer mechanism from primary to secondary side changed to boiler-condenser natural circulation. The downcomer mass flow rate was very low in this mode since condensing is so effective way to transfer heat from the heat source to the heat sink. The test was terminated when the cladding temperatures of the fuel rod simulators stared to increase at 6660 seconds.

4.3.9 REWET-II facility

The REWET-II test facility was an out of pile model of VVER-440 nuclear power plants used in Finland. These power plants are located at Loviisa and they are operated by Imatran Voima Oy. REWET-II was located in a laboratory of Lappeenranta University of Technology and the facility was operated by Technical Research Centre of Finland (VTT), Nuclear Engineering Laboratory. The test results are now owned by VTT Energy.

The REWET-II facility was designed for the investigation of the reflooding phase of a loss of coolant accident (LOCA). The main design principle was an accurate reproduction of the fuel rod bundle construction and the primary system geometry of the reference power plant. Volumetric scaling factor was 1:2333. The reactor pressure vessel was simulated by a stainless steel U-tube construction consisting of a downcomer, a lower plenum, a core and an upper plenum. The primary loops contained a pipe modeling a broken loop and a connection line between the upper plenum and the downcomer simulating five intact loops. The containment was simulated by a pressure vessel which was not scaled using the same scaling factor as the rest of the test facility. Steam generators and the primary circulating pumps were simulated by using flow resistance. The emergency core cooling (ECC) water could be injected to the downcomer and/or to the upper plenum using a pump or from an accumulator.

The rod bundle consisted of 19 fuel rod simulators. The geometry was hexagonal with chopped cosine power distribution. Heating power of the rods were 0-90 kW and the average linear heating power was 0-20 W/cm. The maximum primary side pressure was 1.0 MPa.

The main measurements in the experiments were coolant and cladding temperature measurements by thermocouples at different axial and radial locations. System pressure, differential pressure along the test section, coolant flow rate and heating power were also measured. The data acquisition system consisted of a measurement and control processor (Hewlett-Packard HP 2240A) and a desk-top computer (Hewlett-Packard HP 9835A). All together 96 channels were measured once per second. After a test the results were uploaded to a mainframe computer for storage and hard copies (Calcomp 1039 drum plotter).

4.3.9.1 Description of the Experiments

Test SGI/7
The test facility was heated up by using steam before the actual experiment. The lower plenum was filled with water to the level of the beginning of the core heated section. The core power was turned on and the heating continued until the initial cladding temperature was reached in the middle of the core. Then the injection was started. The test was terminated when the whole core was quenched.

The maximum initial fuel rod simulator cladding temperature was 600 °C. The core power was 30 kW and it was constant during the entire experiment. There was emergency core cooling water injection both to the downcomer and to the upper plenum. The injection mass flow rate was 0.0345 kg/s and temperature 50 °C to both injection locations. The injection mass flow rate (0.0345 kg/s) corresponds to 2 cm/s level rise speed in the core section. The primary system pressure was 0.3 MPa.

4.3.10 PMK-2 Facility

The PMK-2 facility is a scaled down model of the Paks NPP and it was primarily designed for investigating small-break loss of coolant accidents (SBLOCA) and transient processes of VVER-440/213 plants. The volume and power scaling are 1:2070. Transients can be started from nominal operating conditions. The ratio of elevations is 1:1 except for the lower plenum and pressuriser. The six loops of the plant are modelled by a single active loop. The secondary side of the steam generator maintains the steam/water volume ratio. The coolant is water under the same operating conditions as in the nuclear power plant.

It is evident that nuclear safety related experiments have several different types of transients. Facilities designed for a particular transient may have distortions for other types of transients. It means that the design of a multi-purposes facility is a compromise between the strict scaling criteria and the practical and experimental requirements.

In the PMK-2, tests are conducted from nominal initial conditions, thus similarity of thermohydraulic processes should be ensured both during pumps coast-down and in the natural circulation period with either single- or two-phase flow.

Basic agreement in nominal operational conditions between the Paks NPP and PMK-2 is archived by using water at the same parameters is the PMK-2 loop as in the plant. The power ratio is the same as the volume ratio and so is the mass flow rate ratio as well.

Similarity criteria for a natural circulation system are obtained from the integral effects of local balance equations along the entire loop. For both single- and two-phase flow, similarity criteria using balance equations in one-dimensional approximation were used.

The nominal operating parameters of the loop corresponding to the normal operation of the Paks NPP are as follows:

- pressure in upper plenum	12.3 MPa
- loop flow	4.5 kg/s
- core inlet temperature	541 K
- core power	664 kW
- secondary side pressure	4.6 MPa
- feed water flow	0.36 kg/s
- feed water temperature	496 K
- coolant level in pressuriser	9.254 m

- SITs pressure 5.9 MPa
- SITs level (single/double) 10.035/9.635 m
- SITs water temperature 293 K
- HPIS/LPIS temperature 293 K
- SG secondary level 8.4 m

4.3.10.1 Description of the Experiment

Test IAEA-SPE4

The experiment is performed on the PMK-2 facility in the framework of a joint project between IAEA and KFKI-AEKI in 1994. The experiment is characterized as follows:

- a ∅ 3.2 mm cold leg break, modelling a 7.4 % break in the Paks NPP
- starting from full power
- with injection from three hydroaccumulators
- without injection from high pressure injection systems
- the secondary side is isolated after transient initiation
- with injection from low pressure injection systems
- with secondary side bleed
- with emergency feed water injection

The initial conditions of the test correspond to the nominal operating parameters of the reference plant. The test is conducted for a transient of 1800 s.

In the boundary conditions of the test the relevant plant data were used. The head losses and the pressure distribution were measured before the test. In the data processing there is a clear procedure to convert row data into the presented test results. A standard archiving/documentation provides the availability of the data set.

Profile of the SPE-4 experiment. The transient was initiated from full rated power by opening the break valve. At the same time the steam generator was isolated by closing the steam line valve and the control valve in the feed water line. The system started to depressurize and soon reached the saturated condition. The reactor scram was activated in a few seconds, when the pressure in the primary side dropped below 11.15 MPa. The core power was reduced according to a prescribed curve, simulating the decay heat. The pump trip simulation was initiated at 9.21 MPa of the primary pressure and the duration of pump coast-down was 150 s. At the end of pump coast-down, secondary side bleed was started by opening the relief valve on the SG steam line. Hydro-accumulator injection started when the system pressure reached the actual setpoints. During this period, the primary pressure fell slightly below the secondary pressure. The hot leg was cleared for a very short period at 165 s, and final clearance took place at 350 s. Further depressurization was governed by the break flow and the cooldown of the facility due to heat losses to the environment. A gradual decrease of the liquid level eventually led to partial uncovery of the heated section of the core, characterized by a sharp increase in the heater rod surface temperatures. A signal indicating a secondary pressure of 0.93 MPa triggered the injection of emergency feed water into the steam generator, with a flow rate of 0.042 kg/s. Low pressure injection was initiated when the primary pressure dropped below 1.04 MPa. The experiment was terminated at 1800 s transient time.

5. Validation Matrices

In Chapter 3 the Cross Reference Matrices (CRM) have been constructed by inserting the VVER-specific test facilities addressing the different phenomena. In Chapter 4 a number of tests (performed on these facilities) have been selected and described.

The task of the present chapter is to support code validation by filling in the so-called Validation Matrices (VM). The VM represents two submatrices of the CRM: the phenomena versus test facility and the test type versus test facility submatrices, with the difference that data are presented here for the **selected tests** and no more for the test facilities in general.

Each test is evaluated against the phenomena on three levels:

- "simulated", meaning that the phenomenon occurs in the test and is representative for the real plant conditions;
- "partially simulated", when only some aspects of the phenomenon are occurring, or the simulation is only partially representative for plant conditions (e.g. due to scaling);
- "not simulated", with the obvious meaning that the phenomenon is not present in the test.

As a further information for code validation it is also marked which test type is addressed by each of the tests (the test type addressed being marked by a "+").

The validation matrices were filled in on the basis of the experience of the editorial group.

5.1 Validation Matrices

MATRIX I

Phenomena addressed by		**Test facility**	SB	REWET-II	KS	LWL
		Test number	SB/1	SGI/7	KS/19R/ TF84	DNB-D19
1	Break flow		+	-	-	-
2	Phase separation		o	-	-	-
3	Mixing and condensation during injection		-	-	-	-
4	2-phase flow in SG primary and secondary side		-	-	-	-
5	Core wide void + flow distribution		-	-	-	-
6	ECC downcomer bypass and penetration		-	-	-	-
7	CCFL (UCSP)		-	-	-	-
8	UP injection and penetration		o	-	-	-
9	Steam binding (liquid carry over, ect.)		-	-	-	-
10	Pool formation in UP		-	-	-	-
11	Core heat transfer incl. DNB, dryout, RNB		+	-	+	+
12	Quench front propagation		+	+	-	-
13	Entrainment (Core, UP)		o	+	-	-
14	Deentrainment (Core, UP)		o	+	-	-
15	1 - and 2-phase pump behaviour		-	-	-	-
16	Noncondensible gas effects		-	-	-	-

Test versus phenomena	
+	Simulated
o	Partially simulated
-	Not simulated

Test Types		SB/1	SGI/7	KS/19R/ TF84/	DNB-D19
1	Blowdown	x	-	x	x
2	Refill	x	-	-	-
3	Reflood	x	x	-	-

MATRIX II-1

Phenomena addressed by		Test facility	SB	ISB-VVER	ISB-VVER	PACTEL
		Test number	SB/2	UPB-24	NC	ITE-06
1	Natural circulation in 1-phase flow, primary side		-	+	+	+
2	Natural circulation in 2-phase flow, primary side		o	+	+	+
3	Reflux condenser mode and CCFL		-	+	o	o
4	Asymmetric loop behaviour		-	-	-	-
5	Leak flow		+	+	+	+
6	Phase separation without mixture level formation		-	-	-	--
7	Mixture level and entraiment in SG (SS+PS)		-	-	-	-
8	Mixture level and entraiment in the core		-	o	o	o
9	Stratification in horizontal pipes		-	-	-	-
10	ECC-mixing and condensation		o	-	-	-
11	Loop seal clearance (CL)		o	o	o	o
12	Pool formation in UP/CCFL (UCSP)		-	-	-	-
13	Core wide void and flow disbribution		-	-	-	-
14	Heat transfer in covered core		+	+	+	+
15	Heat transfer in partly uncovered core		+	+	+	+
16	Heat transfer in SG primary side		-	-	-	-
17	Heat transfer in SG secondary side		-	-	-	-
18	Pressurizer thermohydraulics		-	-	-	-
19	Surgeline hydraulics		-	-	-	-
20	1- and 2-phase pump behaviour		-	-	-	-
21	Structural heat and heat losses		-	-	-	-
22	Noncondensible gas effects		-	-	-	-
23	Phase separation in T-junct. and effect on leakflow		-	-	-	-
24	Nat. circul., core-gap-downcomer, dummy elem.		+	-	-	-
25	Loop seal behaviour in HL		-	-	-	o
26	Recirculation in the SG primary side		-	-	-	-
27	Boron mixing and transport		-	-	-	-
28	Water accumulation in SG tubes		-	-	-	-

Test versus phenomena	
+	Simulated
O	Partially simulated
-	Not simulated

Test Types		SB/2	UPB-24	NC	ITE-06
1	Stationary Test addressing energy transport on primary side	-	-	-	-
2	Stationary Test addressing energy transport on secondary side	-	-	-	-
3	Small leak overfeed by HPIS, secondary side necessary	-	-	-	-
4	Small leak without HPIS overfeeding, secondary side necessary	-	-	x	x
5	Intermediate leak, secondary side not necessary	x	x	-	-
6	Pressurizer leak	-	-	-	-
7	SG-tube rupture	-	-	-	-
8	SG-header rupture	-	-	-	-

MATRIX II-2

Phenomena addressed by		Test facility	PM-5	PACTEL	PACTEL	PMK-2
		Test number	**6**	**ITE-6**	**LOF-1**	**IAEA-SPE-4**
1	Natural circulation in 1-phase flow, primary side		-	+	?	-
2	Natural circulation in 2-phase flow, primary side		+	+	-	+
3	Reflux condenser mode and CCFL		-	-	-	-
4	Asymmetric loop behaviour		-	+	-	-
5	Leak flow		o	-	-	o
6	Phase separation without mixture level formation		o	o	-	o
7	Mixture level and entraiment in SG (SS+PS)		-	o	o	-
8	Mixture level and entraiment in the core		o	o	-	o
9	Stratification in horizontal pipes		-	o	-	-
10	ECC-mixing and condensation		-	-	-	o
11	Loop seal clearance (CL)		-	-	-	o
12	Pool formation in UP/CCFL (UCSP)		-	-	-	-
13	Core wide void and flow disbribution		-	o	-	-
14	Heat transfer in covered core		+	+	+	+
15	Heat transfer in partly uncovered core		-	o	-	+
16	Heat transfer in SG primary side		o	o	o	o
17	Heat transfer in SG secondary side		-	-	o	-
18	Pressurizer thermohydraulics		-	-	-	-
19	Surgeline hydraulics		-	-	-	-
20	1- and 2-phase pump behaviour		-	-	-	-
21	Structural heat and heat losses		-	-	-	-
22	Noncondensible gas effects		-	-	-	-
23	Phase separation in T-junct. and effect on leakflow		-	-	-	-
24	Nat. circul., core-gap-downcomer, dummy elem.		-	-	-	-
25	Loop seal behaviour in HL		o	+	-	o
26	Recirculation in the SG primary side		-	o	-	-
27	Boron mixing and transport			-	-	-
28	Water accumulation in SG tubes			-	-	-

Test versus phenomena	
+	Simulated
o	Partially simulated
-	Not simulated

Test Types		6	ITE-06	LOF-1	IAEA-SPE-4
1	Stationary Test addressing energy transport on primary side	x			
2	Stationary Test addressing energy transport on secondary side		x		
3	Small leak overfeed by HPIS, secondary side necessary				
4	Small leak without HPIS overfeeding, secondary side necessary				
5	Intermediate leak, secondary side not necessary				
6	Pressurizer leak				
7	SG-tube rupture				
8	SG-header rupture				

MATRIX III

Phenomena addressed by		Test facility	PACTEL	BD	PMK-2	ISB-VVER	LWL
		Test number	LOF-1	BD-1	IAEA-SPE-4	NC	DNB-D19
1	Natural circulation in 1-phase flow		+	-	-	+	-
2	Natural circulation in 2-phase flow		+	-	-	+	-
3	Core thermohydraulics		+	-	-	+	-
4	Thermohydraulics on primary side of SG		o	-	-	-	-
5	Thermohydraulics on secondary side of SG		o	-	-	-	-
6	Pressurizer thermohydraulics		o	-	-	-	-
7	Surgeline hydraulics (CCFL, chocking)		o	-	-	-	-
8	1- and 2-phase pump behaviour		-	-	-	-	-
9	Thermal hydraulic-nuclear feedback		-	-	-	-	-
10	Structural heat and heat losses		o	-	-	-	-
11	Boron mixing and transport		-	+	-	-	-

Test versus phenomena	
+	Simulated
o	Partially simulated
-	Not simulated

Test Types		LOF-1	BD-1	IAEA-SPE-4	NC	DNB-D19
1	ATWS	-	-	-	-	-
2	Loss of feedwater, non ATWS	x	-	-	-	-
3	Loss of heat sing, non ATWS	-	-	-	-	-
4	Station blackout	-	-	-	-	-
5	Steam line break	-	-	-	-	-
6	Feed line break	-	-	-	-	-
7	Cooldown primary feed and bleed	-	-	-	-	-
8	Reactivity disturbance	-	x	-	-	-
9	Over-cooling	-	-	-	-	-

5.2 Discussion and Evaluation of Validation Matrices

As it can be seen from the validation matrices above there is a number of phenomena which have not been adequately addressed by available experimental programs or tests, in particular.

Due to the general similarity between VVERs and PWRs for most of these phenomena, in particular, for 3D effects, integral ECCS behaviour, number of transient phenomena and flow maps in pipes, the code validation may be sufficiently done using CSNI ITF and SETF validation matrices [2, 3].

For the phenomena which are exclusively VVER specific and for which there is no relevance in the CSNI matrices adequate experimental tests should be proposed and carried out. This is in particular true for a large-scale integral simulation of the VVER system behaviour during large break LOCA accidents (including large leak from primary to secondary circuit).

The future experimental program should be effectively based on open international co-operation. In this context also CSNI has discussed future follow up activities such as VVER specific ISPs or International thermalhydraulic research program covering both design basis and beyond design basis accidents.

6. Conclusion

A systematic study has been carried out to select experiments for thermal-hydraulic system code validation. The main experimental facilities for VVERs have been identified and described in Chapter 4 and Appendix D.

Matrices have been established to identify, firstly, phenomena assumed to occur in VVER plants during accident conditions and secondly, facilities suitable for code validation (Chapter 4). Tables identify the experiments selected for validation of computer codes (Chapter 4). The matrices also permit identification of areas where further research may be justified. Compared with [4], a revision and update of the matrices, have been performed in this report.

Additional work has been performed to describe the VVER reactor systems (Appendices A and B), the content of the validation matrices, i.e. the test types (Chapter 2), the phenomena Chapter 2 and Appendix C, and the selected tests (Chapter 4).

A periodic updating of the matrices will be necessary to include new relevant experimental facilities and tests (e.g. investigating boron dilution or behaviour of advanced reactors) and to include improved understanding of existing data as a result of further validation.

To validate a code for a particular LWR plant application, it is recommended that the list of tests in the relevant matrix be viewed as the phenomenological well founded set of experiments to be used for an adequate validation of a thermal hydraulic computer code.

A-1 General Description

The first Soviet standard plant design with pressurised water reactor, designated as VVER-440 NPP model 213, was commercially introduced in 1980s. Over 20 units with model 213 reactor are presently in operation or in final stage of construction in Czech Republic, Finland, Hungary, Russia, Slovakia and Ukraine. The model 213 units are a second generation of the standardised nuclear power plant, called VVER-440. The first design version of VVER-440s, known as model 230, have been in operation since the early 1970s.

All VVER-440 plants have some specific features that affect the overall characteristics of the plant, both from the economic and the safety point of view. The large water inventory in the primary circuit and in the steam generator, relatively low operating parameters and medium reactor output, a primary coolant system layout with six cooling loops, the use of horizontal steam generators and isolation valves on each loop, two medium size turbines of 220 MW(e) each instead of one, have the most considerable impact on economic plant factors, primarily through the large investment costs. However, from the safety point of view, most of these design choices have positive side effects.

The reactor core is composed of hexagonal fuel assemblies with fuel rods arranged in a triangular grid. Control rod assemblies are a combination of a fuel assembly and an absorbing extension. The VVER-440 uses a rack and pinion drive mechanism to move the control rods.

The VVER-440 model 213 substantially differs from the older model 230. The 213 has both additional accident localisation features and the standard ECCS. Reactor coolant pump is equipped with high inertia flywheel to increase the pump coast-down. Safety systems included in the model 213 are, in general, comparable to those used in other plants with pressurised water reactors.

The most significant addition to the accident localisation system involves the inclusion of pressure suppression system, incorporating a large number of water trays serving as suppression pools, in which extensive steam condensation occurs during an emergency LOCA conditions. For each unit, a set of pressure suppression trays is located inside a separate building (the bubbler/condenser tower) constructed adjacent to the reactor building. The tower is connected to the steam generator compartment by a rectangular tunnel. The condenser tower also houses four large receiver volumes serving as air traps.

The model 213 was designed to cope with a double-ended rupture of the largest pipe (500 mm diameter). The design pressure of the model 213 accident localisation volume is about 0.25 MPa absolute. The containment structure integrity relies on bubbler condenser performance that has not been tested on a large scale.
The VVER-440 model 213 incorporates redundant independent emergency core cooling systems including a core flooding system (CFS), a high pressure injection system (HPIS) and a low pressure injection system (LPIS). These systems provide full protection against the entire range of break sizes.

HPIS does not perform any make-up functions, for which a separate system is used. A spray system is provided in the steam generator and pump compartments to condensate steam during emergency conditions.

Some of the main operating parameters of the VVER-440 model 213 plants are provided below [A-1]:

Thermal output	1375 MW(th)
Gross electric output	440 MW(e)
Number of primary circuit loops	6
Reactor pressure	12.26 MPa
Steam pressure before turbine	4.4 MPa
Average reactor coolant temperature	285 °C
Loop flow	1.93 m^3/s
Fuel charge (UO_2)	47 t
Number of fuel assemblies in the core	312
Number of fuel rods in one assembly	126
Average linear load of fuel rods	125 W cm^{-1}
Maximum linear load of fuel rods	325 W cm^{-1}
Number of drives for control rod assemblies	37

This Annex provides basic information related to the design of VVER-440 model 213 unit [A-2, A-4] with particular attention given to Dukovany NPP [A-1, A-3].

A-2 System Design Highlights

A-2.1 Reactor System

The reactor system is composed of the pressure vessel with the reactor internals (Fig.1). The reactor internals include the core barrel, the flow distribution structure, fuel assemblies and control assemblies.

The reactor vessel has 6 inlet and 6 outlet nozzles. The outlet nozzles are located at the higher elevation than the inlet nozzles. The inlet and outlet nozzles are offset. The cold legs are connected to the top of the downcomer, and the hot legs to the upper plenum. A small vessel bypass connection exists between the downcomer and the upper plenum.

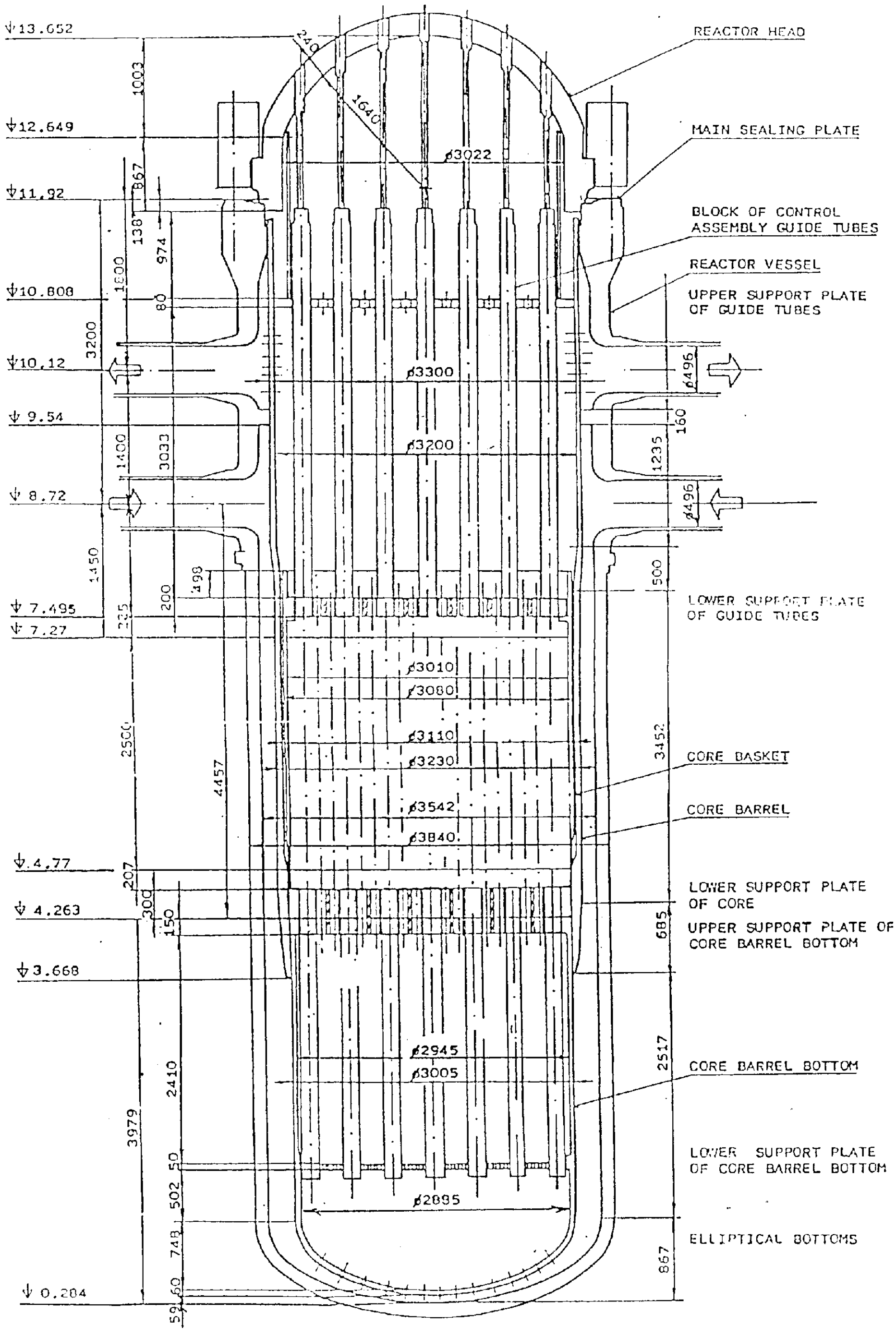

Fig.1. VVER-440/213 reactor pressure vessel

The reactor pressure vessel is fabricated of sections of cylindrical forgings that are welded into a cylinder with an elliptical bottom. The reactor pressure vessel head is bolted on to complete the vessel pressure boundary and to support and locate the control rod drives. Internal surface of the vessel is protected by stainless steel cladding, 9 mm thick.

A cylindrical tank, called core barrel, holds the core basket containing fuel assemblies. At the same time it serves to separate incoming and outcoming coolant and to distribute its flow through the reactor. The cylindrical wall of the core barrel serves also as thermal shield to reduce the neutron flux at the reactor vessel wall. The core basket assembly consist of two plates separated by a series of tube guides for control rods. A flow skirt around the periphery aids in flow distribution. Fixed fuel assemblies are centred at the top assembly arresters and fixed by the core hold-down/ upper internals structure.

Some of the reactor vessel data are listed below [A-1]:

Volumes:

Core fuel region	12.01 m^3
Upper plenum	25.33 m^3
Lower plenum	22.45 m^3
Upper head	15.91 m^3
Inlet nozzle (incl. cold leg piping)	3.25 m^3
Outlet nozzle (incl. hot let piping)	2.76 m^3

Elevations of the volumes (from inside the bottom of the vessel):

Core fuel region	7.411 m
Upper plenum	10.524 m
Lower plenum	4.28 m
Upper head	13.37 m
Inlet nozzle (incl. cold leg piping)	8.44 m
Outlet nozzle (incl. hot let piping)	9.85 m

Weight of steel in contact with primary water:

Reactor head	42 140	kg
Reactor vessel	215 150	kg
Mass of vessel internals (without fuel and control assemblies)	127 432	kg
Total	384 722	kg

The unfavourable feature of the reactor pressure vessel design is its vulnerability to brittle fracture due to the small distance between the core edge and the vessel wall, and the consequential high fluence of fast neutrons [A-5].

The reactor core in the 213 model is composed of 312 hexagonal fuel assemblies and 37 movable control assemblies.

The fuel assemblies contain 126 fuel rods each, arranged in a triangular grid with a rod pitch of 12.2 mm, and a central tube used for instrumentation and control. The fuel bundle is surrounded by the fuel assembly shroud that forms an integral unit. The flow rate is adjusted by means of throttling orifices, installed in the support plate of the core barrel. Eleven honeycomb type grids, fixed on the central tube, are used to space fuel rods and to prevent rod buckling.

The bottom portion of the movable assembly contains fuel and is in tandem with the upper portion containing boron-steel control material. The rack and pinion drive mechanisms used to move the control rods are always coupled. The drive rods and mechanisms are cooled by intermediate component cooling system.

The fuel rod is made of 7.5 mm diameter uranium dioxide pellets (2.4 - 3.6% U-235), contained in 9.1 mm outside diameter tubes made of Zr - 1% Nb alloy, 0.65 mm thick. Fuel pellets have a central hole of 1.5 mm diameter and there is a 56 mm long plenum at the top of the pellet stack. The gap between the pellet and the clad is 0.16 mm at initial conditions. The gap and the plenum are filled with an inert gas under pressure. The active length of the fuel rod for a fixed fuel assembly and the shim assembly is, respectively, 2420 mm and 2320 mm.

A-2.2 Reactor coolant system

Reactor coolant system (RCS) consists of six main circulation loops. Each RCS loop includes a horizontal steam generator (SG), a main circulating pump (MCP) and isolation valves both in the hot and cold legs. The hydraulic diameter of hot/cold legs is 0.496 m. Six primary coolant loops have common flow paths through the reactor vessel, but are otherwise independent in operation. Operation with a loop out of service is possible, due to the use of primary coolant system isolation valves. However, it is not recommended because of asymmetrical distribution of flow in individual loops.

In order to reduce primary coolant elevations the hot leg includes an U-shaped section (Fig.2 [A-8]). This feature provides loop seal effects during a LOCA accident. Also there is a cold loop seal between the SG outlet and a main circulating pump (Fig.3 [A-8]), which may contribute to the reactor level during LOCA conditions.

The main circulating pumps are of a shaft seal type with a special flywheel on the pump shaft to provide the desired flow pump coast down. The data for main circulating pumps [A-1, A-3] are listed below.

Nominal flow	1.94	m^3/s
Pressure rise	0.4	MPa
Volume	0.63	m^3
Temperature	270	°C
Nominal speed	1460	rpm
Nominal torque	8461.8	Nm
Moment of inertia of rotating part	1025	kgm^2
Heat generation	578.8	kJ/kg

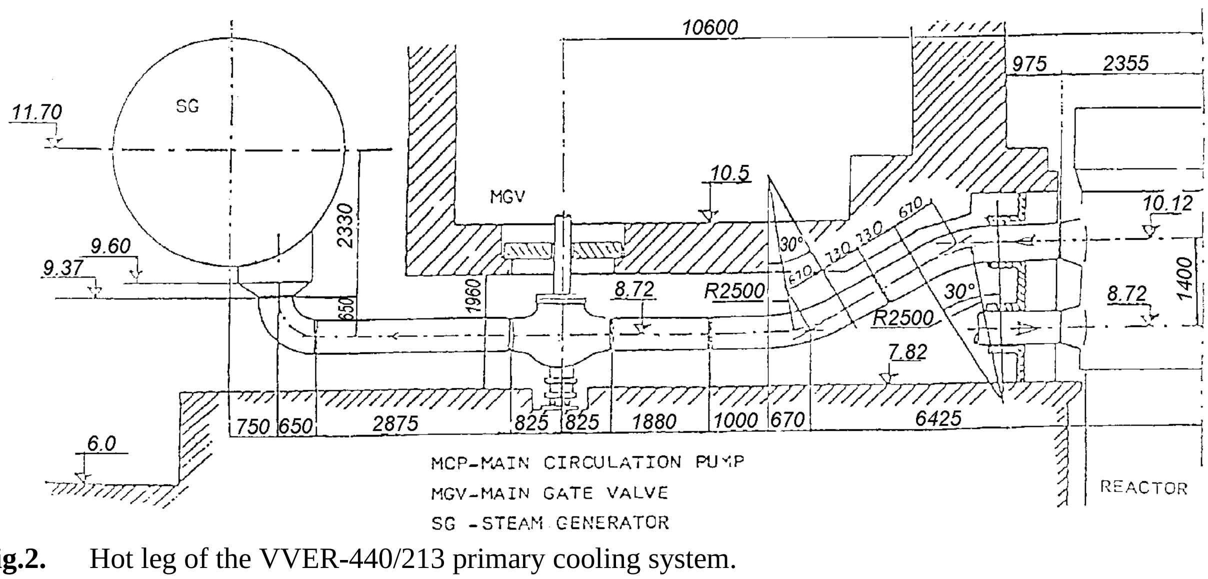

Fig.2. Hot leg of the VVER-440/213 primary cooling system.

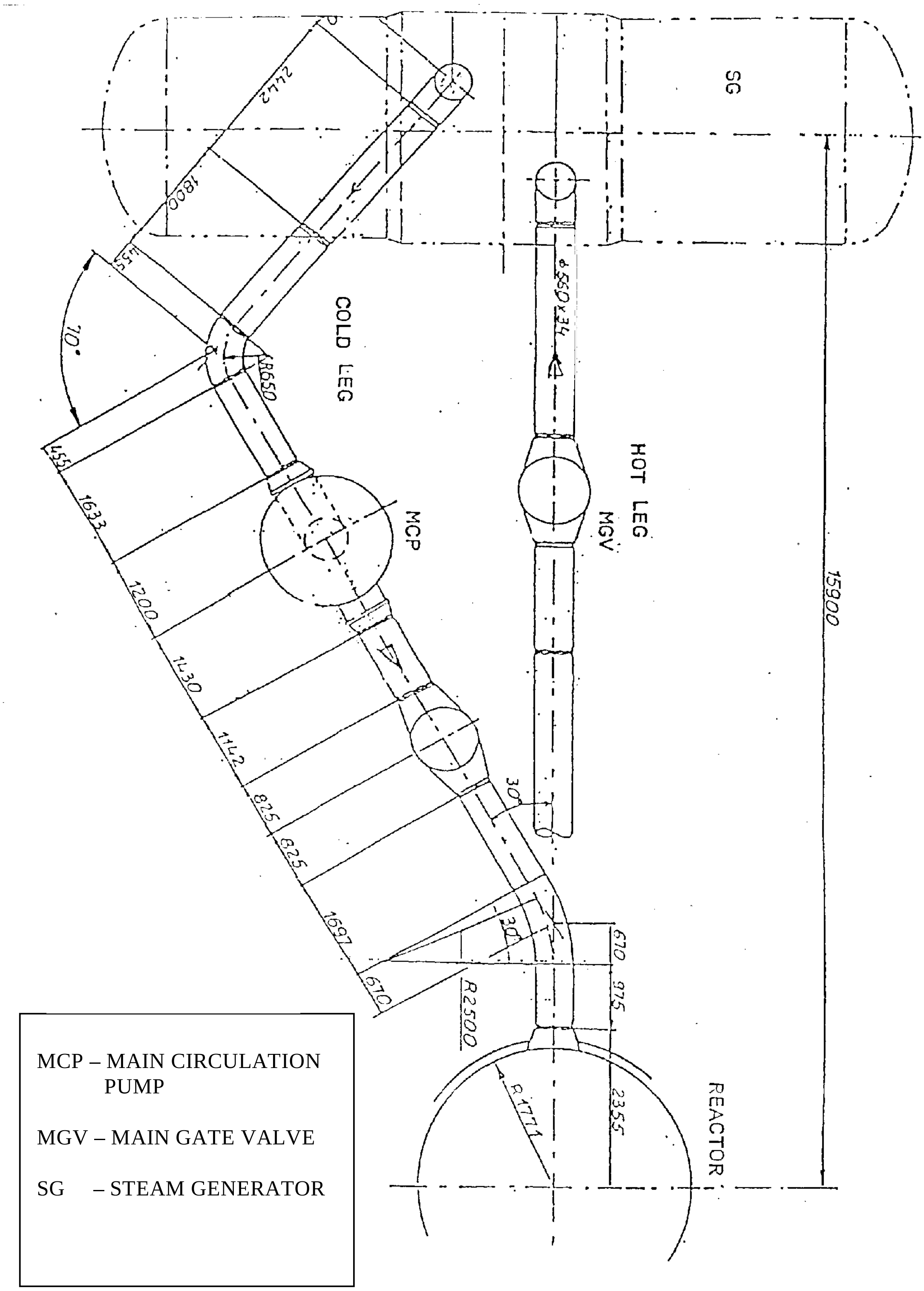

Fig.3. VVER-440/213 primary cooling loop (view from the above)

The main isolation valves used in the primary circuit are of a disk type. Tight closing of the valve is ensured by two disks which close the valve opening tightly in one direction of the fluid flow. Closure time is 36 s and 1927 s, respectively, for remote power operated and manual options [A-1].

