LAPPEENRANNAN TEKNILLINEN KORKEAKOULU
LAPPEENRANTA UNIVERSITY OF TECHNOLOGY

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RESEARCH PAPERS

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Thermal-Hydraulic Studies on the Safety Of VVER-440 Type Nuclear Power Plants

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ABSTRACT

This thesis includes several thermal-hydraulic analyses related to the Loviisa VVER-440 nuclear power plant units. The work consists of experimental studies, analysis of the experiments, analysis of some plant transients and development of a calculational model for calculation of boric acid concentrations in the reactor.

In the first part of the thesis, in the case of simulation of boric acid solution behaviour during long-term cooling period of LOCAs, experiments were performed in scaled-down test facilities. The experimental data together with the results of RELAP5/MOD3 simulations were used to develop a model for calculations of boric acid concentrations in the reactor during LOCAs. The results of calculations showed that margins to critical concentrations that would lead to boric acid crystallization were large, both in the reactor core and in the lower plenum. This was mainly caused by the fact that water in the primary cooling circuit includes borax ($Na_2B_4O_7 \cdot 10H_2O$), which enters the reactor when ECC water is taken from the sump and greatly increases boric acid solubility in water.

In the second part, in the case of simulation of horizontal steam generators, experiments were performed with PACTEL integral test loop to simulate loss of feedwater transients. The PACTEL experiments, as well as earlier REWET-III natural circulation tests, were analyzed with RELAP5/MOD3 Version 5m5 code. The analysis showed that the code was capable of simulating the main events during the experiments. However, in the case of loss of secondary side feedwater the code was not completely capable to simulate steam superheating in the secondary side of the steam generators.

The third part of the work consists of simulations of Loviisa VVER reactor pump trip transients with RELAP5/MOD1-Eur, RELAP5/MOD3 and CATHARE codes. All three codes were capable to simulate the two selected pump trip transients and no significant differences were found between the results of different codes. Comparison of the calculated results with the data measured in the Loviisa plant also showed good agreement.

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Finally, I would like to thank my wife, my mother and grandmother and my sisters for their love and support during all these years.

Garching 15.10.1993 *Jari Tuunanen*

CONTENTS

FIGURES

ACRONYMES

AFW	Auxiliary Feedwater
APROS	Advanced Process Simulator
ATHLET	Analysis of Thermal-hydraulics of Leaks and Transients (code)
ATWS	Anticipated Trancient Without Scram
BWR	Boilig Water Reactor
CATHARE	Code for Analysis of Thermal-hydraulics During an Accident of Reactor and Safety Evaluation
CCTF	Cylindrical Core Test Facility
DBA	Design Basis Accident
DNB	Departure From Nucleate Boiling
DNBR	Departure From Nucleate Boiling Ratio
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
IVO	Imatran Voima Oy
KFKI	Central Research Institute for Physics
LPI	Low Pressure Injection
LPCI	Low Pressure Coolant Injection
LBLOCA	Large-break Loss-of-Coolant Accident
LOCA	Loss-Of-Coolant Accident
LOFT	Loss-of-Fluid Test Facility
LTKK	Lappeenranta University of Technology
LWR	Light Water Reactor
NPP	Nuclear Power Plant
PACTEL	Parallel Channel Test Loop
PKL	Primärkreislaufe
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
REA	Reactor Laboratory
RELAP	Reactor Leak and Analysis Program (code)
RHRS	Residual Heat Removal System
ROSA	Rig of Safety Assessment

RPV	Reactor Pressure Vessel
SCTF	Slab Core Test Facility
SPE	Standard Problem Exercise
SBLOCA	Small-break Loss-of-Coolant Accident
STUK	Finnish Centre for Radiation and Nuclear Safety
UPTF	Upper Plenum Test Facility
VTT	Technical Research Centre of Finland
YDI	Nuclear Engineering Laboratory

PAPERS ON THIS THESIS

1. Kervinen, T., Tuunanen, J., 1987, "Emergency Cooling Experiments with Aqueous Boric Acid Solution in the REWET-II Facility", Trans. Am. Nucl. Soc., Winter Meeting, Los Angeles, pp. 466-467.

2. Tuunanen, J., Kervinen, T., Kalli, H., Tuomisto, H., Komsi, M., Markkanen, E., 1988, "Long-Term Emergency Cooling Experiments with Aqueous Boric Acid Solution with the REWET-II and VEERA Facilities", Proceedings, Int. ENS/ANS Conference on Thermal Reactor Safety, Avignon, Vol. 4, pp. 1428-1436.

3. Tuunanen, J., Kervinen, T., Kalli, H., Tuomisto, H., Komsi, M., Markkanen, E., 1990, "Experiments with the REWET-II and VEERA Facilities on the Behaviour of Aqueous Boric Acid Solution During PWR LOCAs", Proceedings, ENC'90, Lyon, France, Vol. 2, pp. 911-915.

4. Raussi, P., Tuomisto, H., Kervinen, T., Tuunanen, J., 1991, "Mixing of Boric Acid During Long-term Cooling Period of Loss-of-Coolant Accidents", Proceedings, Int. Topical Meeting on Safety of Thermal Reactors, Portland, Oregon, pp. 336-340.

5. Tuunanen, J., Raussi, P., Tuomisto, H., "Experimental and Analytical Studies of Boric Acid Concentrations in a VVER-440 Reactor During Long-term Cooling Period of Loss-of-Coolant Accidents", Nucl.Eng. and Des. (To appear).

6. Tuunanen, J., 1992, "Assessment of RELAP5/MOD3 Code Against Natural Circulation Experiments Performed with the REWET-III Facility", Proceedings, 113th ASME Winter Annual Meeting, HTD-Vol. 209, Anaheim, Cal., pp. 43-51.

7. Tuunanen, J., Nakada, K., 1993, "Analysis of the PACTEL Loss of Secondary Side Feedwater Tests with RELAP5/MOD3 code", Proceedings, ASME National Heat Transfer Conference, HTD-Vol. 245, Atlanta, Georgia, pp. 81-91.

8. Tuunanen, J., Lahti, K., 1993, "Analysis of Loviisa VVER-440 Reactor Pump Trips with the RELAP5/MOD1-Eur, RELAP5/MOD3 and CATHARE Codes", Proceedings, 6th Int. Top. Meeting of Nuclear Reactor Thermal-hydraulics (NURETH6), Grenoble, France, Vol. II, pp. 311-320.

1. INTRODUCTION

About 430 nuclear power plant (NPP) units are now in operation worldwide. Most of the units are comprised of so called light water reactors (LWRs). These reactors use water as a coolant and moderator, with slightly enriched uranium (about 3 % U-235) as fuel. Most of the light water reactors are pressurized water reactors (PWRs), about 240 in number, two of them at the Loviisa power plant in Finland. The other main type is the boiling water reactor (BWR). There are about 90 such reactors in the world, two of them in Olkiluoto on the Western coast of Finland.

Of the over 240 operable PWRs, 44 reactors are of the Soviet VVER type. Most of these reactors are in the former Soviet Union (24 reactors). The rest are located in Bulgaria (6), Hungary (4), former Czechoslovakia (8) and Finland (2). The first generation VVERs were developed by the Soviets between 1956 and 1970. The first prototype reactors were of the VVER-210 and -365 type and they were built in Novovoronezh. These reactors were shut down in 1984 and 1990, respectively. The first standardized model of VVERs was VVER-440 Model V230 and the first power plant units of this type were Novovoronezh 3 and 4. At the moment there are 10 reactors of this first generation in operation, six in Russia and four in Kozloduy, Bulgaria /1/, /2/.

The principal strengths of the VVER-440 Model V230 units are that they have six primary coolant loops (providing multiple paths for cooling of the reactor), each with horizontal steam generators that together provide a large volume of coolant. They also have isolation valves in all six coolant loops that allow plant operators to take one or more of the loops out of service for repair while continuing to operate the plant.

The principal deficiencies of this power plant type include an accident localization system (instead of containments in Western nuclear power plants) designed to handle only 4-inch pipe ruptures. If a larger coolant pipe(s) ruptures, this system vents directly to the atmosphere. Furthermore, no emergency core cooling (ECC) or auxiliary feedwater (AFW) systems exist. The gradual weakening through embrittlement of the pressure vessel surrounding the nuclear fuel causes concern, as well as plant instrumentation and controls, safety systems, fire protection systems and protection of control room which are far below Western standards.

The second generation VVER plants, designated VVER-440 Model V213, were designed between 1970 and 1980. Model V213 plants include two units in Russia, two in Ukraine, six in former Czechoslovakia, four in Hungary and two in Finland. The principal strengths of the VVER-440 Model V213 units include an upgraded accident localization system, comparable to several Western plants, and the use of a vapour-suppression containment structure called a "bubbler-condenser" tower. Furthermore the improvements include addition of the emergency core cooling and auxiliary feedwater systems and a reactor pressure vessel (RPV) with stainless steel lining to alleviate concerns about vessel embrittlement in the first generation model design. Model V213 plants also include improved coolant pumps and continued use of six coolant loops with horizontal steam generators with a large coolant volume. Finally, the standardization of plant components in this design provides extensive operating experience for many parts and makes possible incremental improvements and backfits of components.

The principal deficiencies of this power plant type are the instrumentation and control systems, fire protection, and protection of the control room, which are all well below Western standards, as well as insufficient separation of plant safety systems. Moreover, the weaknesses include the unknown quality of the plant equipment and construction due to poor documentation, and major variations in operating and emergency procedures, operator training and operational safety among the plants.

The third generation VVER (VVER-1000) was developed between 1975 and 1985 based on the requirements of a new Soviet nuclear standard. These reactors include six in Russia, ten in Ukraine and two in Bulgaria. After 1985, derivative versions of the VVER-1000 were developed. The VVER-88 concept is a basic VVER-1000 with post-Chernobyl improvements and it is the basis for the Khmelnitsky-5 in Ukraine, which is scheduled to begin operation in 1994. In 1987, design work began on the VVER-1800, a VVER-1000 upgraded for greater safety and economy. In 1989, Finland and the Soviet Union jointly announced the start of development work on the VVER-91, a VVER-1000 version that would meet stringent Finnish nuclear plant design requirements. Thereafter, a further modified and improved VVER-92 design has been developed.

The VVER-440 power plant units have some unique features differing from Western PWR designs. One of these is the number and type of steam generator. As mentioned earlier, the VVER-440 units have six primary loops with isolation valves and horizontal steam generators. Furthermore, the primary and secondary pressures and temperatures are lower than in Western PWRs and the primary loops have loop seals in both hot and cold legs. The construction of the reactor core is also unconventional. The hexagonal fuel rod bundles are in channels as in BWRs and due to a special control element design the core is shorter than in most PWRs, about 2.5 m long. During reactor shut-down steam generators are used for residual heat removal instead of a separate residual heat removal system (RHRS) as in other PWRs. The special features of the VVER-440 power plants and their effects on safety have been recently discussed by Laaksonen /3/.

