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ON RECRITICALITY DURING REFLOODING OF A DEGRADED BOILING WATER REACTOR CORE

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ABSTRACT

In-vessel core melt progression in Nordic BWRs has been studied as a part of the RAK-2 project within the Nordic Nuclear Safety Programme 1994-1997. A part of this study was the evaluation of possibility and consequences of recriticality in a reflooded, degraded BWR core.

The objective of the study was to examine, if a BWR core in a Nordic nuclear power plant can reach critical state in a severe accident, when the core is reflooded with unborated water from the emergency core cooling system and what is the possible power augmentation related to recriticality. The containment response to elevated power level and consequent enhanced steam production was evaluated.

The first subtask was to upgrade the existing neutronics/thermal hydraulic models to a level needed for a study of recriticality. Three different codes were applied for the task: RECRIT, SIMULATE-3K and APROS. Preliminary calculations were performed with the three codes. The results of present studies showed that reflooding of a partly control rod free core gives a recriticality power peak of a substantial amplitude, but with a short duration due to the Doppler feedback. The energy addition is small and contributes very little to heat-up of the fuel. However, with continued reflooding the fission power increases again and tend to stabilise on a level that can be ten per cent or more of the nominal power, the level being higher with higher reflooding flow rate.

A scoping study on TVO BWR containment response to a presumed recriticality accident with a long-term power level being 20 % of the nominal power was performed. The results indicated that containment venting system would not be sufficient to prevent containment overpressurization and containment failure would occur about 3-4 h after start of core reflooding. In the case of station blackout with operating ADS the present boron system would be sufficient to terminate the criticality event prior to containment failure, but in case of feedwater LOCA and boron dilution to the whole containment water pool, the present boron concentration would not be sufficient to ensure subcriticality in the core.

The project succesfully initiated development of adequate analytical tools for recriticality studies and thus laid foundation for the continued work in the field in the framework of the EU SARA project (Severe Accident Recriticality Analyses) 1997-1998.

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NOMENCLATURE*

A	Flow area
Bi	Biot's number
C_0	$=f(\alpha,D_e)$ = distribution parameter
C_i	Concentration of i'th delayed neutron precursor
c_{pl}, c_{pg}, $c_{p\,wall}$	Specific heat capacity of liquid and gas
D_h	Hydraulic diameter
D_e	Eqiuvalent diameter
$dt, \Delta t$	Time step
G	Mass flux
g	constant of gravitation = 9.81 [m/s^2]
F	Forced convection evaporation factor (assumed ≈ 1 in this study)
$F(D_e)$	Additional term as a function of equivalent diameter
h_{fr}	Effective heat transfer coefficient on wetted side of the quench front
h_{NB}	Nucleate boiling heat transfer coefficient
h_{conv}	Film boiling heat transfer coefficient
h_R	Radiation heat transfer coefficient
h	Effective heat transfer coefficient in nucleate boiling regime
h_g, h_l	Specific enthalpy of steam, water
k_{eff}	Effective multiplication factor
k_l, k_g, k_{wall}	Thermal conductivity of water, steam and wall
K(T)	Coefficient in Urbanic-Heidrich correlation
Nu_l	Liquid Nusselt number
Nu_g	Steam Nusselt number
Pe	Peclet's number
Pr_l	Prantl's number
p	Pressure
P_f	Friction loss
q	Heat flux
Q_1,Q_2	Neutron sources in fast and thermal groups
Q_g, Q_l	Steam, liquid heating power
Re_l	Reynolds number
S	Nucleate boiling suppression factor (assumed ≈ 1 in this study)
S_g, S_l	Liquid, steam source
S_w	Perimeter of quenching
T	Temperature
T_w	Wall temperature
T_s	Coolant temperature
T_g	Steam temperature
T_{clad}	Cladding temperature
T_0	Leidenfrost (quenching) temperature, $T_0=\Delta T_{Leid} + T_{sat}$
t	Time
u_g	Velocity of steam
u_m	Average velocity of steam/water mixture
$\tilde{v}_{Gj}$	Drift velocity = $f(P,\alpha, D_e)$
v_{fr}	Quench front velocity
v_1	Velocity of fast neutrons

v_2	Velocity of thermal neutrons
V_{rod}	Control rod volume
W	Mass of oxidized Zr per unit area
w_m	Mass flow rate of mixture
w	Mass flow rate
x	Steam quality
x_k	Pressure loss coefficient
z	Coordinate