The steam generators are horizontal units (Fig. 4 [A-2]), with submerged tube bundles and a built-in steam separator. Each unit includes a cylindrical horizontal shell, two vertical inlet and outlet manifolds of 750 mm diameter and two horizontal tube bundles of stainless steel, comprising U-shaped tubes of 16 mm inner diameter. The primary coolant flows inside the tubes, while the feedwater is delivered outside them to the shell. The SG secondary side instrumentation includes measurements of the liquid level, temperature and pressure. Fine scale level is used for feedwater controlling. More detailed information on the steam generator is provided below [A-1, A-3]:

SG inlet/outlet water temperature	297/268	°C
SG primary flow rate	7150	m^3/h
SG secondary pressure	4.6	MPa
Steam and feedwater flow rate	125	kg/s
Circulation ratio at rated conditions	5 - 20	
Coolant volume in the tube bundle	6.833	m^3
Total water volume of primary side	15.24	m^3
Number of tubes	5536	
Average tube length	9.02	m
Upper level of the tubing	1.824	m
Nominal steam-water mixture level at secondary side	1.919	m
Water volume corresponding to nominal mixture level	45.45	m^3

The use of horizontal steam generators together with the overall layout of the primary coolant system facilitate the transition to one-phase natural circulation in the primary circuit. Therefore, decay heat removal from the core during the plant cooldown relies on the same heat transfer path as that used during normal power operation. This feature has positive effect on plant risk as compared with other PWRs.

The pressuriser, which maintains overall system pressure (12.5 MPa) and compensates for changes in the volume of the primary coolant, is connected by two pipelines with the hot leg in one of the loops. The pressuriser is equipped with a relief valve and two safety relief valves.

During normal operation the pressuriser maintains RCS pressure within the prescribed limits using pressurise sprays and heaters. The pressuriser spray line originates at the cold leg of RCS at the reactor coolant pump discharge and is connected to a water spray nozzle in the upper section of the pressuriser. The spray depends on the driving force produced by running MCPs and is controlled by the set of motor operated valves. The pressuriser heaters are grouped in five banks and controlled by the pressure controller. Level instrumentation on the pressuriser monitors RCS volume to determine make-up and let-down requirements.

Selected parameters related to the pressuriser control system are as follows [A-1,A-3]:

Relief valve rated pressure	13.23	MPa
Relief valve rated flowrate	5000	kg/h
Flow rate through one safety relief valve fully opened	31.94	kg/s
Number of spray valves	4	
Spray valve capacity (fully opened)	11 - 15	kg/s
Capacities of heater banks	180, 180, 360, 360, 540	kW

Under abnormal conditions the pressuriser perform two additional functions. The relief valves are the only means of external pressure relief for the RCS. Also, the safety relief valves can be used together with HPIS to remove heat from the RCS (primary „feed and bleed“) when heat removal through the secondary system is not available. However, in the VVER-440 this mode of RCS cooling has not been taken into account by the plant designer and at present it is covered by existing emergency procedures only in some plants.

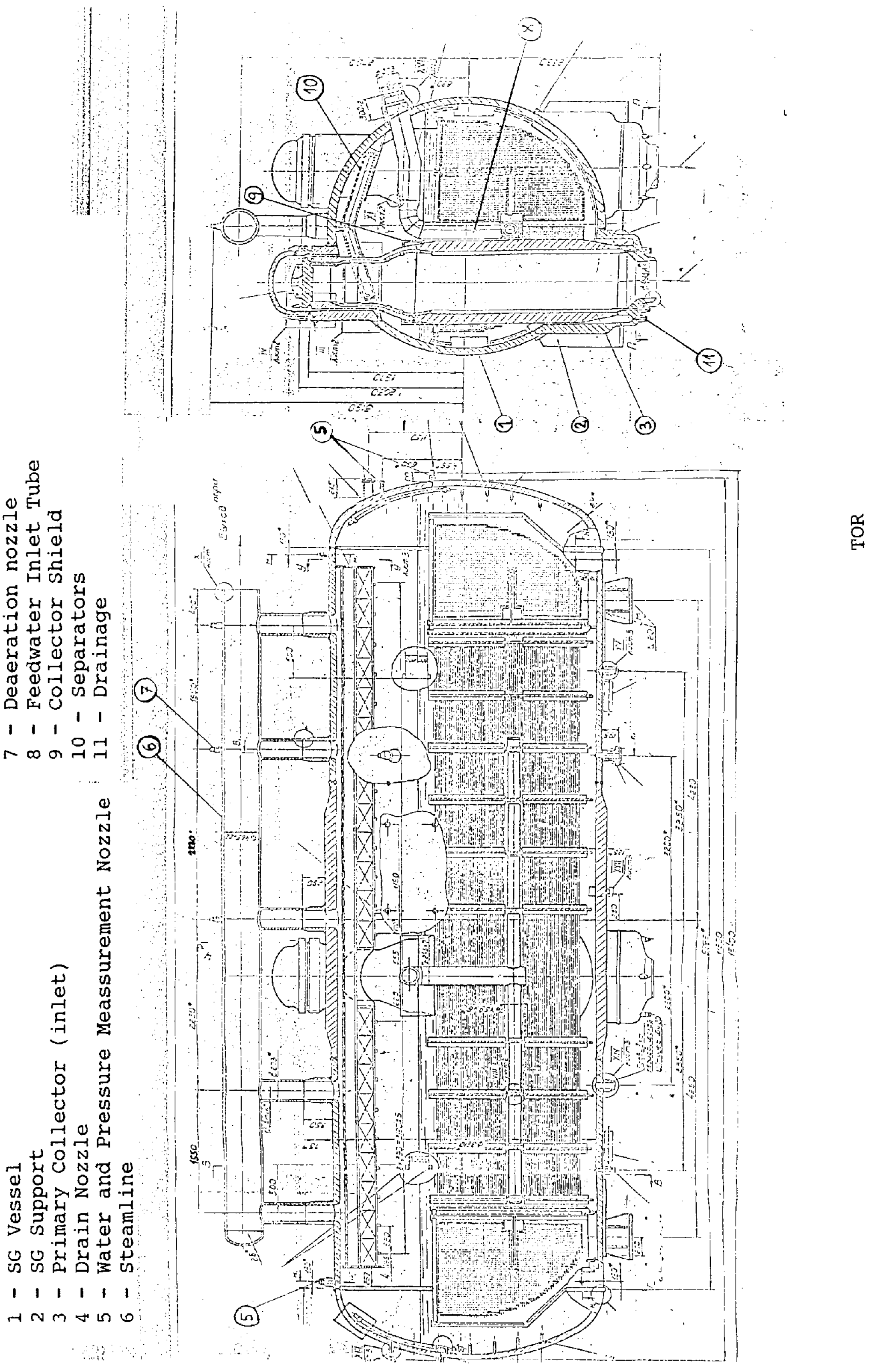

Fig.4. VVER-440/213 steam generator.

A-2.3 Chemical and volume control system (CVCS)

The CVCS controls the volume, purity and boric acid concentration of the reactor coolant during normal operational conditions and transients, including startup, shutdown and changes of the reactor power level.

During normal operation the CVCS compensates uncontrolled and controlled leaks from the RCS. Adjustments in the coolant volume are made automatically to maintain a predetermined level in the pressuriser. Coolant purity is controlled by continuous purification of bypass stream of reactor coolant.. A bleed and feed technique is used to control the boric acid concentration in the reactor coolant. The CVCS also provides seal injection water for the main circulating pumps and collects pump seal leakoff.

In addition to normal operational functions, the CVCS supplies emergency make-up in case of abnormal RCS leaks and emergency boration function during reactivity induced transients. The majority of the system functions related to transient and abnormal conditions require the operator actions.

The CVCS pumps are supplied with power from the essential power source and may be used as alternative means for RCS inventory control under small break LOCA conditions.

A-2.4 Power conversion system

The power conversion system (Fig. 5 [A-2]) is a normal operational system designed to transport saturated steam from the steam generators to the turbines, where thermal energy is converted to mechanical energy required for the generation of electricity in main generator. The system is also used for heat removal through the steam generators during normal plant cooling-down and in post-accident conditions.

The turbine is a condensation, single-shaft turbine with non-controlled steam extractions. The turbine consists of three cylinders, all of two-current flow arrangement. The high pressure cylinder consists of six stages in both flow currents, two low pressure cylinders contain five stages in both flow currents. Basic technical parameters of the turbine at Dukovany NPP [A-2] are as follows:

Rated power at generator terminals	220	MW
Maximum power	235	MW
Rated revolutions	3000	rpm
Rated pressure	4.3	MPa
Rated steam temperature	256	°C
Pressure at HP part outlet	0.5	MPa
Steam flow rate at turbine inlet	1350	t/h

The steam produced in six steam generators (SGs) is distributed through the individual SG piping lines to the main steam collector. The air operated fast-acting valve and the main steam valve are installed on each SG line for isolation of individual SGs.

Each SG is protected by safety valves (SRVs) connected to an SG piping line at the isolable section. The SG safety valves are set to relieve steam at varying setpoints; for one valve the setpoints for opening and closing are slightly lower than for the other (e.g. 5.68/4.8 and 5.78/5.2 MPa, respectively, for relief and operational SG safety valves in Dukovany units). Full capacity of each valve is approximately 42 kg/s at 5.8 MPa.

From the main steam collector the steam is distributed to two turbines. The main steam collector can be separated for two symmetrical parts by the fast-acting air operated valves normally opened.

Each turbine is supplied with steam by two main steam lines equipped with a block of turbine stop valves and of two control valves. In one of the two blocks for each turbine, there is a bypass valve installed ensuring an easier opening of the main control valves by equalising the pressures upstream and downstream the control valve. From the control valve the steam is led via four pipelines into the HP cylinder.

The steam from the HP cylinder outlet is led via two steam lines into two separators - reheaters where moisture is eliminated and the steam is heated above the saturation temperature. From the reheaters, the steam is led via two conducting steamlines into two parallel working low pressure cylinders.

Each turbine has 8 non-controlled steam extraction lines for heating of various consumers. The extractions (except extractions I and II which operate at low pressures) have check extraction valves installed to prevent a backward steam flow into the turbine in case of its trip or a sudden load rejection

The control valves are controlled by a common turbine controller. The axial forces of the rotors are caught in a single two-side axial bearing located between the HP and LP turbine parts. The temperature of the axial bearing is monitored at several points at both sides of the bearing. The position of the disc of the axial bearing and the elongation of both stators and rotors of the turbine are detected by means of inductive sensors and processed by a complex apparatus.

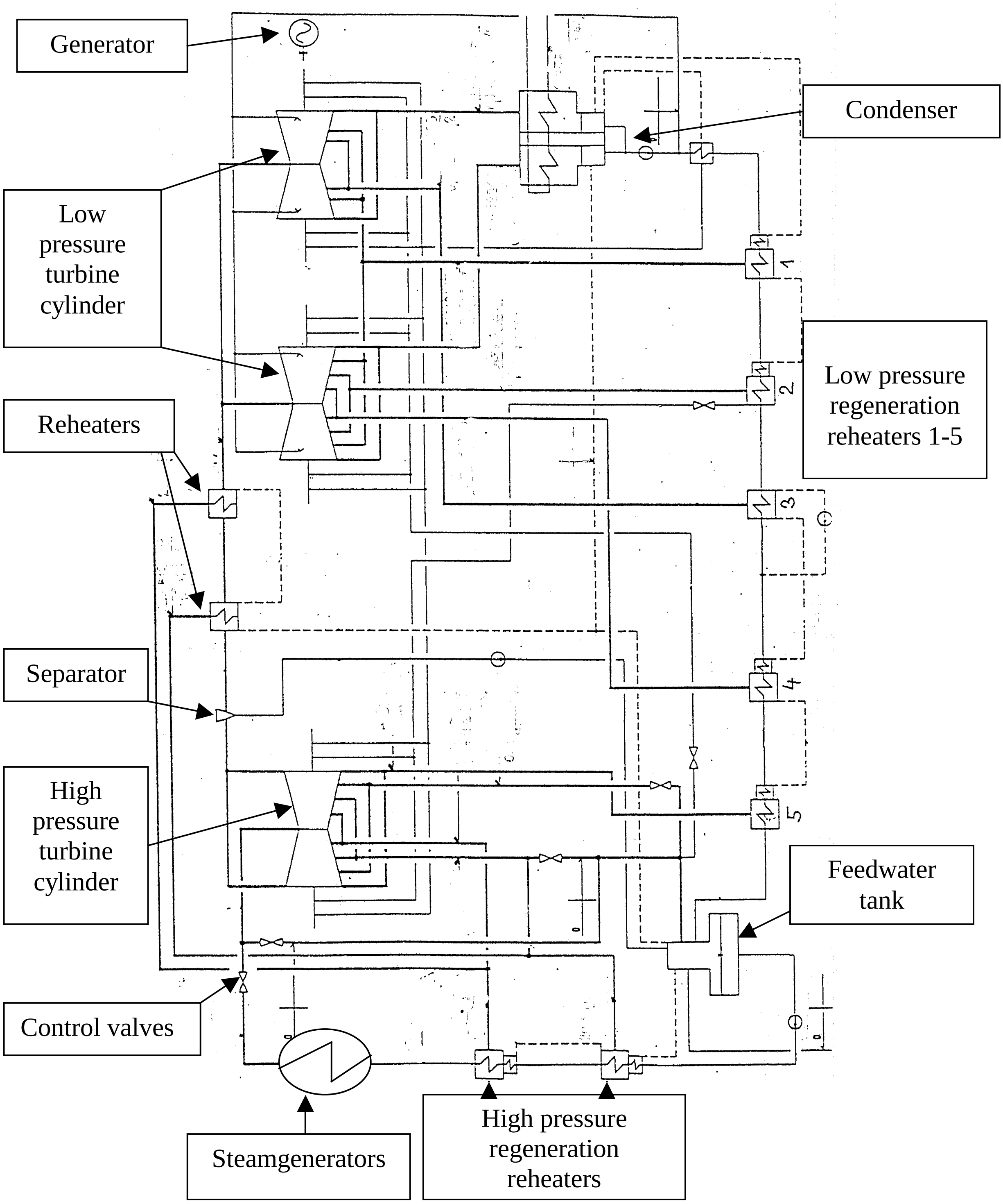

Fig.5. Schematic diagram of VVER-440/213 secondary cooling system (for 1 out of 2 turbines).

The steam expanded in the turbine downstream the LP parts is led into the main condenser. The main condenser is a two-part, two-pressure condenser, both casings are in a single-current arrangement on the side of cooling water, aligned in series. The casings of the condensers are welded by their upper parts to the casings of the LP parts of the turbine. The condenser is protected against a pressure rise by relief membranes (150 kPa).

The coolant from the condenser is pumped to a series of low-pressure feedwater heaters and demineralizers, before entering the deaerators.

Feedwater pumps take suction from the deaerator tanks through a common header. The feedwater is pumped to a common head line, which is split into two trains, each containing three preheaters. Both trains are connected to a common feedwater collector. From this collector the feedwater is distributed to individual steam generators.

There are several connections from the main steam headers to various steam dump facilities: atmospheric steam dump stations (BRU-A), turbine by-pass valves (BRU-K), and steam process condenser dump stations (BRU-TK).

The BRU-TK facilities are a part of the secondary decay heat removal system that is designed for normal cooldown of the plant, but is also a recommended mean of system cooling down in post-accident conditions. The system operates in feed and bleed operational mode until the primary system temperature of 140°C is reached.

SG safety valves and atmospheric steam dump stations create the secondary pressure control system (SPCS). The SPCS is designed to control the secondary side pressure during accident conditions. The SPCS protects the secondary circuit against overpressure and provides means to remove decay heat during various post-accident conditions.

The number of BRU-A stations differs depending on plant design. There is one BRU-A for each turbine in Dukovany NPP. The capacity of each BRU-A is 55.6 kg/s at 5.4 MPa. The BRU-A stations open and close automatically controlling the pressure set up at required level (in the pressure range 4.68 - 5.41 MPa). Below a specified lower limit for pressure, the BRU-A is controlled manually by the operator.

In addition to the BRU-A stations, each steam line to the turbine is equipped with two turbine steam by-pass BRU-K stations that relieve steam to the turbine condenser. Operation of the BRU-K valves is limited to normal operating conditions

A-2.5 Emergency/auxiliary feedwater system

The system includes four feedwater pumps (4 × 100%) providing feedwater from the deaerator tanks (2 pumps) and from water storage tanks (2 pumps); both subsystems are supplied with electric power from independent trains of the essential power subsystem.
Two auxiliary feedwater trains are connected to the feedwater collector. Auxiliary feedwater pumps are used during the startup or shutdown of the plant, as well as in emergency conditions.

A-2.6 Emergency core cooling system

Three independent high pressure injection systems (HPIS) and low pressure injections (LPIS) trains (3 × 100%) and four passive emergency system (PES) trains are designed with relatively high level of independence, achieved by physical/functional separation of redundant trains and independent support features.

The discharge pipe from each PES accumulator tanks is attached directly to the reactor vessel core flooding nozzle. When the RCS pressure drops below 5.9 MPa, two tanks supply borated water into the vessel lower plenum and two other - to the upper plenum. The driving force to inject the water from the accumulators into the reactor vessel is supplied by pressurised nitrogen. Discharge from the hydroaccumulators is automatically stopped when water level drops below the level of 0.5 m above the hydroaccumulators bottom. Due to high pressure setpoint for the accumulators, the passive system may essentially support the function of HPIS during small and intermediate sized leaks.

Two operational modes of the HPIS operation are possible for model 213: an injection mode and a recirculation mode. During the injection phase the system delivers borated water from the borated water storage tanks into the reactor vessel through the reactor coolant inlet lines. In the second phase, the HPIS pumps water from the reactor building emergency sump, supporting low pressure pumps.

The low pressure injection/recirculation system is designed to maintain core cooling in the second phase of a LOCA when the RCS depressurises below 0.7 MPa. The recirculation mode of the LPIS is designed to permit boron concentration control and long term core cooling after a LOCA.

Two trains of the LPIS share common reactor vessel piping with the core flooding system. The third train is attached directly to one of the six primary coolant loops at the cold and hot leg section between the RCS loop isolation valves and the reactor vessel nozzles. Each of the three LPIS trains is rated at 300 m^3/h at pumping pressure of 0.4 MPa.

After the emergency water tank is emptied, the pump suction is switched to the reactor building emergency sumps equipped with a heat exchanger.

A-2.7 Containment system

The design of the VVER-440 model 213 containment system [A-2, A-6] differs from the reactor containments of non-Soviet PWRs, but the design objective is similar. The containment system consists of accident localisation structures, the bubbler condenser and the reactor building spray system (Fig. 6 [A-2]).

The structure of accident localisation compartments, providing confinement function of the containment system, are designated for the 0.15 MPa overpressure. The reinforced concrete walls of the VVER-440/213 are approximately 1.5 m thick. All walls and roofs of the localisation compartments have internal steel lining.

The accident localisation compartments include a sealed set of interconnected compartments surrounding reactor with its cooling system components and an additional building containing the bubbler condenser. The compartments housing the technological systems constitute a part of the reactor building. Bubbler condenser rooms are located in a bubbler condenser tower, connected to the reactor building by a rectangular tunnel.

The bubbler condenser (Fig. 7 [A-2]) provides passive pressure-suppression function. The bubbler condenser comprises twelve levels of water filled trays. Each level contains 163 trays. Total water inventory inside the bubbler condenser amounts to 1250 m^3. The steam discharged from the break during a LOCA is distributed to each level of suppression trays through the vertical shaft. The steam-air mixture enters each tray from the bottom, flows upward through the vertical weirs and then steam condensation takes place. Air and noncondensated portion of the steam collects above all the trays at a particular level. It is directed through doubled check valves into four receiver volumes called air traps where it is localised. When the pressure in the reactor building compartments decreases below that in the outlet plenum of the tray, a reverse flow of water from the condenser trays is initiated. The water pushed out from the trays is sprayed out by the perforated baffles mounted on the roof of each tray level plenum (passive spray). The outlet plenum of each tray level is connected by two redundant check valves to the vertical shaft of the localisation tower. These valves prevent the initiation of passive sprays in the case of SB LOCA or inadvertent initiation of sprays. The valves are blocked under large break LOCA conditions when overpressure in the reactor building exceeds the prescribed level.

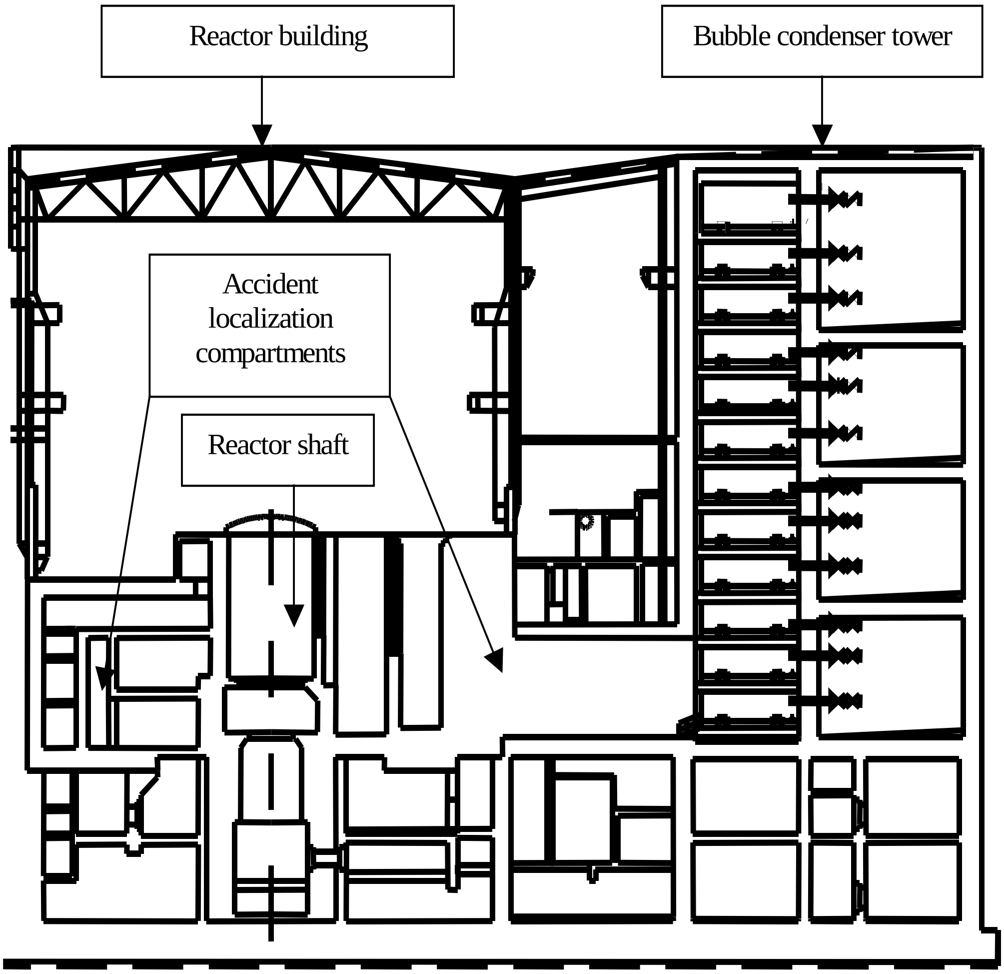

Fig.6. Schematic diagram of VVER-440/213 containment system. (Ref: Rypar, O.:Barbotážní kondenzace páry, Revision 0, December 1998, Archive: MPOTH03-ÚJV-1998-12-006)

The reactor building spray system (RBS) provides water spray to the reactor compartments following a LOCA or a steam line break, to limit the containment pressure and to minimise the release of radioactive materials to the environment.

The reactor spray system is composed of three identical and completely independent trains, each of them with a capacity of approximately 600 m^3/h. The system is composed of the low head pump, chemical addition tank, water ejector pump, piping with their associated valves and a spray header located inside the steam generator room. The pump section line, with the manual valve normally open, is connected to the suction header common for three system - HPIS, LPIS and RBS. The pump draws suction through the ECCS suction header, either from the LPIS tank or from the reactor building sump. The pump charging line contains isolating valves, normally closed.

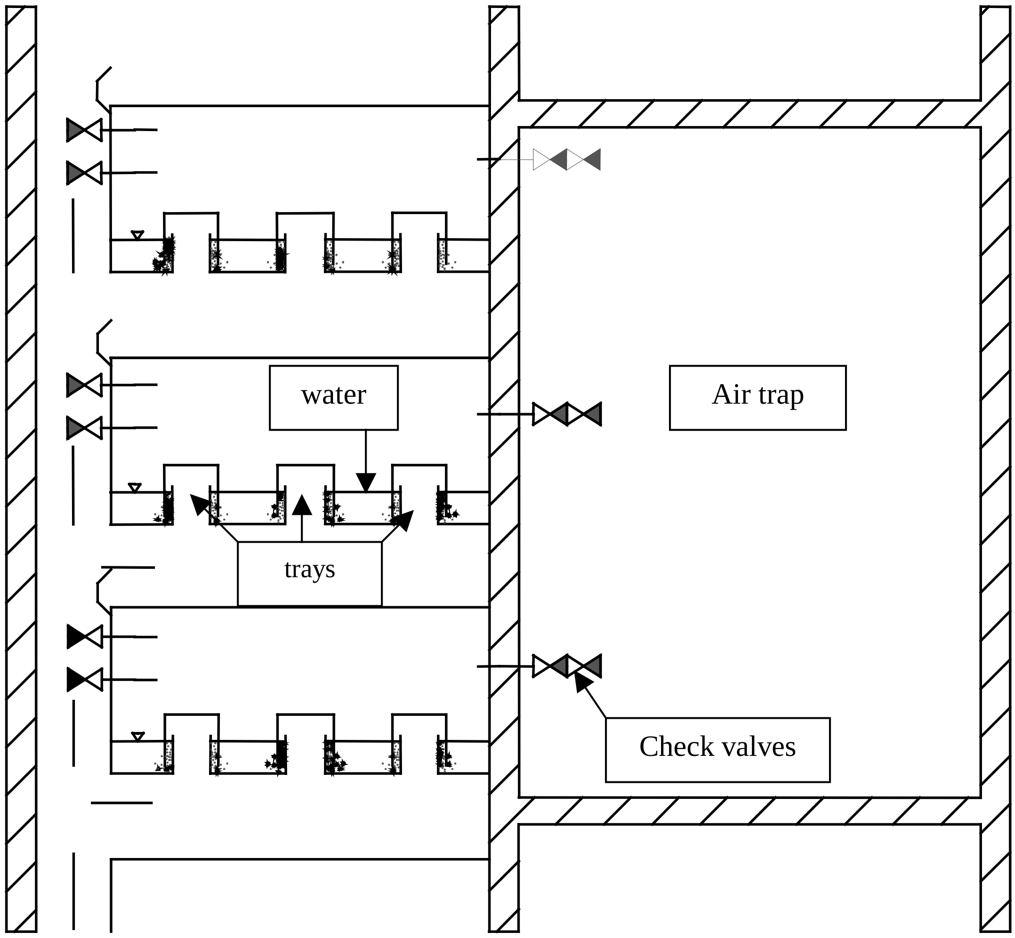

Fig.7. Schematic diagram of the bubbler condenser (3 out of 12 floors). (Ref: Rypar, O.:Barbotážní kondenzace páry, Revision 0, December 1998, Archive: MPOTH03-ÚJV-1998-12-006)

The reactor spray system may be actuated automatically by the protection system or manually. The spray system pumps are switched off when the reactor building pressure is decreased below the prescribed limit. They are switched on again when the pressure reaches the upper limit.

The reactor building pressure suppression system and the reactor building spray system prevent overpressure of the localisation compartments and perform radioactivity removal function across the entire spectrum of LOCAs including a double-ended rupture of the largest pipe (500 mm diameter). Displacement of air from the localisation compartments, followed by condensation of steam due to passive heat removal to cold structures and also due to operation of the spray system, permits the pressure to be reduced to sub-atmospheric values after 10-15 minutes (valid for 500 mm pipe rupture). Continued cyclic operation of the sprays maintains the pressure in the localisation compartments between 0.085 and 0.095 MPa absolute.

Lowering the post-accident pressure in the containment below the atmospheric pressure is a very attractive feature of the design that compensates to some extent other shortcomings.

The relatively large volume of the containment, the large heat capacity of the containment walls, and the considerable amount of water in the pressure suppression trays are positive features influencing post-accident conditions.

A-2.8 Plant control system

The VVER-440 does not have any primary flow control capabilities. The basic control strategy implemented in the control system relies on maintaining constant secondary pressure.

Two main control loops are used for maintaining the neutron flux within acceptable limits and ensuring thermal balance between the primary and secondary circuits: reactor power controller and turbine controller.

The reactor power controller relies on secondary steam pressure and neutron flux signals. The signals from the reactor flux monitor control the position of control rods only when there is a rapid change in the neutron flux. Under steady-state full power operation, the controller responds primarily to changes of secondary pressure.

Turbine controller operation compares the turbine generator limiting output to the actual output of the turbogenerator. This signal controls the turbine valve drive system. The movement of the turbine valve affects the secondary side pressure. This pressure is used as an input for the reactor power controller.

Additional plant controllers are used to control the pressuriser (water level controlling and pressure controlling) and the steam generators (water level controlling).

In addition to the main control loops mentioned above, certain number of signals are generated, based on different limitations with regard to primary pressure, boiling limit, recriticality limit, thermal shock limit, power limit as a function of running RCPs, etc.

A-2.8.1 Unit Power Controller ARM5-S.

The control equipment ARM5-S [A-7] is determined to control unit power in conformance with turbine-generator powers and for stabilisation of reactor power on a given level.
The three basic operation modes are:

- **"T"** stabilisation operation mode of pressure in main steam header with insensitiveness ± p2.
- **"S"** watch operation mode, which assures reactor power decreasing in the case of pressure decrease in primary circuit with insensitiveness ± p2' only.
- **"N"** stabilisation of reactor neutron power on a given power level.

Operation modes "T" and "S" are realised by means of the pressure controller RRT. Operation mode "N" is realised by the neutron power controller RRN. ARM5-S ensures the automatic switch-over from the "N" to the "T" operation mode when the pressure increases by p1.

Maximal allowable power is determined by the form:

$\mathbf{N_{allow} = \boldsymbol{n} / 6}$, where *n* is a number of working main circulation pumps

It means:

when loss of power to 1 MCP occurs, N_{allow} = 83.3 %

when loss of power to 2 MCPs occurs, N_{allow} = 66.7 %

when loss of power to 3 MCPs occurs, N_{allow} = 50.0 %

Turbine Generator Power Controller TVER 03 (Valid for Dukovany NPP only)

A-2.8.2 Turbine Generator Power Controller TVER 03 (Valid for Dukovany NPP Only)

"N" operation mode - power control

The deviation of momentary power from required power is considered as the input value. The power-voltage transposition has a scale 250 MW = 10 V. A positive, a weak pressure corrector and a negative pressure corrector with advanced influence are added to this value. The negative pressure corrector with advanced influence is switched on by loss of power to a MCP. The value obtained in this way goes into the load rate limiter which limits the rate of required power changes so that the rate of load changes does not exceed the setpoint. Negative correctors outputs are added as far as after the load rate limiter. This value continues into the PI controller and into the control valves of turbine.

(For required power, power setpoint can be set in the range of 20÷250 MW.)

"P" operation mode - pressure control

The measured deviation of the main steam header pressure is transposed into voltage (4÷6 MPa = 0÷10 V). This value continues into the PI controller and into the control valves of turbine.

A-2.8.2 Emergency protection system

The emergency protection system monitors parameters associated with abnormal or accident conditions and initiates/controls operation of the proper safely systems.

The protection system [A-7] monitors the following parameters:

- period in source, intermediate and full power range,
- neutron flux in intermediate and full power range,
- core exit pressure,
- coolant temperature at core outlet,
- pressure drop in the reactor loop,
- pressure drop in the reactor core,
- water level in the pressuriser,
- water level in operating steam generators,

- pressure drop in the main steam header,
- position of turbine stop valves,
- power supply to main circulation pumps,
- pressure in the reactor building compartments.

Two main parts of this system are distinguished: the reactor protection system and the engineered safeguards actuation system (ESAS) that provides signals associated with the other safety related plant systems.

Four levels of emergency command signals (AZ-1 to AZ-4) for reactivity control applied in the reactor protection system permit earlier protective actions than the simple reactor scram used in many non-Soviet PWRs and, in consequence, smooth response to minor deviations from the operational state.

A-2.8.4 Reactor Protection System

This system generates four types of emergency command signals, labelled AZ-1 through AZ-4.

1st order emergency command (AZ-1)	This command causes a full immediate and irreversible scram. All control rods fall into the core with the velocity of 0.2 - 0.3 m/s.
2nd order emergency command (AZ-2)	The AZ-2 signal initiates the gradual falling of control rod groups into the core. A next group starts moving when a previous group is 0.5 m above the bottom position. When the signal disappears, only a moving group completes its movement. The velocity of movement is 0.2 - 0.3 m/s.
3rd order emergency command (AZ-3)	The AZ-3 signal initiates a gradual downward movement of groups of control rods with the velocity of 0.02 m/s until the signal disappears. Next group starts moving when previous group is 0.5 m over the bottom position.
4th order emergency command (AZ-4)	This signal simply inhibits upward rod movement. When the signal disappears, rod movement is allowed. When several signals appear simultaneously, a lower order emergency command has higher priority.

In addition to the above mentioned safety related systems, the protection system provides protection/interlock signals to some relevant plant equipment such as turbines, main circulating pumps, pressuriser, certain isolation valves in primary and secondary systems, etc.

A-2.8.5 Automated Protection and Control Signals

Primary and secondary equipment interlocks and protections are activated according to these signals. The vital safety related systems are dependent on ESAS: emergency core cooling system, reactor building spray system, emergency/auxiliary feedwater system and ECCS compartment cooling system.

B-1 GENERAL DESCRIPTION

The VVER-1000 NPP model 320 was introduced in the 1980s. About 20 units with 320 reactor are in operation or in final stage of construction in Ukraine, Russia, Bulgaria and Czech Republic.