The Finnish VVER-440 units differ substantially from the other VVER-440 Model V213 units due to the extensive modifications made by the Imatran Voima Oy (IVO) at the Loviisa power station. These modifications include an ice-condenser containment building of the Westinghouse PWR design. To cope with severe accidents an external containment spray system has been installed and reactor pressure vessel external cooling concept has been planned as an accident management feature into the Loviisa units. On the steam generator secondary side an additional feedwater system have been implemented to secure the heat removal capacity of the steam generators if the normal feedwater systems fail. Furthermore, the plant instrumentation and control systems as well as the control room have been built according to Western standards. The Loviisa plant also has a full-scope training simulator for operator training.

2. A CLASSIFICATION OF NUCLEAR REACTOR ACCIDENTS

The main cause of concern in generating nuclear power is the radioactivity of the fission products and actinides accumulating inside the fuel rods during power generation. In the beginning of the fuel cycle the fission products stay inside the ceramic fuel pellets. Later they will partly accumulate also in the open space left into the rods. The ceramic fuel is thus the first barrier against radioactive release into the environment. In order to reach the environment, the radioactive fission products have to pass through several other barriers. The next two barriers are the cladding and the primary cooling circuit. Furthermore, most PWRs and BWRs have a containment building, which in the case of breaks in the first three barriers, prevents radioactive releases to the environment. In order to ensure the containment integrity during a severe accident, containment venting systems and/or external containment cooling systems have been adopted into some PWR and BWR designs.

The failure of a large number of fuel rods is the first necessary condition for a significant radioactive release from a nuclear power plant into the environment. The events where failures of fuel rods might occur can be categorized into three groups. The first group of events consists of cases where the reactor cooling systems are operating normally but the reactor power for some reason increases and exceeds the reactor cooling capacity. This first category includes reactor overpower transients such as some anticipated transient without scram (ATWS) cases as well as reactivity transients and accidents. Examples of this type of events are control rod withdrawal and boron dilution incidents and some primary system overcooling transients. The Chernobyl accident was a case of an extremely severe reactivity accident.

The second category of events which might lead to fuel failures include situations where reactor power is normal, i.e operating power or decay heat power, but the cooling of the fuel rods somehow fails. This category involves loss-of-flow cases where the flow of coolant to the reactor core or to some individual rod bundle(s) is totally or partially lost. Such an event might be caused by a blockage of flow paths or rod bundles, or due to failures of reactor coolant pumps (RCPs). Loss-of-coolant accidents (LOCAs) are situations where the loss of fluid from the primary system through a leak is so extensive that it can not be compensated with normal makeup water systems. LOCAs are usually further categorized according to the break size (small, medium or large-break LOCAs). The loss-of-heat-sink is a case where, for example, the heat removal capacity of the steam generators of a VVER reactor is lost due to loss of feedwater on the steam generator secondary side.

The third category involves other cases where the reactor systems operate normally, i.e., reactor power is normal, reactor cooling is effective and primary system integrity is not lost. This category includes cases such as fuel bundle failures during refuelling outages or during the transportation of spent fuel inside the plant.

This thesis includes several thermal-hydraulic analyses of Loviisa VVER-440 type PWR. The topic of Papers 1 to 5 of this thesis is boric acid solution behaviour during long-term cooling period of VVER-440 type PWR LOCAs. Boron, usually in the form of boric acid (H_3BO_3), is used in many ways to control reactivity of the core in LWRs. It is used in PWRs as a soluble neutron poison to cope with fuel burnup changes, in the emergency core cooling system to ensure reactor subcriticality during the ECC water injection into the core, and in the ice-condenser (in the form of sodiumtetraborate i.e. borax, $Na_2B_4O_7.10H_2O$) to ensure subcriticality of the reactor in cases where ECC water is taken from the sump. It is also present in both BWRs and PWRs in high concentration boron solution tanks to be injected into the primary circuit to keep the reactor subcritical if the control rods fail to shut down the reactor (e.g., during some ATWS transients). In PWRs boron is also present in the primary coolant to keep the reactor subcritical during refuelling outages, in BWRs the subcriticality during refuelling is realized with control rods.

Two types of boron anomalies might exist during abnormal conditions in PWRs. The first one is a boron crystallization problem, which can occur during LOCAs if borated water is allowed to boil long enough in the reactor core and if the flow out from the reactor pressure vessel is pure steam flow. This event might lead to boric acid crystallisation in the reactor core, blockage of some flow paths in the core and, finally, overheating and failure of some fuel rods. Hence, in the above classification of nuclear accidents the boron crystallization belongs to the category of (partial) loss-of-flow accidents. In this thesis the boron crystallization problem has been studied both experimentally and analytically.

The other possible boron anomaly is a boron dilution incident, which could occur e.g. during small-break LOCAs (SBLOCAs) if the primary coolant flow is in reflux condensation or in boiler/condenser two-phase flow mode. In this mode steam flowing to the steam generators condenses there and the condensate collects to the cold leg loop seals. If this condensate, having almost a zero boron concentration, suddenly flows into the core, it might cause a reactivity transient and lead to a fast power burst in the reactor and damage to some nuclear fuel. The same kind of boron transient might occur during refuelling outage and start-up phase if non-borated water somehow enters the core or if the reactor boron control system fails.

Papers 6 and 7 of this thesis deal with simulation of horizontal steam generators. Horizontal steam generators are one special feature of VVER type PWRs. Steam generators act as a heat sink in PWRs and their performance is needed to transfer heat generated in the nuclear fuel in the reactor core to the secondary circuit during normal operation, reactor shut-down (in VVERs) and abnormal transients. The work described in these two papers deals mainly with verification of computer codes used in transient and accident analyses of PWRs. The work involves computer simulations as well as experimental studies. This work is important since most of the computer codes have been verified against data from experiments simulating PWRs with vertical steam generators. In the above classification of nuclear accidents, the case of Paper 6 belongs to the category of loss-of-coolant accidents, and the case of Paper 7 to the category of loss-of-heat-sink of the primary circuit.

Paper 8 of this thesis deals with pump trip transient analyses for Loviisa VVER-440 reactor. VVER-440 reactors have a special feature connected with these analyses: these units can be used even if up to three of the six coolant pumps are stopped. This is possible since the VVERs have a control system called ROM which, in the case of pump trips, drops the reactor power to a level of operating main coolant pumps. Hence, if one reactor coolant pump is stopped the reactor power is reduced to a level about 17 % below the nominal power level. In other PWRs a pump trip automatically leads to a reactor trip. In the above classification of nuclear power plant accidents a pump trip transient is a typical loss-of-flow case. Paper 8 consists of computer simulations of two selected pump trip transients.

3. METHODS USED IN NUCLEAR REACTOR THERMAL-HYDRAULIC SAFETY STUDIES

Research in the field of nuclear reactor safety deals not only with nuclear and reactor physics, although the research in these areas form a base for the whole use of nuclear power. Outside the field of nuclear and reactor physics, safety research is conducted in many areas. In the area of structural safety of NPPs, research is conducted, for example, in the field of material science, metallurgy, fracture mechanics and seismic engineering. Research in the field of operational safety of NPPs during normal operation conditions, transients and accidents includes studies of power plant and control engineering, electrical engineering, heat transfer and fluid flow (i.e. thermal-hydraulics) as well as probability and reliability theory. The studies of severe accidents include all research areas of operational safety and, furthermore, studies of chemical engineering, aerosol physics, meteorology and health physics. Nuclear waste (disposal) research encompasses the fields of radiochemistry, geology and ground-water flow in soil and bedrock. The main objective of this research is to guarantee safe and economical use of nuclear power plants.

In particular, research in the field of nuclear reactor thermal-hydraulics includes studies of heat transfer and fluid flow during both normal operation conditions and abnormal transients and accidents. During the design phase of a plant, thermal-hydraulic analyses are needed to show that the plant fulfils the design requirements (typically design basis accident (DBA) analysis). Further, thermal-hydraulic analyses are needed to provide information about the state of the plant during normal operation conditions (such as safety margins) and during abnormal events (such as LOCAs and transients) as well as data needed to write operator instructions. Thermal-hydraulic analyses also provide data needed for other safety analyses, such as probabilistic safety analysis (PSA), containment behaviour and severe accident analyses.

The methods used in the thermal-hydraulic safety studies of nuclear reactors can be categorized into four groups: experiments in real power plants, experiments with nuclear test apparatus, experiments with non-nuclear test apparatus and computer analysis. The main organisations performing these safety studies in Finland are the Technical Research

Centre of Finland (VTT) in its different laboratories (Nuclear Engineering Laboratory (YDI), Reactor Laboratory (REA), etc.), Imatran Voima OY, Finnish Centre for Radiation and Nuclear Safety (STUK), Lappeenranta University of Technology (LTKK) and some other universities.

It is only seldom possible to perform experiments in operating nuclear power plants. Some measurements can be made during the commissioning tests and start-up phase of the plants as well as during normal operation. Some measurement data is also available from real plant transients. However, tests to simulate real accidents are only rarely possible since they are extremely expensive and, more importantly, these kinds of experiments are a real safety risk!

Since it is very difficult to perform experiments in real NPPs, many countries have built test facilities to simulate operating nuclear power plants. In these test facilities the main parts of nuclear power plants are simulated on a full-scale or on a smaller scale. These test apparatus can be divided into two groups. The first group include those **test apparatus which use nuclear fuel**. The first group includes experimental apparatus that are real nuclear reactors, such as LOFT /4/ in the U.S.A. Furthermore, there are test facilities that have been built inside nuclear reactors, such as PHEBUS in France /5/ and MARIA in Poland /6/.

The second group of facilities consists of **non-nuclear test apparatus**. These apparatus can be further divided into two groups. In so-called integral test loops all main parts of the NPP primary and secondary circuits are simulated and experiments are performed to study different kinds of accidents and transients, such as SBLOCAs. The largest integral test loops are SCTF and CCTF /7/, ROSA-III /8/ and -IV /9/ facilities in Japan, UPTF (Upper Plenum Test Facility) /10/ and PKL /11/ facilities in Germany, BETHSY in France /12/ and PACTEL in Finland /13/. In the separate effect test loops the aim is to study some particular phenomenon or some individual part of the reactor in either full-scale or scaled-down geometry. Typical examples of these are MARVIKEN in Sweden /14/, PERICLES /15/ and OMEGA /16/ in France and VEERA /17/, REWET-I and REWET-II facilities /18/ in Finland.

The main limitation of thermal-hydraulic experiments is that they are, in most cases, performed in scaled-down facilities. This makes it more difficult to apply the experimental findings in the smaller-scale facilities to the full-scale power plants. Sometimes not only the geometry but also some other quantities, such as time, material properties (pressure or temperature) and working fluid (use of Freon instead of water) are not on the same scale as the reference system, i.e. a real power plant. Problems arise during scaling of the results since some phenomena occurring in small-scale experiments might not happen in full-scale power plants.

Several computer codes, such as RELAP5 /19/, ATHLET /20/, CATHARE /21/ and APROS /22/, can be used as a tool to extrapolate small-scale experimental results to the full-scale. Individual models in the computer programs can be tested by comparing the calculational results with separate effect test results. Verification in the full-scale geometry can sometimes be made by comparing calculational results with measurement data available from real power plants measured during tests or real plant transients. The main problems in the use of the computer codes to scale up test data are some two-phase flow modelling problems, such as modelling of interactions between phases and problems connected with the numerical solution mechanisms of the codes, such as numerical diffusion. Moreover, most computer codes use one-dimensional models to simulate flow and heat transfer processes in three-dimensional geometries of real power plants and experimental apparatus. In particular, this results in problems in modelling some parts, such as the steam generator secondary side of the VVER-440 reactors.