Greek letters

α	Void fraction
β	Fraction of delayed neutrons
ε	Effective emissivity
ε_w	Emissivity of wall
ε_l	Emissivity of water
ϕ_g	Neutron flux in group g
ϕ_1	Fast neutron flux
ϕ_2	Thermal neutron flux
λ_i	Decay constant of i'th group of delayed neutrons
μ_l	Dynamic viscosity of water
ν	Number of neutrons released per fission
ρ_l	Density of water (kg/m^3)
ρ_g	Density of steam (kg/m$^{3)}$
ρ_m	Density of mixture
Σ_f	Fission cross section
Σ_r	Thermalization cross section
σ	Stefan-Boltzmann constant
θ	Temperature term defined in Eq.(25)

• SI units if not otherwise specified

EXECUTIVE SUMMARY

In-vessel core melt progression in Nordic BWRs has been studied as a part of RAK-2 project within the Nordic Nuclear Safety Programme 1994-1997. The work scope has been divided into three subtasks: reflooding, recriticality and late phase melt progression and has been carried out by partners from SKI, Studsvik Eco & Safety AB, Vattenfall Energisystem AB, Risø National Laboratory, Teollisuuden Voima Oy and VTT Energy.

The primary concern in criticality events is a super prompt-critical excursion which would result in rapid disintegration of fuel, rapid molten fuel coolant interaction, and the production of a large pressure pulse capable of directly failing the vessel. The analysis conducted in Pacific Northwest Laboratories [Ref. 1] indicate that the rapid disintegration of fuel is not likely under the conditions of reflooding a hot core, which may or may not be degraded. Doppler feedback is the principle mechanism for terminating rapid transients in low enriched uranium-water systems and is adequate to limit the energetics of reflood recriticality to a level below which the reactor vessel would be threatened by a pressure pulse.

If the the initial power excursion does not fail the fuel, but the reactor still remains critical following the initial pulse at the time of reflooding, it will either reach an oscillatory mode in which water periodically enters the core and is expelled from the core or it will approach a quasi-steady power level [Ref. 1]. The primary concern in the latter case would be the increasing temperature of the suppression pool and the subsequent potential for containment over-pressurization. The capacities of the suppression pool residual heat removal system and the containment filtered venting system are not enough to prevent containment overpressurization during recriticality. The pressure build-up would eventually fail the containment, and probably also cause a failure of core cooling systems. The boiling of the condensation pool would fail the pumps of the core spray system 323, taking suction from the pool. This could ultimately lead to further core damage and create a direct release path to the environment.

The objective of the study was to examine, if a BWR core in a Nordic nuclear power plant can reach a recritical state in a severe accident if reflooded with unborated water and what is the possible power augmentation related to recriticality. The containment response to elevated power level and consequent enhanced steam production would be evaluated.

The approach to the problem was to apply severe accident thermal hydraulic codes to give estimates for the probable control rod/fuel configurations at the start of reflooding and use a separate reactor physics code for calculation of reactivity and fission power during the transient. The reactor physics codes available in the Nordic countries are designed for standard transient calculations and design of fuel management strategies and all of them had to be modified to meet the requirements of the special severe accident problem with recriticality.

The Nordic recriticality studies used three different tools - RECRIT, APROS and SIMULATE-3K - for neutronics related to recriticality. The first version of RECRIT, written at Risø National Laboratory, had simple but sound models for neutronics but carried some significant modeling undershootings in thermal hydraulic parts. These deficiencies were remedied with combining thermal hydraulic models developed at VTT Energy and the neutronics models of the original RECRIT code. The upgraded thermohydraulic model accounts for the quench front movement and the heat transfer below and above the quench front as well as the heat generation due to cladding oxidation. After completion of the code updating, the final recriticality analyses were performed for TVO I/II with RECRIT as a joint effort between Risø and VTT.

The APROS simulation code was applied to study recriticality at VTT. A three-dimensional code model for TVO I/II has been prepared in APROS describing 500 fuel bundles in 250 core channels. Fuel data for ATRIUM10 bundles were processed to the proper format and were implemented into the APROS core model. A set of scoping studies with APROS TVO model was performed taking the initial core damage states for the recriticality calculations as input from the previous MELCOR and MAAP4 calculations.

The third approach, taken by Studsvik Eco & Safety AB, applied the SIMULATE-3K code. SIMULATE-3K is a transient version of the steady-state core analysis computer code SIMULATE-3, which is part of the Studsvik well-established Core Management System code package for steady-state core design and optimisation calculations, which consists of two principal codes; CASMO and SIMULATE.