The general design features of all VVER-1000 units are similar and some of the main VVER-1000 standard reactor parameters are listed below [B-1, B-2, B-4]:

Rated thermal power	3000	MW
Coolant pressure	15.7	MPa
Coolant loops number	4	
Coolant flow through reactor	84800	m^3/hour,
Core inlet coolant temperature	289.7	^oC
Core outlet coolant temperature	320.0,	^oC
Average campaign duration	7000	f.p.d.
Fuel burnup:		
3-year fuel cycle	40 000	MWd/tU
2-year fuel cycle	27 000	MWd/tU

B-2 SYSTEM DESIGN HIGHLIGHTS

B-2.1 Reactor System

Primary circuit consists of four main circulation loops. Each loop has steam generator and main circulation pump. Horizontal part of the hot legs (from reactor outlet nozzle to SG vertical collector) and cold legs (section connected with reactor inlet nozzle) are located on different elevations: hot leg is 1.8 m higher than cold leg.

Steam generator is designated for heat removal from primary coolant and for generation of dry saturated steam. Steam generator ensures primary circuit cooldown to prescribed temperatures in all design basis regimes. Construction of primary circuit together with steam generator provides primary circuit cooldown also at natural circulation.

Active core of the VVER-1000 reactor consists of 163 hexagonal fuel assemblies. Each fuel assembly has 312 fuel rods with fuel pellets from uranium dioxide, central tube and 18 guide tubes for absorber of control rods. Fuel pellets have central coaxial hole. Material of fuel rod cladding is 99%Zr1%Nb alloy. Stainless steel spacer grids are located with interval 255 mm along the core height.

Fuel assemblies of VVER-1000 differ from those of VVER-440. Opposite to VVER-440 reactors fuel assemblies of VVER-1000 have no shroud. Fuel assembly of VVER-1000 has larger size, more fuel rods and control rods of different design. There are 61 assemblies of control rods in the core. Each assembly includes 18 absorber elements. Absorber material is B_4C. Control rod assemblies are splitted into 10 groups one of which is operational group. This group of control rods is partially immersed into the core during operation.

Fuel Rods Data

outer diameter	9.1	mm
inner diameter of cladding	7.72	mm
weight of cladding	0.445	kg
outer diameter of pellets	7.56	mm
fuel rod pitch	12.75	mm
central void diameter	2.2	mm

Reactor includes the following equipment: reactor vessel, internals, upper block, drives of control rods, trains of neutron measurements, active core.

B-2.2 Reactor Vessel Description

The reactor vessel is a cylindrical high pressure container where the fuel assemblies and vessel internal equipment is situated.

There are eight branch pipes (D=8500mm) in two levels (four at each level) in the reactor vessel. The branch pipes are connected with the four primary circuit's loops.

Two additional pipes (D=300mm) connect each level with the Emergency Core Cooling System. At each high level branch pipe, another pipe (D=250mm) for the pulse lines of the control systems is provided.

The pulse lines to the branch pipe with D=250mm are nine: three couples for three independent measurements of the reactor core pressure drop; one couple for level measurement when the reactor is in shut down and one pipe for the absorber concentration measurement.

B-2.2.1 Main Reactor vessel specifications are as follows:

length	10880	mm
diameter, external on the flange	4570	mm
diameter, external on the cylindrical part	4535	mm
maximum dimension on the branch pipe section	5280	mm
thickness of the cladding layer	7	mm
reactor vessel weight	304	t
reactor vessel material	15x2	HMÈA

Axial cross section of the reactor vessel with the internal components and the overall layout of the primary circuit components are presented on Figs 1 and 2.

B-2.2.2 Volumes and masses

Total volume of the downcomer and the lower plenum	34	m^3
Total volume of the upper plenum	61.2	m^3
Coolant volume in the active part of the core	14.8	m^3
Total volume of the vessel	110	m^3
Heat transfer surface of the core	5 130	m^2
Mass of fuel in the core	80 100	kg
Mass of cladding in the core	22 630	kg

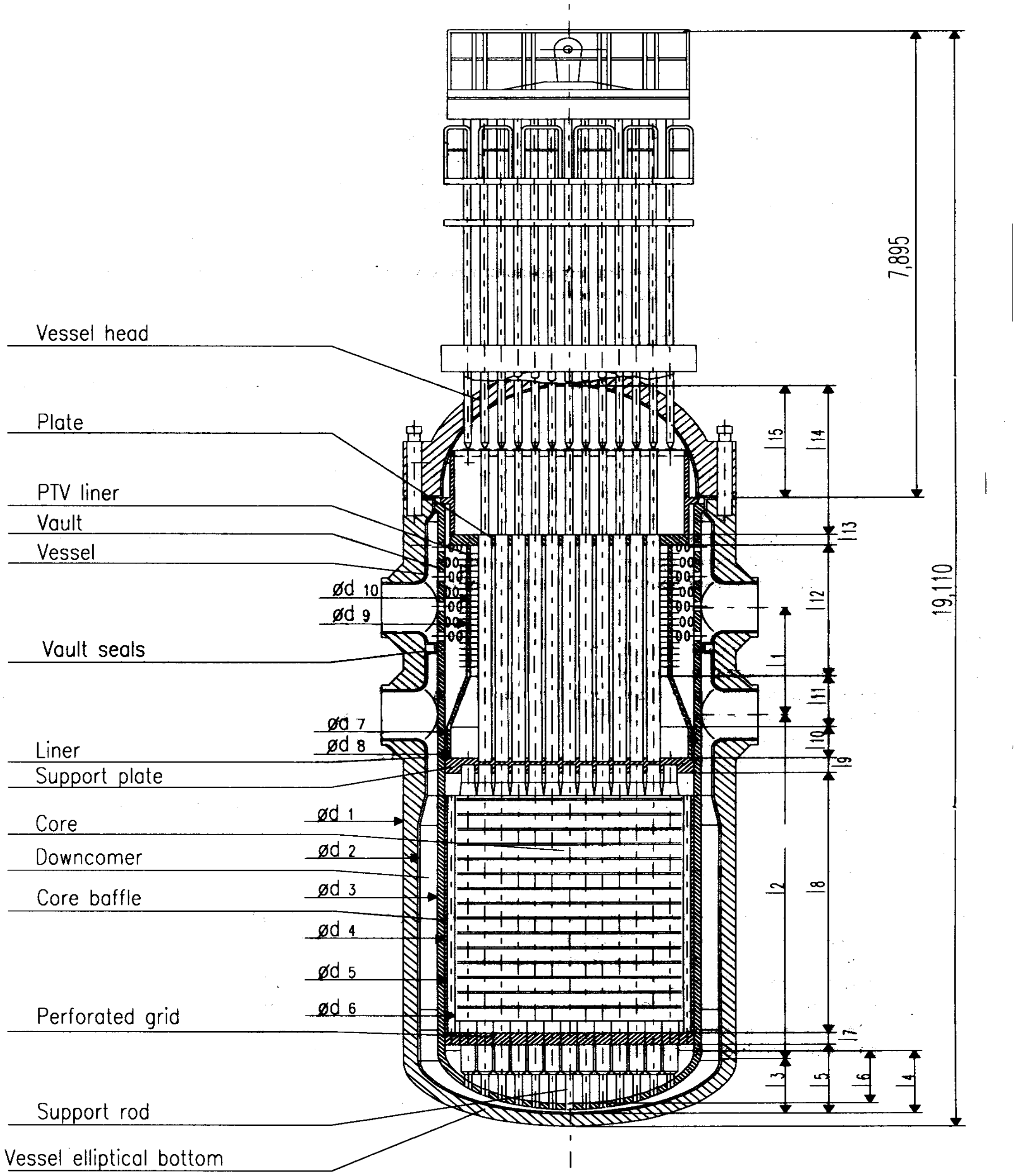

Figure 1: Reactor pressure vessel longitudinal section

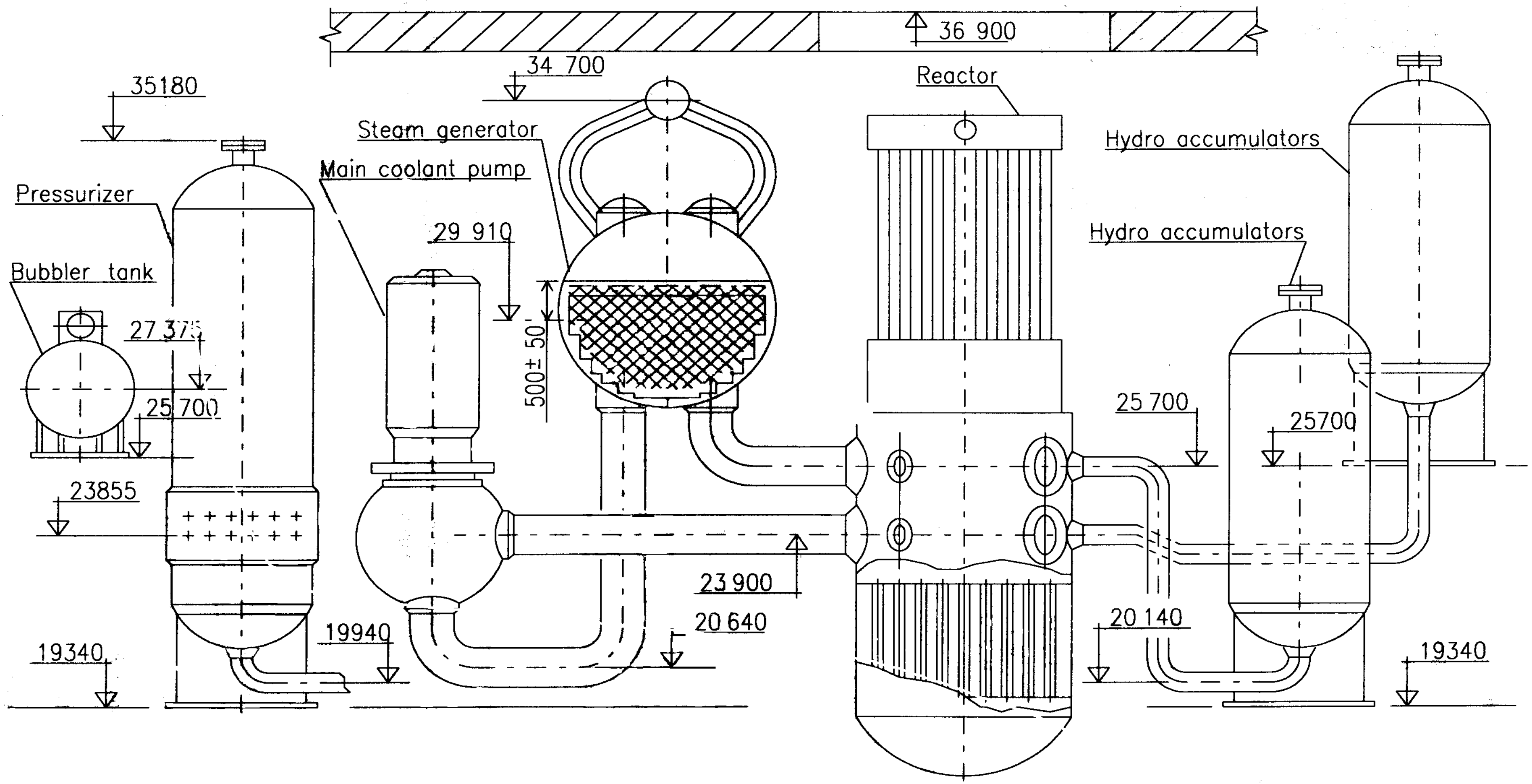

Figure 2: Front view of reactor with altitude marks

B-2.2.3 Geometric dimensions (in mm, Fig 1)

l_1	1800
l_2	5950
l_3	1130
l_4	1757
l_5	3550
l_6	6870

B-2.2.4 Diameters (in mm, Fig.1)

d_1	4635
d_2	4136
d_3	3610
d_4	3490
d_5	2950

B-2.2.5 Weights

upper block	152 000	kg
Zr alloy in the core	20 900	kg
heads and tails of fuel assemblies	12 200	kg
spacer grids	1800	kg

B-2.2.6 Characteristics of materials

density of cladding material (at 20 °C)	6 550	kg/m^3
density of the vessel material (at 20 °C)	7 860	kg/m^3
density of the loop material (at 20 °C)	7860	kg/m^3
thermal conductivity of the gap	32.0	W/cm.K

B-2.3 Reactor coolant system (RCS)

Each RCS loop includes a horizontal steam generator, a main circulating pump, four hot and cold legs. Four primary coolant loops have common flow paths through the reactor vessel, but are otherwise independent in operation.

B-2.3.1 Main circulation pump is vertical one-stage pump with the following main parameters:

Capacity	20 000-27 000	m^3/h
Head	0.74 - 0.54	MPa
Suction side pressure	15.3	MPa
Primary circuit water temperature	300	°C

Main circulation pump is capable to provide special law of flow decrease at accidents with loss of off-site power.

B-2.4 Primary Coolant Pump Data

B-2.4.1 Nominal Data

Heat generations	3640	kW/pump
Nominal speed	995	Rpm
Loss coefficient of stopped pump		
relative to the velocity in the loop		
positive flow direction	27.0	
reverse flow direction	32.7	
Nominal torque		
hot water flow	47 500.7	Nm
cold water flow	64680	Nm
Internal rotor torque	73600	Nm
Power consumption		
hot water flow	5100	kW
cold water flow	3640	kW

B-2.5 RCS Geometry and Hydraulic Data

B-2.5.1 Hot Leg

length	10124	mm
inner diameter	850	mm
outer diameter	990	mm
length from reactor nozzle		
to pressurizer surge line	6860	mm
volume	5.74	m^3/loop
loss coeff. (hot leg without outlet nozzle)	0.41	

B-2.5.2 Cold Leg

length	26 600	mm
inner diameter	850	mm
outer diameter	990	mm
length from p.5 to p.1 (Fig 2)	15 700	mm
volume without MCP	15.07	m^3/loop
volume of MCP	3.0	m^3
loss coeff.. (hot leg without outlet nozzle)	0.88	

B-2.6 Pressurizer Data

A steam pressurizer is connected to the hot leg of the loop No 4 through a pipe (D 426/40mm) of 18000 mm length.

Pressurizer spray pipeline is connected to the cold leg of the loop No 1. Spray can be also performed through makeup pipeline. A set of heaters is installed in pressurizer. The heaters are intended to keep pressure in case of its decrease. The steam dome of the pressurizer is connected to bubble condenser by steam dump pipelines. Steam removal is carried out through three safety valves or through emergency gas removal system.

B-2.6.1 The main pressurizer parameters are as follows:

Total volume	79	m^3,
Nominal water volume	55	m^3
Total height	15 910	mm
Inner diameter	3 000	mm
Outer diameter	3 330	mm
Dry weight	191.5	t
Number of heater blocks	28	
Power per heater block	90	kW
Working temperature	346	°C
Design pressure	17.7	MPa
Working pressure	15,7	MPa
Number of pressurizer relief valves (one control and two operational)	3	
Time for opening/closing of PORV	1	sec
Flow rate through one PORV	180	t/h.

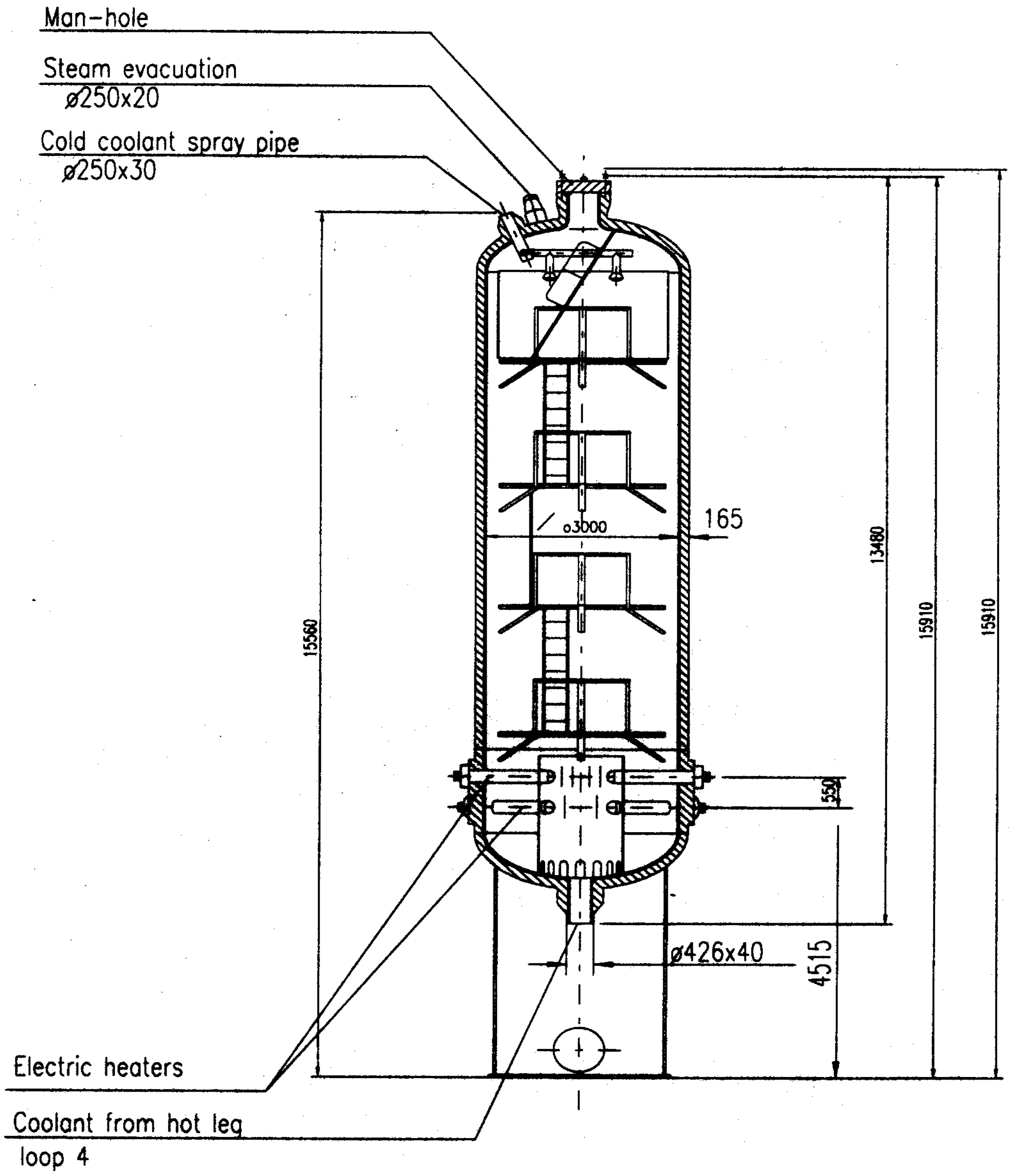

Figure 3: Pressurizer longituninal section

B-2.7 Secondary system

The secondary circuit includes feedwater system, system of fresh steam, turbine generator and condensate system.

B-2.7.1 Steam Generator Data

Steam Generator of the ¤VVER-1000 type is horizontal with built-in separation equipment and practically horizontal heat transfer tubes located in the vessel. Mid of each tube is approximately 20 mm higher than ends of tube.

This SG, at normal working conditions, provides 750 MW heat power removal.

The heat exchange surface is made out of pipes (D 16x1.5mm). The SG's vessel and branch pipes of diameter 500, 850 and 1200 mm are of austenite steel.

The main steam generator parameters are as follows:

Thermal power	750	MW
Primary circuit water flow rate	21200	m^3/h
Primary circuit water outlet temperature	289.8	°C
Primary circuit water inlet temperature	320.1	°C
Primary circuit water working pressure	15.7	MPa
Primary circuit water design pressure	17.7	MPa
Primary circuit water design temperature	350	°C
Saturated steam pressure in the SG	6.3	MPa
Secondary circuit design pressure	7.9	MPa
Feedwater temperature at nominal power	220	°C
Emergency feedwater temperature	5-50	^oC
Steam humidity after SG	< 2%	
Secondary circuit hydrotest pressure	24.5	MPa
Average logarithm temperature head	24.7	°C
Specific heat flow	536 300	kJ/m^2 °C
Heat transfer coefficient	6.03	kW/m^2 °C
Theoretic necessary heat exchange surface	5040	m^2
Real heat exchange surface	6115	m^2
Number of heat transfer tubes	11 000	
Average length of the tubes	11.1	m
Outer/inner diameter of the tubes	16/13	mm
Total cross section area of the tubing (primary side)	1.46	m^2
Cross section area of the collector	0.546	m^2
Nominal secondary side level	2.55	m
Upper level of the tubing	2.19	m
Height of FW inlet nozzle	3.15	m
Equivalent hydraulic diameter (secondary side)	17.4	m
Dry weight	322	t
Water volume in SG	127	m^3
Primary circuit water volume in SG	21	m^3
Nominal steam flowrate	408	kg/s
Number of collectors	2	
Collector volume	2.4	m^3

B-2.7.2 Feed Water System

Main feedwater system includes two feedwater turbine pumps. Auxiliary feedwater system includes two electric pumps for start-up and shutdown regimes. Two feedwater tanks are connected both to main and auxiliary feedwater pumps. Main and auxiliary feedwater delivery is performed through pipeline with internal diameter of 400 mm. Emergency feedwater is supplied separately through tube with 150 mm internal diameter. There are nozzles for permanent and periodical letdown on each steam generator.

Main Feed Water Pump Parameters

Flow rate	3760	m^3/h
Inlet pressure	26.5	MPa
Power	9130	Kw

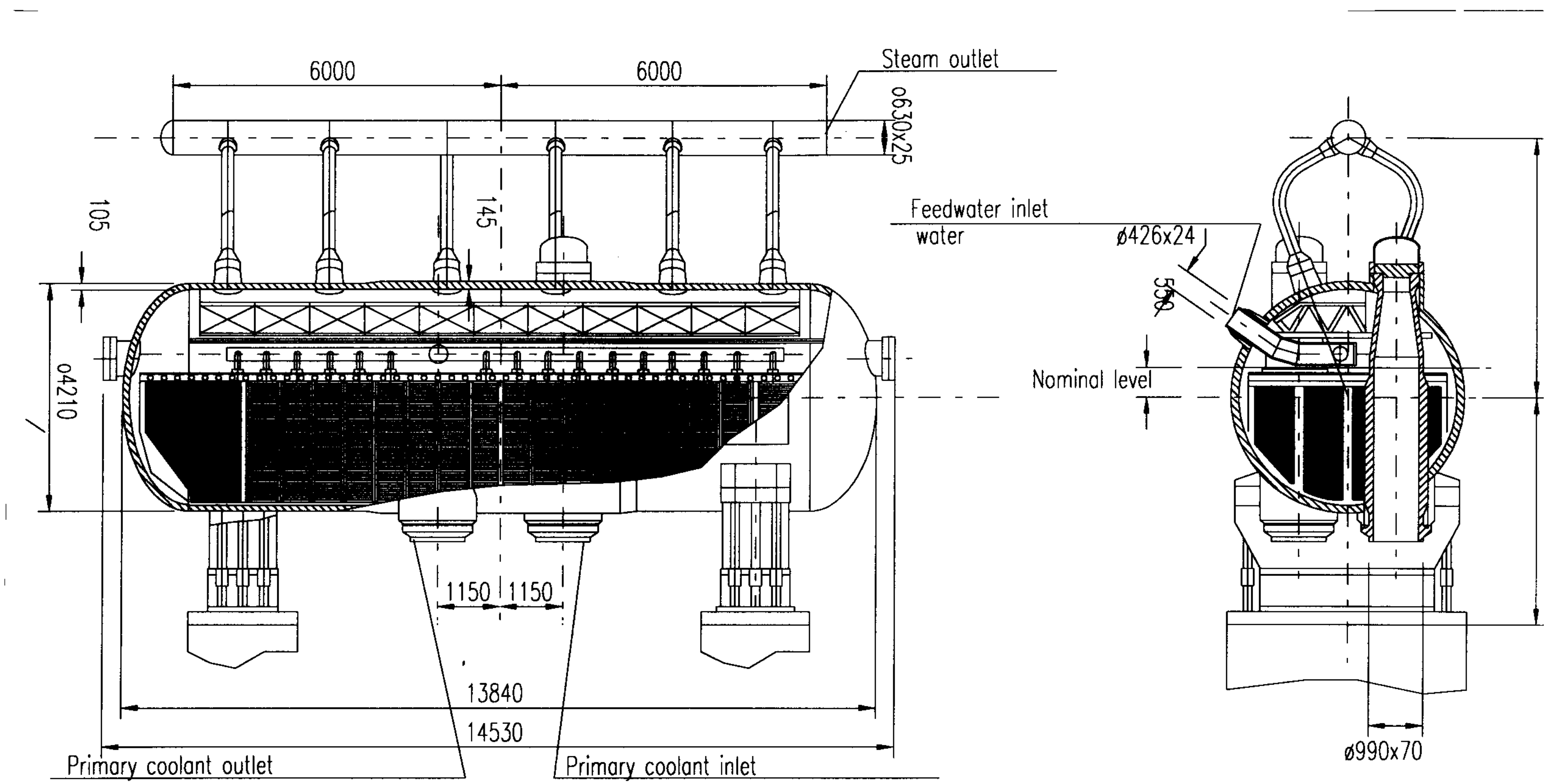

Figure 4: Steam generator PGW-1000 M

B-2.8 Secondary side heat removal system

System of fresh steam includes four steam generators, steamlines, protection and control systems and turbine generator with power of 1000 MW. Steamlines (internal diameter of 600 mm) are connected through main steam collector to equalize secondary pressure. On each steamline there is quick acting isolation valve, two safety valves and quick acting steam dump valve BRU-A. Through the BRU-A dump valve the steam is removed to the environment. Four-channel BRU-K steam dump device is connected to the main steam collector. Through the BRU-K steam is removed to the turbine condensers.

The condensate system is intended for removal of condensate from turbine condensers to the feedwater tanks.

Turbine Steam Bypass Valves (BRU-K)

Steam Dump to the Atmosphere Valves (BRU-A)

Steamgenerator Relief Valves (RV)

Number of BRU-K	4	
Number of BRU-A	4	
Time to fully open/close of BRU-K and BRU-A	15	sec
BRU-K/BRU-A flow rate	900	t\hr
SG relief valve flow rate	1100	t\hr

B-2.8.1 Turbine Basic Technical Features:

Turbine basic parameters:

Nominal power	1000	MW
Maximum power	1103	MW
Number of feed water low pressure heaters	3	
Maximal flow of fresh steam	5870	t/h
Fresh steam pressure	5.89	MPa
Fresh steam temperature	274.3	˚C
Steam humidity	0.5	%
Pressure after intermediate re-heating	1.142	MPa
Steam temperature after intermediate re-heating	250.0	°C
Feed water preheated temperature	224.9	°C
Condenser design pressure	4.0	kPa
Number of preheated steam bleedings	7	
Speed	1500	rpm
Turbine weight	2990	t

The make-up system is designed to change the neutrons chemical absorber concentration and thus to regulate the reactor reactivity (to compensate the reactivity changes during fuel campaigns, reactor star-up and shut down). The system has to assure also:

- the required coolant quantity, during the whole normal operation;
- non organised coolant leakages compensation;
- purification and return to the primary circuit the organised coolant leakages;
- sealing water to the Main Coolant Pumps.

The system is designed according to the following requirements:

- special technical measures were taken in order to avoid pure water inlet to the primary circuit;
- to assure the necessary negative reactivity at reactor shut down regimes;
- to assure reliable system's work during all normal operation regimes;
- in case of accidents, the boric concentration in the make-up system has to be ever higher then the concentration in the primary circuit.

There are three subsystems included in the make-up system:

- Primary circuit blowdown system;
- Blowdown and make-up water de-aeration system;
- Main Coolant Pumps water sealing system.

All subsystems are connected with each other and also there are connections to the drainage and coolant organised leakages system.
Primary makeup pumps are able to deliver water to the primary circuit in the range of pressure from 0 to 17.6 MPa. Nominal flow rate of one pump is 60 m3/hr.

B-2.9 Emergency feedwater system

The system (TX) is intended for feedwater delivery into steam generators at accidents connected with loss of off-site power and with failure of normal feedwater system. The basic layout of the system is given on Fig.3.

The system includes three tanks of desalted water (volume of each tank is 500 m^3) and three emergency feedwater pumps, pipelines, shut-off and control valves. Flowrate of water supplied (150 m^3/hour at secondary pressure 6.3 MPa) ensures decay heat removal. Emergency feedwater pumps are connected to steam generators in such a way that two pumps supply four stem generators (each pump always feeds two different steam generators) while the third emergency feed water pump supplies demineralised water to all four steam generators.

Two steam generators fed by at least one pump are sufficient for the cooling of the primary circuit.

B-2.10 Emergency core cooling system

B-2.10.1 High pressure emergency core cooling system

The system is intended for delivery of boron solution into primary circuit at LOCAs for increase of boron concentration in primary coolant and in case of makeup system failure. The basic layout of the system is given on. Fig.4.

Each of the three system trains includes emergency boron injection pump and high pressure boron injection pump. Each of these pumps supplies water from separate tank with volume of 15 m^3 of emergency storage of boron acid concentrate. The tank for emergency boron injection pump is located inside the containment and the tank for high pressure boron injection pump is located outside the containment. Besides that there is a containment sump with volume of 630 m^3. After emptying the 15 m^3 tanks the pumps of the system take boron solution from the sump.

High pressure boron injection pump gives small flowrate (6.3 m^3/hour) but it operates at pressures up to 17.8 MPa. Emergency boron injection pump allows to ensure boron solution flowrate not less than 130 m^3/hour at pressures below 8.8 MPa.

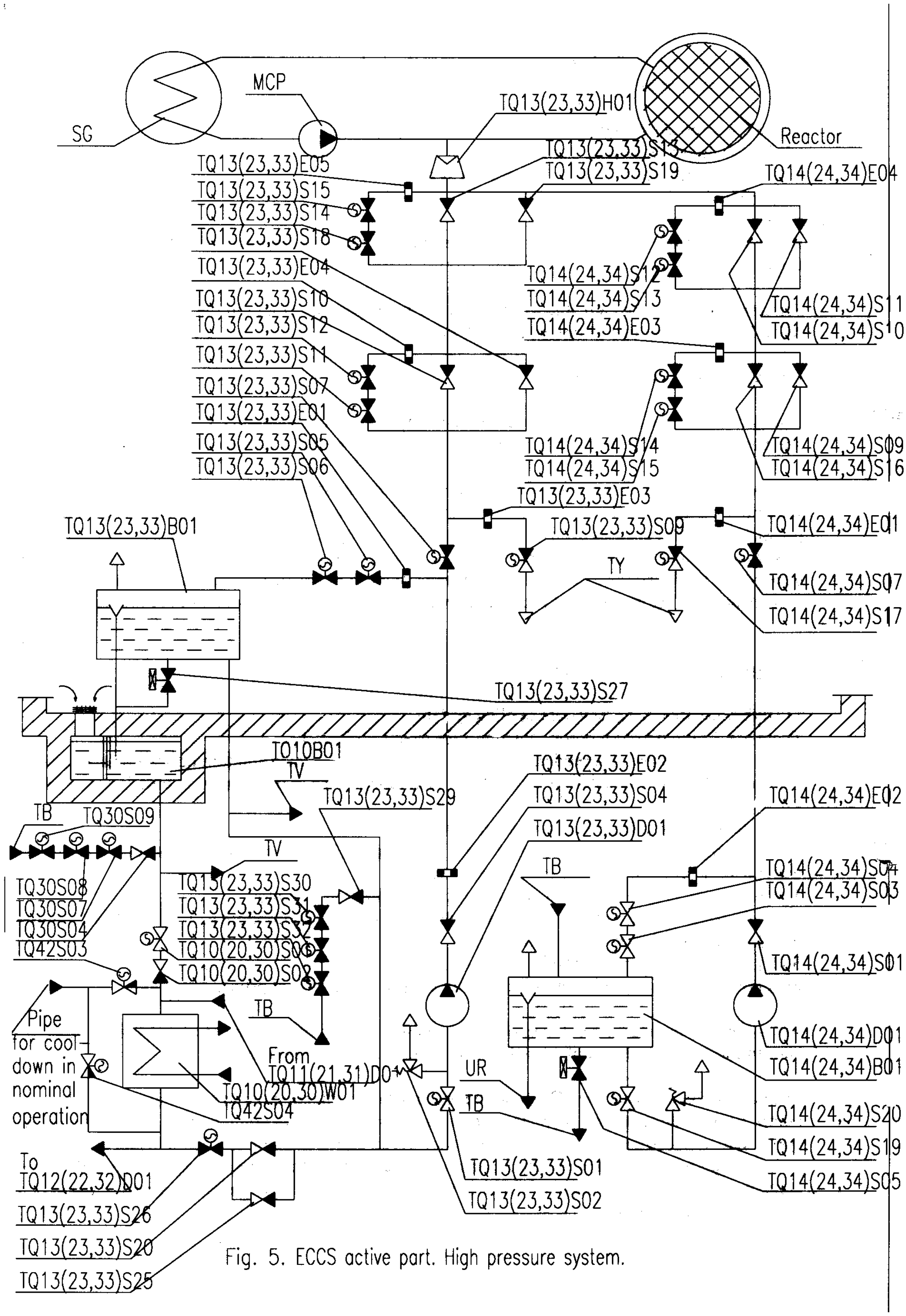

Fig. 5. ECCS active part. High pressure system.

Figure 5: ECCS active part. High pressure System

High pressure injection system (HPIS) Data

number of independent subsystems	3	
water temperature	55-60	°C
boric acid concentration	40	g/kg

B-2.10.2 Passive part of emergency core cooling system

System of hydroaccumulators is designated for quick refill of reactor core with borated water on initial stage of MBLOCAs and LBLOCAs.

Borated water is supplied into reactor from four independent tanks at primary pressure below 5.9 MPa. Cutting in of the accumulator pipelines is performed right into reactor vessel in such a way that two accumulators deliver water into lower plenum and two accumulators deliver water into upper plenum. Each accumulator is connected to the reactor by 300 mm diameter piping that is fitted with two quick-acting valves and two check valves. The quick-acting valves are always open during normal operation. Pressure in accumulators is kept by nitrogen. To prevent entering nitrogen into reactor at decrease of accumulator water level automatic closure of the quick-acting valves occur.

Hydroaccumulator parameters:

Total volume	60	m^3
Water volume	50	m^3
Gas pressure	5.9	MPa
boron concentration	16	g/kg
Temperature	35-60	°C
Solution level above the bottom	6.500	m

B-2.10.3 Low pressure emergency core cooling system

Low pressure emergency core cooling system (TQ12) is intended for removal of core decay heat at significant depressurization of primary circuit and in regime of planned cooldown. The basic layout of the system is given on Fig.5.