4. PAPERS ON THIS THESIS

4.1 Experimental and analytical studies of aqueous boric acid solution behaviour during long-term cooling period of VVER-440 type PWR LOCAs (Papers 1 to 5)

Papers 1 to 5 on this thesis deal with aqueous boric acid solution behaviour during VVER-440 type PWR LOCAs. Boric acid is used in modern PWRs as a soluble neutron poison to control the reactivity of the core. Boric acid concentration in the primary coolant is varied so that the concentration is highest at the beginning of the fuel cycle, typically about 1000 -1500 ppm. At the end of the cycle boric acid concentration is reduced close to zero. In addition, boron is used in the ECC water (in form of boric acid, H_3BO_3) and in the ice-condenser (in Loviisa in the form of sodiumtetraborate (borax), $Na_2B_4O_7 \cdot 10H_2O$) to ensure reactor subcriticality during transients and LOCAs if the ECC water injection is needed.

When using boric acid it should be kept in mind that the solubility of boric acid in water decreases greatly when temperature is decreased. This is illustrated in Figure 1 which shows the solubility values of boric acid in water as a function of temperature. Hence, the flow paths of the high concentration boron injection systems must be maintained at a temperature high enough to avoid crystallisation in the pipelines. So, the piping from the concentrated boric acid tanks to the charging pumps must be heat traced /23/. The same problem might arise in some PWRs during shut-down, if the residual heat removal system fails and the steam generators are used as an alternate means of decay heat removal. If boiling of water in the reactor pressure vessel continues long enough, the restoration of the RHRS may result in the inadvertent precipitation of boric acid in the RHRS lines, preventing its further use /24/. This latter sequence is not possible in VVER-440 type reactors. In these reactors the residual heat is removed through the steam generators and so, there are no external RHRS lines.

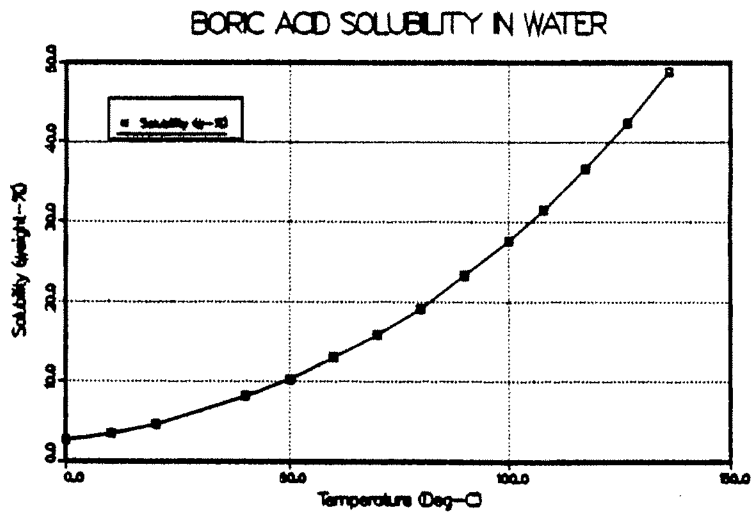

Fig. 1. Solubility of boric acid in water.

During the long-term cooling period of LOCAs, water might boil in the reactor core for an extended time. Boiling increases boric acid concentration in the core since the outflowing steam has a very low boric acid concentration (about 1% of that in water). If water keeps boiling in the core long enough, it would be possible to reach such high boric acid concentrations that boric acid starts to crystallize. This crystallized boric acid might block the flow through the rod bundle(s) and lead to a loss-of-flow type transient, which might lead to overheating and failure of the fuel. In VVER-440 reactors this might be the case in cold leg SBLOCA situations, where collapsed water level in the reactor vessel is maintained near the cold leg connections. In this case the steam generated in the core flows to the steam generators, condenses there and the condensate flows to the cold legs, and partly back to the hot legs. Water level in the core remains stable since the ECC water replaces the water lost through the break. This situation is illustrated in Figure 2.

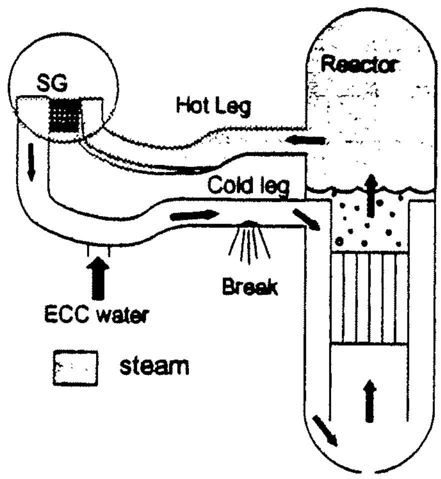

Fig. 2. Schematic representation of a cold leg SBLOCA situation.

In the cold leg SBLOCAs, as described in the previous chapter, there are possibilities for two different types of boron concentration anomalies. First, since the boric acid concentration in the core increases, it is possible to reach such a high boric acid concentration that boric acid crystallizes. On the other hand, steam which flows out from the core condenses in the steam generators, and flows into the cold leg loop seals. If this condensate, which may have almost zero boric acid concentration, flows back to the core when the loop flow starts again it might lead to a severe reactivity transient. In this thesis, the first accidental scenario above, possible crystallization of boric acid during small-break LOCA conditions in a PWR, is studied.

Numerous publications are available about the boron dilution problem, such as by Jacobson /25/, Leach et al. /26/, Vanttola et al. /27/ or Hyvärinen /28/. Much less data has been published about the boron crystallization problem. Boric acid effects on LOCA thermal-hydraulics in case of large-break LOCA (LBLOCA) have been studied by Reeder /29/. Reeder developed a calculational model for the simulation of boric acid concentration changes during early phases of LBLOCA. Reeders analysis dealt with the first 30 s of LOCA and later phases of LOCAs were not discussed. His model was based on boric acid mass balance in different parts of the reactor pressure vessel. In his boron mass balance calculations he used mass flow rates for water and steam calculated by the RELAP4/MOD6 code. His main conclusions were that boron plateout is not expected to occur during this early part of PWR LOCAs. He also proposed that boric acid balances and property calculations should be included in a LOCA simulation model to determine the extent of changes induced on the results by boric acid concentration variations.

Accumulation of boric acid in the reactor core during SBLOCAs (SBLOCA) has recently been studied by Twogood et al. /30/. They calculated boric acid concentrations in San Onofre Nuclear Power Station Unit 1 during a postulated cold leg SBLOCA. Their main intention was to determine the minimum time for boric acid precipitation in the core. This time establishes the minimum time for the initiation of hot leg recirculation to flush the core and terminate concentration. The calculations presented by Twogood et al. were based on boric acid and water mass balances in the primary circuit. The effects of the reactor pressure vessel stored heat on the evaporation of water in RPV was also taken into account. Twogood et al. pointed out that the release of energy stored in RPV and internals boils off water faster than can be made up, the core water level dropps, and the concentration jumps. This effect of stored heat was not taken into account in the previous analysis by the Westinghouse /31/. The calculated concentrations were compared with boric acid solubility limit, which was selected to be 23.5 weight-%. The results of the calculations by Twogood et al. together with the results of earlier analysis are presented in Figure 3. The selection criteria for the solubility limit and the assumptions made in the earlier analysis (8 and 16 hours estimates in Figure 3) were not discussed by the authors of Ref. 31. The data presented in Figure 3 shows that the initial concentration in the beginning of the analysis was about 2.2 weight-%, which is higher than the maximum concentration during normal operation or the concentration of the ECC water in the Loviisa units.

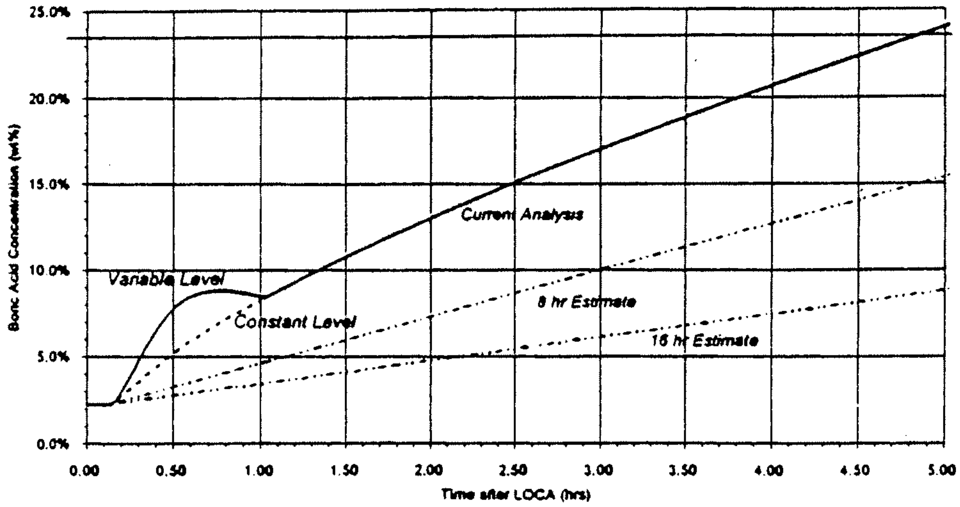

Fig. 3. The RPV boric acid concentration in a cold leg LOCA /30/.

The objective of the work described in this thesis in Papers 1 to 5 was to determine the distribution and concentrations of boric acid in different parts of the reactor pressure vessel of the Loviisa PWR during a cold leg SBLOCA. The work was started with a literature study and it continued with experimental investigations, analysis of the experiments, SBLOCA simulations with the RELAP5/MOD3 Version 5m5 computer code, and with a development of a calculational model for the calculation of boric acid concentration in the reactor. Since understanding of basic phenomena affecting the concentration distribution in the reactor core was considered vital, the experimental investigations were started with long-term boiling experiments with the small-scale REWET-II facility. The experimental arrangements and the results of REWET-II experiments are presented in references /32/ and /33/ and in Papers 1 to 5 on this thesis.

The experiments with the REWET-II facility were aimed to collect general information about boric acid solution behaviour during long-term cooling period of LOCAs. A schematic of the experimental arrangements in the REWET-II experiments is presented in Figure 4. The REWET-II experiments were carried out in such a way that water was boiling in the core section with water level near the core top. Loss of water from the system through boiling was compensated by a feedwater injection into the downcomer. Hence, the collapsed level was almost constant during the experiments. Due to boiling boric acid concentration in the core region increases. During the experiments boric acid concentrations in the core section, condensate sampling tank and in the lower plenum were measured. The main parameters of interest in these tests were the boric acid concentration distribution in the core section and the lower plenum, the maximum concentration in the core and the lower plenum, and the boric acid concentration of the outflowing steam.