The RECRIT code calculations resulted in reasonable trends in power behaviour during recriticality. The power peaks turned down rapidly after reaching criticality and then stabilised or oscillated around an elevated power level. Since the coolant void fraction and the fuel temperature are the principal factors affecting criticality, variations on coolant flow rate were carried out. Both the power peak and the stabilised power level were strongly dependent of the coolant flow rate. With the lowest flow rate of 22.5 kg/s only small power peaks of 30 % could be seen after reaching criticality. With the flow rate of 45 kg/s several power peaks with maximum of 220 % of the nominal power could be seen, but outside the power peaks the fission power was negligible. With the flow rate of 160 kg/s a clear power peak of approximately 400 % of nominal power could be seen followed by several power peaks and the power remained on the level of 30 % from the nominal power. With the highest reflood rate of 540 kg/s the first power peak was about 1730 % of the nominal power followed by another power peak about 8 second later. The power stabilised in this case to a level of 100 % of nominal power.

Scoping studies with APROS resulted in a high initial power peak of about 260 % of nominal power, but the power stabilised in low level, close to decay heat, after the initial peak. The reflooding rate in the scoping studies was low, 10 - 50 kg/s. The calculations with higher reflooding rates terminated in numerical problems soon after start of power peaking. The code will be further developed to overcome these numerical problems in the future work.

Recriticality studies with SIMULATE-3K were started with tentative calculations on a simplified 4 by 4 assembly BWR core, and on a full scale model of Oskarshamn 3 3300 MW_{th} plant. Studied cases started with total station blackout after 3 s with full power followed by scram and stopping of the core flow. The water boil-off and core heat up started and when the core temperatures had increased to 1500 K, half the number of the control rods were withdrawn and reflooding was started. The tentative SIMULATE-3K results showed that the Doppler feedback limited the power pulse to less than one full-power second, although high power peaks may occur (100 - 1500 % of the nominal power). After the peak, the fission power stabilized at an elevated level , the magnitude of which increased with reflooding flow rate. In case of water injection of 625 kg/s corresponding to 5 % nominal recirculation flow rate, the power stabilized at about 35 % of full power.

In the full core calculations for O-3 in the high pressure case the power peak after reaching recriticality was low, but the reactor stabilised at a power level corresponding to about 15 % of the nominal power for a reflooding mass flow of 175 kg/s. In the low pressure case SIMULATE-3K predicted an initial power peak of about 15 % of the nominal power after which, the power settled down close to decay power. SIMULATE-3K is still under development and needs additional updating and modifications. The full-scale calculations could, therefore, only be carried through with some simplified boundary conditions.

The results of present studies showed that reflooding of a partly control rod free core can give a recriticality power peak of a substantial amplitude, but with a short duration due to the Doppler feedback. The energy addition is small andcontributes very little to heat-up of the fuel. However, with continued reflooding the fission power increases again and tend to stabilise on a level that can be ten per cent or moreof the nominal power, the level being higher with higher reflooding flow rate. As only a thin axial control-rod-free core layer is needed to obtain a critical configuration, the axial nodalisation has some impact on the results. With 20 - 40 axial nodes the effect of nodalisation is quite small.

A scoping study on TVO BWR containment response to a presumed recriticality accident with a long-term power level being 20 % of the nominal power was performed. The results indicated that containment venting system would not be sufficient to prevent containment overpressurization and containment failure would occur about 3-4 h after start of core reflooding. In the case of station blackout with operating Automatic Depressurization System (ADS) the present boron system would be sufficient to terminate the criticality event prior to containment failure, but in case of feedwater LOCA and boron dilution to the whole containbment water pool, the present boron concentration would not be sufficient to ensure subcriticality in the core.

The project succesfully initiated development of adequate analytical tools for recriticality studies and thus laid foundation for the continued work in the field in the framework of the EU SARA project (Severe Accident Recriticality Analyses) 1997-1998.

1. INTRODUCTION

In-vessel core melt progression in Nordic BWRs has been studied as a part of RAK-2 project within the Nordic Nuclear Safety Programme 1994-1997. The work scope has been divided into three subtasks: reflooding, recriticality and late phase melt progression and has been carried out by partners from SKI, Studsvik Eco & Safety AB, Vattenfall Energisystem AB, Risø National Laboratory, Teollisuuden Voima Oy and VTT Energy.

The objective of the work has been to evaluate when and how the reactor core is still coolable with water and what are the probable consequences of water cooling. The important phenomena related to rewetting of a degraded core include rapidly increased hydrogen generation inducing further core degradation, possibility of recriticality in a time frame when control rods have melted and disappeared but the fuel still stands in an intact geometry and finally, if the core proves not to be quenchable within the original core boundary, if the core debris reaches coolable configuration in the reactor pressure vessel (RPV) lower head.