Each of the three system trains includes emergency cooldown pump, heat exchanger, pipelines and valves. At accidents the system is connected to the containment sump.

The system ensures boron solution delivery flowrate not less than 250-300 m^3/hour at primary pressure of 2.2 MPa and not less than 700-750 m^3/hour at primary pressure of 0.1 MPa.

Low pressure injection system (LPIS) Data

number of independent subsystems	3	
water volume nominal	630	m^3
water temperature	50	°C
boric acid concentration	16	g/kg

ECCS Heat Exchanger Data

heat exchange area	935	m^2
flow rate of cooling water through pipes	833	kg/s
flow rate of cooled water	417	kg/s

cooling water temperature inlet/outlet	40/70.5 °C
cooled water temperature inlet/outlet	150/90 °C

B-2.11 Containment system

In case of accident with release of radioactive materials (at fuel rod damage and leakage from primary circuit) the system serves as a barrier which limits or prevents the spreading of radioactivity into the ambient environment so that an exceeding of the admissible values cannot occur.

The system consists of reinforced concrete building with cylindrical hermetic vessel of 45m diameter and 54 m height. All parts of the system (building constructions, bushings, hermetic locks) must maintain their full tightness, strength and their functional properties within the entire extent of the working and accident parameters (including MDBA conditions when hermetic space temperature reaches 150 °C and pressure reaches the value of 0.49 MPa). Total volume of the containment is 60000m^3.

B-2.11.1 Control and protection system

The system is intended for reactor control at start-up, operation, planned or emergency shutdown. The system ensures cut-off of chain reaction or power decrease at beginning of accidental situation. Besides automatic system actuation it can be started by the personnel in the main control room.

Fulfilment of system functions is performed by means of displacement of control rods in the core on the signals which are formed depending on combinations of nuclear steam supply system parameters. The following types of protections can be singled out: emergency protection, preventive protections of two types and accelerated preventive protection.

Emergency protection is performed by dropping of all control rods into the core during time interval not more than 4 s.

Preventive protection of the first type is performed by successive movement in certain order of all groups of control rods downwards with velocity of 2 cm/s.

Preventive protection of the second type means prohibition to withdrawal of the control rods from the core. Movement downwards is permitted.

Accelerated preventive protection is performed by quick partial power decrease by means of dropping of one group of control rods into the core.

The scram signals only are given below [B-3]:

- reactor period less than 10 sec;

- neutron flux level (in the power range) 107%;

- difference between saturation temperature and maximum coolant temperature in any of the four loops 10 °C;

- liquid level in pressuriser less than 4000 mm;

- steam line depressurisation rate in any of the SGs more than 0.15 MPa/sec and steam line pressure less than 5.1 MPa;

- steam line pressure less than 4.9 MPa and primary and secondary saturation temperature difference more than 75 °C;
- decreasing MCP P from 0.4 to 0.25 MPa for less than 5 sec;
- MCP pressure difference less than 3.5 MPa for 3.5 sec;
- loss of power for two out of three or four MCP in operation for a period more than 70 sec at reactor power more than 5%;
- SG liquid level more than 500mm above nominal with the respective HPH in operation;
- MCP power supply frequency less than 46 Hz;
- containment pressure more than 0.03 MPa
- primary pressure more than 17.65 MPa
- coolant temperature at any place in the hot legs more than nominal plus 8 °C;
- pressure in the upper plenum more than 13.7 MPa and reactor power more than 75%;
- secondary SG pressure in any SG more than 7.84 MPa;
- UP pressure less than 13.7 MPa and hot leg temperature more than 260 °C.

B-2.11.2 Emergency gas removal system

The system is intended for removal of gas-steam mixture from primary circuit (reactor, pressuriser, steam generators) at accidents connected with core dry-out and beginning of metal-steam reaction.

The system consists of pipelines that connect main equipment with bubble condenser and valves on pipelines.

The system can be used for forced decrease of primary pressure at accidents.

B-2.11.3 Spray system

Spray system is intended for accident localization by means of condensation of evaporated part of coolant ejected into containment. The function of the spray system is to decrease containment pressure as soon as possible to the value of environment pressure with injection of cold borated water. Spray system is used also to fix iodine. For this purpose a special solution is delivered to suction side of spray pumps.

Each of the spray system three trains consists of a pump with capacity of 700 m^3/hour, water-jet pump for special solution delivery and tank of inventory of this solution. Spray pumps take water from the containment sump. Each train of the spray system has 25 nozzles through which spray water is pulverized with certain degree of dispersion under the containment cupola.

Spray System Data

pump flow rate	700	m^3/h
pressure	1.37	MPa
water temperature	10-100	°C

CO-1 Description of Phenomena

This appendix provides descriptions of phenomena identified in the VVER cross-reference matrices. VVERs are pressurised water reactors and their system response and phenomena in accident conditions are similar to PWRs. Work summarised in this report expands and complements the findings and evaluations of phenomena in PWRs with phenomena typical for VVERs. Therefore, the following description of the phenomena are based on phenomena descriptions published in the "CSNI Integral Test Facility Validation Matrix for the Assessment of Thermal-Hydraulic Codes for LWR LOCA and Transients" (NEA/CSNI/R(96)17, July 1996) report. Differences between VVER and PWRs are highlighted and specific phenomena or system response are addressed. For detailed description of underlying thermal-hydraulic phenomena the reader is directed to the report "Separate Effects Test Matrix for Thermal-Hydraulic Code Validation", Volume I "Phenomena Characterisation and Selection of Facilities and Tests" (NEA/CSNI/R993)14/PART1.

This appendix is organised in three chapters addressing large break LOCA, intermediate and small break LOCA, and transients and accident management processes. For each identified phenomenon a description and discussion of safety relevance is provided. The descriptions provide common basis for understanding and interpretation of the phenomena. They indicate were in the reactor system, and under what conditions the phenomena are likely to occur. The descriptions provide also an aid to the identification of suitable and representative experimental data for inclusion in the validation matrices. The discussion on relevance to reactor safety highlights the impact of the individual phenomena (in isolation as much as possible from other phenomena) on safety parameters such as pressure or fuel cladding temperature. Identification of the relevance to reactor safety is important for assessment of the adequacy of the experimental database.

C1-1 Break flow (B1.1)

C1-1.1 Description of the phenomenon

When a break occurs in a separating wall structure between a high and low pressure system the flow through the break will be dependent on conditions upstream the break and on the break area and shape. Critical flow through a break is similar to critical flow through a nozzle, but because the geometry of the break can encompass any shape, location and size from a small crack to a complete 200 percent guillotine break in a flow pipe, multidimensional effects and losses are important but also difficult to account for accurately.

For a guillotine break in a large pipe the flow will pass across the entire break area cross section in a rather homogeneous way but for a small break the geometry of the break and the upstream flow pattern close to the break is of great importance. At low flow rates stratification effects upstream of the break are crucial to the flow rate. Under such conditions, the orientation of the break relative to the liquid-vapour interface governs the actual flow rate and this is also likely to change during the course of drainage through the break. Two-phase flow processes influencing the flow rate through a break in a horizontal pipe include upstream flow regime transitions, liquid entrainment in the break flow and vapour pull-through.

In a constant cross-section pipe choking is expected to occur at the pipe exit. In this case the flow rate is still influenced by entrance losses and wall friction in the pipe since these produce a fall in pressure before the choke point is reached. Thus length to diameter is an important parameter in determining critical flow from pipes. Pipe length is also important in determining the degree of thermal and mechanical non-equilibrium in critical flow.

Measurement of break flow in large Break LOCA is evaluated by instantaneous measurements of density, mean velocity and/or fluid momentum. The mass flow rate is obtained by proper combination of these measurements with some assumptions on the flow pattern. In small break experiments catch tanks are generally available to give the discharged mass and by deriving this quantity the evaluation of the mass flow rate is obtained. This phenomenon has no specific in comparison with PWRs. Relevance to nuclear reactor safety.

The critical flow through breaks is a very important element in analysing plant behaviour during the course of the complete spectrum of LOCAs. The break flow determines the depressurisation rate of the system and the time to core uncovery which in turn are of, major concern for when and how different mitigation auxiliary systems will be initiated and function. Because of the empirical nature of critical flow models as well as the arbitrariness in location, orientation, shape and size of a break, the licensing requirements of a plant include that a spectrum of postulated breaks is analysed. It is important in such analyses that the envelope of the influences from the break conditions can be found.

For a pipe break the pipe involved could be a main coolant pipe, in which case complete severance leads to a large break LOCA, or a pipe connected to the main coolant loop (e.g. an ECCS line) which could lead to an intermediate or small break LOCA, or broken steam generator tube.

C1-2 Phase separation (B1.2)

C1-2.1 Description of the phenomenon

Vapour and liquid phases remaining at rest or flowing with minimal inertial effects in vertical and horizontal channels, which means when flow velocities and accelerations are low, tend to separate owing to gravity force driven by the different densities of the two phases. Gravity leads to separation or stratification of the liquid below the vapour. The situation arises in the reactor pressure vessel, when dynamics of the blowdown phase during a large- or intermediate-break LOCA have ended. It also occurs in the primary system piping during a small break LOCA, when the main coolant pumps are tripped. Flow stratification was found to occur in LOFT horizontal pipes at reduced primary inventory even while the primary pumps were still running. Essentially, the only physical aspect to be considered in modelling this situation is the time required to reach a stable condition starting from mechanical non-equilibrium (e.g. liquid in the upper part of a voided system). This will depend on vapour bubble size, geometry and drag (form, interfacial, wall), which control the velocity of rising bubbles in the liquid and the stability of the separated flow interface.

The situation is much more complex in a flowing system. Various flow configurations may occur as a function of system parameters (e.g. equivalent diameter, flow area, geometric flow path, wall heat .transfer, etc.), of the fluid thermalhydraulic parameters (e.g. pressure, liquid and steam velocities, void fraction, etc.) and of the fluid properties. The well known flow regime zones are the result of the interaction of the parameters mentioned. The formation of a mixture level is also a consequence of the interaction between the phases in flowing and wall-heated two-phase fluid systems. Finally, special flow regimes may occur in the case of strong mechanical and thermodynamic non-equilibrium such as in the case of abrupt area change or reflood.

Several experiments investigating large break scenarios start at the end of blowdown, the refill or the reflood phase. Therefore, the suitability for code validation has to consider the test types performed in the different experimental facilities (see cross reference matrix I). The “transition” of phase separation means the simulation of the whole accident sequence including complete blowdown, refill and reflood, whereas “condition” means the simulation of the phase separation phenomenon under the specific experimental sequence performed, e.g. reflood only.

This phenomenon has no specifics in comparison with PWRs and defining criteria for phase separation may be the same.

C12.2 Relevance to nuclear reactor safety

The phenomenon is relevant to nuclear reactor safety because of the significant dependence of the values of pressure drops and heat transfer upon the flow regimes. The heat transfer from the fuel cladding to the fluid is highly dependent on the location of the mixture level in the core region. Below the level heat transfer can be quite high, resulting in moderate cladding temperatures, whereas the heat transfer above the level can be much lower and cladding temperatures much higher.

The prediction of flow regime may be important in the case of *WWER* (conditions of accidents without scram) in the evaluation of local void fraction, and as a consequence, in predicting neutron flux.

The downcomer water collapsed level is the driving force for core reflood and consequently core cooling conditions. The phase separation in the downcomer influences the water inventory loss during blowdown via a cold leg break due to its influence on the pressure loss mainly in the broken loop and on the critical flow rate at the break.

The collapsed and mixture level in the upper plenum has a significant influence on the break flow in case of a hot leg break. Besides that, phase separation in the upper plenum, hot leg and steam generator inlet header has an impact on the steam binding phenomena.

C1-3. Mixing and condensation during injection (B1.3)

C1-3.1 Description of the phenomenon

Direct contact condensation of steam on cold water is a very efficient heat removal mechanism which often takes place at very rapid heat and mass transfer rates. Violent pressure oscillations due to the rapid condensation of steam and the resulting volume reductions have been observed in several situations when a subcooled liquid is brought into intimate contact with steam. For example, such oscillations take place at the ECC injection points. The mixing of liquid and vapour has an effect on the magnitude of the interfacial area and consequently, in conjunction with temperature differences between steam and water, it determines the condensation efficiency. The mixing is mainly influenced by the void fraction and the mass fluxes of steam and water as well as steam and water temperatures and other properties.

In VVERs ECCS water is delivered directly to downcomer and upper plenum. Besides, in downcomer of VVER-440 reactor the vertical guide fins are installed that reduce mixing of downward flow of ECCS water and upward flow of steam-water mixture. So formation of water plugs in cold legs of VVER RCS is less probable than in PWRs. These features facilitate entering ECCS water into the core from the downcomer. Steam condensation in upper plenum and downcomer of VVER facilitates decrease of pressure inside the vessel and decreases loss of coolant rate. This process also influences phase separation, CCFL and entrainment.

Condensation may be reduced if non-condensable gases, e.g. nitrogen from the accumulators, are present, see “non-condensable gas effects “.

C1-3.2 Relevance to nuclear reactor safety

ECC water delivery from the downcomer and upper plenum injection locations towards the core is essential for effective performance of the ECCS.

Direct contact condensation on the ECC water which is injected may have an influence on the countercurrent flow in the downcomer region, and at the upper tie plate if ECC water is injected also at the hot side of the reactor vessel. Upward flowing steam is reduced by the condensation rate. Consequently, the water downflow rate towards the core can increase. The condensation rate again is dependent on the downward water mass flow rate.

Steam condensation causes a warm-up of the subcooled ECC water which may reduce pressurised thermal shock in the reactor vessel wall in the case of high pressure ECC water injection.

C1-4 2-PHASE FLOW IN SG PRIMARY AND SECOND11ARY SIDE

C1-4.1 Description of the phenomenon

During blowdown phase the SG hydraulic resistance limits flow via long leg of broken loop. This defines the stagnant point position in the reactor vessel. During refill and reflood phase entrained liquid evaporates in the SG tubes that leads to pressure increase in upper plenum (see "steam binding"). These phenomena are similar to that in PWRs, but quantitative characteristics may be different.

Due to termination of steam generation in the horizontal SG secondary side and the level collapsing a part of the tube bundle is dried out and practically does not participate in heat transfer. As concerns for outside surface of submerged tubes the heat transfer from them occurs in conditions of free convection of single-phase liquid.

C1-4.2 Relevance to nuclear reactor safety

Only 2-phase flow in SG primary side is relevant to LBLOCA (blowdown, refill, reflood). It has some effects on the core cooling.

C1-5 Core wide void and flow distribution (B1.4)

C1-5.1 Description of the phenomenon

Core wide void and flow distribution addresses mainly the three-dimensional flow behaviour in the core region of VVERs. Core wide means large scale behaviour of the void and flow distribution within the core.

During normal operation of a VVER the core is completely covered with water. In the case of an accident the water level may drop and the cooling of the core has to be re-established by means the ECC system.

During the blowdown phase some local top-downwater flow occurs from the pressuriser and the steam generator primary sides into the core before the actuation of the ECC systems. The flow the upper plenum is three-dimensional.

Downcomer injection leads to a core refill from the bottom upwards. The combined downcomer and upper plenum ECC injection leads to a local top-down cooling of the core in addition to refilling from the bottom. The upper plenum ECC injection water flows down within preferred flow paths. The precursory cooling in these water downflow locations is higher than in the steam dominated core regions. The precursory cooling may reduce the possible fuel rod damage during the heat-up period.

A three dimensional distribution of water downflow and steam upflow establishes itself above the swell level in the core. After the start of reflood the void fraction and flow distribution determines the further core cooling.

In VVER-1000 these phenomena are similar to that in PWRs, but in VVER-440 there is significant dfference due to boxed fuel assemblies which prevent radial flows within the core.

One of the major sources of uncertainty in current thermal hydraulic computer code simulation is the fact that it is not adequately known for all flow situations, how to describe the local interfacial area density and the momentum, mass and energy transfer rates at those interfaces, irrespective of the computational cell size.

C1-5.2 Relevance to nuclear reactor safety

The cooling of the core by means of available remaining water or by injected subcooled ECC water is the most essential safety aspect. The increase of water inventory by means of ECC injection determines the heat transfer from the fuel rods.

The core wide three-dimensional flow is important for *downcomer and upper plenum* injection during reflood in large break LOCAs . For *such* injection this phenomenon is important for the whole course of an accident including blowdown, refill and reflood .

C1-6 ECC downcomer bypass and penetration (B1.5)

C1-6.1 Description of the phenomenon

ECC penetration means the partial or complete water delivery to cool the core. In VVERs part of the ECC water is injected directly into the downcomer. ECC water flows down the downcomer partially or completely at lower steam upflow or low azimuthal steam flow. Injected water which does not contribute to core cooling because it is lost through the break before reaching the core is called “bypassed”.

Complete ECC bypass occurs if the steam upflow rate is so high that it does not permit any liquid downflow to the core. All the injected ECC water is carried out by the steam flow through the broken cold leg connected to the downcomer. This may be caused by high steam upflow in the downcomer, increased steam upflow due to pressure decrease at the ECC injection location due to condensation, as well as high azimuthal steam velocities from the intact cold legs to the broken cold leg during large break LOCA blowdown (entrainment). The steam upflow in the downcomer may be increased by evaporation of ECC water at the hot downcomer walls (hot wall effect).

UPTF results indicate that the larger the size of a flow passage such as the downcomer, combined with non-uniform geometric distribution of water inlet and flow outlet, the more multidimensional the flow distribution becomes. Fluids of different density and different direction of flow find it much easier to establish their own preferred flow paths in which the flow is predominantly concurrent rather than countercurrent.

In VVERs bypass is reduced due to direct injection into the downcomer. Besides, in VVER-440 reactor there are vertical guide fins that reduce interaction between ECC water and steam-water mixture and also reduce bypass.

C1-6.2 Relevance to nuclear reactor safety

ECC water injected directly into the downcomer has to flow downward through the downcomer in order to refill the lower plenum and subsequently reflood the total core to prevent overheating. This water flow can be limited by steam flowing upwards through the downcomer. Consequently, the core reflood can be delayed. Downcomer ECC bypass and penetration is one of the most important phenomena during large break LOCA refill in VVERs.

C1-7 UP injection and penetration

C1-7.1 Description of the phenomenon

In VVERs ECCS water is delivered not only into downcomer but also into upper plenum. On the upper core plate countercurrent flow of water and steam occurs that hinders ECCS water penetration into the core. In experiments the penetration depends essentially on facility scale. If VVER core is simulated with small number of rods then conditions for ECC water penetration are the least favourable during refill and reflood. If the core is simulated with full-scale fuel assembly then ECC water penetration is much better due to additional degrees of freedom. In real reactor the ECC water penetration conditions are even more favourable.

Steam condensation in upper plenum reduces velocity of steam outflowing from the reactor. This reduces entrainment, makes less steam binding and facilitates upward movement of the quench front.

C1-7.2 Relevance to nuclear reactor safety

ECC water injection into upper plenum as well as downcomer injection has direct influence on cladding temperatures.

C1-8 Countercurrent flow limitation (B1.6)

C1-8.1 Description of the phenomenon

Countercurrent flow through the upper tie plate or the upper core support plate (UCSP) can occur due to water being de-entrained in the upper plenum flowing back to the core region, or because ECC water is injected into the upper plenum.

Countercurrent flows of steam (or steam/water mixtures) upward and water downward play an important role regarding the distribution of coolant in the primary system. In countercurrent flow a limiting condition appears first when the downward flow of liquid is influenced by the upward flow of vapour. Any increase of vapour flux then leads to a reduction in the liquid counter-flow until, eventually, this becomes zero and the incoming liquid may be carried upward with the gas flow. This countercurrent flow limitation (CCFL) tends to occur at flow cross section restrictions, like the tie plate or upper core support plate, where the gas velocity is a maximum.

For steam/water CCFL the situation may be influenced by steam condensation on subcooled ECC water resulting in a reduced steam upflow rate. Steam condensation reduces the vapour flux, allowing a greater liquid flux which in turn increases condensation. This process may lead to a complete water “break through”.

If the upflow consists of a two-phase gas/droplet flow at the CCFL location the momentum of the entrained droplets opposes the liquid downflow leading to more limitation of the liquid downflow rate.

Experiments indicate that the larger transverse scale of test facility, the more free penetration of liquid into the core takes place. E.g., UPTF results indicate that the larger the size of a flow passage such as the reactor vessel upper plenum and core, combined with non-uniform geometric distribution of water inlet and flow outlet, the more multidimensional the flow distribution becomes. Fluids of different density and different direction of flow find it much easier to establish their own preferred flow paths in which the flow is predominantly concurrent rather than countercurrent.

VVER-1000 has no specifics in comparison with PWRs, but VVER-440 has due to boxed fuel assemblies (see also “UP injection and penetration”).

C1-9.2 Relevance to nuclear reactor safety

Either de-entrained water from the upper plenum, or ECC water injected into the upper plenum, has to flow down through the upper tie plate countercurrently to steam generated within the reactor core or to steam generated by evaporation due to depressurisation or to steam expansion due to depressurisation. The steam flows towards the broken loop hot leg. The injected water is needed in the core region to prevent overheating. The limited water delivery into the core can delay the core reflood. Upper tie plate CCFL has a high ranking of importance for *upper plenum* ECC injection during large and intermediate break LOCA refill and reflood.

C1-9 Steam Binding (Liquid Carry over etc) (B1.7)

C1-9.1 Description of the Phenomenon

Droplet entrainment is a form of phase separation and is related to interface friction dependent on the flow regime. Water, can for example, be entrained by sufficiently high steam velocities from the bottom of the reactor vessel and be carried out of the vessel. This water can be lost from the liquid inventory in the pressure vessel.

The main source of entrainment, in the case of core uncovery, is the core, where steam is produced. Another place closely connected to the core is the upper plenum where both entrainment and de-entrainment can occur. *Steam binding denotes the generation of backpressure in the steam generators by the evaporation of droplets which were entrained by steam from the liquid in the core and UP. This phenomenon is similar to that in PWRs with ECC injection into hot legs. However, the magnitude of backpressure generated in horizontal SG may be different than that is in vertical SG with U-tubes. This phenomenon is weakened due to condensation of part of steam in upper plenum that leads to steam velocity decrease and entrainment decrease.*

C1-9.2 Relevance to nuclear reactor safety

Entrainment/De-entrainment in steam generator *inlet collectors* contributes to steam binding for large breaks which influences the mass distribution in primary circuit.. *Backpressure in SG due to evaporation of entrained droplets may affect the down-top core cooling.*

Entrainment/De-entrainment in steam generator tubes also contributes to the so-called steam binding for large break LOCAs.

C1-10 Pool Formation in Upper Plenum (B1.8)

C1-10.1 Description of the phenomenon

This phenomenon is closely related to the entrainment and de-entrainment in the core as well as to UP injection. Some of the carryover out of the core will be captured in a water pool, which may form on the upper core support plate, and by upper plenum structures. Pool formation refers to the collection of water in the upper plenum during reflood. Main source of this water is UP injection Another source of this water is entrained water carried up from the core which is then de-entrained in the upper plenum. The upper plenum contains a great deal of internal structures that act as a steam separator to de-entrain liquid in the two phase flow. In addition, the drop in flow velocities from the core to the upper plenum allows gravitational separation of liquid from steam. The flow behavior of a collected pool may be highly three-dimensional because of the tendency of the water pool to collect in low flow area and in regions near walls and structures.

VVERs have no specifics but features of in-vessel geometry should be taken into account (see also CCFL).

C1-10.2 Relevance to Reactor Safety

Upper plenum pool formation is important to ECCS performance during a LOCA. The reasons can be summarized as: the pool, particularly due to its multidimensional nature, provides a source of additional cooling water for the core; de-entrainment and pool formation reduce carryover to the steam generator, the pool can reduce the down-top flooding rate by creating a static head pressure drop in the upper plenum.

Pool formation in UP delays refill of the core and therefore highly relevant to reactor safety.

C1-11 Core Heat Transfer including DNB, Dryout, RNB (B1.9)

C1-11.1 Description of the phenomenon

The heat transfer between wall structures and fluid is an essential heat transport mechanism in light water reactors for transporting the heat generated in the core to the turbines. In the first step the heat is transferred in the core from fuel to fluid. The steam generator tubes transmit the heat between the high pressure primary side and low pressure secondary side. The safety aspects of the heat transfer are mostly related to the core region.

The fuel cladding temperature is usually near the saturation temperature of water. During an accidental increase in power or decrease of flow and pressure a deterioration in the heat transfer process is possible. The surface temperature increases to such a high level that the heated surface can no longer support continuous liquid contact. This phenomenon is called boiling crisis or dryout depending on the mechanism leading to the event. The key question is, how efficiently the deteriorated heat transfer mechanism can remove heat generated in the core uncovery and local for high power rods.

In VVERs these phenomena in general do not differ from those in PWRs but quantitative behaviour is governed by geometry of specific reactor.

C1-11.2 Relevance to Reactor Safety

Forced convection or natural convection can occur when single-phase liquid, vapour or two-phase flow is present. During normal operation of a PWR the forced convection to single-phase water is the main heat transfer regime in the primary coolant system. In the core subcooled boiling may also occur.

Cooling of the core and the primary side fluid inventory via the steam generator has the highest priority in nuclear reactor safety. Transition from single-phase liquid heat transfer to nucleate boiling takes place once the wall temperature exceeds the temperature for Onset of Nucleate Boiling. This may occur at accident or transient conditions.

In the VVER core no bulk boiling is expected, but subcooled nucleate boiling can occur in the top third of the core. The possible local voiding thus generated has the unwanted side effect of disturbing the neutron power profile.

For steady state operation of the reactor the avoidance of CHF is one of the main criteria in acceptance of the plant power level. Taking into account the peaking factors for the maximum power fuel bundle and for the maximum power fuel rod, no CHF is allowed during steady state conditions. Rods experiencing CHF may be damaged. The criterion is also used in anticipated transients, when the plant protection systems are operating normally. In accidents including LOCA, or in transients with an additional failure of plant protection systems, CHF at least in the hottest rods may be expected, and as a consequence the rods may be damaged causing a release of radioactivity to the coolant. During LOCA the occurrence of CHF initiates the core heat up period.

The post CHF regime means in general the regime where the hot wall temperature or lack of: coolant does not allow liquid to contact the surface. Above the dryout type of CHF the coolant in the flow channel consists of a mixture of droplets and steam when the annular liquid film has evaporated. After the DNB-type of CHF the hot surface temperature prevents the turbulent mixture of coolant and steam wetting the wall. During quasi-stationary conditions in the core after core uncovery only steam exists above the swell level. All these modes are considered as post CHF heat transfer.

During the emergency core cooling after a large break LOCA the cooling by the droplet-steam mixture is essential both for bottom and top reflooding.. Quite soon after the reflooding water enters the core, the temperature increase in the upper part may be stopped due to the steam cooling caused by the droplets. Also, UP injection leads to the cladding temperature decreasing. Thus the post CHF heat transfer is the main contributor to the limiting maximum cladding temperatures in the core and preventing damage by oxidation.

Thermal radiation will transfer energy both from surface to surface and from surface to the two-phase flow. Exchange of thermal radiation with the two-phase flow will mainly consist of absorption in the two-phase flow, mainly in the droplets. Due to the relatively low temperature the emission of the two-phase mixture is negligible. The radiation heat transfer becomes important in addition to other heat transfer modes when the structure temperature locally exceeds the saturation temperature by 200K. These land of high temperatures are possible only in the core with pure steam, with droplet-steam mixture or with inverted annular flow regime. Typically the net radioactive heat flux streams from the highest temperature regions into colder parts, being partially absorbed in the fluid.

Before emergency cooling water enters the core during a LOCA during which the core has become uncovered, the radiation heat transfer between structures is the most significant heat transfer balancing the local temperature differences. The net heat flux is directed from the hottest rods to the cooler ones. Due to its strong dependence on the fuel temperature the radiation may effectively prevent the temperature rise and prevent cladding oxidation.

C1-12 Quench Front Propagation - Up and Down (B1.10)

C1-12.1 Description of the Phenomenon

The reflooding and quenching phenomenon are important for Light Water Reactors. Indeed, it is usually postulated that the core is uncovered and overheats due to decay heat from the fission products and the energy stored prior to LOCA during the so-called blowdown phase of the LOCA. In VVERs, the core is recovered, i.e. the overheated rods are quenched and adequate heat transfer is reestablished, by reflooding from downcomer and upper plenum. Water is delivered into the core by the ECCS via bottom and top flooding in order to stop overheating of the fuel rods and re-establish cooling. Otherwise cladding oxidation, zircaloy-water chemical reaction or clad melting and the consequent release of radioactive products can occur.

Reflooding refers to a particular mode of post-burnout cooling of a hot channel by refilling it with coolant. Quenching or rewetting of the hot surface occurs during reflooding. Quenching refers to the transition from a mode of heat transfer characterized by total or almost total absence of liquid contacts with the wall to one where the wall is essentially wetted by the liquid. The teat transfer coefficient increases dramatically following quenching.

As a result of the high temperatures attained by the clad before the emergency coolant arrives, water does not initially wet the hot clad surface. Rewetting or quenching of hot surfaces occurs when the coolant re-establishes contact with the dry-surface. The surface temperature corresponding to the achievement of liquid-solid contact is the rewetting or quenching temperature. The temperature of the fuel pellets and fuel clad is reduced by heat conduction and convection to the coolant. As the coolant rises in a hot channel or in the overheated nuclear fuel rod bundle, complex heat transfer and two-phase flow phenomena take place and also the succession of regimes moves radially up the rod bundle channels. The hot surfaces along the channels experiences in turn free-or forced-convection cooling by steam, dispersed-flow film boiling, inverted-annular film boiling, transition boiling, nucleate boiling, and finally single-phase convection to the liquid. Almost, all the heat transfer modes are encountered during reflooding and quenching phase.

During reflood multi-dimensional flow patterns occur in the core and upper plenum due to: Flow rates and flow regimes are such that gravitational forces are of the same order as inertial forces; non-uniformities in core power distribution and differences in resistance to flow through the intact and broken hot legs tend to promote multidimensional effects; as the flow passes through the upper plenum it must flow around several structural elements and additionally the flow behavior of a collected pool in the upper plenum may be highly three-dimensional.

At experiments on heated rod bundles it was observed several quench fronts due to droplets separation on spacer grids.

Another important phenomenon, observed in the VVER integral system large break LOCA experiments, was core wide cooling of the fuel cladding and quench during the blowdown phase. This phenomenon is important regarding heat up of the core, because it removes a large part of the stored energy from the fuel during the early phases of the transient.

Quench phenomena are similar to those in PWRs with combined ECC water injection (into hot and cold legs), however differences in geometry should be taken into account.

C1-12.2 Relevance to nuclear reactor safety

The rewetting characteristics of the overheated core after the large LOCA was one of the most interesting research topics in the 70-s and still has a significant influence on acceptance criteria in licensing and PSA safety analyses. The main interest is related to the maximum temperature in the core, but this turn-over temperature is determined by the liquid dispersed flow well before quenching. Depending on the amount of water available the cooldown takes place earlier or later.

The large temperature gradient in the cladding gives rise to a mechanical stress on the cladding and it may affect fuel damage and radioactivity leakages. The rapid temperature drop is also associated with strong steam generation and this may have an effect on system characteristics including:

- liquid entrainment rate
- counter current flow limitation in the upper tie plate
- steam binding in the steam generator
- multidimensional flow distribution in the core

The rewetting of fuel box walls (inner and outer) in VVER-440 gives an important contribution to core reflood.

C1-13 Entrainment in the Core and Upper Plenum (B1.11)

C1-13.1 Description of the phenomenon

Droplet entrainment in the core and upper plenum is a form of phase separation. Water can, for example, be entrained by sufficiently high steam velocities from the bottom of the reactor vessel and be carried out of the vessel. Thus water can be lost from the liquid inventory in the vessel. Droplets carried out of the reactor core by rising steam during reflooding can be de-entrained by impinging on the various structures situated in the upper plenum and, consequently, may fall back into the core.

Entrainment of liquid droplets by the flow of steam has considerable influence on the effectiveness of emergency core cooling. The importance of the entrainment has been especially recognized associated with heat and mass transfer processes during LOCAs, in particular, during the recovery phase of these accidents through reflooding of a core. In this case entrainment improves heat transfer, since droplets act as a heat sink through droplet evaporation. This will lead to lower vapour superheat and improved cooling of fuel rods. Indeed the hot portion of the fuel rods in the core is cooled by the mixture of steam and entrained droplets. Liquid entrainment affects the liquid core inventory and is also important in the determination of the critical heat flux point. Another indirect effect of the entrainment is that the vaporisation of the droplets carded to steam generators pressurizes the upper plenum and prevents the core water level from rising (i.e. steam binding effect).

This phenomenon is similar to that in PWRs with combined ECC water injection, however, differences in geometry should be taken into account.

C1-13.2 Relevance to nuclear reactor safety

This is discussed below in conjunction with de-entrainment.

C1-14 De-Entrainment in the Core and Upper Plenum (B1.12)

C1-14.1 Description of the phenomenon

Liquid de-entrainment occurs when water entrained in the steam generated in the core is removed from the steam flow at other places in the core, upper plenum, or beyond. De-entrainment occurs from gravitational and inertial forces. It is enhanced when flow slows down because of a flow area increase or when it changes direction to pass around flow obstructions or structures or to turn out through a nozzle. De-entrainment removes entrained water from the two-phase flow mixtures. De-entrainment water that accumulates in the core and upper plenum provides a source of water for core cooling supplementing the cooling by bottom reflood. This enhances core cooling near fuel assembly grid spacers and is also important for the upper regions of the core. Clearly, such entrainment and de-entrainment phenomena are controlled by interfacial shear. Correct modelling of the interfacial shear force is crucial for the adequate prediction of a number of phenomena and interfacial shear models also vary according to the flow regimes. In VVERs conditions for this phenomenon are more favourable than those in PWRs due to partial steam condensation in upper plenum and decrease of steam velocity.

C1-14.2 Relevance to reactor safety

The importance of the entrainment and de-entrainment in a core has been recognized associated with heat and mass transfer processes during LOCAs, in particular, during the recovery phase of these accidents through reflooding of a core and also under boil-off situations. Since these phenomena will lead to improved cooling of the fuel rods, the knowledge of the extent of entrainment and de-entrainment is necessary to predict the hydrodynamic and thermal response to the core.

C1-15 One and Two Phase Pump Behavior (B1.13)

C1-15.1 Description of the phenomenon

In single phase conditions the dimensionless pump characteristics are almost independent of the fluid (liquid, gas, steam). The small variations are mainly due to viscous properties connected to recirculation losses, shock losses on the impeller, and fluid friction. In two phase flow the torque and head characteristics are affected by so called degradation which is a function of void fraction, pressure and flow rate.