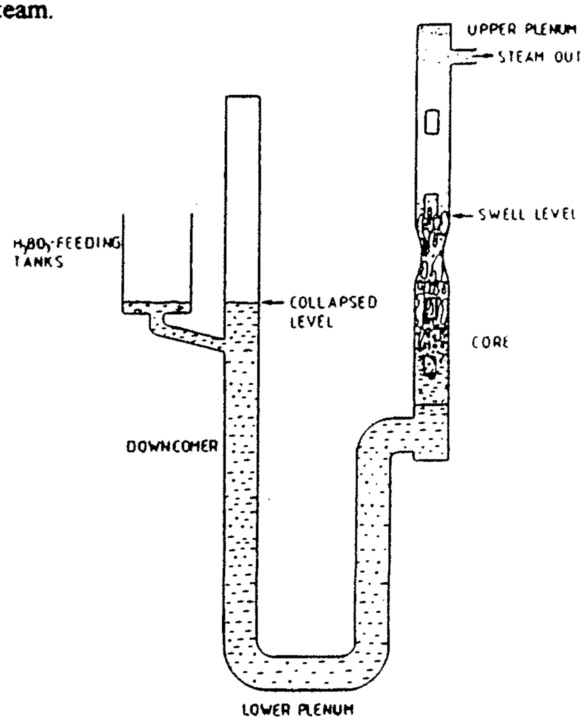

Fig. 4. Experimental arrangements in REWET-II boric acid experiments.

The REWET-II experiments demonstrated that it is possible to reach such high boric acid concentrations in the core simulator that boric acid crystallizes and a flow blockage is formed. Conditions for crystallization were favourable when pressure (temperature) in the system was low, rate of droplet entrainment (flow of boric acid out from the core in the steam) was limited, water level in the core was low and boric acid concentration of feedwater was high. In these experiments, crystallization of boric acid took place near the collapsed water level. If the collapsed water level was below the top of the core section, crystallization of boric acid took place in the core simulator where boric acid crystals gathered below spacer grid(s) and, finally, blocked the rod bundle there. This led to a rapid rise of rod simulator temperatures since the flow of water through the core ceased.

Boric acid concentration measurements in the REWET-II experiments showed a clear concentration difference between the top and the bottom of the core simulator. They also indicated that mixing between the core and lower plenum was rather limited. The literature study performed and the measurements made by Tuunanen /34/ showed that an addition of sodiumtetraborate (Borax, $Na_2B_4O_7.10H_2O$) into an aqueous boric acid solution greatly increases solubility of boric acid in water (Figure 5). The effect of droplet entrainment on the rate of increase of boric acid concentration in the core was found to be significant. In the experiments where a water separator was not used to limit droplet entrainment, concentrations in the core simulator reached a level of equilibrium below the saturation value and no crystallization of boric acid was observed. This happened since the entrained water removed the same amount of boric acid from the system as was added with the feedwater. The effects of droplet entrainment on the core concentrations is illustrated in Figure 6, which shows the measured boric acid concentrations in the core simulator from two REWET-II tests. In the test where the water separator was used in the steam exit line the boric acid concentration in the core simulator reaches the saturation concentration after about 14 hours. In the test without water separator the concentration in the core stays clearly below the saturation value.

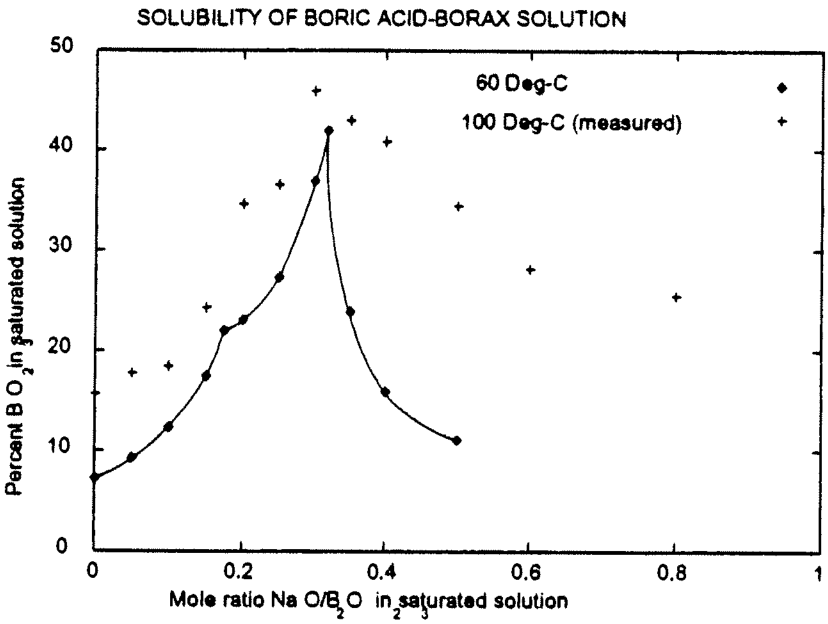

Fig. 5. Solubility of boric acid - borax solution on water.

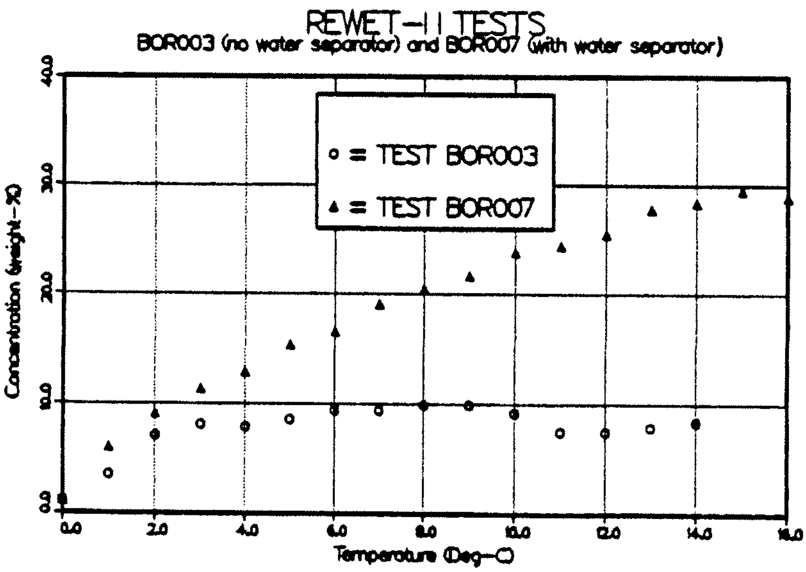

Fig. 6. The effect of droplet entrainment on the rate of increase of boric acid concentration in the core.

Balashov et al. /35/ have recently published results of experiments performed with the AMT facility in Russia to study boric acid solution behaviour in SBLOCA situations. The experiments were performed, as the REWET-II tests, in a test facility with a 19 heater rod core simulator. The experimental set-up in these tests was similar to that in the REWET-II boric acid test. The main measurements in the AMT tests included steam and water temperatures, pressures and pressure differences in the test section as well as water level measurements. In these tests no concentration measurements were performed. The blockage of the core due to crystallized boric acid was detected by the thermocouples. The main experimental parameters in the AMT tests were the core heating power (5 to 15 kW), water level in the core (0.4 to 0.6 x core height), pressure and boric acid concentration. The main intention of the tests performed by Balashov et al. was to study boric acid crystallization process in the case of low core water level.

The main results of the AMT tests were similar to the results of the REWET-II tests. The test section was blocked by crystallized boric acid in the same manner as in the REWET-II tests. In the tests with low water level the blockage of the core was slower than in the tests with higher level, and the blockage occurred gradually. In some tests no core blockage was observed at all. No reason for that was given by the authors. Another Russian test series to simulate boron solution behaviour have been performed at the Zuevka H&P Station /36/[1]. In these test deposit of boron was observed "on condensing but not on the steam generating surfaces". This finding is in accordance with the results of the tests performed by the author of this thesis with the VEERA facility, as will be described later in this chapter.

The core simulator of the REWET-II facility, as well as the core simulator of the Russian AMT facility, included 19 heater rods, instead of 126 in the real reactor bundle. In order to verify the effects of scale of the test apparatus on the mixing processes of boric acid in the core and between the core and the lower plenum, a larger VEERA facility was built. The VEERA facility includes one full-scale copy of a VVER-440 reactor rod bundle with 126 full-length rod simulators. In the original VEERA facility, all the structures near the top of the core section of the reactor (i.e. core outlet nozzles) were accurately simulated. This was done because the aim of the first experiments was to study crystallization process, and that always happened at the top of the core in the REWET-II experiments.

[1] This paper has not been available to the author of this thesis. These tests were mentioned in Ref. 35.

The experimental arrangements of VEERA experiments and the results of the experiments are presented in references /17/ and /37/ and Papers 2 to 5 of this thesis.

The experimental set-up in the VEERA tests was similar as in the REWET-II tests, as shown in Figure 4. So, the main difference between these two test series was the scale of the test apparatus (volumetric scaling ratio 1:2333 in the REWET-II facility and 1:349 in the VEERA facility). The experiments with the REWET-II and VEERA facilities clearly demonstrated that the scale of the experimental apparatus has a significant effect on the results of the experiments. The main difference between the results of the small-scale REWET-II experiments and the larger-scale VEERA tests was that the mixing in the core section of the VEERA facility was much more effective. Hence, boric acid concentration distribution in the VEERA rod bundle was almost uniform. Mixing between the core section and the lower plenum, as well as in the lower plenum, was also more effective than in the REWET-II tests. The main reason for that was the larger scale of the test apparatus.

Experiments with the VEERA facility showed that crystallization of boric acid and blockage of the rod bundle may also take place in this larger-scale rod bundle. In fact, crystallization conditions in the facility were achieved in three different ways. In the experiments with a constant pressure crystallization started, after several hours of boiling (typically 7-8 hours with 2.5 weight-% feedwater boric acid concentration[2]), at the core top near the collapsed water level where supersaturation was largest. In the experiments where supersaturation was reached by a fast depressurization of the system, crystallization took place in the whole upper part of the core section. The third mode of crystallization was observed in the lower plenum. Unsaturated, highly concentrated and hot boric acid solution drifted down from the core to the lower plenum, cooled down, reached supersaturation and crystallized there on the cold inner surface of the lower plenum pipe (this is probably the mechanism of crystallization observed in the tests mentioned in Ref. 36 i.e. crystallization of boric acid outside the core on a cold surface). In the first two cases above, crystallization took place in the core simulator and a flow blockage was formed. In the first case above, if the water level was raised above the rod

[2]This concentration value, which was used in the most of the REWET-II and VEERA experiments, is higher than the ECC water concentration in the Loviisa plant. This higher value was selected for the experiments in order to make the experiments faster i.e. to shorten time needed to reach saturated concentration in the test apparatus.

bundle, crystallization took place in the upper plenum and, hence, no flow blockage in the bundle was observed.

After the first experimental series the VEERA facility was slightly modified. This modified version included an exact simulation of the core inlet nozzles of the VVER-440 reactor. This was necessary since one of the aims of these experiments was to evaluate the rate of mixing of boric acid between the core section and the lower plenum. This data was needed later in the final reactor analysis.