Investigation of reflooding of an overheated BWR core was carried out by calculating similar accident scenarios with three different computer codes; risk analysis codes MAAP4 and MELCOR and detailed best-estimate code SCDAP/RELAP5. The reasonability of the calculated results was evaluated in light of experimental observations from different available results.

The base scenario for the study was station blackout with successful depressurization of the reactor coolant system (RCS) and with failure of automatic depressurization system (ADS). The restoration of power and subsequent start of coolant injection was assumed to occur at different times, when the maximum core temperature varied from 1400 K to 2000 K.

The results showed that each code model predicted a different end state of the core after reflooding: MAAP4 predicted formation of melt pool in the core, MELCOR resulted in formation of rubble bed and SCDAP/RELAP5 predicted material melting and/or fuel fragmentation due to mechanical stresses caused by temperature difference of coolant and hot fuel rods. However, fast core cooling was obtained in cases where reflooding was started at the maximum core temperature of < 1600 K even with half a capacity of high pressure injection system. The core was slowly coolable if the cladding temperature at the beginning of the reflooding was < 1800 K. The results of the high pressure cases showed that large local temperature differences are created in the core due to slow boiloff of coolant and respective efficient oxidation in the uncovered parts of the core.

The variations of the water injection location showed that the core top spray provides more efficient cooling of core than the downcomer injection. In all cases the hydrogen production was increased after start of reflooding. Typically the oxidation heat exceeded the decay heat generation in all reflooding cases.

All codes predicted some time window between melting of control rods and fuel. The lenght of the time window varied with different codes being 1-2 min with MELCOR and 3 - 40 min with MAAP4. All codes predicted that there is at least a metastable period, where part of the core (10-70 %) is without control material. The fuel pellets may be in a rubble bed form or in intact fuel geometry. Studies of possible recriticality seem to be most interesting in cases, where fuel rods are in intact geometry and the control rods have partially melted.

The primary concern in criticality events is a super prompt-critical excursion which would result in rapid disintegration of fuel, rapid molten fuel coolant interaction, and the production of a large pressure pulse capable of directly failing the vessel. The analysis conducted in Pacific Northwest Laboratories [Ref 1] indicate that the rapid disintegration of fuel is not likely under the conditions of reflooding a hot core, which may or may not be degraded. Doppler feedback is the principle mechanism for terminating rapid transients in low enriched uranium-water systems and is adequate to limit the energetics of reflood recriticality to a level below which the reactor vessel would be threatened by a pressure pulse.

If the the initial power excursion does not fail the fuel, but the reactor still remains critical following the initial pulse at the time of reflooding, it will either reach an oscillatory mode in which water periodically enters the core and is expelled from the core or it will approach a quasi-steady power level. The primary concern in the latter case would be the increasing temperature of the suppression pool and the subsequent potential for containment over-pressurization. The capacities of the suppression pool residual heat removal system and the containment filtered venting system are not enough to prevent containment overpressurization during recriticality. The pressure build-up would eventually fail the containment, and probably also cause failure of core cooling systems. Boiling of the condensation pool would fail the pumps of the core spray system 323, taking suction from the pool. This could ultimately lead to further core damage and open a direct release path to the environment.

2. BACKGROUND

Reactivity accidents in degraded cores have been addressed in some studies during the past decade with somewhat varying conclusions.

In the earlier work by Scott [Ref. 1] it was concluded that recriticality is possible but the probable power excursion would be moderate not being able to fragment the fuel. If the reactor remains critical following an initial excursion at the time of reflooding, and the reflooding is conducted without boration, it will either enter an oscillatory mode in which water periodically enters and is expelled from the core or it will approach a quasi-steady power level. In either case, the power level achieved will be determined by the balance between the reactivity added and the feedback mechanisms (reactor water level, void fraction, fuel temperature, primary system pressure). Furthermore, a recriticality event is likely to produce core power less than about 20 % of normal power and probably not much more than 10 % of normal power, but may be significantly above the decay heat level (circa 2 % after 30 minutes).

Shamoun et al. [Ref. 2] present the importance of change in void fraction on recriticality calculations. During reflood, when the fuel temperature is still high before quenching, and void fraction is below 20 %, recriticality is possible only if at least 95 % of the control rod material is lost from the control cell and the core is reflooded with unborated water. Recriticality is not possible under any circumstances during reflooding phase before fuel quenching, even if 100 % of control material is lost, if the void fraction exceeds 20 %. Reflooding the core with borated water at a boron concentration of 1200 ppm is sufficient to prevent recriticality under short- and long-term recovery conditions corresponding to zero void fraction and high (525 K) and low (325 K) moderator temperature.