The main causes of the degradation seem related to phase separation in the impeller due to centrifugal forces, with liquid phase acceleration and relative deceleration of the gas or steam. The pre- and post-rotations of fluid in the inlet and outlet pipes play a part in this problem. All these phenomena depend on the type (centrifugal, axial, mixed flow) and geometry of the pump. *VVERs have no features in this respect.*

C1-15.2 Relevance to nuclear reactor safety

For all LOCAs with running pumps the pump performance is important. For large LOCAs the behavior of the intact loop pumps plays a role in determining the position of the stagnation points in the primary circuit.

For small break LOCAs the pump behavior plays a role in steam-water mixing and core cooling.

C1-16 Non-Condensable Gas Effect (B1.14)

C1-16.1 Description of the phenomenon

The presence of a non-condensable gas in the primary circuit plays a role in mechanical and thermal fluid behavior. Non-condensables are mainly nitrogen from the accumulator injection or hydrogen from clad oxidation.

In the case of nitrogen injection the mechanical effects are related to gas expansion in the upper part of the downcomer and upper plenum. The momentum transfer between the gas, steam and liquid increases the liquid velocity. This has a some impact on the downcomer water penetration and on water bypass to the break. Both in the upper plenum and downcomer gas decreases condensation rate.

Within the primary circuit the gas migrates from its injection or generation point. There are several mechanisms for migration: the entrainment of gas by flowing steam, the gravity forces arising from density differences and diffusion driven by the concentration gradient. In the horizontal steam generator, due to possible condensation in the tubes gas may accumulate in outlet ends and outlet primary collectors when the break occurs in the hot leg. At the break in the cold leg gas is expelled out of circuit after loop seal clearance. Gas accumulation in SG may deteriorate heat transfer and limit natural circulation.

C1-16.2 Relevance to nuclear reactor safety

This phenomenon could have some influence on the general thermal hydraulics behaviour for large breaks but is of most relevance to SBLOCAs and accidents with natural circulation phases if vents on SG collectors are not used.

C2-1 Single Phase Natural Circulation (B2.1)

C2-1.1 Description of the phenomenon

After stop of reactor coolant pumps, natural circulation caused by temperature differences within the loops is the mechanism for heat removal from the core. The flow rate depends on the relative elevation of major components as well as on the temperature difference between primary and secondary side that in turn depends on the core power. In general, single phase natural circulation is well understood, VVER specifics concerning code validation does not exists. Compared to PWRs, special design features of VVERs have to be taken into account on the level input data deck, e.g. horizontal instead of vertical steam generators (lower geodetic elevation differences in VVERs) and the different elevations of the reactor inlet and outlet nozzles.

C2-1.2 Relevance to Nuclear Reactor Safety

If heat removal from the secondary side is available during the accident, single phase natural circulation is a reliable mechanism for decay heat removal.

C2-2 Two Phase Natural Circulation (B2.2)

C2-2.1 Description of the phenomenon

The void creation in the hot leg determines an increase of the buoyancy forces because the density difference between the hot and cold leg increases. As a consequence an increase in the mass flow rate at core inlet occurs. In these conditions steam produced in the core may separate under reactor cover and in upper part of the inlet primary collector and condense in the SG tubes. The removal capacity in these conditions is very efficient since high heat transfer coefficients are present in both sides of the tubes. At definite mass inventory there is a maximum flow rate at the core inlet. This maximum, normally referred as two-phase peak flow is dependent on many parameters (the primary inventory, the core power, the pressure, the geometry) but all the facilities show however a similar behavior. Further decrease of the mass inventory leads to gradual decrease of the flow rate up to full interruption.

Existence of hot leg loop seals in VVER-440 has an important impact on 2-phase natural circulation with a possibility of oscillatory behavior due to separation in descending side of the hot loop seal and condensation of steam in the SG.

C2-2.2 Relevance to Nuclear Reactor Safety

The two phase natural circulation is a very effective mechanism of decay heat removal. Interruption of this process may lead to the core temperature increase. The degradation or interruption the two phase natural circulation may occur also due to phase separation in the loop seals.

C2-3 Reflux Condenser Mode (B2.3)

C2-3.1 Description of the phenomenon

As the two-phase natural circulation is interrupted a new circulation mode is established in which the steam coming from the reactor pressure vessel is condensed within the SG tubes. The liquid comes back through the tubes to the hot leg and to the core flowing in counter current flow with steam. The ability of this mode to remove the decay heat has been provided in many experimental facilities (SEMISCALE, LOBI, PKL, BETHSY. etc). Conditions for the establishment of the reflux condensing mode are therefore: high void fraction in the primary system and in the SG tubes in particular, good cooling capabilities of the secondary side, not very high inlet steam flow that could determine CCFL in the tubes or in the hot leg.

Presence of non-condensable gases and their accumulation in exit part of tubes promotes to condensate flow to the hot leg. These gases greatly affect heat removal capacity and induce rather complex circulation in the tubes due to density difference of steam-gas mixture in the SG inlet and outlet collectors. Accumulation of light gas (hydrogen) leads to full blockage of the upper tube rows. The reflux circulation is not expected in the VVER-440 because of the hot loop seals.

C2-3.2 Relevance to Nuclear Reactor Safety

In a small break LOCA in which the secondary side is available reflux or condensation mode can be an effective heat removal mechanism. CCFL in the tubes can not occur because condensate may flow both to inlet and outlet tube ends. The conditions for decrease of the removal effectiveness are depending on the plant configuration. For example, in VVER-440 reflux condensation has not any significance but boiler condensation mode is possible due to loop seal in hot leg. The presence of non-condensable gas can greatly affect this heat removal mechanism.

Accumulation of boron-free condensate in the cold leg loop seal may form slug, which may be displaced into the core after natural circulation re-establishment. This phenomenon is called as local inherent boron dilution and requires special consideration (See item C2.24).

C2-4 Asymmetric Loop Behavior (B2.4)

C2-4.1 Description of the phenomenon

The adoption of symmetric boundary conditions to the different primary system loops should guarantee that a symmetric behavior should occur in the phenomena. However, this condition is never fulfilled due, for example, to presence of pressuriser in one loop, break located in one loop, injection ECC water into some loops, non-uniform distribution of non-condensable gases between loops or unavailability of certain systems. Also, due to non-linear nature of the phenomena asymmetries may occur. The asymmetric behavior is manifested with different mass and flow distribution in different loops. For example, loop seal clearing may occur in some loops, but not in others. These asymmetric effects lead to different heat removal capabilities of different loops and may influence the overall system behavior. This phenomenon is similar to that in PWRs .

C2-4.2 Relevance to Nuclear Reactor Safety

Asymmetric conditions in primary system may reduce the overall heat removal capacity because some of steam generators may not take a part in the heat removal process. Also, the system effects caused by asymmetries are important while simulating the accident conditions.

C2-5 Break Flow (B2.5)

C2.5.1 Description of the phenomenon

During IB and SB LOCAs break flow is controlled by the balance of break size, break geometry and its location , static pressure loss over the loop and ECC injection rate. Stratification under low flow rates is of great importance. Integral break flow rate adjusts itself to these conditions. See also item C1.1 for LBLOCAs.

C2.5.2 Relevance to Nuclear Reactor Safety

See item C1-1.2 for LBLOCAs

C2-6 Phase Separation without Mixture Level Formation (B2.6)

C2-6.1 Description of the phenomenon

This phenomenon takes intermediate place between homogenious and fully stratified two-phase flow. Occurence of this phenomenon is governed by steam and liquid velocities. In VVERs such mode of separation can occur in the SG primary inlet collectors during 2-phase natural circulation phase and may lead to non-uniform steam distribution at the tubes inlet. Also, this phenomenon can occur in downflow side of the hot leg loop seal of VVER-440 and can lead to decrease of driving head.

In other parts of primary system there is no specifics in comparison with PWRs.
See also item C1-2 for LBLOCAs.

C2-6.2 Relevance to Nuclear Reactor Safety

Separation in the core controls the cladding temperature. Separation in downflow parts of primary system (downcomer, hot leg loop seal) controls driving head at two-phase natural circulation. Separation in other primary components influences coolant distribution (in particular, in the horizontal SG tubes) and break flow during SBLOCA.

C2-7 Mixture Level and Entrainment in SG (Primary and Secondary Sides) (B2.7)

C2-7.1 Description of the phenomenon

In some transients a mixture level may be formed in the SG collectors defining the inlet conditions to the SG tubes. Some of the tubes will be completely uncovered and filled only with steam, some tube entrances will be covered with two-phase mixture, and other with water.

In normal operating conditions the tube bundles of the horizontal SG are covered by steam-water mixture. In the whole, the mixture circulates upward through the tube bundles and in VVER-1000 additionally through a submerged perforated sheet. As separation between the steam and water occurs, the water descents in downcomer gaps between the bundle sections, at both sides and the ends of the SG. However, in detail the mixture circulation is rather complex due to non-uniformity of heat transfer in longitudinal and transversal directions. Voids and velocities distribution has some features. At some transients under full power swollen level may drop in the large degree but all tubes of the bundle remain wetted due the droplets entrainment. At SBLOCAs the droplets entrainment in secondary side has not any significance.

These phenomena are quite different in VVERs and PWRs.

C2-7.2 Relevance to Nuclear Reactor Safety

In some phases of SBLOCA the mixture level in both sides of the SG will determine the heat transfer through the SG tubes and therefore the decay heat removal from the core.

C2-8 Mixture Level and Entrainment in the Core (B2.8)

C2-8.1 Description of the phenomenon

Due to boiling a two-phase mixture level may be formed in the core. At sufficiently high steam velocities water can be entrained from the core region in form of droplets. This entrainment of water may results in reduction of coolant inventory in the core. However, some of the droplets by impinging on various structures in upper plenum may be de-entrained and fall back into the core.

Entrainment of liquid droplets has significant influence on the effectiveness of emergency cooling and is discussed in more detail in C1.13. It should be recognized that the VVER core geometry (pitch, rod diameters, grid spacers shrouds) is quite different in VVERs than in PWRs, therefore the magnitude and condition of entrainment may be different from PWRs.

C2-8.2 Relevance to the reactor safety

Core mixture level affects directly core cooling. Also, entrainment and de-entrainment affects core cooling. These phenomena are important to safety as they control fuel and cladding temperature in some phase of the SBLOCA.

C2-9 Stratification in Horizontal Pipes (B2.9)

C2-9.1 Description of the phenomenon

At high velocities the flow of a two-phase mixture in a horizontal pipe can be fairly homogeneous. As the velocity falls gravitational forces tend to cause phase separation across the direction of the flow and become more important than the inertial forces. The ratio between these two forces is represented by the Froude number. The occurrence of the stratification can be correlated with the value of the Froude number. Stratified flow conditions are a prerequisite for counter current flow in a horizontal pipe.

The prediction of the behavior within the stratified regime requires a knowledge of wall friction inter-phase friction and inter-phase heat transfer relationships. These are less difficult to determine than other more complex flow regimes, although they depend on whether the stratified surface is smooth or wavy.

In small break scenarios the stratification can occur in the hot legs, cold legs and loop seals. The development of stratified conditions in the hot legs is closely bound up with the transition from two-phase natural circulation to reflux condensation. When the stratification occurs in the pipe containing the break the break void fraction is dependent on the relative location of the inter-phase. This dependence is not simply dictated by the break elevation. In fact even if the level is higher than the upper side of the break a quantity of vapor can be entrained in a vortex and reach the break (vapor pull through). Analogously as the level is below the lower side of the break, liquid can be transported to the break by inter-phase shear stress (liquid entrainment). Both phenomena becomes less relevant as the break size decreases.

Stratification may occur in single-phase conditions when flows with different temperatures (densities) enter the pipe. It also is correlated with the value of Froude number. As an example, one may point out ECC water injection into cold leg and accumulation of the boron diluted water in the cold leg.
This phenomenon is similar to that in the PWR pipes.

C2-9.2 Relevance to Nuclear Reactor Safety

The occurrence of the stratification is important in the case it happens in correspondence of the break location. In this case it can affect the break void fraction and thus the inventory discharged by the primary system. In the LOFT facility the stratification in the hot leg in a hot leg small break occurred even while the pumps were still running, determining an unpredicted primary system inventory behavior.

Stratification in single-phase conditions has an importance in viewpoint of thermal stresses on the primary pressure boundary, in particular, on the reactor vessel (pressurized thermal shock) and non-homogeneous boron dilution in the core.

C2-10 ECC-Mixing and Condensation (B2.11)

C2-10.1 Description of the phenomenon

After primary pressure decrease and actuation high pressure injection system (HPIS) cold water enters downcomer and flows there as a cold plume mixing with coolant. As it has been shown in many studies, there is good mixing (in spite of guiding fins in the VVER-440 downcomer), that prevents excessive thermal stresses in irradiated part of the reactor pressure vessel. However, long-term HPSI under high pressure (without LPSI) can lead to pressurized thermal shock (PTS) due to overcooling entire downcomer region. To prevent PTS accident management is need.

Another problem related to mixing arises at the main steam line rupture which leads to overcooling of the emerged loop. This event leads to introduction of positive reactivity into the core.

Also, mixing of ECC water is important, when the boron diluted slug formed in the cold leg loop seal under SBLOCA conditions, enters the reactor at re-establishment of natural circulation.

As for condensation, see C1-3.

C2-10.2 Relevance to Nuclear Reactor Safety

Mixing in the downcomer, lower plenum and adjacent pipes is very important to mitigate undesirable consequences: excessive thermal stresses, insertion of positive reactivity into the core.

C2-11 Loop Seal Clearing (Cold Leg) (B2.12)

C2-11.1 Description of the phenomenon

In most current VVER designs, the primary piping from the outlet of the steam generators descends to a level below the mid level of the core before rising to enter the pump inlet from below. During a LOCA, when the primary circuit is partially voided, liquid present in the U- bend of this intermediate leg can form a barrier to the flow of steam around a loop. This liquid plug is said to form a loop seal. A pressure difference across the loop seal which is greater than the hydrostatic pressure arising from the height of the pump inlet side of the intermediate leg is necessary before steam can displace the loop seal.

The manner and extent of the displacement of the liquid from the loop seals depend on the steam supply characteristics. At reduced steam velocities and low liquid subcoolings, the expulsion of fluid will be incomplete. This aspect of loop seal behavior is closely connected with the phenomenon of flooding (CCFL). However the pipe sizes of interest are very much larger than those studied in most separate effect tests.

When the loop seals clear, their liquid contents are displaced into the reactor vessel, producing a rapid refill. The number of loop seals clearing, which depends for instance on the size of the break, affects the total amount of water added. How many loops need to clear is determined by the balance between the frictional pressure drop around the cleared loops and the hydrostatic pressure difference corresponding to the loop seal.

It is possible for loop seals to reform after they have cleared. Back flow of ECCs from the cold legs through the pump is limited by the height of the impeller and the diffuser geometry, but can occur if there is a general flow reversal, or if the level in the cold leg is high enough. Condensate from the down side of the steam generator tubes also can provide a contribution to the reforming of the seals.

C2-11.2 Relevance to Nuclear Reactor Safety

The loop seals have a particularly significant effect on the behavior of a plant in a small cold leg break LOCA. Until at least one of the loop seals has cleared, steam produced in the core cannot reach the break, and the pressure in the primary system remains high, at a level determined by the secondary side pressure. Furthermore, as the liquid content of the primary side falls, the hydrostatic head of the liquid trapped in the pump inlet causes the liquid level in the core region to be depressed by an amount equal to the depth of the loop seal. This can lead to uncovering of the core.

This phenomenon is similar to that in PWRs.

C2-12 Pool Formation in UP/CCFL (UCSP) (B2.13)

See item C1-10 for LBLOCA

C2-13 Core Wide Void and Flow Distribution (B2.14)

C2-13.1 Description of the phenomenon

Upper plenum ECC injection and shrouded fuel assemblies in VVER-440 create some specifics in comparison with PWRs. See also item C1.5 for LBLOCA.

C2-13.2 Relevance to Nuclear Reactor Safety

Affects directly cladding temperatures. See also item C1.5 for LBLOCA.

C2-14 Heat Transfer in Covered Core (B2.15)

C2-14.1 Description of the phenomenon

The thermal hydraulic situations included in this phenomenon are representative of pre-accident conditions as well as the initial or less critical phases of an accident. In increasing priority from safety point of view, it is possible to mention:

- forced convective heat transfer
- natural circulation heat transfer and subcooled nucleate boiling
- nucleate boiling heat transfer

C2-14.1.1 Forced convective heat transfer

This is the normal situation, present in the core when the main coolant pumps are running. The heat transfer coefficient is primarily function of the fluid velocity. Single phase flow is present in the core even if locally subcooled boiling can exist. There are situations of two phase forced convection (small break LOCA with pumps running) but this case is a transition versus conditions similar to natural circulation heat transfer, due to the pump head degradation.

C2-14.1.2 Natural circulation heat transfer

As the velocity in the core decreases the heat transfer coefficient decreases and the core temperature difference increases. It is possible that at a certain point in the core saturated nucleate boiling conditions are achieved. The transition between forced convective boiling and natural circulation heat transfer has to be accompanied by a core power reduction, due to the reduced heat removal capabilities of this condition.

In natural circulation, and to some extent also in forced convective heat transfer, some subcooled boiling can occur. This can be explained with temperature profiles within the liquid and locally the temperature can be higher than the average in the bulk. The subcooled natural circulation cannot exist for a long time in small break because the saturation temperature decreases due to pressure reduction and therefore soon or later conditions of saturated nucleate boiling are achieved.

C2-14.1.3 Nuclear boiling heat transfer

Saturated nucleate boiling starts when the bulk liquid temperature equals saturation temperature.

In saturated boiling the heat transfer consists of two contributions, a nucleate pool boiling and a convective term. The heat transfer coefficient is generally quite effective to remove the heat. This condition can exist up to a maximum void fraction near to 1 (0.95-0.98). If the void fraction exceeds this value dryout condition is established in the core (see *Heat Transfer Partially Covered Core)*
In VVERs this phenomenon is very similar to that in PWRs

C2-14.2 Relevance to Nuclear Reactor Safety

The importance of this phenomenon is related to the fact that it represents the initial condition for the partially covered core situation. As it is it does not represent a problem but the prediction of the end of the so called ‘covered’ core situation is of extreme importance to safety. In order to correctly predict this occurrence the prediction of the mixture level is important and therefore the void fraction distribution within the core.

C2-15 Heat Transfer in Partially Uncovered Core (B2.16)

C2-15.1 Description of the phenomenon

A partially covered situation is characterized by the formation of a mixture level within the core region above which steam single phase flow exists and thus core heats up. This situation, in small break LOCAs can be originated by two mechanisms: an increase of the pressure difference between the upper plenum and the downcomer, or a progressive loss of primary inventory.

During a small break LOCA the formation of a level in the core is balanced by a water level in the loop seal. This determines a pressure difference between the hot and cold leg that causes a pushing and decrease of level within the core. The loop seal clearing can cause a reduction of the pressure difference and thus a core level recovery.

The second situation can occur even in absence of pressure difference and is caused by a progressive water depletion in the core due to boiling and inventory decrease through the break. In the same small break LOCA both situations can occur. The first one is generally shorter in time. The extent of the core temperature excursion depends on the plant geometry and the ECCS setpoints.

When the 'partially covered core' situation occurs the section below the mixture level is generally well cooled in nucleate boiling while the section above initiates an heat-up, controlled by the actual power.

As it has been shown, heat transfer above the mixture level in VVERs is very similar to that in PWRs. In VVER-440 boxed fuel assemblies should be taken into account for correct calculation of steam velocities.

C2-15.2 Relevance to Nuclear Reactor Safety

The phenomenon is very important for Nuclear Reactor Safety. After Three Mile Accident the importance of small break issues has increased. The correct prediction of the mixture level and the onset of dryout conditions is very important for the following evolution of the heater rods surface temperature. The type of correlation that can be used in these situations are mainly steam natural convection types, since the amount of water contained above the mixture level is very limited. Therefore the correct mass distribution within the primary system is of extreme importance for this phenomenon.

The real concern is the detection of the conditions for the core uncovery situation. When the situation is correctly detected it can be well predicted by computer codes.

C2-16 Heat Transfer in SG Primary Side (B2.17)

C2-16.1 Description of the phenomenon

The heat transfer in the SG primary side covers the following type of situations:

- forced convective heat transfer;
- natural circulation heat transfer;
- condensation.

The first two are conditions basically similar to the ones described under Heat Transfer in Covered Core (lee that chapter) phenomenon, with the difference that in this case the heat is going from the fluid to the wall. The third situation can occur in the presence of two-phase flow or single phase steam flow in the SG tubes, during natural circulation periods when the secondary side pressure is below the primary one, so that the tube wall temperature is below the fluid saturation temperature and condensation is possible in the SG tubes. The condensation is a quite effective mean to remove heat from the primary coolant system. In slightly inclined tubes the condensation produces a thin stream of liquid continuously removed by gravity. The rate of heat transfer is dependent on the subcooling of the surface. The presence of non-condensable gas in the vapor can result in a drastic reduction of the condensation rate due to reduction of the local saturation temperature. The reflux (VVER-1000) and boiler (all VVERs) condensation natural circulation, described as a phenomenon presents this type of heat removal mechanism.

C2-16.2 Relevance to Nuclear Reactor Safety

All the heat transfer modes here considered are of high importance in small break scenarios because the reduction of the heat removal capability from the steam generator can determine critical conditions within the reactor, particularly for smaller break sizes. As soon as the break size increases the heat removed by the steam generator becomes negligible in comparison with the break energy and soon the heat transfer reverses from the steam generator to the primary side.

C2-17 Heat Transfer in SG Secondary Side (B2.18)

C2-17.1 Description of the phenomenon

In normal operating conditions tube bundles of horizontal SG are covered by steam-water mixture, which guarantees high heat exchange rates. The mixture circulates through the tube bundles and in VVER-1000 through submerged perforated sheet. Water from top of the sheet and the bundle descends in downcomer gaps between sections of the bundle and between the bundle and SG vessel.

During events with reactor scram or RCP trip the tube bundle may be partially uncovered due to void collapse. Emergency feed water system restores the level. This water has low temperature and is sprayed above level. Therefore, secondary water inventory has almost uniform temperature. In SGs of VVER-440 main feedwater is delivered to lower part of the bundle forming economizer zone. In early designs of VVER-440 emergency feed water is supplied under the level through the main feedwater distribution collector

C2-17.2 Relevance to Nuclear Reactor Safety

Heat transfer in the SG secondary side is of high importance in small break scenarios because the reduction of the heat removal capability from the steam generator can determine critical conditions within the reactor, particularly for smaller break sizes. As soon as the break size increases the heat removed by the steam generator becomes negligible in comparison with the break energy and soon the heat transfer reverses from the steam generator to the primary side.

C2-18 Pressuriser Thermalhydraulics (B2.19)

C2-18.1 Description of the phenomenon

During a small break LOCA within the primary system the pressuriser is initially emptied by the same amount of water lost by the break. No special emphasis should be given to this phenomenon. The only aspect that should be considered is the behavior of the gas space that can be superheated by the heat released by the pressuriser heat structures.

If the break is located in the pressuriser itself (PORV stuck open) the scenario is quite different as well as the phenomena occurring. In this case the pressuriser initially is filled with subcooled water coming from the primary system and as the pressure reduces the flow becomes saturated and then two-phase. The development of a mixture level in the pressuriser depends on the valve size, and it is quite crucial for the determination of the break void fraction and thus of the primary system inventory. The prediction of the time of mixture level formation (if occurring)is important in the determination of the outlet mass flow rate. This phenomenon in VVERs does not differ from that in PWRs.

C2-18.2 Relevance to Nuclear Reactor Safety

If the break is located in the primary system, this phenomenon has no relevance to Nuclear Reactor Safety. If. however the break is represented by a pressuriser valve stuck open the relevance for the Safety is very high since it can determine critical conditions in the primary system. It should be mentioned that the accident of Three Mile Island can be considered like a PORV stuck open and the misunderstanding of the pressuriser thermohydraulics had a dominant importance on the course of the accident.

C2-19 Surgeline hydraulics (B2.20)

C2-19.1 Description of the phenomenon

During a small break LOCA within the primary system the pressuriser is initially emptied by the same amount of water lost by the break. The mass flow rate flowing through the surge line can determine high friction losses that can keep high the pressure in the pressuriser and thus in the primary system. A higher pressure in the primary system determines a higher mass being discharged through the break.

If the break is located in the pressuriser itself (PORV stuck open) the scenario is quite different as well as the phenomena occurring. In this case the pressuriser initially is filled with subcooled water coming from the primary system and as the pressure reduces the flow becomes saturated and then two phase flow. The flow can create friction losses and determine a great pressure difference with the primary system. In the case in which separation would occur in the pressuriser (small valve flow area), the surge line could experience a counter current flow limitation (CCFL) with steam coming from the primary system and liquid descending from the pressuriser. In this case the surge line characteristics (diameter, length and layout) can greatly affect this phenomenon. Differences between VVERs and PWRs in this respect are possible.

C2-19.2 Relevance to Nuclear Reactor Safety

If the break is located in the primary system, the only relevance to the safety is the ability to drain pressuriser. Pressurisers in VVERs are large and contribute to primary system coolant inventory and cooling. If however the break is represented by a valve stuck open the relevance for the Safety is moderate since in case of CCFL it can affect the possibility of recuperating the inventory remaining in the pressuriser.

C2-20 1- and 2-phase pump behavior (B2.21)

See item C1-15 for LBLOCA

C2-21 Structural Heat and Heat Losses (B.22)

C2-21.1 Description of the phenomenon

The thermal energy stored in the metal structures is released to the fluid when the temperature of the fluid starts decreasing as effect of depressurization; since the amount of the stored heat is not negligible it has to be accounted in the overall energy balance. The mechanism of heat transfer from the structures depends on the temperature difference and the void fraction. The heat losses to the surrounding is realized by radiation heat transfer; as effect of the heat losses the condensation of vapour could be determined inside the pressure boundary and some mass redistribution could be implied.

C2-21.2 Relevance to nuclear reactor safety

The heat losses (usually much bigger in the test facilities) are more relevant in the long-term accidents (SBLOCAs) while the structure heat is more important in the short term (LBLOCAs). However, structural heat input to the fluid from reactor vessel structures may not be negligible and may affect the system behavior. In general, the structural heat and the heat losses are important factors contributing to the course of SBLOCA and have to be considered in safety analysis.

The advanced computer codes have the capability to evaluate the heat losses and the structure heat.

C2-22 Non Condensable gas effects (B2.23)

See item C2-17 above

C2-23 Phase separation in T-junction and effect on break flow (B2.10)

C2-23.1 Description of the phenomenon

The concern is to determine the liquid-steam partition at the break according to upstream conditions in the main pipe and break characteristics (break device, break size and orientation).

At low velocities, where the flow tends to stratify, phase separation at branches is dominated by gravity effects. In this case, the flow through the branch will be liquid rich if the branch is at the bottom of the pipe and gas rich if it is at the top. In general the branch flow quality will be greatly dependent on the liquid level in the main pipe. Liquid entrainment and vapor pull-through, which are primarily inertial effects, both exert a powerful influence on this relationship.

At high velocities, the effect of inertial forces on branching flow becomes dominant: the heavier phase (liquid) tends to flow in the straight direction, while the lighter one (steam) is more easily diverted by the low pressure in the side junction of the branch. This causes in most cases a larger void fraction in the side pipe with respect to the value at the branch inlet The mechanism is multidimensional and is significantly affected by all the geometric peculiarities of the system.

The flow regime also plays an important role in this connection. As an example, in bubbly or slug flow, the steam phase can prefer to flow in the side branch as mentioned above, but in annular flow, the liquid on the walls may be diverted more easily to the side branch.

This phenomenon in VVERs is the same as in PWRs.

C2-23.2 Relevance to nuclear reactor safety

In LOCA scenarios, the prediction of the phase separation at branches is important for evaluating the mass and energy lost from the system. Vapor pull-through and liquid entrainment are examples of phenomena expected.

C2-24 Natural circulation core-gap-downcomer, dummy elements

C2.24.1 Description of the phenomenon

This phenomenon is specific for VVER-440. In the core of this reactor there are such unheated channels as gaps between shrouded fuel assemblies, between dummy (shield) assemblies and core barrel and inside shield assemblies. While the core is covered these channels serve as organized downcomers and enhance natural circulation within the core. They also enhance penetration of ECC water into the core from upper plenum and boron mixing within the core.

C2-24.2 Relevance to Nuclear Reactor Safety

Natural circulation of the mentioned type is not very significant to safety but contributes to energy distribution through the reactor vessel.

C2-25 Loop seal behavior in hot leg

C2-25.1 Description of the phenomenon

VVER–440 has loop seal not only in cold leg but in hot one approximately of the same height. Hot loop seal leg prevents reflux condenser mode, i.e. condensate from SG can not flow back into reactor. Condensate plugs which may accumulate in the loop seal have to be expelled periodically to cold leg via

SG if break occurs in the cold leg. At break in the hot leg the loop seal can restrain steam flow to SG that can initiate primary pressure rise and extend the time before accumulators start to inject.

Hydrostatic pressure drop in the hot leg loop seal may be added to the drop in the cold leg loop seal, thus, core uncovery may occur.

The volumetrically scaled integral test facilities tend to exaggerate the effect of hot leg loop seal in some cases. See also items C2.2 and C2.12 above.

C2-25.2 Relevance to Nuclear Reactor Safety

Behaviour of the hot leg loop seal in VVER-440 affect system parameters and may contribute to the core incovery.

C2-26 Recirculation in SG primary side

C2-26.1 Description of the phenomenon

Recirculation in the primary side of horizontal SG takes place at single- and two-phase natural circulation when the friction losses are less than the hydrostatic pressure difference between vertical collectors of SG in their lower part (approximately 1/3 of full height). When inventory in primary or secondary side decreases this phenomenon is not observed.

C2-26.2 Relevance to Nuclear Reactor Safety

In principle, recirculation in the SG primary side affects heat transfer capability increasing in some degree primary pressure and temperature. However, it remains quite enough for decay heat removal. When loss of feed water occurs increased primary pressure serves as initial condition for further pressure increase up to safety relief valves actuation. Nevertheless, secondary water inventory allows to avoid this actuation for a long time.

C2-27 Boron mixing and transport

C2-27.1 Description of the phenomenon

In VVERs boron is added to the coolant and its concentration compensates excess reactivity of a fresh core. As the fuel burn-up increases, this concentration is gradually decreased. Because of high liquid velocity at forced or well-developed natural circulation in primary system the added boron or deborated water may be mixed quite homogeniously without steep gradients. In these conditions inadvertent changes of boron concentration are very slow and may be detected in time.

However, there are possible low flow or stagnant flow conditions, when the local boron concentration in primary system may differ from average one. These situations are related to different scenario at some SBLOCA, primary-to-secondary leak, hot and cold shut down conditions, when slugs of deborated water may form. After re-establishment circulation (forced or natural) slug enter the reactor inserting positive reactivity into the core, that may lead to reactivity initiated accident (RIA).

In this section it will be considered inherent boron dilution (without contribution from external systems), which may occur at SBLOCA or primary-to secondary leak.

When natural circulation has been interrupted and steam from the core condenses in the SG tubes, boron-free condensate accumulates in the cold leg loop seals. After circulation re-establishment diluted slugs are expelled into reactor inserting positive reactivity into the core. Mixing of the slug is very important to mitigate this consequence. In mixing there are involved coolant in the reactor, HPSI borated water and boron solution which is delivered from the SGs to loop seals at early phase of the circulation re-establishment.

Another local inherent boron dilution may occur at final stage of primary-to secondary leak (partial case of LBLOCA), when reversal flow from the SG to primary loop arises. This case is less dangerous because volume of deborated slug is less than in the previous case. Besides, slug may be removed by RCP startup in intact loop when the slug is expelled into upper plenum by reversal flow in emergency loop and mixes well there. In VVER-440 there is possibility to cut off failed SG by main gate valves.

On the contrary, at some SBLOCAs during long-term boiling in the core without natural circulation there is possibility for progressive boron accumulation up to boron deposition threshold. This leads to flow area blockage and overheating of the fuel rods. It should be pointed out that such situation may be reached only after many hours provided:

- no operator actions during this period;

- break size in the cold leg of the loop is small enough, so primary pressure can not decrease naturally down to the actuation threshold of low pressure safety injection for a long time (SGs remain hot);

- high pressure safety injection pumps are connected cold leg.

The boron accumulation is slowed down by excessive ECC water which is partially mixes with the downcomer and lower plenum inventory due to temperature difference and then flows out into break. Mixing may exist for several hours providing moderate rise of the concentration to equilibrium value. At zero temperature difference which may be ultimately reached, this mixing can not continue. Therefore boron accumulation accelerates. Boron carry-over by droplets entrainment is assumed to be negligible because of their separation in upper plenum. Boron dissolving in steam is very small and it has some importance only at very high concentrations. However, it does not prevent crystallization.

When LPSI actuates after forced SG cooling boron accumulated is removed from the core.

C2-27.2 Relevance to Nuclear Reactor Safety

Boron mixing and transport within reactor coolant system is of crucial importance for power control in VVERs especially for fresh core. Mixing mitigates danger of RIA.
Boron accumulation in the core at long-term boiling without operator actions may last up to threshold of crystallization that leads to flow area blockage and fuel rods overheating.

C2-28 Water accumulation in the SG tubes

C2-28.1 Description of the phenomenon

Because of the small inclination of the steam generator tubes there is a possibility that condensate may accumulate in the tubes reducing the heat transfer capability of the steam generator. This is a postulated phenomenon without experimental evidence.

C2-28.2 Relevance to Nuclear Reactor Safety

The phenomenon could affect coolant distribution and reduce heat removal capability of steam generators.

C3-1 Natural Circulation in 1-phase flow (B4.1)

See C2-1.

C3-2 Natural Circulation in 2-phase flow (B4.2)

See C2-2.
In some transients with reactor coolant pump out of operation coolant boils up. Then two- phase natural circulation is an efficient mechanism of removing decay heat.

C3-3 Core Thermalhydraulics (B4.3)

See also C1-11, C2-14 and C2-15.

C3-3.1 Description of the phenomenon

During normal operation the core is cooled by subcooled coolant with surface boiling for VVER-1000 but not for VVER-440. At transients, the conditions for heat transfer may change in a very wide range from forced or natural circulation with or without boiling to critical heat flux and post critical heat flux conditions. In different zones of the core, different heat transfer modes can exist simultaneously.

Most transients start with a decrease of the heat removal from the fuel rods. In consequence, the temperature and pressure rise and the safety margin to boiling crisis at the rod surface decreases.

The VVER-1000 has no specific features in comparison with PWRs, except of different fuel rod pitch and spacing grids. VVER-440 has shrouded fuel assemblies that prevent coolant mixing between assemblies.

C3-3.2 Relevance to nuclear reactor safety

The heat transfer in the core is of high safety importance, because it defines the compliance of the safety criteria.

C3-4 Thermalhydraulics on Primary Side of Steam Generator (B4.4)

C3-4.1 Description of the phenomenon

See also C2-16.
During some transients, such as ATWS with loss of feedwater or station blackout, primary coolant can boil. In this case steam is condensed inside steam generator tubes.