The tests with the modified VEERA facility included two tests with different core inlet orifice. These two tests showed that the change of the diameter of the core inlet orifice from 74 to 50 mm did not affect on the concentration distribution in the core or in the lower plenum. Hence, these experiments demonstrated that U-tube oscillation plays an important role in transferring boric acid from the core into the lower plenum. This observation was later confirmed in the tests made in the IVO Hydraulic Laboratory /38/. The tests in the IVO Hydraulic Laboratory showed that the change of the core inlet orifice diameter from 74 to 50 mm significantly restricts the buoyant mixing between the core and lower plenum. So, the U-tube oscillation was the main mechanism for transporting boric acid from the core to the lower plenum in the VEERA tests.

The main problem of the VEERA experiments was that the lower plenum of the facility did not simulate accurately enough the lower plenum of the reactor. The reason for this was that volumetric scaling preserving the elevations but reducing the flow area was used in the design of the VEERA facility. Hence, three-dimensional mixing processes in the lower plenum of the simulated reactor were not reproduced in these experiments.

To gain more detailed information about the boric acid mass transfer in the reactor, a separate test loop was built in the IVO Hydraulic Laboratory. The experiments in the IVO Hydraulic Laboratory were designed so that the mixing in the lower plenum was considered as a buoyant mass transfer leaving oscillations out from the simulation. The experimental apparatus used in these tests and the experimental arrangements are presented in reference /38/ and Papers 4 and 5 of this thesis. After these experiments it was concluded that the buoyant mixing process is quite effective in the three-dimensional flow pattern of the lower plenum of the reactor, equalising concentration differences in the lower plenum. As mentioned earlier, the effect of oscillations is more important for the mass transfer through the inlet orifice section in the core.

Before starting to develop a model for the calculation of boric acid concentrations in the Loviisa PWR, the boron model of the RELAP/MOD3 code was tested and verified against an experiment performed in the VEERA-facility. The boron tracking model included in the RELAP5 code assumes, that boron is moving together with water (with the same velocity) and that the concentration of steam is zero. The model does not take into account the effects of concentration on the water properties, such as density of water. So, the model calculates only the convection of boron together with water in the simulated system.

The verification calculations clearly demonstrated one of the problems of using computer codes to scale up experimental data. In these calculations, numerical diffusion together with U-tube oscillation was observed to transfer concentrated boric acid solution from the core section to the lower plenum and downcomer, and to decrease calculated concentration differences around the test loop. The calculated boric acid concentrations were higher than those measured in the downcomer and feedwater tank, and lower in the lower plenum and the core section. The calculations clearly showed that the boron model of the RELAP5/MOD3 code should not be used in the calculations of boric acid concentrations in a system including strong flow oscillations and/or concentration gradients /39/. The results of this analysis were the main reason why a separate calculational model for the reactor boric acid concentration calculations was developed, instead of using the model of the RELAP5/MOD3 code.

Before the calculations of the boric acid concentrations in the reactor were started, the RELAP5/MOD3 code was used to determine the boundary conditions for LOCAs under which concentrating of boric acid might occur. Experiments with the REWET-II and VEERA-facilities had shown that crystallization of boric acid could be achieved if the pressure in the reactor was low, the collapsed water level in the reactor pressure vessel was near the top of the core and if the rate of droplet entrainment was low. These conditions apply to cold leg SBLOCAs when the low pressure coolant injection (LPCI) is not effective i.e. the pressure is slightly above the actuation pressure of the low pressure injection (LPI) pumps.

The RELAP5/MOD3 SBLOCA analyses were made with three different break sizes (25 cm^2, 20 cm^2 and 15 cm^2). The effects of the reactor secondary side cooling and ECC capacity available were also studied. The results of these analyses are presented in detail in reference /40/ and Paper 5 of this thesis. The RELAP5/MOD3 calculations showed that, in the case of cold leg SBLOCA without secondary side cooling, the water level in the core stabilizes near the level of the bottom of the cold leg pipes, leading to a stable long-term cooling situation. If the reactor secondary side is cooled the water level in the reactor pressure vessel rises near the hot leg connections.

The RELAP5/MOD3 analyses also demonstrated that there are two types of internal by-pass flows in the reactor. First, there is a flow through the dummy elements from the upper plenum to the lower plenum (typically about 20 kg/s). Second, water circulates between the rod bundles inside the core section (typically about 180 kg/s). The by-pass flow through the dummy elements enhances mixing between the core and lower plenum and decreases the concentration difference between these two sections. The by-pass flow between the bundles improves mixing inside the core section and decreases the concentration differences between individual rod bundles. These by-pass flow rates calculated with the RELAP5/MOD3 code were not directly used in the final reactor analysis. However, from the RELAP5/MOD3 analyses it was concluded that there will be only two mixing regions in the reactor with almost uniform boric acid concentrations: the core section with a high concentration and the lower plenum with a lower concentration. The only concentration difference existing in the reactor will be found between these two sections.

The development of the model for the calculations of the reactor boric acid concentration was made by the author of this thesis, together with Mr P. Raussi (co-author of Papers 4 and 5). The model developed was based on the boric acid mass balance in the reactor primary circuit. In the calculations the reactor pressure vessel was divided into two mixing regions (core section and lower plenum) and boric acid concentrations were calculated separately for these two sections. In some cases, the concentration for the hot bundle of the core, i.e. for the channel with the highest power was also calculated. This was done by assuming that the concentration in the hot sub-bundle is directly proportional to the ratio of the power of the hot sub-bundle to the power of the averaged bundle (the calculated core average concentration was multiplied by this ratio).

The flow chart of the simulation model is presented in Figure 7. The simulation model also includes a calculation of borax concentration in the lower plenum and in the core. The borax concentration calculation was based on the borax mass balance in the system. The borax enters the reactor when ECC water is taken from the sump. The borax concentration is needed since it affects significantly on the solubility of boric acid in water, as shown in Figure 5.

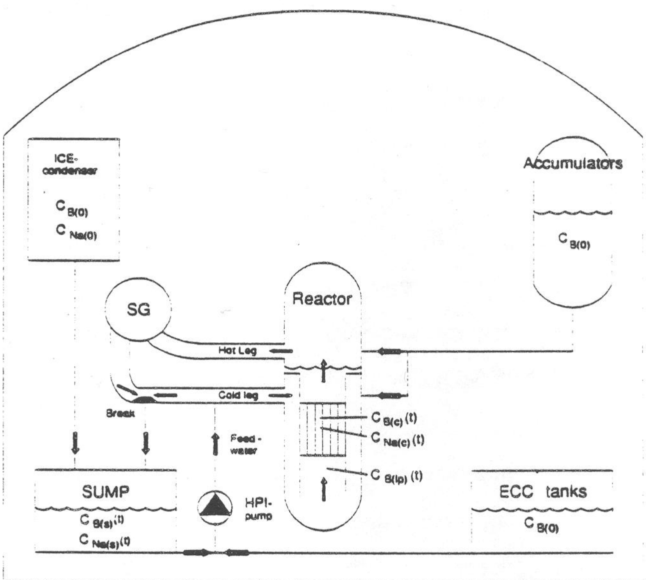

Fig. 7. Flow chart of the simulation model for the calculation of boric acid concentrations in a VVER-440 reactor.

The calculation of the mixing rate of boric acid between the reactor core section and the lower plenum was based on the results of the VEERA tests and the tests performed at the IVO Hydraulic Laboratory. The developed calculational model was not verified against experimental data since there is no useful data for verification available (only a large-scale test facility with multiple rod bundle simulators and a large lower plenum can provide such data). Hence, parametric studies with concervative boundary conditions were performed to cope with the lack of full-scale experimental data about boron mixing in the RPV. These parametric studies included an analysis of hot sub-bundle concentration, as mentioned earlier in this chapter, as well as studies of the effects of the core water level, boric acid concentration of steam, and the mixing rate of boric acid between the core and the lower plenum on the concentrations in the RPV. The calculation of the accumulation of boric acid in the RPV was started from the time when the level at the RPV was stabilized after a cold leg SBLOCA. This time was predicted by the RELAP5/MOD3 code. In the case of 15 cm^2 cold leg break this happened after about 5000 seconds. Before that time it was assumed that boric acid flows through the break with water out from the primary circuit and no concentrating in the core takes place. The heat stored in RPV and internals was assumed to be released before that time and, so, their effects were not taken into account in the calculations.

The calculated concentration values in the reactor core and in the lower plenum were compared with the critical concentrations that would lead to boric acid crystallization. The critical concentration for the reactor core was the solubility of boric acid in water at 100 °C. This value was selected to ensure that there is no possibility of boric acid crystallization even if the pressure in the primary circuit suddenly drops, for example, due to secondary side cooling. For the lower plenum, solubility value of boric acid in water at 70 °C was used as the critical concentration. This value was used since the temperature of pressure vessel internals in the lower plenum might be near the ECC water temperature and, thus, at a lower temperature than the saturation temperature in atmospheric pressure. In the calculations of the critical concentrations, the effects of borax on boric acid solubility in water were taken into account.

The reactor analysis showed that the concentrations in the reactor stay clearly below the critical values during the first 50 hours of SBLOCAs. Borax in the ice-condenser ice increases boric acid solubility in water and greatly increases the margin to the critical concentrations. This is demonstrated in Fig. 8 presenting the calculated reactor concentrations together with the calculated critical concentrations during the first 50 hours of SBLOCA for one of the parametric study cases. In the case shown in Fig. 8 the margins to the critical concentrations are clear, both in the lower plenum and in the core as well as in the hot sub bundle.

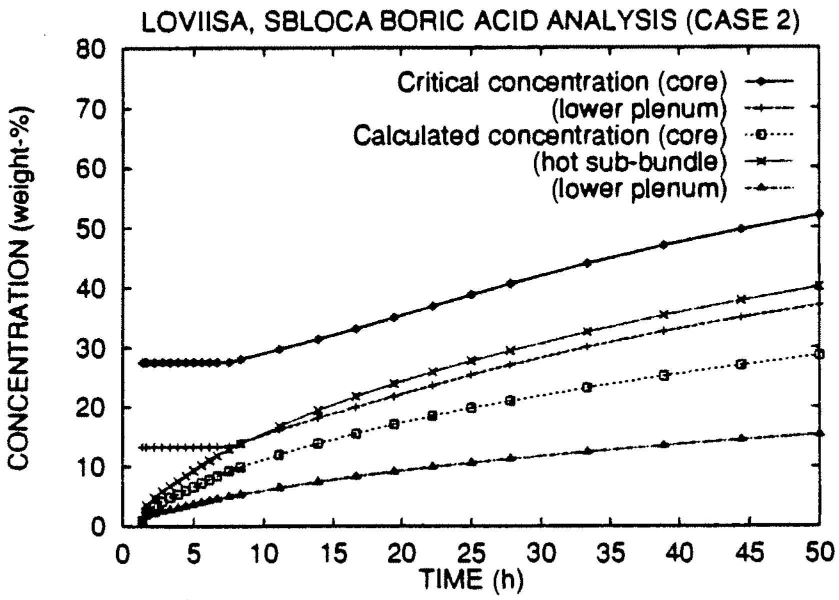

Fig. 8. Calculated boric acid concentrations in the Loviisa VVER-440 RPV during a cold leg SBLOCA.

The reactor analyses showed that the risk of crystallization in the reactor core is extremely low. In the long-term cooling period of LOCAs, further attention should be directed to the studies of the other accidental boric acid sequence connected to the SBLOCAs, i.e possible power burst due to flow of pure condensate from the cold leg loop seals to the reactor.