Two-dimensional four-bundle studies by Mosteller et al. [Ref. 3] suggest that retention of 10-20% of control poison may prevent recriticality. However, three-dimensional effects may affect the result, since for example all the present severe accident models predict non-uniform melting of the control rods.

Okkonen [Ref. 4] on the other hand proposes that a degraded BWR core will reach recriticality if at least a length of one meter of fuel is intact and the coolant void fraction in the core is below 0.6.

Earlier studies based on hand calculations [Ref. 5] suggest that 850-1000 ppm boron in the coolant would be sufficient to prevent recriticality in intact fuel geometry of TVO I/II. Currently emergency core cooling water in the Nordic BWRs does not contain boron.

The basic characteristics of the possible recriticality during reflooding can be evaluated by looking at the variation of reactivity parameters as a function of void fraction, liquid temperature, fuel temperature and control rod concentration (occupation) in the core. The following analysis is based on the results of the burnup and reactivity coefficient calculations carried out with the CASMO-4 fuel assembly burnup code [Ref. 16, 17] for a SVEA-64 bundle having an average enrichment of 3.12 % and six burnable absorber rods (Gadolinia content of 3.95 %). Three cases were defined as follows:

Case	Burnup of the bundle (MWD/kgU)	Void history of the bundle (%)
BOC	10	40
MOC	20	40
EOC	30	75

In Figure 1 the BOC data, as used in the RECRIT, are shown. When the RECRIT data sets were generated, it was assumed that the void content was constant everywhere in the bundle, ie. both inside and outside the flow channel.

The assumption of equal void fraction inside and outside the flow channel could be discussed. During the normal operation the voids exist only inside the channel. During reflooding the equal collapsed water levels can be expected in the fuel rod and the bypass regions. The swell level in the fuel can be higher due to more evaporation inside of the box but on the other hand, the friction of the gas flow tends to drop the fuel side water level. Thus the selected approach is a good compromise.

The results indicate that the Doppler reactivity coefficient is around -2.3 pcm/K. The fuel temperature rise of 1000 K can compensate the excess reactivity of 2300 pcm in the BOC core. For the concentration of the control rods the reactivity response is -300 pcm/%. The relation between reactivity and the changes in void fraction depends strongly on the void fraction range. No criticality is possible in the void fraction range of 0.85 -1.00 with BOC data, in the void fraction range of of 0.70 - 1.00 with MOC data and in the void fraction range of 0.50 - 1.00 with EOC data. In the BOC core criticality is possible in the cold conditions with any control rod concentration 0 - 100 %, when the infinite multiplication factor is used as the criterion. With MOC data the criticality is possible with 0 - 50 % control rod concentrations and with EOC data with 0 - 35 % control rod concentrations.