C3-4.2 Relevance to nuclear reactor safety

The thermalhydraulics on the primary side of the steam generators leads to no restricting conditions during transients except for the cases with a lowered secondary level.

C3-5 Thermalhydraulics on Secondary Side of Steam Generators (B4.5)

C3-5.1 Description of the phenomenon

See also C2-17

During some transients, e.g. loss of feedwater or station blackout, the steam generator tube bundle may be partially uncovered and dry out completely. This leads to degradation of the steam generator efficiency. Tube bundle uncovery causes a partial loss of the heat sink and consequent primary pressure and temperature increase. Subsequently it can lead to actuation of pressuriser safety valves and a partial loss of the primary coolant inventory.

During such transients as steam line rupture, the loss of secondary side inventory reduces the overcooling in the loop concerned.

C3-5.2 Relevance to nuclear reactor safety

The thermal hydraulics of the steam generator secondary side is of high importance because it determines the behaviour of the parameters in the primary system and the core.

C3-6 Pressuriser Thermalhydraulics (B4.6)

C3-6.1 Description of the phenomenon

Pressuriser thermal hydraulics is important during

- Feed and bleed operation:
  Steam-water mixture passes through the pressuriser and its safety valves. Phase separation and, finally, level formation in the pressuriser is possible. The outlet mass flow rate via safety valves depends on the degree of separation and the valve flow area. As a result, the primary pressure is stabilised at a defined liquid level.
- Fast pressure increase in the primary system:
  Competing phenomena as steam compression in the pressuriser, which leads to pressure increase and fountain effect of subcooled water entering the pressuriser, which leads to pressure decrease, should be taken into account.

The thermalhydraulic behaviour of the pressuriser in VVERs is almost the same as in PWRs.

C3-6.2 Relevance to nuclear reactor safety

Pressuriser behaviour affects the thermalhydraulic parameters in the primary system and the core.

C3-7 Surgeline Hydraulics (CCFL, choking) (B4.7)

C3-7.1 Description of the phenomenon

During transients the surge line may restrict the flow rate between pressuriser and primary circuit.

- CCFL can occur in the surge line when the coolant in primary circuit is boiling and steam is removed through pressuriser valves. This situation is relevant to feed and bleed operation and other transients, for which the coolant boils.

- Due to the hydraulic resistance of the surge line, the pressures in the primary circuit and pressuriser cannot equalise. This is possible, when the primary pressure rises rapidly (e.g. at ATWS with station blackout or at loss of feedwater).

The thermalhydraulic behaviour of the surge line in VVERs is almost the same as in PWRs although the connection of the surge line to the bottom of the hot leg loop seal in VVER-440s may create specific phenomena.

C3-7.2 Relevance to nuclear reactor safety

CCFL in the surge line may decrease coolant inventory in the primary circuit and can affect the thermalhydraulic parameters in the core during a transient. Code validation against experiments originating from scaled test facilities, must take into account the hydraulic resistance of surge line. The pressure drop in the test facility may exceed the value in the real plant.

C3-8 1- and 2-Phase Pump Behaviour (B4.9)

C3-8.1 Description of the phenomenon

See C1-15

C3-8.2 Relevance to nuclear reactor safety

For all transients without station blackout and with running pumps the pump performance is important because it defines the thermalhydraulic parameters in the core. During transients listed in Matrix III the reactor coolant pump operates in the single phase region. When the pumps are out of operation the natural circulation is important.

C3-9 Thermalhydraulic - Nuclear Feedback (B4.10)

C3-9.1 Description of the phenomenon

An increase in the void fraction in the core means a lower moderator density and reduces the nuclear power generated in the core. Data for reactivity coefficients of specific core loadings are calculated by nuclear design codes. Values at normal operating conditions are confirmed by nuclear measurements and are well founded. VVERs have a different core geometry from PWRs (e.g. hexagonal lattice).
At present the thermalhydraulic system codes contain 1D- and point kinetic models. In code developments underway, the thermalhydraulic codes are coupled with 3D neutron kinetic codes.

C3-9.2 Relevance to nuclear reactor safety

The feedback of the neutron kinetics and thermal hydraulics due to the negative void reactivity coefficient influences the fission heat production in the core and therefore the core cooling.

C3-10 Structural and Heat Losses (B4.11)

C3-10.1 Description of the phenomenon

See C2-21.

C3-10.2 Relevance to nuclear reactor safety

Thermal energy stored in metal structures can be transferred to the fluid in some transients. For long transients, heat losses from different parts of the structure can be significant.

C3-11 Boron Mixing and Transport (B4.12)

C3-11.1 Description of the phenomenon

See also C2-27:

Boron acid in VVERs as in PWRs is used for reactivity control during the core lifetime. Boron concentration in the primary system is at a maximum for fresh core and gradually decreases as the fuel burn-up increases.

When the reactor coolant pumps are in operation there is good mixing of boron in the primary system with no significant concentration changes. However, when the pumps are out of operation the concentration gradients may develop. This situation may be arise in such transients as

1. Steam line break with HPIS actuation. When the HPIS is connected to an intact loop, effectivity of boron solution injection depends on mixing with subcooled water from broken loop.

2. Inadvertent injection of pure water into primary system at shut down conditions with or without reactor coolant pumps start-up.

In this section local external boron dilution in the primary circuit is described. It may occur under different operational conditions when coolant in the loops is stagnant or circulation is very weak. It is assumed that water of low or zero boron concentration may be inadvertently injected to the primary circuit from external sources and form unmixed slugs. There are provided administrative and technical measures to prevent this event, but it is possible due to failures and operator errors. Examples of such may be an eventual injection of deborated water by make-up system and some others at hot or cold shut-down conditions. After start-up of the first RCP deborated slug could be transported into the core during several seconds. This may lead to severe core degradation, especially at cold shut-down. Coolant mixing in downcomer and lower plenum provides an inherent safety mechanism and most important mitigative feature against diluted slugs. When one RCP is in operation, complex 3-D flow pattern arises in downcomer and lower plenum. It can be described by 3-D computational fluid dynamics (CFD) codes, which provide an effective tool for mixing calculations. Both experiments and calculations show that at the core inlet there is well mixing.

There are no principal differences between PWRs and VVERs, but differences in the geometry of primary system need to be taken into account

C3-10.2 Relevance to nuclear reactor safety

The formation of steep boron concentration gradients in the primary system could lead to local criticality with the hazard of reactivity initiated accidents.

There are no principal differences between PWRs and VVERs, but differences in the geometry of the primary system need to be taken into account. The shrouding of the fuel elements in the VVER-440 will prevent crossflow mixing between fuel elements.

Facility	Test	Brief description
SB, EDO, Russia	SB/1	100% double break in cold leg
SB, EDO, Russia	SB/2	1% cold leg break
BD, EDO, Russia	BD-1	Boron dilution
SVD-2, IPPE, Russia	2	Dryout at low pressure
PM-5, IPPE, Russia	6	Loop seal clearance
KS, RRC-KI, Russia	KS/19R/TF84	DNB, dryout
KS-1, RRC-KI, Russia	KS-1/05-91/N024	Heat transfer in covered and partially covered core
ISB-WWER, EREC, Russia	UPB-2.4	2.4% upper plenum break
ISB-WWER, EREC, Russia	NC	Natural circulation
GWP, Skoda, Czech Rep.	DNB-B	DNB in 7-rod bundle
FLORESTAN, KfK-IATF, Germany	FDNR 18	Gravity forced flooding
REWET-II, VTT Energy, Finland	SGI/9	Reflood
REWET-III, VTT Energy, Finland	NC/6	Natural circulation
PACTEL, VTT Energy, Finland	ITE-06	Natural circulation
PACTEL, VTT Energy, Finland	LOF-1	Loss of feedwater
PMK-2, KFKI-AEKI, Hungary	IAEA-SPE4	CLB with secondary bleed and feed

IT FACILITY		No. 1-1
Subject	**Description**	
Test Facility	**PACTEL, VTT Energy, Finland**	
Operating Period Availability	1990 - available	
Objectives **Test Period**	- SBLOCA and Transients in PWR with loop seals and horizontal steam generators 1990 -	
Facility Geometry	- parallel channel test loop (PACTEL) • volumetrically scaled model of WWER-440 PWR, scaling factor 1 : 305 • 3 separate loops with loop seals and horizontal steam generators • 144 full length (2.42 m), 9.1 mm OD, indirectly electrically heated fuel rod simulators, arranged in three parallel channels, nine-step chopped cosine axial power distribution, stainless steel cladding, MgO (Boron nitride in 6 rods) insulation; rod bundle spacers number and form identical to reference plant • reactor vessel simulated by U-tube configuration, i. e. external downcomer • primary pumps simulated by flow resistances (pumps will be added later) • steam generators with 38 U-tubes each, average U-tube length = 8.8 m, tube diameter (13 mm ID) and spacing 1 : 1 • full length (8.8 m) pressurizer (140 mm ID) connected to one hot leg, provided with heater and spray system • low and high pressure injection systems (LPIS, HPIS) and 1 accumulator; 1 LPIS and 1 HPIS pump • injection location: LPIS and accumulator: downcomer and/or upper plenum • HPIS: cold legs • lower plenum: 139.7 mm ID, 2.8/2.6/1.3 m length • core: 182.5 mm ID, 4.54 m length • upper plenum 188.5 mm ID, 5.33 m length • hot leg: 52.5 mm ID, 0.97/3.36/1.28/2.84 m length	
Facility Geometry	• cold leg: 52.5 mm ID, 2.84/0.56/3.17/1.10/1.59/0.32 m length • downcomer: 150 mm ID/0.94 m length, and 73.7 mm ID/4.96 m length • pressurizer line: 27,3 mm ID, 2.02/3.93/1.33/0.48 m length • accumulator line: 27.3 mm ID, 12.62 (to upper plenum)/10.7 (to DC) m length • LPIS line: 27.3 mm ID; HPIS line: 13.0 mm ID • spray line: 27.3 mm ID; feedwater line: 18.3 mm ID	

IT FACILITY			No. 1-2
Subject	**Description**		
Experimental Conditions and Parameter Range	- max. electr. heating power:	1 MW	
	- max. fuel rod simulator temperature:	800 •C	
	- max operating temperature:	8.0 MPa	
	- max. operating pressure:	295 •C	
	- max secondary side pressure:	4.65 MPa	
	- max. secondary side temperature:	260 •C	
	- feedwater tank:	pressure: 2.5 MPa temperature: 225 •C	
	- accumulator::	pressure: 5.5 MPa temperature: 105 •C	
	- ECCs water pressure:	LPIS: 0.7 MPa HPIS: 8.0 MPa	
	- ECCs water temperature:	60 •C	
	- ECCs water flow rate:	LPIS: 0 - 14 I/s HPIS: 0 - 23 I/min	
	- ECCs injection location:	upper plenum and/or downcomer	
Measurement Instrumentation	- temperature: cladding, fluid, structure steam generators: fluid (primary, secondary), tube wall	Chromel-Alumel TCs, mineral insulated =D = ?, sheat =?	
	- diff. pressure: also for water level; ?		
	- pressure: ?	with electromagnetic flow meters (?)	
	- mass flow rate: single-phase: two-phase:	impedance probe + drag disk (?)	
	- 330 temperature, 4D pressure + diff. pressure measurement channels		
	- measurement transducers, range, uncertainty = ?		
Data Acquisition	- Hewlett Packard HP 3852A data aquisition unit, HP9000/360 Work Station		

IT FACILITY		No. 1-3
Subject	**Description**	
Data Documentation	- system description and first results • 2 conference papers 1989, 1990 • Technical Report of Technical Research Centre of Finland, to be published	
Data Availability	- available	
Use of Data	foreseen for: • development and verification of computer codes • development of operator instructions for accidents - basic phenomena involved: • natural circulation as function of power, water inventory • thermal behaviour of PWR components under transients • small and medium break LOCAs	
Special Features	- WWER-440 like triangular rod bundle geometry - PWR-like hexagonal parallel channels - horizontal steam generators - loop seals in hot and cold legs	
Correctness of Phenomena	suitable for code assessment	
Comments	- integral system test facility for WWER-440 reference plant, without pumps, three loops, three parallel channels as core simulator; LPIS, HPIS and accumulator as ECCS; pressurizer with heater and spray system; - primary test objective: natural circulation, SBLOCAs, Transients - interesting and important integral and separate test facility for WWER-440 simulation, with potential for upgrading - information to be completed, see "?"	

IT FACILITY		No. 2
Subject	**Description**	
Test Facility	**PMK-2, KFKI-AEKI, Hungary**	
Operating Period	1985 -	
Availability	available, facility is still in operation	
Objectives	main goals: LOCA and plant transients to study the WWER-specific phenomena with special respect to loop seal behaviour and horizontal steam generator	
Test Period	1985 -	
Facility Geometry	- reference plant: Paks Nuclear Power Plant of WWER-440/213-type, 1375 MWt, hexagonal fuel arrangement	
	- general scaling factor: Power, volumes 1 : 2070, elevations 1 : 1	
	- primary coolant system (1 loop representation)	
	• pressure: 16 Mpa	
	• core inlet temperature: 540 K	
	• nominal core power: 664 kW	
	• nominal flow rate: 4,5 kg/s	
	- special features	
	• 19 heater rods uniform axial and radial distribution	
	• 2,5 m heated length	
	• external downcomer	
	• pump is accommodated in by-pass line: flow rate 0 to nominal value, NPP coast down simulation	
	• loop piping 46 mm ID	
	- secondary system	
	• pressure: 4,6 Mpa	
	• feed water temperature: 493 K	
	• nominal steam mass flow: 0,36 kg/s	
	- special features	
	• horizontal steam generator	
	• controlled heat removal system	

IT FACILITY		No. 2
Subject	**Description**	
Facility Geometry	- safety injection system • high Pressure Injection System (HPIS) • low Pressure Injection System (LPIS) • hydroaccumulators (SIT) • emergency feed water system	
Experimental Conditions and Parameter Range	- initial conditions are the nominal operational conditions of the plant • primary pressure: 12.35 MPa • core inlet temperature: 540 K • core power: 664 kW (nominal), 2000 kW (available) • primary mass flow rate: 4,5 kg/s (nominal) • secondary pressure: 4,6 MPa • feedwater inlet temperature: 493 K • feedwater / steam mass flow: 0,36 kg/s • SIT pressure: 5.9 MPa - HPIS pressure: actirated by SI signal • LPIS pressure: 0.75 MPa - running procedure of experiments - experiments are started from nominal operating parameters. In accordance with the type of experiment, the test is controlled by a process computer. - test fluid is water	
Measurement Instrumentation	- temperatures • coolant: Pt resistance thermometers • heater rods, structures, SG tube walls: chromel-alumel thermocouples - pressures, differential pressures • P and DP transducers - coolant levels • DP transducers (collapsed) • micro void-probes (swell)	

IT FACILITY		No. 2
Subject	**Description**	
Measurement Instrumentation	- flow rate • turbines, ventures - void fraction • gamma densitometer • micro void-probes - measurement accuracy • thermocouples: less than $\pm$ 2 K • Pt thermometers: less than $\pm$ 1 K • pressure transducers. 0.02 to 0.05 MPa • differential pressure transducers: 1 kPa • differential pressure transducers (level measurements): $1.5 \cdot 10^{-2}$ to $5 \cdot 10^{-2}$ m • flow rate: 0.02 to 0.06 kg/s	
Data Acquisition	- HP6942A + IBM AT PC - number of channels: 138 - scanning rate per channel: 1 s	
Data Documentation	data storage: disc, tables, plots specification and measurement reports data qualification	
Data Availability	available	
Use of Data	already used for code validation by Hungarian and international groups	
Special Features	WWER specific phenomena can be investigated.	
Correctness of Phenomena		
Commens		

IT FACILITY		No. 3
Subject	**Description**	
Test Facility	**REWET-III, VTT ENERGY, Finland**	
Operating Period	1985 - 1989	
Availability	dismantled	
Objectives	natural circulation natural circulation with noncondensible gases compensated leak single phase natural circulation	
Test Period	1985 - 1989	
Facility Geometry	- reference reactor: WWER-440 • 19 rod bundle, 9.1 mm OD rod, 2.4 m heated length, triangular • array with 12.2 mm pitch, nine-step chopped cosine power distribution - reactor vessel internals volume scaled 1:2333, elevations scaled 1:1 - emergency core cooling systems (ECCS: • low pressure injection system • accumulator system • injection points in UP and DC - hot/cold leg: 22.0 mm ID - downcomer: 30.7 and 39.4 mm ID, 5.92 and 2.56 m length - core: hexagonal shroud, dhydr. 7.2 mm, 2.547 m length - lower plenum: 56 and 50 mm ID, 0.74 and 2.42 m length - upper plenum: 65 and 83.9 mm ID, 3.54 and 0.8 m length - core inlet tube steam generator • 12 horizontal U-tubes 13 mm ID, 7.7 m length • one tube in a layer, pitch 21 mm	

IT FACILITY			No. 3
Subject	**Description**		
Experimental Conditions and Parameter Range	- natural circulation test:	core power: 15 - 30 kW primary coolant inventory: 60 - 100 % noncondensible gases present in some tests pressure: 0.1 Mpa	
	- compensated leak tests (single phase natural circulation)	core power: 4.5 - 12.0 kW injection flow rate: 0.5 - 2.5 1/min pressure: 0.1 - 0.35 MPa	
Measurement Instrumentation	- 96 measurement channels temperature:	coolant rod cladding	
	pressure:	system pressure differential pressures flow meter in downcomer heating power	
Data Aquisitation	Hewlett-Packard HP 2240A Measurement and control processor IIP 9835A desktop Computer		
Data Documentation **Data Availability**	system description: Research Notes 929, 1989, Technical Research Centre of Finland availability		
Use of Data	verification of RELAP and SMABRE computer codes		
Special Features	horizontal U-tube steam generator, hot leg loop seal		
Correctness of Phenomena			
Comments	data only available on paper		

IT FACILITY		No. 4
Subject	**Description**	
Test Facility Operating Period Availability	**KMS, NITI, Russia** commencement of operation – 1996 construction	
Objectives **Test Period**	- computer codes verification; • algorithm processing of reactor cooldown in emergency mode; • operational capacity and effectiveness validation; emergency core and containment cooling optimization; • complex study of thermohydraulics and physical-chemical processes in emergency simulation modes; • technical and mathematical features checking of process control NPP system ; 1996 -	
Facility Geometry	- volumetrically scaled facility, scaling factor 1:27 - linearly scaled: • high-altitude reactor scale 1:1 • containment scale 1:3 - 4-loop number - reactor model: • core simulator – 7-rod full-scaled non-shroud rod bundle • 2184 –total rod bundles simulators • direct and indirect heating • 9,15 mm outer diameter cladding • 3530 mm-heated length - containment model: • 6,0 m diameter • 20 m height • ~ 2000 m^3 volume	

IT FACILITY		No. 4
Subject	**Description**	
Experimental Conditions and Parameter Range	- simulator reactor power up to 30 Mw_t - primary side pressure up to 18 MPa - primary side temperature up to 350 °C - secondary side pressure up to 12 MPa - feedwater temperature up to 235 °C - containment pressure up to 0,6 MPa	
Measurement Instrumentation **Data Aquisitation**	- about 3000 discrete and analogy parameters, including: • temperature: 1008 measurements • local void fraction: 420 measurements • coolant level: 38 measurements • pressure: 123 measurements • coolant flow, flow velocity: 65 measurements • coolant density: 12 measurements - long-time logging with following magnetic disks archiving • information processing and real-time process diagnostics • reference frequency 1:10 Hz - automatic data acquisition and processing system (NITI working out)	
Data Documentation **Data Availability**	- facility design project - available	
Use of Data	-	
Special Features	- full-scald integrated facility modelling interconnected reactor, safety and containment processes	
Correctness of Phenomena		
Comments		

IT FACILITY		No. 5
Subject	**Description**	
Test Facility Operating Period Availability	**PSB-WWER, EREC, Russia** 1996 - It is constructing	
Objectives **Test Period**	SBLOCA and Transients in WWER-1000 1996 -	
Facility Geometry	- integral model of WWER-1000 • height scale 1:1 • volume-power scale 1:300 • 4 identical loops with SG, pumps and CL loop seals • 168 indirectly heated rods: 3.53 m length, 9.1 mm OD, thriangular array with 12.75 mm pitch, stainless steel cladding + 1 unheated central rod • top rod assembly simulator • reactor model: reactor vessel with core simulator: 4.96 m height, hexagonal form for 168 mm spanner; upper plenum (UP): 6.42 m height, 181 mm ID, with inner structure • external downcomer with lower plenum: 7,07 m total height, 116 mm ID; connecting line, 100 mm ID • core heated by-pass: 4.72 m height, 41 mm ID • primary pumps installed in each loop • steam generator with 34 full length 16 x 1.5 mm diameter slightly inclined (51 mm to outlet header) tubes, secondary side height preserved, SG vessel vertical, 440 mm ID • pressurizer: 12.77 m height, 163 mm ID with electrical heater (80 kW) and spray system • hot leg: 2.43/1.44 m length, 76 mm ID • cold leg: 6.5/1.91/2.62/3.68 m length, 76 mm ID with pump and 2 valves	

IT FACILITY			No. 5
Subject	**Description**		
Facility Geometry	• surge line, 43 mm ID, 1,48/1.08/4.02/2.3/1.1/0.47 m length (connected with loop 4) and 1.48/1.2/4.0/1.4/1.25/0.27 m length connected with loop 2)		
	• 4 accumulators, 6.78 m height, 195 mm ID, connected in pairs with DC an UP		
	• break unit in 9 points of primary system		
	• blowdown tank with 2 measurement parts		
Experimental Conditions	- max. electr. Heating power:	15 MW	
And Parameter Range	- max. fuel rod simulator temperature:	1000 °C	
	- max. operating pressure:	20 MPa	
	- max. operating temperature:	350 °C	
	- max. secondary side pressure:	10 MPa	
	- max. feed water temperature:	270 °C	
	- max. coolant flow:	280 m^3/h	
	- accumulator pressure:	6 MPa	
	- temperature:	60 °C	
Measurement	- 850 measurement channels:		
Instrumentation	• pressure:	in 58 points	
	• temperature:		
	• fluid, structure:	87 Chromel-Copel TCs	
	• fuel rod cladding, SG tubes:	470 Chromel-Copel TCs	
	• diff. Pressure, levels:	at 66 parts of system (including SG, PRZ, AC, condensate tank)	
	• mass flow rates (1-phase):	19 venture and orifices (loops, secondary side, LPIS, HPIS) 1 turbine probe with eddy current detector (downcomer) “?”	
	• mass flow rates (2-phase):	2 full-flow turbine meters (vert. Section of loops) + 2y-densitometers “?”	
	• local void fractions:	28 conductivity probes	
	• heat fluxes:	In 120 points	

IT FACILITY		No. 5
Subject	**Description**	
Data Aquisitation	ADCs: 20 times/s, 12 bits, VME bus with MC 68020, 3 x PC/AT 486, 66 MHz, 16 MB, 540 MB	
Data Documentation	- system description • Research Report 2.387, ENIS, 1991 - Measurements • Research Reports 2.403, 1993; 2.410, 1994, EREC	
Data Availability	Available	
Use of Data	- Foreseen for:	
	• Development and verification of codes	
	• Development of operator instructions for transients and accidents	
	- basic phenomena involved:	
	• 1- and 2-phase natural circulation	
	• thermal behaviour of WWER-1000 components under transients	
	• small and medium break LOCA	
	• large break LOCA “?”	
Special Features	- SGs of special design eight slightly inclined full length tubes	
Correctness of Phenomena		
Comments	- Russian 1st large scale integral test facility for WWER-1000 reference plant with potential for upgrading	

IT FACILITY		No. 6
Subject	**Description**	
Test Facility	**PM-5, IPPE, Russia** (Slab Model with 5 bundles)	
Operating Period	1991 - 1994 (in 1994 it will be reconstructed for full height model with representative models of steam generator and internals of reactor vessel, about 50 % of equipment is already manufactured)	
Availability	available	
Objectives	countercurrent flow limitation (influence of nonuniform heating, fuel assembly shroud, verification of RELAP program) loop seal influence on the instability of processes (frequency and amplitude of pressure, water level fluctuation, verification of programs); heat transfer in partly uncovered core (influence of lateral mixing, water levels, noncondensible gases natural circulation under stable and unstable condition in pressure vessel and primary circuit system safety poisoning by air in case of ht leg breaking boron acid solution behaviour during LOCA (1994-expected)	
Test Period	1991	
Facility Geometry	- integral model of WWER-1000 type reactor. One primary loop with loop seal and heat exchanger. - scales. Elevation scale: m=1:5. Cross-sectional scales (for loop, reactor model): M=1:730 - as number of rods ratio in model and reactor - heated assembly: 5 parallel rod bundles (each of them comprises 14 heated rods arranged in pressure vessel 38*340 mm², downcomer - external; upper tie plate and other internal elements can be used. As front and back walls can be used transparent (up to 500 °C and 1 bar) or metal plates - loop - 60 mm ID, loop seal of two types: 60 and 100 mm ID, elevation scale 1:5, transparent section in horizontal and vertical parts of loop seal (model of 100 mm ID); - steam generator - is not modelled (for full height model SG incorporates 20 full length tubes - other elements: pump (5000 kg/h), water tank, surge lines etc.	

IT FACILITY		No. 6-2
Subject	**Description**	
Experimental Conditions and Parameter Range	- uniform and nonuniform power distribution between 5 bundles; elements; - electrical heating power: 0 - 350 kW, single rod power 5 and 20 kW for two types of elements: - pressure: 1-3 bar (1-1.5 bar for transparent model); - temperature: up to 800 °C for rod cladding; - flow rate: natural circulation in the primary circuit (5000 kg/h, forced circulation), injection with necessary - flow rate in hot and cold legs and pressure vessel	
Measurement Instrumentation	- pressure: top and bottom of the heated section, - pressure drop: axial - 0.5 and the whole heated section, bottom - top of the pressure vessel, 15 pressure half tapping around the primary circuit including loop seal; lateral - on the opposite walls of the pressure vessel at a and full distance of the heated section: - flow rate. orifice pressure drop flowmeters primary and secondary circuits, feed lines), thermocouple signal correlation flowmeter (local velocities at the inlet and outlet of test section) and volumetric (for integral measurement loss of coolant); - temperature: rod cladding (7 elevations), temperature distribution above the test section (5 elevations, traversing in lateral direction), 10 thermocouples around the primary circuit, inlet and outlet temperatures in the secondary circuit of heat exchanger - void fraction: 3 elevations with traversing in lateral direction in loop seal and 5 elevation above test section (resistance probes); - electrical power of 5 bundles (individually)	
Data Aquisitation	data acquisition system includes amplifiers, transducers, etc. It is based on personal computer PC/386 (286). scanning rate - 2 kH, number of channels - 140, the data storage - discs	
	- IPPE reports in 1992 - 1994, - International Heat Transfer Conference TF-92 (Obninsk)	
Data Availability	restricted	

IT FACILITY		No. 6-3
Subject	**Description**	
Use of Data	- qualitative and quantitative analysis of influence of different factors on physical processes: • fuel assembly shroud influence on flooding and heat transfer in uncovered core, • radial power distribution influence on the above mentioned processes, • water injection, geometry and other parameters influence on loop seal behaviour, - development and verification of the RELAP5 code for natural circulation and flooding (1994)	
Special Features	• flexibility of experimental facility (rod bundles with or without shrouds, changeable lateral power distribution etc.); • transparent walls in pressure vessel and transparent section i loop seal	
Correctness of Phenomena		
Comments		

IT FACILITY		No. 7
Subject	**Description**	
Test Facility Operating Period Availability	**SB, EDO, Russia** 1971 - 1985 not available	
Objectives	- core heat transfer during LBLOCA and SBLOCA - break flow - cooldown of rod bundle during refill and reflood phase - critical flow and reactive forces	
Test Period	1971 - 1976 (for WWER-440 and 1976 - 1985 for WWER-1000)	
Facility Geometry	- volumetrically scaled model of WWER-1000, scaling factor 1:3000, elevations 1:1 • 2 separate loops without steam generators, • 7 or 19 rod bundle, 9.1 mm OD, 3.5 m heated length, rods with indirect electrical heating in triangular lattice with 12.75 mm pitch • break size 2 x 30 mmd - volumetrically scaled model of WWER-440, scaling factor 1:300, elevations 1:1, 485 unheated rods in the core, break size 2 x 80 mm	
Experimental Conditions and Parameter Range	- max. power 1000 kW - max. heat flux 1.8 MW/m² - heat flux distribution - uniform and stepwise - pressure - up to 16 Mpa - coolant temperature - up to 320 °C - max. fuel rod simulator - 700 °C - HPIS ECCS water pressure - 6 Mpa - ECCS water temperature - 60 °C - ECCS injection location - upper plenum and/or downcomer	

IT FACILITY		No. 7-2
Subject	**Description**	
Measurement Instrumentation	- temperatures: rod cladding, fluid (chromel-alumel TCs) - pressure: upper and lower plenum, downcomer, break point, some locations in loops - flow rate: single- and two-phase flow through break - power - reactive forces	
Data Aquisitation		
Data Documentation	- THERMALHYDRAULICS" seminar in Warsaw, 1988 - research reports of EDO "Gidropress"	
Data Availability	- restricted	
Use of Data	- development and verification of TECH-M and DYNAMICA computer codes - safety studies for WWER-440 and WWER-1000	
Special Features	- triangular rod bundle geometry of WWER type	
Correctness of Phenomena		
Comments		

IT FACILITY		No. 8
Subject	**Description**	
Test Facility Operating Period Availability	**ISB-WWER, EREC, Russia** 1992 - available	
Objectives **Test Period**	- SBLOCA and Transients in WWER-1000 - 1992 -	
Facility Geometry	- integral model of WWER-1000: • height scale 1:1 • volume-power scale 1:3000 • 2 separate loops with volume ratio 3:1 • intact loop includes 3 SG, broken - 1 SG • 19 electrically heated rods (both directly and indirectly heated ones available): length: 3.53 m, 9.1 mm OD, triangular array with 12.75 mm pitch, stainless steel cladding • reactor vessel simulated by 4 sections: • core simulator (CS), 4.99 m height, hexagonal form for 58 mm spanner • downcomer (DC) with lower plenum, 6.655 m total height, 30 mm ID; 2 parallel connecting lines with CS, 1.04/1.16 m length, 20 mm ID • core by-pass, 4.89 m height, 14 mm ID • upper plenum (UP), 6.43 m height, 56 mm ID, with inner structure; 2 parallel connecting lines with CS, 1.13/1.07 m length, 20 mm ID • primary pumps installed in by-passes of each loop • steam generator: 11 vertical U-tubes, 2.71 m average length, 16 x 2.5 mm diameter, 2.19 m maximum tube • bundle height; SG vessel: vertical, 0.156 m ID, 4.2 m height • pressurizer: 10.9 m height, 56 mm ID (65 mm ID in heater space), may be connected with hot leg of each loop • intact loop:	

IT FACILITY		No. 8-2
Subject	**Description**	
Facility Geometry	• hot leg, 2.596/0.64 m length, 56 mm ID and 3 connecting tubes (with SGs) 0.86 m length, 20 mm ID • cold leg, 5.18/1.43/3.21/3.22 m length, 41 mm ID and 3 connecting tubes, 1.7 m length, 20 mm ID • broken loop: • hot leg, 1.57/1.56 m length, 25 mm ID • cold leg; 6.62/1.1/3.24/2.55 m length, 25 mm ID • surge line, 10 mm ID, 0.39/2.94/1.42/0.24 m length (connected with small loop) and 0.39/2.94/1.42/0.22 m length (connected with lumped loop) • spray line, 10 mm ID, 12.06 m length (with small) and 12.33 m (with lumped) • accumulator (AC) line, 10 mm ID, 3.79/2.78 m length (connected with CS/DC) • connecting line between UP and DC, 3.5 m length, 10 mm ID • break unit in 8 points of primary system • blowdown tank with 2 measurement parts	
Experimental Conditions and Parameter Range	- max. electr. heating power: 1,8 MW - max. fuel rod simulator temperature: 1000 °C - max. operating pressure: 25 MPa - max. operating temperature: 400 °C - max. secondary sode pressure: 13 MPa - max. feed water temperature: 260 °C - max. coolant flow: 8 kg/s - accumulator: pressure 6 MPa, temperature: 60 °C	
Measurement Instrumentation	- 208 measurement channels • temperature: • fluid, structure: 63 Chromel-Copel TCs, • fuel rod cladding, SG tubes: 78 Chromel-Alumel TCs • diff. pressure, levels: at 43 parts of system (including SG, PRZ, AC, condensate tank) • mass flow rates (1-phase): 2 venture (loops) orifices (secondary side, LPIS, HPIS) - local void fractions: 16 conductivity probes • pressure 20 transmitters	

IT FACILITY		No. 8-3
Subject	**Description**	
Data Aquisitation	- PC/AT with DGSh3.036.006 PS	
Data Documentation	- system description and first results • 4 preprints, EREC, 1990, 1993, 1994 • Research Report 3.397, EREC, 1992 • 2 conference papers, 1994	
Data Availability	- restricted	
Use of Data	- foreseen for: • development and verification of codes (RELAP/5Mod3 - Kurchatov inst., DINAMIKA, TECH-4 - EDO "Gidropress", DJIP -NITI, RLAP - FZR-Rossendorf) • development of operator instructions for transients and accidents - basic phenomena involved: • 1 - 2 phase natural circulation • thermal behaviour of WWER-1000 components under transients - • small and medium break LOCA	
Special Features	- vertical SG with primary/secondary sides elevations 1:1 (will be remodelled) - auxiliary electrical heater for stationary tests	
Correctness of Phenomena		
Comments	- 1st integral test facility for WWER-1000 reference plant with potential for upgrading	

IT FACILITY		No. 9
Subject	**Description**	
Test Facility Operating Period Availability	**BD, EDO, Russia** 1994 - available	
Objectives **Test Period**	- a study dispersion of accumulated in cold loop seal pure condensate with coolant on inlet of WWER reactor core during switching circulating pump	
Facility Geometry	- in scale 1:5 model of reactor WWER-1000 with simulation downcomer and lower plenum - one experimental circulating loop with scale 1:5 loop seal and circulator - three working loops without circulator	
Experimental Conditions **and Parameter Range**	- flow rate in experimental loop 800 - 1100 m/h - temperature of coolant up to 80 C - pressure up to 1,0 MPa - flow rate in each working loops up to 10 % from experimental loop flow rate	