Papers 1 to 5 of this thesis include multiple authors. The author of this thesis was responsible for performing and analyzing the experiments with the REWET-II and VEERA facilities, simulating the experiments and Loviisa SBLOCA transients with the RELAP5/MOD3 computer code, and, together with P. Raussi, performing and analyzing the experiments in the IVO Hydraulic Laboratory, developing a calculational model for the reactor boric acid concentration analyses and using the model for the Loviisa PWR boron concentration simulations. Part of the experimental work formed the diploma thesis and the licenciate thesis of the author. The diploma thesis was done under supervision of Mr. T. Kervinen (co-author in Papers 1 to 4) and Prof. H. Kalli (co-author in Papers 2 and 3) and the licensiate thesis under supervision of Prof. Kalli and Dr. H. Tuomisto (co-author of Papers 2 to 4). Most of the work presented in Papers 1 to 5 was financed by the Imatran Voima Oy, which is represented in the publications by Dr. H. Tuomisto, Mr. E. Markkanen and Mr. M. Komsi.

4.2 Assessment of RELAP5/MOD3 against natural circulation experiments performed with the REWET-III facility (Paper 6)

During normal operation of PWRs the primary coolant is circulated by the primary pumps. During abnormal conditions these pumps are often tripped and the primary coolant flow is reduced to the natural circulation flow caused by density differences around the loops. Depending on the amount of coolant in the loops and the power of the reactor core, different natural circulation modes will occur. Single-phase natural circulation, where subcooled or saturated single-phase water circulates in the loops, takes place when the primary circuit is at full inventory. When the water inventory is reduced, steam is produced in the primary system and the transition to two-phase circulation takes place. In this mode a mixture of steam and water flows from the core to the steam generators and cools there. If the water inventory is further reduced, more voiding will take place and the flow to the steam generators turns to a single-phase steam flow. Depending on the type of the flow this mode is called reflux condensation or boiler-condenser natural circulation. These different natural circulation modes have been illustrated in Figure 9 /41/.

For many years natural circulation cooling of Western LWRs has been a topic of extensive experimental and analytical work in different countries. In the case of VVER-type reactors the amount of experimental data is much more limited. The main organisations performing VVER-440 reactor experiments in this field in the world are KFKI in Hungary with their PMK/NVH facility /42/ and VTT/YDI and LTKK in Finland with the REWET-III /18/ and PACTEL loops /13/. In the case of VVER type PWRs, these facilities are the main sources of data used in the verification of the calculational models implemented in the thermal-hydraulic computer codes. Some experimental apparatus have been built in the former Soviet Union but financial support from the Western countries is needed for continuation of these programs /43/.

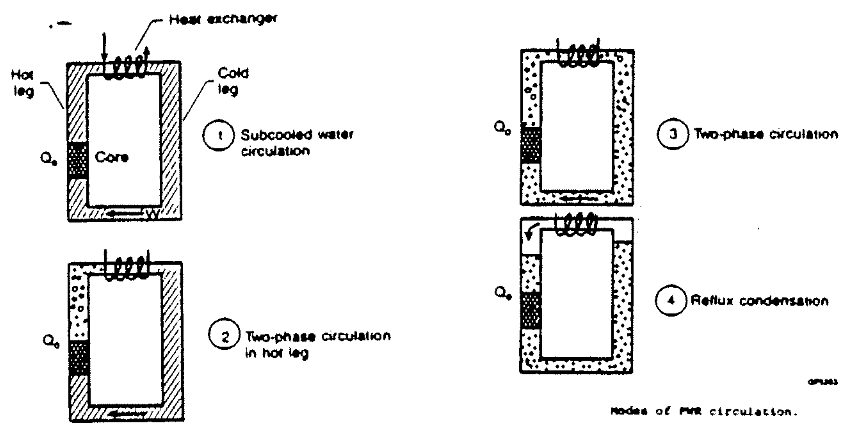

Fig. 9. Different natural circulation modes.

The natural circulation cooling of VVER type reactors has earlier been studied by Kervinen and Hongisto /44/, Hyvärinen, Kalli and Kervinen /45/ and recently by Lomperski and Kouhia /46/. Natural circulation in the PAKS VVER-440 Model V213 plant unit has been studied by Bandurski et. others /47/ and Toth /48/. Moreover, the International Atomic Energy Agency (IAEA) has organized several Standard Problem Exercises (SPEs) to study VVER reactor thermal-hydraulics /49/.

The topic of Paper 6 on this thesis is verification of RELAP5/MOD3 computer code against natural circulation experiments performed with the REWET-III facility. REWET-III was the main experimental apparatus used by VTT/YDI in thermal-hydraulic experiments before the new and larger PACTEL facility was built in 1989. For the code verification purpose, two natural circulation tests performed with the REWET-III facility were calculated using the RELAP5/MOD3 Version 5m5 code. The first experiment was a single-phase case at 20 kW thermal power corresponding to about 3 % decay heat level in the reference reactor. The second case involved two-phase flow with a fluid inventory of 80% of full capacity and 30 kW thermal power. One of the main objectives in these analyses was to verify the capability of the RELAP5/MOD3 code to simulate horizontal steam generators.

The calculations presented in Paper 6 showed that the RELAP5/MOD3 code was capable of predicting the main events occurring during the experiments. The RELAP5/MOD3 calculations indicated that the presence of air in the steam generator is the reason for low water temperatures measured in the uppermost steam generator tube in the single-phase natural circulation test. In both tests analyzed, the calculated temperature distribution for the steam generator hot collector was unstable, i.e the code assumed that there was hot water below cold in the collector. The main reason was found to be the one-dimensional structure of the code not allowing a proper mixing of water in the collector. The calculated primary water temperatures in the single-phase case were slightly higher than the measured values. The reason for this was the slightly under-estimated heat losses. In the two-phase case the calculated mass flow and temperatures were close to the measured values.

The heat losses from the primary loop to the environment were modelled assuming a constant boundary volume temperature as well as a constant heat transfer coefficient from the insulator surface to the environment. The same heat transfer coefficient was used throughout the primary loop. This assumption probably underestimated the heat losses, as was seen from the slightly over predicted lower plenum, downcomer and core inlet temperatures. On the other hand it gave good values for the hot and cold leg fluid temperatures, which was important since the main interest in these analyses was in the behaviour of the horizontal steam generator. The use of the form loss coefficients calculated by the code itself gave good results in the prediction of the primary loop flow rates. No tuning of the loss coefficient was needed to obtain calculated results that were comparable to the measured flow rates.

In the analyses described in Paper 6 of this thesis the steam generator secondary side was modelled with only a few nodes. In this case the model was accurate enough to describe the secondary side conditions. The most problematic part of the modelling in these calculations was the steam generator hot collector, where an unstable water temperature distribution was predicted. Due to a code error the calculated fluid temperatures in two uppermost steam generator hot collector volumes showed unrealistic, low values when flow regime transferred from slug to annular-mist regime. Unfortunately, a comparison with the measured hot collector temperature distribution was not possible since there was only one temperature measurement point in the collector in the tests. The analyses showed, however, that the RELAP5/MOD3 code can be used in the simulation of the horizontal steam generators with good accuracy. Moreover, it should be mentioned that in the calculations presented in Paper 6 of this thesis a very detailed steam generator model was used on the steam generator primary side. In this simulation model all 12 heat exchange tubes of the steam generator of the REWET-III facility were modelled separately. In the simulations of the real power plants with the RELAP5 code it is not possible to use such a detailed steam generator models due to the large number of heat exchange tubes in a real NPP. Hence, several rows of heat exchange tubes, (say 20-30 rows having about 200 tubes each) must be combined in the simulation models of the real steam generators.

The author of this thesis was responsible for performing the analyses described in Paper 6.

4.3 Analyses of the PACTEL loss of secondary side feedwater tests with RELAP5/MOD3 (Paper 7)

Steam generators are the main heat sink of the primary circuit in PWRs. In order to transfer heat from the primary to the secondary circuit there must be a sufficient amount of water on the steam generator secondary side. During normal operation a constant steam generator water level is maintained with the feedwater system. In case of failures of the main system the auxiliary feedwater system(s) are used.

In a PWR with vertical steam generators the complete loss of electrical power, i.e. station blackout, leads to loss of feedwater on the steam generator secondary side which might lead to core damage within 1 to 3 hours. In VVER-440 reactors with horizontal steam generators having a large secondary side water inventory the core temperatures may be acceptable even after 6 hours /3/.

Paper 7 of this thesis describes an experimental and analytical study of loss of feedwater transient in PACTEL test rig simulating VVER-440 type PWRs in Loviisa. In this paper the main intention was to study the capability of the RELAP5/MOD3 code to simulate the horizontal steam generators in cases where the steam generator secondary side water level varies. This information is of great importance since the RELAP5/MOD3 code has been verified earlier only against experiments in test rigs having vertical steam generators.

For the purpose of code verification, a series of loss of secondary side feedwater tests was conducted with the PACTEL facility. The tests performed were then analyzed with the RELAP5/MOD3 Version 5m5 code. The test results described in Paper 7 clearly demonstrated that the horizontal steam generators of the VVER-440 type reactors provide large safety margins in the loss of secondary side feedwater case. In an experiment with about 3.8 % decay heat power and a total loss of feedwater in all steam generators, the time before the primary system pressure started to rise and the pressurizer safety valve opened was about two hours. In the case of loss of feedwater in one PACTEL steam generator and a 3.8 % decay heat power, the primary system pressure did not rise at all. The reason was that the two other PACTEL steam generators, operating normally, were capable of transferring all heat from the primary to the secondary side. It should be mentioned that one steam generator in the PACTEL simulates two steam generators in the Loviisa plant.

The analyses with the RELAP5/MOD3 code showed that the code was capable of predicting the main events in the experiments. However, the code underestimated the rate of steam superheating on the steam generator secondary side. Underestimation of the steam superheating was caused by a too effective cooling of steam in the upper parts of the steam generator secondary side and in the steam line. This also effected the primary side temperatures.

The reasons for the underestimation of the steam superheating in the steam generator secondary side were studied by changing the nodalization in the upper part of the steam generator secondary side. The calculations with a more detailed steam generator model showed that the division of steam generator secondary side into seven, relatively thin layers, each layer simulating one layer of heat exchange tubes, smoothed the effects on the primary side temperatures and flow of the water no longer covering the secondary side tubes. The original model, where three layers of tubes were combined into one secondary side node, was also capable of predicting the effects on the primary side parameters of the water ceasing to cover the tubes. In the simulations of steam superheating on the steam generator secondary side, both models underestimated the rate of steam superheating.

The simulation of the horizontal steam generators with the RELAP5/MOD3 code (or other large system codes) does have some problems. On the steam generator primary side, it is not possible to model all heat exchange tubes separately. Hence, several tubes must be combined in the model. Normally, this is done so that the heat exchange tube bundle is divided into one to five parallel channels, each simulating several rows of horizontal U-tube heat exchange tubes. Determination of the flow rates from the collectors into these combined heat exchange tubes is somewhat difficult. In a real steam generator, the flow direction might be reversed in some heat exchange tubes, a phenomenon which is not necessarily correctly simulated with these combined models. On the steam generator secondary side, problems arise due to the large coolant volume and due to the three-dimensional geometry of the steam generator.