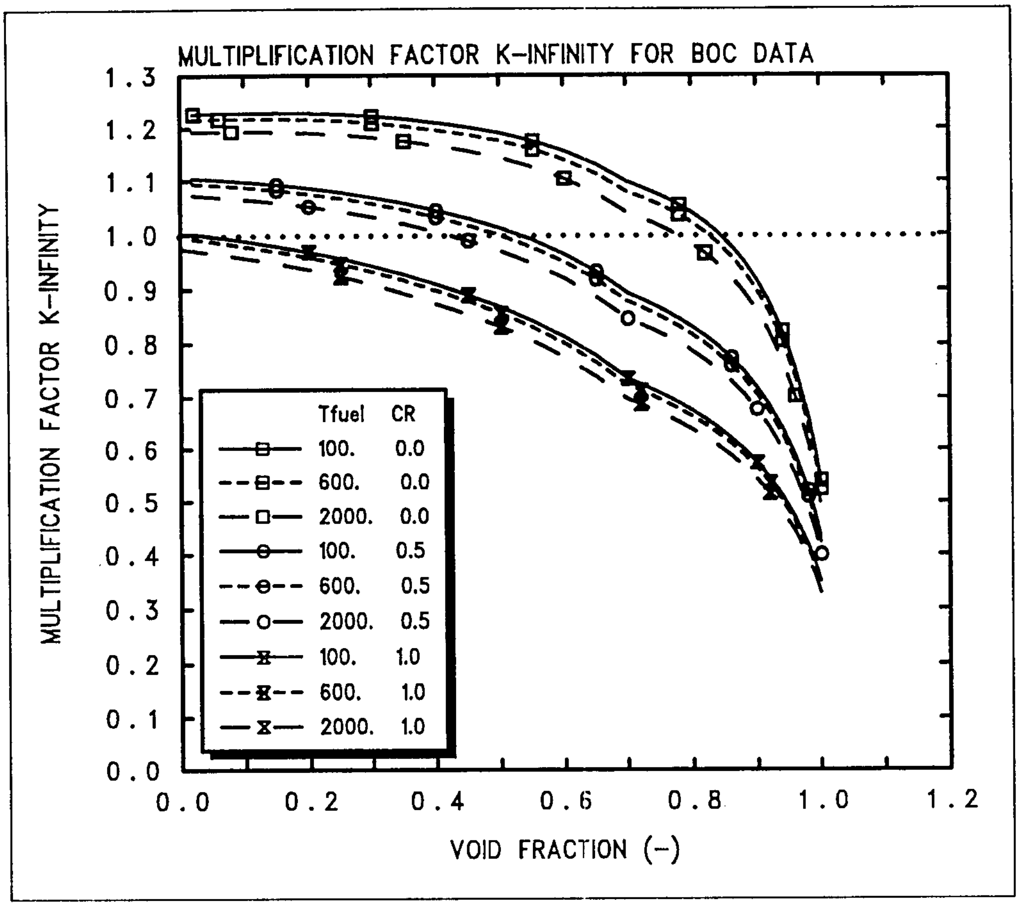


Figure 1. *Characteristics of the reactivity behaviour on the expected parameter range of reflooding.*

The strongest responses of void fraction and control rod concentrations on reactivity may be discussed against expected conditions during reflooding. The void fraction of 0.9 ... 1.00 is expected in droplet dispersed flow pattern above quenching front. The void fraction of 0.0 ... 0.70 can be expected below the quenching front in bubbly or churn turbulent flow pattern. Therefore, based on the infinite multiplication factor (BOC data in the Figure 1 and MOC data examined as well) three different zones can be defined: 1) no criticality is possible with control rod concentrations 100% for BOC data, 50% for MOC data and 35% for EOC data, 2) no criticality is possible above quenching front and 3) the criticality is possible below the quenching front in the zone with control rod concentrations of less than 100% in BOC data case, 50 % in MOC data case and 35% in EOC data case.

The first power peak will appear within a couple of seconds after the reactor becomes critical. Because the power level of the reactor initially is very low initially, there is a possibility to reach prompt criticality in this first peak. If the power peak is high enough, it pushes the water down for a short time. Later on new power peaks are possible, but in these cases the prompt criticality is less probable, because the base neutron flux level is reasonaly high. The power reacts quickly to the increased reactivity but the the growth of reactivity is compensated by the Doppler effect.

If the core, especially in the BOC conditions, is further reflooded after the first power peak, the infinite multiplication coefficient indicates, that the core with melted control rods may be difficult to recover, although the risk for prompt criticality can be excluded. However, this period was not studied within the current NKS work.

3. OBJECTIVES

The objective of this study is to examine, if a BWR core in a Nordic nuclear power plant can reach a recritical state in a severe accident when the core is reflooded with unborated water from the emergency core cooling system and what is the possible power augmentation related to recriticality. The containment response to elevated power level and subsequent enhanced steam production will be evaluated.

However, there are no standard code systems applicable to address the problem of recriticality. The only available tools will be a combinations of thermal hydraulic codes giving estimates of the probable control rod/fuel configurations at the start of reflooding and of a separate reactor physics code calculation of the fission power and the thermal hydraulics during the transient. The reactor physics codes available in the Nordic countries are designed for standard transient calculations and for design of of fuel loadings and all of them had to be modified to meet the requirements of this special severe accident problem. The first subtask was to upgrade the existing models to a level needed for the study of recriticality. The second task was to update or build plant inputs for the codes. The third task was to perform selected base case analyses.

This work will produce base analyses and analysis tools for assessment of recriticality. Furhermore, the code development and acquisition of analysis routine in the area will form the necessary background for the future work in the EU project SARA in 1997-1998.

4. TECHNICAL APPROACH

The Nordic recriticality studies used three different tools for reactor physics related to recriticality. Firstly, a simple model RECRIT has been created at Risø National Laboratory. The first version of RECRIT has simple but sound models for neutronics but might have carried some significant modeling undershootings in thermal hydraulic parts. The reflooding physics was not described in detail and the location of the quenching front was considered to be equal to the collapsed water level. No liquid entrainment was calculated either. These deficiencies were remedied with implementation of thermal hydraulic models developed at VTT Energy with the neutronics of the original RECRIT code. The upgraded thermohydraulic model accounts for the quench front movement and the heat transfer below and above the quench front as well as the heat generation due to cladding oxidation. After completion of the code updating, the final recriticality analyses were performed for TVO I/II with RECRIT.