APPENDIX D2

SET FACILITY		**No. 1-1**
Subject	**Description**	
Test Facility Operating Period Availability	**KS, EREC, Russia** 1975 - 1989 in stand-by	
Objectives **Test Period**	- steady state critical discharge of water coolant through channels of different geometry - effect of scaling factor on critical flow rates - 1975 - 1989	
Facility Geometry	- 65 experimental channels: - orifices, cylindrical channels, straight or bent tubes with sharp entrance - converging nozzles and Laval nozzles with round entrance - annular channels, long cylindrical channels with intermediate step-shaped contraction - branched channels - valve of special design	
Experimental Conditions and Parameter Range	- initial pressure 0.15 – 9.3 MPa - initial water temperature 10 – 305 °C - initial mixture quality 0 - 1 - initial steam temperature 110 – 400 °C - outlet pressure 0.08 – 9.3 MPa - maximum flow rate 23 kg/s - narrowed cross section diameter 9.79 – 48.5 mm - relative channel length 0 – 500.2 m - channel length 0 – 10.02 m	

SET FACILITY			No. 1-2
Subject	**Description**		
Measurement	- initial pressure	at channel inlet	
Instrumentation	- initial temperature	TCs or special manometric device	
	- initial mixture quality	heat and mass balance or throtting equation	
	- local pressure	manometers or diff. manometers at different points along channel including outlet channel cross section and outlet chamber	
	- flow rates	orifices, nozzles and venture	
	- void fraction	one-beam diverging γ-densitometers	
Data Acquisition	- manual		
Data Documentation	- system description, experimental procedures and results (plots and tables): • 21 research reports, 1975 – 1990, ENIS/NIKIET/IAE/ENIN • 16 reports and papers - results (disc and summary tables): - Summary report on Experimental Data Bank 1.375, 1990, ENIS		
Data Availability	- restricted as a whole		-
Use of Data	- phenomenological and quantitative analysis: summary • methodical direction MU 34-70-142-86, 1986, WTI • Research Report 1.378, 1990, ENIS/NIKIET		
Special Features	- precise measurements during long steady-state tests - large volume of experimental data (5.985 tests)		
Correctness of Phenomena	- test suitable for code assessment		
Comments	- test results useful for critical flow models qualification - test results useful for code assessment		

SET FACILITY		No. 1-3
Subject	**Description**	
Test Facility Operating Period Availability	**TF-102, EREC, Russia** 1978 – 1990 in stand by	
Objectives **Test Period**	- PRZ steam compression during water level increasing - non-condensable gas (nitrogen) effect - 1984 - 1985	
Facility Geometry	- pressure vessel, 1.2 m ID, 10 m height - internal steam heater in lower (water) space - water (saturated or slight subcooled) injection point in lower space (2.1 m from bottom) - nitrogen injection point in upper (steam) space	
Experimental Conditions and Parameter Range	- system pressure: 0.8; 1.0; 1.2; 1.6; 1.8 MPa - steam space height: 2.5; 3.0; 3.5; 4.0 m - nitrogen volume content: 0 %; 5 %; 10 %; - water level growth rate: 0.06 – 0.12 cm/s	
Measurement Instrumentation **Data Acquisition**	- s20 measurement channels: • pressure: system • water level: system • nitrogen content: 2 points in steam space (rotameter and condensate tank) • temperature: diff. TCs at different elevations • water flow rate: 6 – 12 kg/s - HO43.1 oscillographs; KSP-4 devices	

SET FACILITY		No. 1-4
Subject	**Description**	
Data Documentation **Data Availability**	- system description, experimental procedures and results (plots and tables): • Research report 12.307, 1985 ENIS/ENIN - restricted	-
Use of Data	- qualitative analysis and extrapolation on PRZ of AST-500 2nd system • Research report 12.307, 1985 ENIS/ENIN	
Special Features	- AST-500 operational parameters	-
Correctness of Phenomena	- test results suitable for code assessment	
Comments	- test results useful for PRZ-WWER thermalhydraulic model qualification	

SET FACILITY		No. 1-5
Subject	**Description**	
Test Facility Operating Period Availability	**24-MT, EREC, Russia** 1987 - 1988 not available	
Objectives **Test Period**	- thermalhydraulic behaviour of WWER-440 core simulator with free coolant level conditions during cold leg LOCA 1987 - 1988	
Facility Geometry	- 7 rod bundle, 9.1 mm OD, 2.5 m heated length, triangular array with 12.2 mm pitch, 11 grid spacers - reactor vessel, 41 mm ID - direct electrical heating with uniform linear power - core simulator is coupled with auxiliary vessel for coolant level controlling -	
Experimental Conditions and Parameter Range	- bundle power: 0 – 80 kW (0 – 60 kW/m² - system pressure: 0.5 – 3.0 MPa - inlet mass flow rate: 1- 25 kg/m² s - inlet coolant temperature: 50 – 220 °C - coolant mass flow level: 0.75 – 1.5 m	
Measurement Instrumentation	- 34 measurement channels: • temperature: inlet water, outlet steam – 2 TCs; rod cladding – 24 TCs embedded in cladding at different locations • void fraction: 3 one-beam γ-densitometers at different locations • pressure: inlet and outlet • coolant level: diff. pressure along auxiliary vessel water 0 – 0.09 kg/s steam 0 – 0.09 kg/s: condensate tank	
Data Acquisition	- IWK-2 (SM-1420)	

SET FACILITY		No. 1-6
Subject	**Description**	
Data Documentation **Data Availability**	- system description, experimental procedures and results (plots and tables): • Research Report 2.339, 1987 ENIS • Research Report 2.354, 1988 ENIS - restricted	-
Use of Data	- comparison with available correlations for prediction of heat transfer coefficient in nonwetted core zone: - Research Report 2.354, 1988 ENIS	
Special Features	- stationary heat transfer conditions	-
Correctness of Phenomena	- test results suitable for code assessment	
Comments	- test results useful for code verification	

SET FACILITY				No. 2-1
Subject	**Description**			
Test Facility	**GWP, Skoda, Czech Republic** (Great Water Loop)			
Operating Period	1982 -			
Availability	test facility is ready to operation			
Objectives	- core heat transfer (DNB) - quench front propagation (QFP)			
Test Period	DNB: 1982 - 1986; 1993 - 1994 QFP: 1977 - 1984 (small loop); 1990 - 1992 (GWL)			
Facility Geometry	- experimental channels	DNB	QFP	
	- reactor vessel, 41 mm ID inner diameter	120 mm	150 mm	
	- total length	7100 mm	1150 mm	
	- distance between inlet and outlet nozzle	5000 mm	1420 (959) mm	
	- max. number of heated rods	19	37	
Experimental Conditions and Parameter Range	- maximal overpressure	18 MPa (DNB), 16 MPa (QFP)		
	- maximal temperature	320 °C		
	- delivery head	2,2 MPa		
	- test section electrical Input	3,75 MW		
	- voltage	150 V		
	- current	25 kA		
	- max. Flow rate	200 m^3/hour		
	- volume of ECCS Water tank	2 x 100 l (QFP)		
	- break imitation system	dia 23 mm (QFP)		

SET FACILITY		No. 2 - 2
Subject	**Description**	
Measurement Instrumentation	- resistance thermometer Type E - cable Thermocouples NICr - Ni (type K) - dia 1 mm: 0,5 mm - pressure Transducers -ROSSEMOUNT 1151 - shunts - METRA	
Data Acquisition	- number of measurement channels depend of type of experiment - Hewlett Packard 4397A - computer CPU 386/33 - Model IPC 610 - computer CPU 386/33 - Model IPC 610	
Data Documentation	- means of data storage: Hard discs, floppy discs, report form - additional information) FT-88 Seminar: CHF measurement in bundles on GWL SKODA, WARSAW, 1988 (in Russian) Research Report Skoda - Ae 7657: Experimental Results of Quench Phenomena, November, 1991	
Data Availability	- DNB data from test period 1982 - 1986 are available - DNB data from test period 1993 - 1994 are not available - QFP data are available	
Use of Data	- phenomenological and quantitative analysis - comparison with available correlations - development of quench velocity correlations	
Special Features	- DNB - WWER-1000 like triangular rod bundle geometry	
Correctness of Phenomena	- phenomena have been checked	
Comments	- QFP data from test period 1977 - 1984 were performing on "small loop"	

SET FACILITY		No. 3-1
Subject	**Description**	
Test Facility Operating Period Availability	**VEERA, VTT Energy, Finland** 1987 available	
Objectives **Test Period**	- aqueous boric acid solution behaviour during PWR LOCA - refill and reflood phase phenomena of LOCA (after changes to the test facility) • starting at 5 bar effect of various ECC modes (upper plenum, lower plenum, downcomer injection) - 1987	
Facility Geometry	- 126 rod bundle, 1:1 dimensions to reference plant (Loviisa PWR: WWER-440) • 9,1 mm OD, 2,4 m heated length, triangular pitch (12.2 mm, 11 grid spacers, hexagonal shroud • nine-step chopped cosine power distribution • electrical heating: indirect - LPIS to be installed for reflooding tests - components dimension • boron tank: 0.64 m ID, 1.0 m length = 317 l vol- • downcomer (external): 83.9 mm ID, 7.85 m length • lower plenum: 134.7 mm ID, 3.9 m length • core inlet tube: 134.7/214.1 mm ID, 519/440 mm length • core: hexagonal shroud, d_{hydr} = 7.7 mm, 2.56 m length • core outlet tube: 163.3 mm ID, 495 mm length • upper plenum: 162.3 mm, 2.85 m length • hot leg: 56,3 mm ID, 1.025 m length + moisture separator 30.4 l • condenser: 135.7 mm ID, 1.46 m length = 21.1 l vol. • condensate sampling tank: 750 mm ID, 1.5 m length = 663 l vol.	

SET FACILITY		No. 3-2
Subject	**Description**	
Experimental Conditions and Parameter Range	- bundle power: 0 – 120 kW (average linear power 0 – 4 W/cm) - system pressure: 0.5 MPa - max. cladding temperature: 1000 °C - reflood rate: 1 – 10 cm/s water level rise in a cold core - boric acid solution concentration: 0 – 25 g/kg - ECCS pressure, temperature: 20 – 100 °C	
Measurement Instrumentation	- temperatures: 20 coolant temperature measurements (TC, 0.5 mm, 1.5 mm); 26 cladding temperature measurements (TC, 0.5 mm) - absolute pressure: top of the test section - differential pressure: along the downcomer, core plenum and upper plenum - heating power: normal power analyser - boric acid concentration: samples from 12 locations - single phase flow: ECC line, magnetic flow meter 0.05 m/s – 10 m/s	
Data Acquisition	HP 3852 A Data acquisition unit; HP 9000 series 310 micro computer	
Data Documentation	- system description = to be published 1993 - first results: 2 conference papers 1988 and 1990 (aqueous boric acid solution behaviour) - quick look report of first reflood experiments in 1994	-
Data Availability	- restricted	
Use of Data	- development and verification of computer codes: RELAP5 - safety studies of Loviisa PWR: (aqueous boric acid solution behaviour during long term cooling of PWR LOCA) - basic phenomena involved: • mixing of boric acid at core section and between core and lower plenum • crystallisation of boric acid and its behaviour	
Special Features	- WWER-440 like triangular rod bundle geometry	-
Correctness of	- boric acid tests: suitable for code assessment	

SET FACILITY		No. 3-2
Subject	**Description**	
Phenomena	- reflood tests: limited suitability for code assessment	
Comments	- more recent test facility with WWER-440 core geometry - main objectives: boric acid distribution and behaviour during LOCA - reflood experiments will start after the changes to the test facility have been completed (autumn 1993	

SET FACILITY		No. 4-1
Subject	**Description**	
Test Facility Operating Period Availability	**REWET II, VTT Energy, Finland** 1981 - 1989 not available	
Objectives **Test Period**	- reflooding phenomena - simultaneous injection of ECC water into upper plenum and downcomer - effect of spacers on reflooding 1981 - 1989	
Facility Geometry	- 19 rod bundle, 9.1 mm OD rod, 2.4 m heated length, triangular array with 12.2 mm pitch - 3 fuel rod simulator bundles • 2 bundles with nine-step chopped cosine power distribution • 1 bundle with uniform linear power - reactor vessel internals: volume scaled 1 : 2333 elevation preserved 1 : 1 - emergency core cooling systems (ECCS): • lower pressure injection system (LPI) • accumulator system - injection points in upper plenum and downcomer - hot/cold leg: 22.9 mm ID, 5.4/9.47 m length - downcomer (external): 30.7 and 39.4 mm ID, 5.92 and 2.56 m length - core 1 = hexagonal shroud dhydr. = 7.2 mm and 2.547 m length - core 2 = round shroud dhydr. = 14.2 mm - lower plenum : 56 and 50 mm ID, 740 and 2420 mm length - upper plenum : 65 and 83.9 mm ID, 3540 and 800 mm length	-

SET FACILITY		No. 4-2
Subject	**Description**	
Facility Geometry	- core inlet tube: 65 mm ID, 750 mm length - containment simulator: 2000 l volume capacity - water separator: 5.7 l volume capacity	-
Experimental Conditions and Parameter Range	- bundle power: 0 - 90 kW - system pressure: 0.1 - 1.0 MPa - max. cladding temperature: 1000° C - coolant temperature 120° C - flooding rate: 0-15 cm/s - ECC injection location: upper plenum and/or downcomer	
Measurement Instrumentation	- 78 measurement channels • temperature: coolant, rod cladding — TCs at different radial and axial location; rods with cosine power shape: TCs embedded in cladding; rod with uniform power: TCs soldered on cladding • pressure: system; diff. pressure along core simulator • coolant flow rates: 0 - 1.0 kg/s • heating power: 90 kW max	
Data Acquisition	- HP2240 A, Measurement and Control Processor; HP 9835 A, Desktop Computer	
Data Documentation	- system description: Research Notes 9.29, 1989, Techn. Research Centre of Finland - experimental procedures and results: • Acta Polytechnica Scandinavia PH 138, 1983 • Research Report YDI 615, 1986, Techn. Res. Centre of Finland • 6 conference papers 1982 - 1987	-
Data Availability	- available	

SET FACILITY		No. 4-3
Subject	**Description**	
Use of Data	- development and verification of computer codes CATHARE, FLUT, NORCOOL-I - basic phenomena involved: • combined upper plenum and downcomer ECC infection • spacer grid effect on reflooding process • dissolved boron effect	
Special Features	- DNB - WWER-1000 like triangular rod bundle geometry	
Correctness of Phenomena	- suitable for code assessment	
Comments	- reflood and ECC injection data relevant to WWER-440 core geometry	

SET FACILITY		No. 5-1
Subject	**Description**	
Test Facility Operating Period Availability	**IVO-CCFL, IVO, Finland** 1984 – 1988 dismantled	
Objectives **Test Period**	- counter-current flow limitation (CCFL) behaviour for full-scale fuel bundle top area structures of pressurised water reactor WWER-440 and WWER-1000 - 1984 for WWER-1000, January 1986 to December 19888 for WWER-440	
Facility Geometry	- vertical flow channel of transparent acrylic material for visualisation connected to upper plenum and lower collection chamber - different internals of flow channel (test section): • hexagonal perforated upper tie plate in full scale of WWER-1000 (889 circular holes, diameter of holes 5 mm, thickness of plate 20 mm, hydraulic diameter of test section 234 mm) • half-size upper tie plate of WWER-1000 (440 circular holes, diameter of holes 5 mm, thickness of plate 20 mm, hydraulic diameter of test section 165 mm) • full size upper tie plate with elongated holes of WWER-1000 (33 elongated holes, hole dimension 6.4 x 102 mm, thickness of plate 20 mm, hydraulic diameter of test section 234 mm) • round perforated plate (52 circular holes, diameter of holes 5 mm, thickness of plate 20 mm, diameter of tests section 60 mm) - height of test section 1,250 mm, hydraulic diameter of test section 234 mm: • fuel bundle top area structure of WWER-1000 in full scale (combined structure of hexagonal perforated upper tie plate and coons-shaped screen with cylindrical part with control rod guides above it; coons with 88 circular holes, diameter of holes 18 mm, diameter of cylindrical part 170 mm, 18 control rod guides) • combined structure of full-size upper tie plate of WWER-1000 and coons-shaped screen with cylindrical part with control rod guides above it (coons with 22 elongated holes, hole dimensions 18 x 95 mm, diameter of cylindrical part 170 mm, 18 control rod guides)	

SET FACILITY		No. 5-2
Subject	**Description**	
Facility Geometry	- height of test section 3,500 mm, hydraulic diameter of test section 141 mm: • hexagonal perforated upper tie plate in full scale of WWER-440 (21 elongated holes, hole dimensions 7 x 49 mm, diameter of circular hole 12 mm, hydraulic diameter of test section 141 mm: combined structure of full size upper tie plate of WWER-440 and fuel rod bundle of WWER-440 below it (length of unheated rod bundle used in tests 1,220 mm, 126 rods, diameter of rods 9.1 mm, pitch of rods 12.2 mm, 6 honeycomb-type grid spacers for triangular array) • fuel rod bundle of WWER-440 • fuel bundle top area structure of WWER-440 in full scale (combined structure of upper tie plate, rod bundle and upper structures above of full-size upper tie plate of WWER-440 and upper structures above it • full-size upper tie plate with thickness of 20 mm of WWER-440 at height of bottom of fuel rod bundle - height of test section 3,500 mm: • tube (diameter of test section 60 mm) • free-flow channel of hexagonal cross-section (hydraulic diameter of test section 141 mm) - upper plenum (diameter 300 mm, height 3,000 mm for height of test section, height 2,000 mm for height of test section 3,500 mm) - lower collection chamber (diameter of chamber 300 mm, height of chamber 1,000mm)	
Experimental Conditions and Parameter Range	- atmospheric pressure, ambient temperature - air injection by compressor into lower collection chamber - water injection by pump into upper plenum - water drainage through valve from bottom of lower collection chamber: • air flow rate: 0.15 – 312 x 10^{-3} m^3/s • liquid heat: 0.2 mH_2O or 0.8 H_2O - measurements after prolonged steady-state conditions were reached: • registration of average injected water and air flow rates • registration of average mixture level heights above perforated plate or above coons-shaped screen	

SET FACILITY		No. 5-3
Subject	**Description**	
Measurement Instrumentation	- air and water flow rates (rotameters installed at entrance of flow channel) - liquid heads (water manometer) - mixture level heights (visual scale) - flow pattern observations (visual scale)	
Data Acquisition	manual	
Data Documentation	- system description, experimental procedures, results: • Nucl. Eng. Design 102, 171 – 176, 1987 • Fourth International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Karlsruhe, Germany, pp. 82-87, October 1989 • IVO Research Report IVO-A-07/89, Imatran Voima Oy, October 1989 • IVO Research Report IVO-A-07/89, Imatran Voima Oy, October 1989 - Exp. Thermal Fluid Sci3, 581-587, 1990	
Data Availability	- report IVO-A-07b/89 only confidential	
Use of Data	- basic phenomena involved - CCFL across perforated plates, fuel bundle, different core structures - comparison with RELAP/MOD3 calculations: - First Code Application and Maintenance Program Meeting, Villingen, Switzerland, June 1992 (RELAP2/MOD3 Assessment Results in Finland)	
Special Features	- large scale counter current flows in WWER-specific core upper structures - stationary CCFL conditions	
Correctness of Phenomena	- phenomena have been checked - test results suitable for code assessment	
Comments	- interesting experiments	

SET FACILITY		No. 6-1
Subject	**Description**	
Test Facility Operating Period Availability	**IVO-Loop Seal, IVO, SF** (Air/Water) 1984 – 1989 dismantled	
Objectives **Test Period**	- two-phase flow phenomena with loop seal (U-tube between steam generator and reactor coolant pump (RCP) during cold leg break loss-of –coolant accidents (LOCA) of pressurised water reactor (PWR) - 1985 – 1987, 1989	
Facility Geometry	- full scale model of entire cold leg of WWER-1000 PWR • 850 mm I. D. pipe, original length and loop seal depth - break location simulated between RCP and reactor pressure vessel (RPV) - pump simulator with same single-phase pressure drop and overflow edge height as in real pump	
Experimental Conditions and Parameter Range	- atmospheric pressure and room temperature tests - steam flow modelled with air flow (high capacity max 3 m³/s, speed controlled fan): • buffer tank (10 m³) and throttling for control of oscillations • maximum superficial air velocity 9 m/s within cold leg pipe - initial water level height in loop seal ≤ diameter of horizontal pipe section (first test series): • water outlet through valve from bottom of RPV tank • maximum make-up water injection into loop seal 30 kg/s - three test series • onset of slug flow in horizontal pipe section residual water mass in loop seal • two-phase pressure drop over loop seal - initial water level in loop seal at same height as horizontal section between RCP and RPV (second test series): • no water outlet from bottom of RPV tank • air injection by auxiliary compressor (max. 0.25 m³/s) into bottom of vertical pipe section of loop seal • two phase pressure drop over loop seal - break as varied parameter between tests	

SET FACILITY		No. 6-2
Subject	**Description**	
Measurement Instrumentation	- pressure in vertical loop seal legs (top end) - differential pressures in horizontal loop seal leg, three locations (one location for second test series), over pipe cross section (vertical direction) for average void fraction (level height) - air flow measurements: • air velocity in pipe between buffer tank and loop seal with Pitot tube • auxiliary air flow rate with rotameter • make up water flow measurement with magnetic flow meter visual observations of flow regimes through windows in both horizontal and vertical parts of loop seal	
Data Acquisition	Watanabe Multicorder Model MC 641 (six channels) plotter, in conjunction with data acquisition system	
Data Documentation	- system description, experimental procedures, results: • Nucl. Eng. Design 107, 295 – 305, 1988 • Test Report DLV1-G380-0006, Imatran Voima Oy, Hydraulic Laboratory, November 1989 (in Finnish)	
Data Availability	- available	
Use of Data	- basic phenomena involved: • onset of slugging • inclination of horizontal liquid level • droplet entrainment • CCFL in vertical two-phase flow • two phase pressure drop in U-tube • residual water in loop seal - verification of RELAP5 code against data from loop tests: • Third International Topical Meeting on Nuclear Power Plant Thermal Hydraulics and Operations, Seoul, Korea, pp. 180 – 187, November 1988 • International Code Assessment and Application Programme Meeting, Bethseda Maryland, USA October 1989 (Developmental Assessment of RELAP/MOD3 Against Data of IVO Loop Seal Experiments) • USNRC Report NUREG/IA-0082, April 1992	

SET FACILITY		No. 6-3
Subject	**Description**	
Special Features	- WWER-1000 Specific cold leg including loop seal and mock-up for RCP in full-scale	
Correctness of Phenomena	- phenomena have been checked - test results suitable for code assessment	
Comments	interesting data on loop seal phenomena (oscillation, stratification, droplet entrainment pressure drop, CCFL etc) from 1:1 experiments; important for code/model verification since no scaling effects	

SET FACILITY		No. 7-1
Subject	**Description**	
Test Facility Operating Period Availability	**HORUS-II, TH Zittau, Germany** (**HOR**izontal **U-**tube **S**team generators) started in 1991 for investigation of the condensation process in a single horizontal heater tube (WWER-NPP) during SBLOCA available	
Objectives **Test Period** **Facility Geometry** **Experimental Conditions** **and Parameter Range**	- analysis of flow direction and condensation rate - influence of non-condensing gas, mixed into steam - 01/91 – 02/96 - intensive scored single heater tube of WWER-440 (original geometry) in a pressure vessel: • length 9.065 m • diameter 16 x 1.5 m • material DIN 1.4541 (X10CrNiTi18.10) - horizontal secondary pressure vessel: • length 4.70 m • diameter 219 x 8.2 m • material DINB 1.4541 (X10CrNiTi18.10) - temperature up to 300 °C (primary system); up to 300 °C (secondary system) - pressure up to 80 bar (primary); 64 bar (secondary)	
Measurement Instrumentation **Data Acquisition**	- thermocouples - pressure and differential pressure measurements - ultrasonic measurement - steam flow rate instrumentation - gas mass flow controller - Scanner LSB 36 (LINSEIS corp.), PC-data acquisition (PCI system)	

SET FACILITY		No. 7-2
Subject	**Description**	
Data Documentation	- technical reports (no. THZ-61202-01 … 09)	
Data Availability	- available as converted ASCII-files for IBM 6000/Mod 320 workstations) - various data formats (floppy disks, 3.5", 1.44 MB, 5.25", 1.2 MB)	
Use of Data	- analysis of the SBLOCA behaviour - test of heat transfer correlations (condensation in horizontal tubes) - verification of thermalhydraulic computer codes (e. g. ATHLET)	
Special Features	- original geometry and parameters	-
Correctness of Phenomena		
Comments	- variation of the primary and secondary conditions - single effect test facility of WWER-440 and WWER-1000 - counterpart experiments for the integral test facility PACTEL (Fin.) available	

SET FACILITY		No. 8-1
Subject	**Description**	
Test Facility Operating Period Availability	**4-Tube Model, EDO, Russia** 1982 – 1983 available	
Objectives **Test Period**	- non condensable gases effects 1982 – 1983 (with hydrogen), 1990 – 1993 (with nitrogen)	
Facility Geometry	- 4-tube model with heat exchanger of horizontal arrangement - full-length tubes, headers of natural height - model of WWER-1000 reactor: • height: 9.48 m • ID: 90 mm - model circulation loop • length: 8.99 m • ID: 26.0 mm - natural height of loop seal	
Experimental Conditions and Parameter Range	- coolant pressure: up to 3.0 MPa - temperature of steam supplied: up to 234 °C - flow rate of stream – up to 30 kg/h - volumetric concentration of non condensable gas in gas-steam mixture: 0 – 0.98 - secondary pressure: 5.0 – 6.0 MPa	
Measurement Instrumentation **Data Acquisition**	- temperature: surfaces of SG tubes, pipelines, reactor, stream and fluid - pressure in the reactor model - level in the reactor model and in the circulation circuit - flowrate of steam and fluid - IMPACT 3590 Data Acquisition unit	

SET FACILITY		No. 8-2
Subject	**Description**	
Data Documentation	- "Thermalhydraulics" seminar, 1984 - 3 papers on conferences, 1983 - 1990 - research reports of EDO "Gidropress", 1983, 1992	
Data Availability	restricted	
Use of Data		
Special Features		
Correctness of Phenomena		
Comments		

SET FACILITY		No. 9-1
Subject	**Description**	
Test Facility Operating Period Availability	**Thermal Mixing, EDO, Russia** 1982 - 1986 dismantled	
Objectives **Test Period**	- thermal mixing of cold high pressure injection (HPI) water with hot primary coolant legs, downcomer and lower plenum of WWER-440	
Facility Geometry	- geometric scaling 1:7.4 in Fraud number model with 6 short-circulated loops, downcomer and lower plenum - annular downcomer circumference - injection cold water from high-pressure vessel in cold leg N 6 - cold legs: without loop seals and reactor coolant pumps, inner diameter: 67 mm • downcomer: height: 1,020 mm, gap from 20 mm to 36,5 mm • HPI nozzles: from bottom of cold leg perpendicular< to pipe centre, distance from reactor vessel model nozzles 180 and 710 mm, inner diameter 33.5 mm each	
Experimental Conditions and Parameter Range	- operating pressure 12.3 MPa - operating temperature up to 300 °C - flow rate of flooded water from 0.5 to 1.6 m/h - temperature of flooded water from 20 to 90 °C - initial water level in the model from 0 (model is completely filled) to –0.5 m	
Measurement Instrumentation **Data Acquisition**	- 75 Chromel-Alumel TC's 1.0 mm OD in downcomer at pressure vessel wall and wall layer - inaccuracy of used thermocouples e 3 C - flow rate of flooded water: by a set from orifice plate, a differential pressure gauge - computer complex M-6000	

SET FACILITY		No. 9-2
Subject	**Description**	
Data Documentation	- experimental procedures and results: • research reports EDO "Gidropress" - 2 conference papers	
Data Availability	available	
Use of Data	- basic phenomena involved: thermal mixing and stratification of HPI water in cold legs and downcomer	
Special Features		
Correctness of Phenomena		
Comments		

SET FACILITY		No. 10-1
Subject	**Description**	
Test Facility Operating Period Availability	**SKN, EDO, Russia** 1969 - available	
Objectives **Test Period**	- Core heat transfer, including CHF, dryout and post-CHF conditions 1969 – 1986 (for WWER-440 and WWER-1000)	
Facility Geometry	- 7 rods and 119 rod bundles, OD: 9.1 mm; heated length: 1.75-3.5 m; triangular array with 12.2 and 12.75 pitch - annular channel, • rod OD: 9.1 mm, • ID channel: 14 mm; • heated length: 3.5 m - 19 fuel rod simulator bundles: • 16 bundles with uniform axial power distribution • 3 bundles with non-uniform axial power distribution	
Experimental Conditions and Parameter Range	- bundle power: 0 – 1,000 kW, max. heat flux 2.5 MW/m² - max operation pressure 16.7 MPa - max. fuel rod simulator temperature: 800 °C - max. coolant temperature: 350 °C - coolant mass flow rate: 700 – 4,200 kg/m²s	
Measurement Instrumentation **Data Acquisition**	- temperatures of rods (chromel-alimel TCs) - pressure at inlet and outlet of test section - mass flow rate: single phase fluid power - IMPACT 3590 Data Acquisition unit	

SET FACILITY		No. 10-2
Subject	**Description**	
Data Documentation	- 4 papers in 1974 – 1984 on "THERMALHYRAULICS" seminars - "Teploenergetika", 1976, No. 2	
Data Availability	available	
Use of Data	- development and verification of ALPHA computer module - safety studies for WWER-440 and WWER-1000	
Special Features	- triangular rod bundle geometry of WWER type	
Correctness of Phenomena		
Comments		

SET FACILITY		No. 11-1
Subject	**Description**	
Test Facility Operating Period Availability	**KS, RRC-KI, Russia** since 1970 available	
Objectives **Test Period**	- DNB, dryout - since 1970	
Facility Geometry	- 7, 13, 19, 37 rod bundles • OD rod: 9 mm • heated length: 2.5 m • spacer pitch:240, 250 mm • triangular array with 12.2 mm pitch • uniform axial power distribution • uniform, non-uniform radial power distribution	
Experimental Conditions and Parameter Range	- heat fluxes: up to 3 MW/m^2 - system pressure: 3 - 10 MPa - mass velocity: 200 - 40000kg/(m^2s) - outlet quality: 0 - 0.35	
Measurement Instrumentation **Data Acquisition**	- heating power - inlet mass flow - outlet pressure - coolant temperature at inlet and outlet - temperatures of interside surface of rod cladding at outlet for different radial locations - manual	

SET FACILITY		No. 11-2
Subject	**Description**	
Data Documentation	- TF-74 Seminar: Investigations of CHF in Fuel Bundles. Moscow, 1974 (in Russian) - TF-84 Seminar: Thermotechnic Safety of WWER nuclear reactors. Varna, 1984 (in Russian)	-
Data Availability	- available	
Use of Data	- development of RRC KI CHF-correlation - verification of Groeneveld method used in RELAP5/MOD3 code	
Special Features	- wide array of CHF data for WWER core modelling test bundles	-
Correctness of Phenomena	- test results suitable for CHF models assessment	
Comments	- CHF test data relevant to WWER core geometry (with triangular array)	

SET FACILITY		No. 12-1
Subject	**Description**	
Test Facility Operating Period Availability	**KS-1, RRC-KI, Russia** since 1979 available	
Objectives **Test Period**	- heat transfer in covered core - heat transfer in partially uncovered core - since 1990	
Facility Geometry	- core: 19 rod bundle • OD rod: 9 mm • heated length: 2.5 m • triangular array with 12.75 mm pitch • 11 spacers • uniform axial power distribution • uniform radial power distribution - downcomer (external): 80 mm ID - closed circuit available does not model WWER circulation circuit; heat transfer goes by heat losses and because of steam drainage; coolant circulation goes by natural convection	
Experimental Conditions and Parameter Range	- heat fluxes: 3 - 60 kW/m² - system pressure: 1 - 8 MPa - inlet temperature: 50 - t_S °C	

SET FACILITY		No. 12-2
Subject	**Description**	
Measurement Instrumentation	- 100 measurement channels: • temperature of coolant at different axial locations (TC) • temperature of rod cladding at different radial and axial locations (TC) • absolute pressure • differential pressure • water mass flow • heating power	
Data Acquisition	- CAMAC + IBM PC/AT	
Data Documentation	- system description, experimental procedures, quick-look analysis and first results: Report RRC KI no. 32 / 1-1386-91 (in Russian)	
Data Availability	- available	
Use of Data	- verification of computer codes ATHLET and RELAP5/MOD3 - verification of models for heat transfer in partially uncovered core	
Special Features	- WWER-1000 like triangular rod bundle geometry with 12.75 mm pitch - test results suitable for code assessment	
Correctness of Phenomena	- test results suitable for code assessment	
Comments	- thermalhydraulics of partially uncovered core relevant to WWER-1000 geometry (12.75 mm pitch)	

SET FACILITY		No. 13-1
Subject	**Description**	
Test Facility Operating Period Availability	**SVD-1, IPPE, Russia** functioning facility yes	
Objectives **Test Period**	- study of thermalhydraulic characteristics (local and average cross-sectional) under nominal and accident conditions (small LOCA; power and flow rate variation according to given law); uncovery and reflooding in rod models of WWER-type reactor fuel assemblies. ?	
Facility Geometry	- high pressure facility N1 (SVD-1) • two-loop scheme • circulation loop comprises 2 circulation pumps, coolers, pre-heater, air separator, mechanical filter, steam superheater, pipelines and valves • heat removal from the circulation loop is implemented by technical water after magnetic treatment and hydrocyclonic cleaning • distilled water subjected to ionite treatment is used as coolant in circulation loop - test arrangements are bundles (from 1 to 19 each of them) of either: • rods OD: 9.1 mm • heated length 3,500 mm - or 9 rod bundle with • rods OD: 9.5 mm • heated length: 3,658 mm - both direct (due to passing direct heating of rods is used; uniform and non-uniform axial power distribution	
Experimental Conditions and Parameter Range	- maximum electrical power supply to test arrangement • without steam superheater: 2.4 MW • with steam superheater: 1.2 MW - voltage: to 250 V - current: to 20 kA - power generated in preheater: 600 kW - power generated in steam superheater: 800 kW	

SET FACILITY			No. 13-1
Subject	**Description**		
	- maximum operating pressure	19.6 MPa	
Measurement Instrumentation	- maximum operating temperature of collant at the experimental apparatus inlet	to 400 °C	
	- maximum operating flow rate of coolant	12 m³/h	
	- maximum operating pressure	1.0 MPa	
	- temperature of heater rod wall and coolant:	Chromel-Alumel thermocouple with diameter of 1 mm, Chromel, Aumel or Copel thermocouples with diameter of 0.2… 0.3 mm	
	- coolant flow rate:		
	- local coolant velocity:	constricting devices; turbine transducers-flowmeters	
	- pressure difference, coolant level:	correlative analysers, isokinetic probes	
	- local quality	differential manometer with electrical unified output signal	
	- coolant pressure:	conductometric probes + secondary transducer	
	- electrical power:	manometers with electrical unified output signal	
	- current, voltage:	Hall-effect-based measuring transducers measuring transducers with electrical unified output signal	
	- total amount of channels is to 200, about 40 of them – are for measurement of pressure, flow rate, pressure difference, electrical parameters; individual calibration of each measuring transducer		
Data Acquisition	acquisition of experimental data: measuring terminal + PC AT with displaying the current operating information to operator and outputting alarm and protection signals; measuring terminal is modelled for tasks of a particular experiment; speed of response is to 3,000 measurement/s		
Data Documentation	- under preparation for the press	-	
Data Availability	?		
Use of Data	- designated for: • development of closing equations, development and verification of computer codes • improvement of fuel assembly and spacer grid designs		