The steam generator model used in these calculations was based on the work by Karppinen /50/. Another way to simulate the steam generator secondary side behaviour is to divide it horizontally into more nodes and to connect these nodes together with cross flow junctions. An attempt to do this and to simulate the three-dimensional flows on the secondary side of the VVER-440 reactor steam generator is presented in reference /51/. It should be remembered, however, that the RELAP5/MOD3 code is a one-dimensional code and the use of it to simulate three-dimensional flows is somewhat questionable.

The author of this thesis was responsible for performing the experiments and final computer simulations with the RELAP5/MOD3 code, as described in Paper 7. The co-author of Paper 7, Mr. K. Nakada, assisted in performing the experiments and pre-test calculations with the RELAP5/MOD3 code.

4.4 Analyses of Loviisa VVER-440 reactor pump trips with RELAP5/MOD1-Eur, RELAP5/MOD3 and CATHARE codes (Paper 8)

If a reactor coolant pump trips in a large PWR the reactor scrams to avoid a heat transfer crisis in the reactor core. In a VVER-440 a pump trip is routinely accommodated by an automatic power reduction. The results of analysis discussed by Laaksonen /3/ show that if the pump stops instantly due to a severe mechanical failure and the reactor continues to operate at full power the reactor core would stay far away from heat transfer crisis. A similar analysis shows that a Westinghouse plant reaches heat transfer crisis in less than 1 second /3/.

Some differences can be found if a pump trip transient in the Loviisa plant is compared with the same transients in other VVER-440 type PWRs. For example, in the Kozloduy VVER-440 Model V230 plant in Bulgaria the reactor coolant pumps will stop in less than 10 seconds if the pump power is lost from all six loops /52/. In the Loviisa plant it will take more than 200 seconds before the pumps stop, if the power is lost for all reactor coolant pumps. The reason is that the reactor coolant pumps in Loviisa are of a high inertia type instead of the low inertia pumps in the other VVER-440s.

Paper 8 of this thesis describes the results of calculations of two pump trip transients for the Loviisa VVER-440 type PWR with the RELAP5/MOD1-Eur, RELAP5/MOD3 and CATHARE codes. The first was a trip of one reactor coolant pump and the second one a reactor coolant pump shaft seizure combined with a failure in the reactor power control system (ROM). The calculations were verified against data measured in the Loviisa plant during the plant start-up tests and during a pump trip transient which occurred in Loviisa in 1991.

The minimum departure from nucleate boiling ratio (DNBR) was calculated with five different correlations. Two correlations (Gidropress and Smolin /53, 54/) were especially developed for VVER-440 fuel bundles. The three other correlations were those in the RELAP5/MOD1-Eur, RELAP5/MOD3 and CATHARE codes (Tong and Hsu & Beckner correlations in the MOD1-Eur /55/, AECL-UO Critical Heat Flux lookup table in the MOD3 /19/ and Groeneveld Tables in the CATHARE /56/). One aim of the original MOD1-Eur analysis was to compare their results with the results of the analysis made by the reactor vendor.

In the RELAP5/MOD1-Eur and RELAP5/MOD3 analysis the control variables were used to model the Gidropress and Smolin DNB-correlations. DNBRs were calculated separately for the average fuel rod, averaged fuel rod in the hot bundle as well as for the rod with the highest power in the hot bundle. All together 517 control variables were needed to calculate DNBRs in 15 different positions.

In the case of a trip of one RCP, there were no significant differences between the calculated results of different codes. The secondary side pressure was controlled by the turbine control valve and all three codes gave nearly equal results for this phenomenon. All three codes calculated almost identical reactor thermal power behaviour, but the primary loop mean temperature was slightly lower in MOD1-Eur simulations. This difference was caused by slightly different initial mean temperature values.

The calculated minimum DNBRs in general, varied in a narrow band between these three codes' simulations. However, the calculated MDNBRs were far above the value of heat transfer crisis in all simulations.

In the case of reactor coolant pump shaft seizure together with a failure in ROM, the MOD3 code calculated about 1 % lower reactor thermal power and slightly lower primary side mean temperature than in the MOD1-Eur and CATHARE calculations. The differences in core power and primary side mean temperature simulations are caused by differences in the secondary side behaviour. In this second case, it was assumed, that the ROM was not in operation and, hence, that the turbine control valves did not control secondary side pressure. Due to this, the secondary side pressure behaviour was not identical in all three simulations.

The calculated primary pressure behaviour was somewhat different in all three simulations. In the MOD3 calculations, the primary pressure rose to the actuation pressure of the pressurizer spray system on two occasions, and in the MOD1-Eur calculations this happened only once. In the CATHARE simulations, the pressure remained below the spray system actuation pressure during the whole transient.

Differences in the simulated primary pressure behaviour in the RELAP5 simulations were observed to be connected with the core power calculations. The calculated reactor thermal power affected the pressurizer level setpoint value. In the RELAP simulations, the pressurizer level was controlled by two makeup pumps. Due to different simulated core thermal power behaviour these pumps were not started and stopped at the same time in different calculations. This, together with the pressurizer heaters, affected the primary pressure behaviour in the RELAP5 simulations.

In the CATHARE simulations only one makeup pump was modelled. Furthermore, a constant pressurizer level setpoint value was used. These were the reasons for the slightly lower calculated primary pressure behaviour at the end of the transients in the CATHARE simulations than in the RELAP5 calculations.

Although the differences between the results were larger in this second case than in the first one, the calculated minimum DNBR values were still, in all simulations, well above the value of the heat transfer crisis.

The author of this thesis was responsible for performing the computer simulations with the RELAP5/MOD1-Eur and RELAP5/MOD3 codes as presented in Paper 8. The co-author of the Paper 8, Mr. K. Lahti was responsible for the CATHARE analysis.

5. CONCLUSIONS

This thesis includes several thermal-hydraulic analyses related to the Loviisa VVER-440 Model V213 type nuclear power plant units. The thesis consists of experimental studies, analysis of the experiments, analyses of some plant transients and development of a calculational model for reactor boric acid concentration calculations. The experiments and analyses once again showed that the VVER-440 reactor units have inherent safety features with positive effects on the performance of these reactors during LOCAs and transients. They also demonstrated the effects of certain improvements on the original VVER-440 design that were made in the Loviisa plant by Imatran Voima Oy (IVO) .

In the case of studies of boron solution behaviour during the long-term cooling period of LOCAs the margins to critical concentrations that would lead to boric acid crystallization were found to be large both in the reactor core and in the lower plenum. This is mainly caused by borax ($Na_2B_4O_7 . 10H_2O$), which enters the reactor when ECC water is taken from the sump and greatly increases boric acid solubility in water. The ice-condenser containment with borax on ice is one specific feature of the Loviisa plant and one of the design improvements made by IVO.

In the case of simulation of horizontal steam generators, the RELAP5/MOD3 code was capable in predicting the main events during the experiments. However, in the loss of secondary side feedwater case the RELAP5/MOD3 code was not capable in simulating steam superheating in the steam generator secondary side. This problem was not solved by changing the secondary side nodalization at the top of the steam generator.

In the both analyzed cases of primary coolant pump failures the margins to heat transfer crisis were large. No significant differences were found between the results of the RELAP5/MOD1-Eur, RELAP5/MOD3 and CATHARE calculations. The main reasons for the large margin to heat transfer crisis during pump trips are the low core power level of the VVER-440 reactors and the high inertia type main coolant pumps. The high inertia main coolant pump is one of the design improvements made by IVO to the Loviisa plant.

An important <u>**inherent safety feature**</u> of the VVER-440 reactors is the **large primary coolant volume to core power ratio**. The advantages of this safety feature were well seen in all analyzed cases of this thesis. In the case of boron solution behaviour during long-term cooling period of LOCAs the large primary coolant volume provides a large effective mixing volume for concentrated boric acid solution, which together with the low core power level, affects the rate of accumulation of boric acid in the reactor pressure vessel. In the case of SBLOCAs, such as those simulated in the REWET-III natural circulation experiment, the large primary coolant volume gives time for corrective actions and delays possible core heatup. In the case of loss of secondary side feedwater the large primary coolant volume to core power ratio slows down the rise of the primary system temperature and pressure when the heat transfer to the steam generator secondary side is deteriorated, and delays the core heatup when the primary water inventory is decreasing through the opening pressurizer relief valves. In the case of primary coolant pump failures this feature smooths the primary pressure and temperature behaviour during the transient. **The large coolant volume on the secondary side of the steam generators** gives time for corrective actions in the case of loss of secondary side feedwater, for example, during a station blackout and slows down the heatup of the primary system. **The six coolant loops in the primary system** provide multiple cooling paths in the case of loss of feedwater and moderate pump failure transients.

REFERENCES

/1/ "World Nuclear Industry Handbook 1993", Supplement to Nuclear Engineering International magazine (CISSN 0029-5507), January 1993.

/2/ "Soviet Design Nuclear Power Plants in the Former Soviet Republics and Czechoslovakia, Hungary and Bulgaria", Source Book of the U.S. Council for Energy Awareness (USCEA), April 1992.

/3/ Laaksonen, J., 1992, "It Ain't Necessarily So: Reassessing VVER-440 Safety," Nuclear Engineering International, pp. 22-25.

/4/ "The OECD/LOFT Project, Achievements and Significant Results", 1990, Proceedings of an Open Forum, Madrid, Spain.

/5/ PHEBUS- announcement, CEA/Service d'essais de surete, CEN Cadarache, France.

/6/ Strupczewski, A., Moldysz, A., Filipczak, N., Spasskov, V., Shmelev, V., Shumski, A., 1987, "Stand der Arbeiten der Versuchanlage MARIA und Algorithmen der Ersten Drei im Programm Vorgesehenen Experimenten", Kernenergie 30, No. 8, pp. 313-320.

/7/ Tuunanen, J., Kervinen, T., 1986, "Experimental Facilities for the Simulation of Nuclear Reactor Thermal-hydraulics", ATS Ydintekniikka, No. 3, pp. 36-40 (in Finnish).

/8/ Kumaru, H., Tasaka, K., 1985, "ROSA-III Test Facility for BWR Integral Simulation Test", Proceedings, Specialists Meeting on SBLOCA Analysis in LWRs, Pisa, Italy, Vol 1, pp. 539-554.

/9/ Koizumi, Y., Tasaka, K., 1985, "ROSA-IV Large-scale Test Facility for PWR SBLOCA Integral Simulation", Proceedings, Specialists Meeting on SBLOCA Analysis in LWRs, Pisa, Italy, Vol 1, pp. 555-569.

/10/ Weiss, P., Sawitzki, M., Winkler, F., 1986, "UPTF - a Full-scale PWR Loss-of-Coolant Accident Program", Atomkern-Energie Kerntechnik, Vol. 49.

/11/ Umminger, K., Brand, B., Mandl, R., Watzinger, H., 1985, "The PKL Facility Status and Future Plans", Proceedings, Specialists Meeting on SBLOCA Analysis in LWRs, Pisa, Italy, Vol 1, pp. 507-526.