Secondly, the APROS simulation code was applied to study recriticality at VTT. A three-dimensional code model for TVO I/II has been prepared in APROS describing 500 fuel bundles in 250 core channels. Fuel data for ATRIUM10 bundles was processed to the proper format and was implemented into the APROS core model. A set of scoping studies with

APROS TVO model was performed taking the initial core damage states for the recriticality calculations as input from the previous MELCOR and MAAP4 calculations.

The third approach to recriticality applied the SIMULATE-3K code for neutronics and the studies were performed by Studsvik Eco & Safety AB. SIMULATE-3K is a transient reactor physics model with more detailed neutronics but simpler thermohydraulics than e.g. APROS. SIMULATE-3K was used with Oskarshamn 3 plant data. A tentative calculation was performed with a simple core model having 4 x 4 BWR assemblies. Studied cases started with total station blackout after 3 s with full power followed by scram and stopping of the core flow. The water boil-off and core heat up started and when the core temperatures had increased to 1500 K, half the number of the control rods were withdrawn and reflooding was started.

5. RECRIT CODE

The RECRIT computer code is based on joint inter-Nordic work between Risø and VTT Energy for producing a practical recriticality analysis tool without dramatic undershootings on any relevant areas, but without complicating the task too much. The code combines a neutronics model, based on a two-dimensional cylinder symmetric core description, with a novel code developed at VTT for BWR reflooding thermohydraulics.

User-friendliness and the following guidelines have been applied in the development of the thermohydraulics:

The needed thermohydraulic input data were minimized by hard-wiring the nodalization to fixed sections for core, bypass, lower plenum, upper plenum, steam separator, steam dome and downcomer. The calculation level subdivision of nodes is generated automatically from the data of the seven main control volumes by assuming equal node heights and areas in the vertical direction inside a main control volume. In the core the thermohydraulic axial and radial node divisions are adjusted to match the nodalization selected for the neutronics.

One base rule was to avoid the complexity of the former NORCOOL-1 computer code [Ref. 6] in the modelling of the reflooding related phenomena. This precludes any complex water level modelling as a separator between liquid continuous and steam continuous regimes. No detailed radiation model between different rod groups inside a bundle can be included. The droplet field is accounted for by drift flux model instead of using particle distribution modelling. On the other hand, the inconvenient limitations of NORCOOL-1 had to be avoided. This brought in the possibility to model parallel core channels.

Experiences have shown that the code structure should be simple and include a lot of comments. That is why the following principles have been chosen in development of RECRIT thermohydraulics code structure:

1. The integral momentum equations are applied only, where momentum is solved, i.e. in the inlet of each core channel and the bypass.
2. The steam table data are given as rational functions with reasonable accuracy.
3. No circulation pumps are described.

4. Only limited boundary conditions can be used for water injection into the vessel and for steam release from the vessel.

Some thermohydraulic phenomena are of particularly great importance for recriticality studies and the accurate modelling of these phenomena is emphasized in the development of the RECRIT thermohydraulics. Especially, by low reflooding rate the reactor power excursion may be expected within a few tens of centimeters in the region, where the top of the melted control rods exist. In this area the spatial fuel temperature and void fraction profile histories have a strong contribution to spatial dynamics of the neutron flux peaking. The important phenomena include the entrainment of water above the quench front, boiling below quench front and related void fraction, movement of quench front, post-dryout heat transfer above the quench front, oscillations of water level between core (bypass) and downcomer, water level oscillations between different core channels and effect of system pressure changes on steam dump rate, water injection rate into the vessel, water swell behaviour due to flashing and volume collapse due to condensation.

The decay heat and the fission power of the fuel pellets is integrated by the heat conduction model, which integrates the fuel pellet , the gas gap and the cladding structure for getting the Doppler effect on the neutronics solution. The heat generation rate (fission and decay heat) determined by the neutronics model is transferred to the thermalhydraulic part of the code as a heat source of the fuel pellet. The thermohydraulic model calculates also the heat generation from zirconium oxidation reaction in the cladding. Presently, the oxidation of fuel boxes is not accounted for in RECRIT. The control rod melting is simulated assuming the temperature of a control rod to follow closely the temperature of the fuel rods. The thermohydraulics model solves the heat balance of fission/decay heat, oxidation heat and reflood heat transfer to coolant and updates the local coolant densities (or void fraction and liquid temperature) for the neutronics model. If the coolant were borated, the boron concentration could be solved as well. The thermohydraulic part of the code calculates the radiation heat transfer between heat structures in axial and radial directions as well as the radiative heat losses from the peripheral zones to the surroundings.

5.1 Neutronics modelling in RECRIT

The neutron flux distributions, reactivities etc. are calculated by the subroutine TWODIM, which is a standard diffusion theory, multigroup, difference approximation code [Ref 15].