SET FACILITY		No. 13-3
Subject	**Description**	
Special Features	- experimental arrangements simulating WWEWR-type reactor core fuel assemblies with triangular and square spacer grids - validated measurements procedures of local longitudinal and lateral parameters by using probes of different kinds - devices for probe travelling on the height and over the cross-section of rod bundle and output of required number of transducers from the high pressure zone - possibility of supplying single-phase (water, overheated steam and two-phase coolant to the inlet)	
Correctness of Phenomena		
Comments		

<table>
<tr><th colspan="2">SET FACILITY</th><th>No. 14-1</th></tr>
<tr><th>Subject</th><th>Description</th><th></th></tr>
<tr><td>Test Facility
Operating Period
Availability</td><td colspan="2">SVD-2-I, IPPE, Russia
functioning facility
yes</td></tr>
<tr><td>Objectives

Test Period</td><td colspan="2">- thermalhydraulics of multirod (to 37 rods) full height fuel assembly models for WWER-type reactor under nominal and accident conditions (small LOCA; large LOCA, reflooding, primary circulating pump with current cut-off Thermalhydraulics of loops with natural circulation, including two parallel channels under nominal and accident conditions. Thermalhydraulics of WWER reactor core spational model with seven 19-rod bundles under nominal and accident conditions. Development and verification of closing equations, computer programs and codes.
1992 - 1994</td></tr>
<tr><td>Facility Geometry</td><td colspan="2">- two independent circulation loops, each having two fuel assembly models, both of which can be connected to the one core model simultaneously. Natural circulation loop including two parallel loops
• full height (3.530 m long) WWER 19-37 rod bundle (rod diameter of 9.1 mm) with direct and indirect heating
• full simulation of the option and arrangement of spacer grids and non-fuel elements for directly heated models
• in addition (for indirectly heated models), absolute simulation of outlet pipeline, with perforated grid and RCS tube location simulated
• 5-7-step axial distribution of power and 2-4-step radial one
• stainless steel cladding

- full height natural circulation loop for two full scale 18-rod channels with directly heated heater rods (5 m in length and 13.6 mm in diameter), with separators, jet pumps, in relation to MKER-reactor design

- WWW-1000 (500) model cross-section reduced to a scale of 1:260, including 133 full length heater rods with indirect electrical heating (heater rod outer diameter 9.1 mm, heated length is 3.530 m)
• use of free loops as the secondary circuit and emergency cooling systems</td></tr>
</table>

SET FACILITY			No. 14-2
Subject	**Description**		
Experimental Conditions and Parameter Range	- max. electrical power supplied to experimental arrangement		9.0 MW 8 m³/h
	- voltage of rectified current		225 V
	- current		40 kA
	- max. operating pressure		26 MPa
	- max. coolant temperature at the model inlet		400 °C
	- max. flow of coolant (water)		70 m³/h
	- max. coolant temperature at the model outlet		510 °C
	- power generated in the preheater		1.5 MW
	- max. feedwater flow rate		8 m³/h
	- fast-response electrohydraulic flow rate control system with arbitrary law of flow rate variation		
	- power control thyrister system in secondary independent channels with power changing according to laws being adequate to accident ones		
	- max. pipeline diameter at the model inlet		50 mm
	- max. pipeline diameter at the model outlet		56 mm
	- max. diameter of feedwater pipelines		40 mm
Measurement Instrumentation	- temperature of: heat rod wall:	Chr-Al, Chr-Co thermocouples with diameter of 1 mm	
	coolant:	Chr, Al, Co wire thermocouples with diameter of 0.2, 0.3 mm	
	- coolant flow rate:	constricting devices; turbine transducers-flowmeters	
	- local coolant velocity:	correlative analysers, isokinetic probes	
	- pressure difference, coolant level:	differential manometer with electrical unified output signal	
	- local quality:	conductometric probes + secondary transducer	
	- coolant pressure:	manometers with electrical unified output signal	
	- electrical power:	shunts + measuring amplifiers	
	- voltage:	measuring transducers with electrical unified output signal	
Measurement Instrumentation	- measurement accuracy	for pressure	0.5 %
		for flow rate	1.0 %
		for temperature	1.0 %
		for power	0.5 %

	SET FACILITY	No. 14-2
Subject	**Description**	
Data Acquisition	- total number of measuring channels is to 400, to 80 of them are designated for measurement of coolant pressure, flow rate and temperature, electrical parameter; individual calibration of each of the measuring transducers. - Acquisition of experimental data: measuring terminal + PC AT with displaying the current operating information to the operator and outputting alarm and protection signals; measuring terminal can be modified for tasks of a particular experiment; speed of response is to 3000 measurement /s	
Data Documentation **Data Availability**	- data storage: disk, plots, tables - under preparation to the press are reports dealing with study of: • temperature behaviour of central heater rod of full height fuel assembly< model • critical power and temperature behaviour of 19-rod bundle model at emergency shut-off of 1-4 circulating pumps • critical power of full height 19-rod bundle at a law (to 0.3 bar) pressure and mass flow rates from 0 to 300 kg/(m^2s)	
Use of Data	- designated for: • development and verification of closing equations, computer programs and codes • optimisation of fuel assembly and spacer grid designs, with basic processes considered • natural circulation round the core, depending on coolant level, pressure and power • natural circulation in the presence and absence of the jet pump, hydrodynamic instability in parallel channel systems • design basis transient and accident conditions (primary circulating pump cut-off, small and large LOCA, refloods)	
Special Features	- experimental arrangements simulating fuel assemblies with triangular and square spacer grids for VVER-type reactor cores - measurement procedures for local axial and cross-sectional rod bundle parameters by using different types of probes - devices for probe movement on the height and over the cross-section of a rod bundle - transducer outputs from the high pressure zone - possibility of supplying single-phase and two-phase coolant to the inlet	
Correctness of Phenomena		
Comments		

SET FACILITY		No. 15-1
Subject	**Description**	
Test Facility Operating Period Availability	**EVTUS, IPPE, Russia** 1980 - available	
Objectives **Test Period**	- CHF - reflooding phenomena 1991 - 1993	
Facility Geometry	- tube: 11 mm ID, 13 mm OD, 3 m length	
Experimental Conditions and Parameter Range	CHF - power: 0 – 140 kW - pressure: 1.1 – 16.7 MPa - coolant flow rates: 0 – 0.1 kg/s - coolant inlet temperature: 9 – 50 °C - Reflood - power: 0 – 120 kW - pressure: 0.1 – 1.3 MPa - coolant flow rates: 0. – 0.3 kg/s - inlet cooling temperature: 5 - 20 °C - max. tube temperature: 1,000 °C	
Measurement Instrumentation	- temperature: coolant: TCs (1.0 mm) on inlet and outlet outside tube wall: 20 TCs (0.5 mm) along the test section (1 %) - pressure inlet and outlet (1 %) pressure drop over the test section (1 %) - flow rate: orifice pressure drop pressure drop flowmeters (2 %) turbine flow (2 %) - electrical power 150 kW max. (2 %)	
Data Acquisition	- Multiplexer CAMAC 752, ADC 712, PC HP Vectra ES. - Number of channels: 100; • scanning rate: 3 kH; - storage means • disc.	

SET FACILITY		No. 15-2
Subject	**Description**	
Data Documentation	- means of data storage: disc., plots, tables - additional information: • IPPE reports: 1992, 1993 • CAMP Meeting, October 19 – 21, 1992, Washington, USA • Heat Transfer Conf. TF-92, October 200 – 22, Obninsk, Russia • Technical Committee Meeting on Thermalhydraulics of Cooling Systems in Advanced Water Cooled Reactors, May 25 – 27, 1993, Villingen, Switzerland	-
Data Availability	restricted	
Use of Data	- phenomenological and quantitative analysis of CHF and re-wetting - verification of RELAP 5 code reflood model	
Special Features	- influence of inlet and outlet throttling on CHF - archive (data bank) evaluated data	
Correctness of Phenomena		
Comments	- interesting data regarding relation between CHF and critical flow	

SET FACILITY		No. 16-1
Subject	**Description**	
Test Facility Operating Period Availability	**TOPAZ, MPEI, Russia** 1974 – 1991 available	
Objectives **Test Period**	- reflood phenomena - quench front velocity and heat transfer above quench front bottom injection - effect of spacer grids at reflood - effect of top. simultaneous top/bottom ECC water injection - 1974 – 1991	
Facility Geometry	- 7 rod bundle, • rod OD: 9.1 mm • heated length: 2.16 m • triangular array with 12.2 mm pitch, • hydraulic diameter 8.5 mm • rods from zirconium alloy - two types of shroud: • optically clear quartz glass tube with OD: 9.11 mm; thickness: 0.65 mm • steel tube with separate source of electrical power supply - spacer grids: of WWER-440 type with distance between them of 250 mm - ECCS system: LPIS with bottom, top or combined injection	
Experimental Conditions and Parameter Range	- atmospheric pressure - bundle power: 0 – 890 kW; heat flux 0 – 40 kW/m² - maximal cladding temperature 1,200 K - inlet coolant temperature 20 – 95 °C - quench front velocity 0.010 – 0.075 m/s - ECC water injection: directly into bundle (bottom, top, top/bottom)	

SET FACILITY		No. 16-2
Subject	**Description**	
Measurement Instrumentation	25 channels: - coolant temperature: • TCs at inlet • TCs at outlet beyond the separator for vapour temperature measurements - tube temperature: 114 TCs embedded in tube at 0.5 mm depth • 8 TCs at central tube at the following elevations (0 mm – inlet): 125 mm, 375 mm, 625 mm, 875 mm, 1.125 mm, 1.375 mm, 1.625 mm, 1.875 mm • 6 TCs in other tubes at the following elevation:375 mm, 625 mm, 875 mm, 1.125 mm, 1.375 mm, 1.625 mm - power: wattmeter, ampermeter, voltmeter - flow rate: volumetric method, delta P transducer, rotameter at the entrance line - flow regime observation: visual method, high speed camera, movies	
Data Acquisition	- records on magnetic tape of 16 temperatures (260 cycles per second) - processing with EC IBM-compatible computer and generation of plots of cladding temperatures, heat fluxes, heat transfer coefficient versus time	
Data Documentation	- Transaction of ANS, Vol. 57, p. 384, 1988 - NURETH-4 Topical International Meeting, Karlsruhe, V. 1, pp. 584-590 October 1989 - Preprint IAE-4118, 1985 (in Russian) - Final MPEI Report, No. 80070956, 1986 (in Russian) - Dynamics of thermal hydraulic processes in energy facilities Proceedings, Irkutsk, pp. 39 – 56, 1989 (in Russian)	
Data Availability	- available excepting Final MPEI Report	
Use of Data	- verification of CATHARE, RELAP4 and RELAP5 codes - basic phenomena involve: • quench front velocity at bottom injection • heat transfer coefficient above quench front • heat transfer enhancement due to spacer grids for bottom reflood • criteria for transition form inverted annular flow dispersed flow • water drop size measurements	

SET FACILITY		No. 16-2
Subject	**Description**	
	• comparison of different types of ECC water injection (top, bottom, top/bottom)	

SET FACILITY		No. 16-3
Subject	**Description**	
Special Features	- WWER-440 rod bundle geometry	
Correctness of Phenomena	- phenomena were checked with the tests of k. Rust, P. Ihle at KfK carried out with different fuel rod simulators - test results are suitable for code assessment to WWER-440 and probably to WWER-1000	
Comments	- the reflood data received are interesting due to WWER-440 real geometry	

SET FACILITY		No. 17-1
Subject	**Description**	
Test Facility Operating Period Availability	**TVC-440, EDO, Russia** 1975 - available	
Objectives **Test Period**	- refill and reflood phase phenomena of LOCA 1975 - 1980	
Facility Geometry	- 126 rod bundle; 1:1 dimension to reference plant (WWER-440) • OD: 9.1 mm • heated length: 2,5 m • triangular pitch 12.3 mm • 11 grid spacers • hexagonal shroud • uniform power distribution • electrical heating: indirect - LPIS: for reflooding test - downcomer (external): • ID: 80 mm • length: 4.28 m - upper plenum: • ID: 152 mm • length 3.0 m • hot leg ID: 26 mm • cold working leg with loop seal	
Experimental Conditions and Parameter Range	- bundle power: 50 – 500 kW - system pressure: 0.1 – 0.5 MPa - max. cladding temperature: 690 °C - ECCS, temperature: °C	

SET FACILITY		No. 17-2
Subject	**Description**	
Measurement Instrumentation **Data Acquisition**	- temperature: 48 cladding temperature measurements (TC, 0.5 mm) 13 shroud temperature measurements (TC, 0.5 mm, 0.1 mm) - absolute pressure: to9p of the test section - two-phase mass flow: in hot leg and working loop by turbine flowmeter and gamma density unit	
Data Documentation **Data Availability**	- experimental procedures and results: • 2 research reports EDO "Gidropress" ?	
Use of Data	- verification of computer subroutine "ZALIV"	
Special Features		
Correctness of Phenomena		
Comments		

SET FACILITY		No. 18-1
Subject	**Description**	
Test Facility Operating Period Availability	**SG.NPP, NPP NV-AEP, Russia** put in operation in 1992 available	
Objectives **Test Period**	- determination of steam generator thermal power under condensation of steam produced (generated) in reactor core - influence of non-condensable and radiolytic gases on heat exchange - validation of methods and selection of locations of gas removal from primary circuit - heat exchange between steam generator piping and secondary side water - steam-gas mixture recirculation inside the piping and its influence on heat power - steam and condensate flows hydrodynamics in steam generator piping - every year (during May-October)	
Facility Geometry	- full scale facility is obtained by using of one circulation loop of 1 unit of NVAES NPP • it contains steamgenerator and primary side pipeline of Dnom = 500 mm • heated surface area of 1,290 m² is produced by 2,074 heat exchanging pipes with size 21 x 1.5 mm, steamgenerator vessel diameter is 3,000 x 65 mm • The facility is equipped with a source of heating steam and pipelines with Dnom = 200 mm that deliver it separately to hot and cold legs of the loop; cylinder-stored gas system for delivery of Nitrogen, Helium and Nitrogen-Helium mixture to the flow of heating steam; steamgenerator feedwater system and secondary steam condensation system • cold and hot legs of the loop are provided with condensate removal system • cold leg of the loop is equipped with steam-gas mixture sampling system	
Experimental Conditions and Parameter Range	- heating steam flowrate: 0 – 500 kg/hour - heating steam temperature: 158 °C - pressure 0.6 MPa - feedwater temperature 15 – 20 °C - before the start-up of the experiment a needed water reserve s produced and the it is heated up to saturation temperature and during the experiment no feeding is available	

SET FACILITY		No. 18-2
Subject	**Description**	
Measurement Instrumentation	- 8 low-inertion thermocouples with output through capillary tubes are located inside every steamgenerator header • steam and water volumes of steamgenerator each contain 1 low-inertion thermocouple - surface thermocouples are located on the perimeter of the steamgenerator and circulation loop pipelines cross-section (minimum 6 thermocouples in every cross section) - pressure in heating and secondary steam paths is measured by reference manometer - heating steam flowrate is measured by orifice plate - per cent concentration of Nitrogen and Helium in the mixture is measured by chromatography	
Data Acquisition	- temperature is registered by automatic recording devices	
Data Documentation	- two scientific reports concerning the results of experiments are issued by institute "ATOMENERGOPROJEKT"	
Data Availability		
Use of Data	- calculation codes for investigation of accident processes	
Special Features	- the results are intended for B-320, B-440 reactor types as well as for new generation reactor units	
Correctness of Phenomena		
Comments		

SET FACILITY		No. 19-1
Subject	**Description**	
Test Facility Operating Period Availability	**IF-NC, NITI, Russia** 1993 - available	
Objectives	- WWER-640 emergency pool cool down with loss of primary coolant (natural circulation) - Experimental investigations of core space thermalhydraulic processes with non-shroud rod bundle at low working coolant parameters and low pressures (0.1 – 0.5 MPa)	
Test Period	1993 -	
Facility Geometry	- Volumetric scaled facility, scaling factor 1:1,000; - high-attitude scale of natural circulation circuits elements 1:1; - core model: • 80-rod (5 x 16) rectangular cross-section rod bundle with in-direct heating; • square pitch 13 mm • 9.1 mm outer diameter; • 1.5 m heated length - emergency pool model: • 510.0 mm diameter; • 6.0 m height; - fuel pool model: • 170.0 mm diameter • 6.0 m height • natural circulation circuit pipeline (15 – 40 mm diameter)	
Experimental Conditions and Parameter Range	- bundle power 0 – 300 kW - system pressure: 0.1 – 0.5 MPa - coolant flow: -500 … +500 kg/m²s - temperature: 20 – 150 °C	

SET FACILITY		No. 19-2
Subject	**Description**	
Measurement Instrumentation	- 60 coolant temperature measurement (TC, 1.0 mm); - 12 core model rods temperature measurements (TC, 1.0 mm); - single-phase coolant flow: 4 measurements; - absolute pressure: 2 measurements - pressure drop: 5 measurements; - coolant level: 5 measurements - void fraction: 2 measurements - heating power	
Data Acquisition	Module data collection system (NITI working out) IBM/AT 386	
Data Documentation	- facility description is given in reports; experimental results are not published	-
Data Availability	- available	
Use of Data	- WWER-640 safety studies: - Verification of PARNAS, DJIP (NITI); RELAP5 computer codes	
Special Features		
Correctness of Phenomena		
Comments	Unique data are received on natural circulation low pressure free level coolant in complex geometry loops (vapour separations, oscillations, stratification)	

SET FACILITY		No. 20-1
Subject	**Description**	
Test Facility Operating Period Availability	**FLORESTAN, KfK-IATF, Germany** 1985 - 1991 available	
Objectives **Test Period**	- forced and gravity reflooding phenomena - ECC water injection into lower plenum / downcomer - effect of spacers (helical fins / grid spacers) on reflooding behaviour - burst tests (Zircaloy claddings) 1985 - 1991	
Facility Geometry	2 hexagonal 61-rod bundles: - p/d = 1.06, 10.1 mm OD rod, heated length: 2.024 m, 6 integral helical fins, stainless steel claddings - p/d = 1.24, 9.5 mm OD rod, heated length: 2.024 m, grid spacers, Zircaloy claddings - total length of fuel rod simulator bundle: 5.6 m - upper plenum: approx. 90 l volume capacity - lower plenum: approx. 60 l volume capacity - containment simulator: approx. 150 l volume capacity - water separator: approx. 78 l volume capacity	
Experimental Conditions and Parameter Range	- total electrical bundle power: up to 170 kW - system pressure: 0.28 MPa - 0.41 MPa - max. cladding temperature: approx. 1000 °C - coolant temperature: 130 °C (110 °C) - flooding rate: 2.5 - 16.0 cm/s - ECC injection: lower plenum / downcomer	

SET FACILITY		No. 20-2
Subject	**Description**	
Measurement Instrumentation	- 256 m measurement channels - temperature: approx. 180 rod temperatures, coolant temperatures - pressures - coolant flow rates - heating power	
Data Acquisition	- DEC PDP 11 - IBM 3090	
Data Documentation	- IBM magnetic tape called IRB229.FDWR 1B.DATA - KfK Report: KfK 5121, November 1993	
Data Availability	available on request (Dr. F.J. Erbacher, KfK/IATF, P.O. Box 3640, D-76021 Karlsruhe)	
Use of Data	- development and verification of computer codes: e. g. ATHLET, RELAP	
Special Features	- data base for hexagonal fuel elements of innovative reactor designs, e. g. WWER-type, high converters	
Correctness of Phenomena		
Comments	- data base for reflooding phenomena relevant to WWER-core geometry	

SET FACILITY		No. 21-1
Subject	**Description**	
Test Facility Operating Period Availability	**Mixing Model, FZR, Germany** Started in 1992 for vibration measurements Available, facility has to be completed for mixing tests	
Objectives **Test Period**	- coolant mixing in WWER-440 pressure vessel	
Facility Geometry	- WWER-440/230 pressure vessel + internals • Scale: 1:10 (in all directions), volume scale: 1:1,000 • 6 loops, not true geometry • circulation pumps • plane pressure vessel head • core barrel support skirt • core simplified fuel elements (pressure drop kept) • upper core structure • control elements inlet/outlet nozzles	
Experimental Conditions and Parameter Range	- forced circulation, variation of flow rates in separate loops - temperature up to 100 °C - atmospheric pressure - max. flow rate m^3/h	
Measurement Instrumentation **Data Acquisition**	- facility not equipped with instrumentation for coolant mixing experiments, facility has to be completed with thermocouples/concentration sensors	
Data Documentation **Data Availability**		

SET FACILITY		No. 21-2
Subject	**Description**	
Use of Data	- calibration and verification of 3D coolant mixing models (boron- and cold plug problem	
Special Features	- unique facility in respect to the volumetric scaling	-
Correctness of Phenomena	-	
Comments	the facility was planned, erected and is used for pressure vessel internals vibration measurements under consideration of fluid-structure interactions	

SET FACILITY		No. 22-1
Subject	**Description**	
Test Facility Operating Period Availability	**7 Assembly, EDO, Russia** 1978 - available	
Objectives **Test Period**	- comprehensive study of hydraulic and vibration characteristics of WWER-1000 fuel assemblies of various modifications 1978 - 1992	
Facility Geometry	- 7 assembly column with internals and main circulation pipelines with 400 mm ID - 2 symmetrical loops - each fuel assembly consists of 312 fuel rods, 9.1 mm OD and 18 channel for absorber elements height of column: 13,800 mm, OD: 1,130 mm	
Experimental Conditions and Parameter Range	- coolant pressure: up to 1.0 MPa - coolant temperature: up to 80 °C - coolant flow rate: 3 through the column: up to 4,200 m³/h	
Measurement Instrumentation **Data Acquisition**	- 7 stationary Pito tubes - 10 pressure measurements - 10 pressure drip measurements - measurement of fuel assembly vibrations	
Data Documentation **Data Availability**	research reports of EDO “Gidropress” - restricted	

SET FACILITY		No. 22-2
Subject	**Description**	
Use of Data	- hydraulic and vibration studies of the WWER-1000 fuel assemblies of various modifications	
Special Features	- triangular rod bundle geometry of WWER-1000 type	
Correctness of Phenomena		
Comments		

Appendix E

Requirements for VVER Thermalhydraulic Code Validation Matrix Database

The Support Group is of the opinion that for efficient and effective code validation a VVER Thermalhydraulic Code Validation Matrix Database similar to the OECD/NEA TH Databank should be established. Documentation required for inclusion of a test facility/test data the VVER TH Code Validation Matrix Database should be sufficient to prepare independent input models for thermal-hydraulic computer code analysis of the test and for data analysis. For this documentation and the data following structure is recommended:

Abstract

An abstract is necessary for searches and available information review. The abstract shall contain the following information:

1. Name of facility
2. Test designation
3. Description of the test facility (type of facility, i.e. integral, separate effects; reactor type which is simulated; main scaling parameters).
4. Description of the test (purpose and scope of the test, main initial and boundary conditions, results).
5. Experimental limitations and shortcomings.
6. Phenomena tested.
7. Special features of the experiment.
8. Counterpart experiments.
9. References.
10. Name and address of organization (s) responsible for the test, name and address of organization which operates the facility.
11. Full description of all data files, including format and size specifications.

Facility Description

Facility description must be provided to give future users of the data sufficient understanding of layout and size and allow developing code model of the facility. The facility description shall include:

1. Drawing showing the overall facility layout.
2. “As built” drawings showing major components and pipe connections in the facility (isometrics if possible).
3. Component and pipe sizes; hydraulic characteristic of special components such as valves, pumps etc. Calculated volumes, surfaces, areas, effective thickness, mass etc. are highly desirable.
4. Details of test sections were appropriate.

For plants, the above data may be made available in form of well-commented code input models.

Instrumentation and Data Acquisition System

Experimental facility or plant instrumentation should be discussed in general. Both types and number of instruments available should be described on facility drawing including:

1. Location and type of instrumentation.
2. Uncertainty bands given by vendors or calculated.
3. List of instruments from which recordings will be available (qualified data).
4. Description of the data acquisitions system including recording equipment, response time and sampling frequency.

Initial and Boundary Conditions

Documentation shall be provided that will allow independent understanding of the test conduct and allow adequate modelling using computer codes.

1. Detailed list of initial conditions (power, system pressures, temperatures, pressure differentials, flows, pump speeds, component conditions, valve open or close, heaters on or of, etc.)
2. Test procedure, i.e. timing of events, configuration changes and all actions taken during the test. Also all deviation of the test procedure and data acquisition procedure should be listed.
3. Evaluation of heat losses and their distribution.

Test Data

The experimental result data should be stored in electronic form for easy access, distribution and processing. Digitized computer readable data should be submitted (plots and graphs will be provided as additional information only). The computer readable data should be provided in ASCII or EBCDIC formats. Following materials should be provided:

1. Data report containing data in graphical and/or tabular form. The data report should clearly identify directly measured parameters and derived (computed from other measurements) parameters. Data uncertainty should be identified. Conversion algorithms should be provided.

2. Detailed description of the content of the electronic information. Each file should be described, a tape, diskette, or other medium map should be provided showing format, coding and file size. This should include a cross-reference identifying instrument, recording channel and location in files.

3. The data encoded in ASCII or EBCDIC should be provided on floppy disks, cartridge tapes or provided directly via FTP.

   - raw data

   - qualified data in engineering units

Annex 1

List of Support Group Members involved in Preparation of the Report

Bulgaria

Mr POPOV, Emilian Ljubomirov	ENERGOPROEKT-PLC 51 James Boucher 1407 Sofia Tel.: +359 2 682 175 Fax: +359 2 689 311 Email: nep@enpro.bg

Czech Republic

Ms KRHOUNKOVA, Jelena	Thermal-Hydraulic Analyses Department Nuclear Research Institute Tel: +420 2 6617 2447 Fax: +420 2 685 7954 email: krj@nri.cz
Dr MACEK, Jiri	Thermal-Hydraulic Analyses Department Nuclear Research Institute Rez plc 25068 Rez Tel: +420 2 20940 954 Fax: +420 2 20940 954 email: mac@nri.cz

Finland

Dr HOLMSTROM, Heikki	VTT Energy Nuclear Research Institute P.O. Box 1604 Espoo 02044 Tel: +358 9 456 50 50 Fax: +358 9 456 50 00 email: heikki.holmstrom@vtt.fi
Mr KOUHIA, Jyrki	VTT Energy Nuclear Research Institute P.O. Box 20 Lappeernranta 53851 Tel: +358 5 621 23 75 Fax: +358 5 621 23 79 email: jyrki.kouhia@vtt.fi

France

Dr REOCREUX, Michel

IPSN/DRS/DIR
Bat 250
CEA-CEN Cadarache
13108 St. Paul-lez Durance Cedex
Tel: +33 4 42 25 31 48
Fax.: +33 4 42 25 70 80
email: michel.reocreux@ipsn.fr

Mr SAINT-RAYMOND, Philippe

Directeur adjoint
DSIN
99, rue de Grenelle
75353 Paris 07 SP

Germany

Dr GLAESER, Horst

Gesellschaft für Anlagen- und Reaktorsicherheit
(GRS) mbH
Forschungsgelände
85748 Garching
Tel: +49 89 3 20 04-4 08
Fax: +49 89 3 20 04-5 99
email: gls@grs.de.de

Mr KREPPER, E.

Zentralinstitut für
Kernforschung Rossendorf e. V.
Postfach 51 01 19
01314 Dresden
Tel: +49 351 260 34 60
Fax: +49 351 260 23 83
email: krepper@fz-rossendorf.de

Mr LIESCH, Klaus

Gesellschaft für Anlagen- und Reaktorsicherheit
(GRS) mbH
Forschungsgelände
85748 Garching
Tel: +49 89 3 20 04-4 03
Fax: +49 89 3 20 04-5 99 or – 90 14
email: lie@grs.de

Hungary

Dr SZABADOS, Laszlo

KFKI Atomic Energy Research Institute
P.O. Box 49
1525 Budapest
Tel: +36 1 395 90 50
Fax: +36 1 395 92 93
email: szabados@sunserv.kfki.hu

Mr TOTH, Ivan
AEKI Atomic Energy Research Institute
P.O. Box 49
1525 Budapest
Tel: +36 1 395 90 41
Fax: +36 1 395 92 93
email: tothi@sunserv.kfki.hu

Dr VÖRÖSS, Lajos
Deputy Director General
Hungarian Atomic Energy
Authority HAEC
P. O. Box 676
1539 Budapest 114
Tel: +36 1 356 55 66
Fax: +36 1 355 15 91

Italy

Prof. D'AURIA, Francesco
Universita degli Studi di Pisa
Costr. Meccaniche e Nucleari
Via Diotisalvi, 2
56126 Pisa
Tel: +39 050 58 52 53
Fax: +39 050 58 52 65
email: dauria@ing.unipi.it

Poland

Dr SZCZUREK, Jan
Institute of Atomic Energy
05 400 Otwork-Swierk
Tel: +48 22 77 79-85 90
Fax: +48 22 77 79-38 88
email: jansc@cxl.cyf.gov.pl

Russian Federation

Dr BEZRUKOV, Yuri
EDO Gidropress
Ordzhonikidze St. 21
Moscow Region
14103 Podolsk
Tel: +7 09675 2 12 01
Fax: +7 09675 4 25 16
email: bezrukov@grpress.msk.ru

Dr Prof. BLINKOV, Vladimir N.
First Deputy Director
Electrogorsk Research & Engineering Center
EREC Nuclear Plant Safety
Bezymaynnaya 6
Moscow 142530
Tel: +7 096 4 33 30 74
Fax: +7 096 4 33 23 22
email: blinkov@ami.rospac.ru

Dr DEVKIN, A.	RRC Kurchatov Institute NSI Kurchatov Square 1 Moscow 123182 Tel: +7 095 196 93 20 Fax: email:
Dr ELKIN, Ilya	RRC Kurchatov Institute NSI Kurchatov Square 1 Moscow 123182 Tel: +7 095 196 762 Fax: email: elkin@nsi.kiae.ru
Mr LIPATOV, I	Electrogorsk Research & Engineering Center EREC Bezymaynnaya 6 Moscow 142530 Tel: +7 096 4 33 15 45 Fax: +7 096 4 33 05 15 email: psb@ami.rospas.ru
Mr LOGVINOV, Serguey	EDO Gidropress Ordzhonikidze St. 21 Moscow Region 14103 Podolsk Tel: +7 096 7 52 12 01 Fax: +7 096 7 54 25 16 email: bezrukov@grpress.msk.ru
Dr OUGOLEVA, Inna	Science Engineering Safety Center under Gosatomnadzor 14/23 Avtozavodskaya St. Moscow 109280 Tel: +7 095 2 75 55 48 Fax: +7 095 2 75 55 48 email:
Dr POMETKO, R.	IPPE Bondarenko Square 1 Kaluga Region Obninsk 249020 Tel: +7 084 3 99 86 91 Fax: +7 095 2 30 23 26 email: pometko@ippe.obninsk.com
Dr SUSLOV, Alexander	RRC Kurchatov Institute NSI Kurchatov Square 1 Moscow 123182 Tel: +7 095 196 91 76 Fax: +7 095 196 61 72 email: ais@inr.kiae.ru

Dr ZAITZEV, S.	EDO Gidropress Ordzhonikidze St. 21 Moscow Region 14103 Podolsk Tel: +7 096 137 91 08 Fax: +7 096 75 4 25 16 email: bezrukov@grpress.msk.ru
Slovakia	
Mr HATALA, Branislav	VÚJE - Výskumný ústav jedrových elektrárni Trnava a.s Okruzna 5 91864 Trnava Tel.: +421 (805) 5 99 11 72 Fax: +421 (805) 5 99 17 08 hatala@vuje.sk
Mr HUSARCEK, Jan	Director Safety Analysis and Technical Support Dept. NRA SR Bajkalska27 P.O. Box 24 820 07 Bratislava 27 Tel: +421 7 572 32 68 Fax: +421 7 293 603 email: Jan.Husarcek@ujd.gov.sk
Mr KLIMENT, Tomas	VÚJE - Výskumný ústav jedrových elektrárni Trnava a.s Okruzna 5 91864 Trnava
Ukraine	
Dr GROMOV, Gregory	Head of Div. – Ministry of Environmental 35 – 37 Radgospna 03142 Kiev Tel: +380 44 452 99 35 Fax: +380 44 452 99 25 email: gromov@sad.gan.kiev.ua
Dr SHIKHABUTINOV, Valery	Safety Assessment Division Scientific & Technical Centre Nuclear Regulatory Admin. 17 Kharkovskoye Shosse Kiev 253160 Tel: +380 44 417 71 75 Fax: +380 44 417 71 75 email: xxx@sad.gan.kiev.ua

United Kingdom

Dr KIMBER, George R.	AEA Technology plc Performance and Safety Services Dept. A32, Winfrith Techn. Center Dorchester, Dorset DT2 8DH Tel: +44 1305 20 23 54 Fax: +44 1305 20 25 08 email: george.kimber@aeat.co.uk
Dr LILLINGTON, John N.	AEA Technology plc Performance and Safety Services Dept. 364/A32, Winfrith Techn. Center Dorchester, Dorset DT2 8DH Tel: +44 1305 20 21 06 Fax: +44 1305 20 25 08 email: john.lillington@aeat.co.uk

United States of America

Dr MODRO, Slawomir Michael	Idaho National Engineering & Environmental Lab. P.O. Box 16 25 Idaho Falls, IDAHO 83415-3895 Tel: +1 208 526 94 02 Fax: +1 208 526 05 28 email: mod@inel.gov
Dr SIENICKI, James	Reactor Engineering Division Argonne National Lab. 9700 South Cass Ave Building 206 Argonne, ILLINOIS 60439 Tel: +1 630 252 48 48 Fax: +1 630 252 60 80 email: sienicki@jacobi.re.anl.gov

OECD/NEA

Mr DROZD, André	OECD Nuclear Energy Agency Le Seine Saint-Germain 12 Boulevard des Iles 92130 Issy-les-Moulineaux France Tel.:+33 1 45 24 10 58 or 82 00 Fax: +33 1 45 24 11 10 email: drozd@nea.fr
Mr HREHOR, Miroslav	OECD Nuclear Energy Agency Le Seine Saint-Germain 12 Boulevard des Iles 92130 Issy-les-Moulineaux France Tel.:+33 1 45 24 10 58 Fax: +33 1 45 24 11 10 email: miroslav.hrehor@oecd.org

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