/12/ Deruaz, R., 1985, "BETHSY Facility Description and Test Matrix", Seminaire CATHARE, Grenoble: EDF-CEA-FRAMATOME, Centre d'Etudes Nucleaires de Grenoble.

/13/ Kervinen, T., Purhonen, H., 1991, "PACTEL, Facility for Nuclear Reactor Thermal-hydraulics," Proceedings, Jahrestagung Kerntechnik '91, Annual Meeting of Nuclear Technology '91, Bonn, F.R.G, pp. 89-92.

/14/ Ericson, L., et al., 1979, "The Marviken Full-Scale Critical Flow Tests", Interim report: results from test 24, MXC224.

/15/ Giri, A., Attal, P., 1988, "Framatom Reflood Code Assessment on PERICLES Tests", Proceedings, International ENS/ANS Conference on Thermal Reactor Safety, Avignon, France, Vol. 3. pp. 1127-1135.

/16/ OMEGA. Départment des Réacteur á Eau (DRE/STT). C.E.N. Grenoble. Janvier 1985.

/17/ Tuunanen, J., 1989, "Experiments with VEERA Facility on the Simulation of Long-term Cooling of a Nuclear Reactor", Licenciate Thesis, Lappeenranta University of Technology (LTKK), Department of Energy Technology (ENTE), Lappeenranta, Finland.

/18/ Kervinen, T., Purhonen, H., Haapalehto, T., 1989, "REWET-II and REWET-III Facilities for PWR LOCA Experiments", Research Notes 929, Technical Research Centre of Finland (VTT), Espoo, Finland.

/19/ Carlson, K.E., Riemke, R.A., Rouhani, S.Z., Shumway, R.W., Weaver, W.L., 1990, "RELAP5/MOD3 CODE MANUAL, Volume 1: Code Structure, System Models, and Solution Methods," EG&G Idaho, Inc., Idaho Falls, Idaho.

/20/ Burwell, M, J., Lerchl, G., Miró, J., Teschendorf, V., Wolfert, K., 1989, "The Thermal-hydraulic Code ATHLET for Analyses of PWR and BWR Systems", Proceedings, 4rd International Topical Meeting on Nuclear Reactor Thermalhydraulics, Karlsruhe, F.R.G., Vol. 2, pp. 1234-1239.

/21/ Barre, F., Bernard, M., 1990, "The CATHARE Code Strategy and Assessment", Nuclear Engineering and Design, 124, pp. 257-284.

/22/ Porkholm, K., Hänninen, M., Puska, E. K., Ylijoki, J., 1992, "APROS Code for the Analysis of Nuclear Power Plant Thermal-hydraulics Transients", Proceedings, ANS 1992 Winter Meeting, Chicago, Illinois, U.S.A.

/23/ Lish, K., C., 1972, "Nuclear Power Plant Systems and Equipment", Industrial Press inc., New York, N.Y.

/24/ L. W. Ward, 1992, "Evaluation of the Loss of Residual Heat Removal Systems in Pressurized Water Reactors with U-tube Steam Generators", Nuclear Technology, Vol 100, pp. 25 -38.

/25/ Jacobson, S., 1989, "Some Local Dilution Transients in a Pressurized Water Reactor", Thesis no. 171, LIU-TEK-LIC-1989:11, Linköping University, Department of Mechanical Engineering, Linköping, Sweden.

/26/ Leach, C., Cheng, A. C., 1989, Westinghouse internal letter, "Review of Local Dilution Transients in Pressurized Water Reactors", NS-SAT-TSA-89-116.

/27/ Vanttola, T., Valtonen, K., 1993, "Boron Dilution During Refuelling Outage in a Pressurized Water Reactor", Proceedings, ASME Winter Annual Meeting, Third

International Power Plant Transient Symposium, FED-Vol. 140, Anaheim, California, U.S.A, pp. 129-134.

/28/ J. Hyvärinen, 1992, "An Inherent Boron Dilution Mechanism in Pressurized Water Reactors", Proceedings, 5th International Topical Meeting on Nuclear Reactor Thermal-hydraulics, NURETH-5, Salt Lake City, USA, pp. 98-109.

/29/ Reeder, P. L., 1979, "Boric Acid Effects on Water Properties and LOCA Thermal-hydraulics", Idaho Falls National Engineering Laboratory, LTR 20-91, Idaho Falls, U.S.A., 42 p.

/30/ Twogood, F. J., Strong, B., Lew, B. S., Kramer, C., 1993, "Boric Acid Precipitation Following a Cold Leg LOCA", Trans. Am. Nucl. Soc., 68, pp. 286-288.

/31/ "Long-Term Core Cooling - Boron Considerations", 1975, CLC-NS-309, Westinghouse.

/32/ Tuunanen, J., 1986, "The Behaviour of Boric Acid During Long-term Cooling of a Nuclear Reactor", Diploma Thesis, Technical University of Lappeenranta (LTKK), Department of Energy Technology (ENTE), Lappeenranta, Finland (in Finnish).

/33/ Kervinen, T., Tuunanen, J., 1986, "The Behaviour of Boric Acid During Emergency Cooling of the Light Water Reactor Core", Research Report YDI615, Helsinki, Finland.

/34/ Tuunanen, J., 1986, "Solubility of Boric acid - Borax solution on Water", Technical Research Centre of Finland (VTT), Nuclear Engineering Laboratory (YDI), Report TEKO 8/86 (In Finnish).

/35/ Balashov, S. M., Videneev, E. N., Nigmatulin, B. I., 1992, "The Effects of Boric Acid on the Thermalhydraulic Characteristics of a Partially Uncovered Core", Thermal Engineering, 39 (9), pp. 509-512.

/36/ Gordon, B. G., Grigorev, A. S., Pomelnikov, V. N., 1990, "The Physical Mechanism of Core Blocking with Evaporation of Orthoboric Acid", Paper to the All-Union Confeerence: Two-Phase Flow in Power Plant Machines and Apparatus, Leningrad, Soviet-Union.

/37/ Tuunanen, J., Kervinen, T., 1990, "Boric Acid Experiments with Modified VEERA Facility", Technical Research Centre of Finland (VTT), Nuclear Engineering Laboratory (YDI), Research Report YDI07/90.

/38/ Raussi, P., Tuunanen, J., 1992, "Buoyant Mixing of Boric Acid in a Core of a Nuclear Reactor: Experiments in the IVO Hydraulic Laboratory", Technical Research Centre of Finland (VTT), Nuclear Engineering Laboratory (YDI), Research Report YDI 12/92 (in Finnish).

/39/ Tuomisto, H., et.others, 1992, "RELAP5/MOD3 Assessment Results in Finland", Paper presented at The First Code Application and Maintenance Program Meeting, Paul Scherrer Institute, Switzerland.

/40/ Tuunanen, J., 1992, "Simulation of VVER-440 Reactor SBLOCAs with RELAP5/MOD3 Code", Technical Research Centre of Finland (VTT), Nuclear Engineering Laboratory (YDI), Report PROPA 1/92 (in Finnish).

/41/ Duffey, R. B., Sursock, J. P., 1985, "Natural Circulation Phenomena Relevant to Small-breaks and Transients", Proceedings, Specialists Meeting on SBLOCA Analyses in LWRs, Pisa, Italy, Vol 1, pp. 87-118.

/42/ Szabados, L., Almeida, C., Maroti, L., 1985, "Scaling and Instrumentation of the PMK/NVH Facility", Proceedings, Specialists Meeting on SBLOCA Analyses in LWRs, Pisa, Italy, Vol 1, pp. 527-537.

/43/ Hicken, E. F. et. others, 1992, "Thermal-hydraulic WWER Test Facilities in Electrogorsk/Russia and in Zugres/Ukraine", Final Report of OECD/NEA/CSNI expert group.

/44/ Kervinen, T., Hongisto, O., 1986, "Natural Circulation Experiments in the REWET-III Facility," Proceedings, Int. ANS/ENS Topical Meeting on Thermal Reactor Safety, San Diego, Calif., Vol. 3, Paper XIII.4.1.

/45/ Hyvärinen, J., Kalli, H., Kervinen, T., 1989, "RELAP5 Assessment with REWET-III Natural Circulation Experiments," Proceedings, 4th Int. Topical Meeting on Nuclear Reactor Thermal-hydraulics, Karlsruhe, F.R.G, Vol. 1, pp. 510-515.

/46/ Lomperski, S, W., Kouhia, J., "Natural Circulation Experiments with a VVER Reactor Geometry", Nuclear Engineering and Design (to appear).

/47/ Bandurski, T., Ezsoel, G., Maroti, L., Toth, I., 1989, "Modelling of Two-Phase Natural Circulation in a WWER-plant: PMK Experimental Results", Proceedings, 4th International Topical Meeting on Nuclear Reactor Thermal-hydraulics (NURETH-4), Karlsruhe, F.R.G., Vol. 1, pp. 478-483.

/48/ Toth, I., 1988, "Loop Seal Effects on Core Cooling in VVER Type Reactors", Proceedings, International ENS/ANS Conference on Thermal Reactor Safety, Avignon, France, Vol. 3. pp. 1049-1058.

/49/ Szabados, L., Ézsöl, G., 1991, "Experiments and Future Research Needs", International Seminar on Horizontal Steam Generator Modelling, Lappeenranta University of Technology, Research Papers 18, Lappeenranta, Finland, Vol II, pp. 287-320.

/50/ Karppinen, I., 1991, "Modelling of Horizontal Steam Generator with RELAP5/MOD2 and MOD3", International Seminar on Horizontal Steam Generator Modelling, Lappeenranta University of Technology, Research Papers 18, Lappeenranta, Finland, Vol II, pp. 182-194.

/51/ Tuunanen, J., 1993, "Analysis of the VVER-440 Reactor Steam Generators Secondary Side with the RELAP5/MOD3 Code", Second International Seminar of

Horizontal Steam Generator Modelling, Lappeenranta University of Technology, Research Papers 30, Lappeenranta, Finland, pp. 175-184.

/52/ Sabotinov, L., 1989, "Thermal-hydraulic Parameters in the Reactor Core of VVER-440 (B-230) at Loss of Main Pump Power", Paper presented in the Technical Committee / Workshop on the IAEA Programme on Computer Aided Safety Analysis, Berlin, GDR.

/53/ Bezrukov, Yu. A., et al., 1977, "Experimental Investigation and Statistical Analysis of Data on Burnout in Rod Bundles for Water-Moderated Water-Cooled Reactors", Thermal Engineering, Vol 23, No. 2, pp. 67-70.

/54/ Smolin, V. N., et al., 1982, "Experience of Correlating Data on Burnout in Rod Assemblies by the Subchannel Method", Thermal Engineering, Vol. 29, No. 1, pp. 16-18.

/55/ Städtke, H., Kolar, W., 1986, "Prediction Capabilities of RELAP5/MOD1-Eur; An Improved Version of the LWR Safety Code RELAP5/MOD1", Proceedings, European Nuclear Conference, Genova, Vol. 3, pp. 535-542.

/56/ Groenevelt, D. C., Cheng, S. C., Doan, T., 1986, "Critical Heat Flux Look-up Tables", Heat Transfer Engineering 7, pp. 46-62.