TWODIM solves the multigroup diffusion equation. In RECRIT the two group formalism is applied and the equations for the fast and thermal neutrons are written as:

$$D_1\nabla^2\Phi_1 - \Sigma_1^r\Phi_1 + \lambda(\nu_1\Sigma_1^f\,\Phi_1 + \nu_2\Sigma_2^f\,\Phi_2) + Q_1 = 0 \;\left(= \left(\frac{1}{v_1}\frac{d\Phi_1}{dt}\right)\right) \tag{1}$$

$$D_2\nabla^2\Phi_2 - \Sigma_2^r\Phi_2 + \Sigma_{12}\,\Phi_1 + Q_2 = 0 \;\left(= \left(\frac{1}{v_2}\frac{d\Phi_2}{dt}\right)\right) \tag{2}$$

$$\lambda = \frac{1}{k_{eff}} \tag{3}$$

The original version of TWODIM was aimed for stationary solutions and a modification of the code making dynamical solutions possible has been presented in [Ref.14]. As a result of the analysis in [Ref 14] one may conclude, that if timesteps equal to the prompt neutron lifetime are applied, then the procedure makes a quasi correct time integration for the neutronics in the critical and prompt critical core.

The numerical solution includes an inner and an outer iteration. In the outer iteration the eigenvalue problem is solved for the discretized set of equations, which in matrix notations may be written as

$$A\Phi = S\Phi + \lambda F\Phi \tag{4}$$

where A represents the leakage, the absorption and out-scattering from the individual energy groups, S contains the transfer cross section between groups and F the fission spectrum and neutron production.

In the inner iteration the matrix inversion is done for the diagonally dominant band matrix A from the equation

$$A\Phi = k \tag{5}$$

by using the block successive over-relaxation iterative method. For iteration the optimum relaxation factor has been calculated.

TWODIM is a versatile code, capable of doing neutronics calculations using rectangular, cylindrical or polar coordinates in 3 modes:

1. In the "k_{eff}-mode", k_{eff} is determined for any source-free configuration as a function of the parameters contributing to reactivity, i.e. fuel temperature, coolant temperature, coolant void fraction and control rod configuration. The control rods may be partially drawn or melted.
2. In the "poisoning-mode", a critical distribution of a poisoning cross section is determined
3. In the "source-mode", the flux corresponding to a distribution of source neutrons is determined.

By using the modified version described in [Ref 14] it is possible to use TWODIM for dynamic calculations as well. In the RECRIT code the subroutine TWODIM is used in cylindrical geometry only, but in both the "k_{eff}-mode" and the "source-mode".

During the time from the beginning of the accident to the point, when recriticality is reached, alternating calls are made to the subroutine TWODIM. A call to TWODIM with the "k_{eff}-mode" is made to secure that the reactor is still subcritical, and a call to TWODIM with the "source-mode" gives the amplification of the neutron source (the spontaneous fissioning of U^{238}) for calculation of the fission power.

The updated power distribution, the sum of the decay heat and the fission power, are transferred to the thermo-hydraulic subroutine as heat sources.

When recriticality is achieved, the calculation proceeds in time steps of the length of the prompt neutron lifetime, i.e. of the order of 0.5 ms.

As described in [Ref 14], the fissions caused by thermal neutrons are assumed to take place at a time equal to the prompt neutron lifetime since these thermal neutrons were borne as fast neutrons in fissions. Thus, the TWODIM (static) subroutine may be used in a dynamic mode by a few simple manipulations:

The subroutine TWODIM is called in its "source-mode" (with 2 neutron energy groups) and the sources in the fast and thermal groups are given by Equations (6) and (7) respectively:

$$Q_1 = \phi_2 \cdot \left(\nu \Sigma_f\right)_2 + \sum_i \lambda_i C_i \tag{6}$$

$$Q_2 = 0 \tag{7}$$

As the thermal fissions are treated as source neutrons we must set $(\nu\Sigma_f)_2 = 0$ in the fast diffusion equation. (This will also secure the necessary formal "subcriticality").

In the Equation (6) for Q_1, the term $\sum_i \lambda_i C_i$ is the contribution from the delayed neutrons. Correspondingly, these delayed neutrons must be subtracted from the prompt neutron production, by modifying the $(\nu\Sigma_f)_i$ (in both fast and thermal groups) and thus we must write:

$$\left(\nu \Sigma_f\right)_i = \left(\nu \Sigma_f\right)_i \cdot (1 - \beta) \tag{8}$$

where β is the fraction of delayed neutrons.

After calculation of the new neutron flux, the updated concentrations of the delayed neutron precursors, C_i, are calculated from Equation (9):

$$C_i = C_i e^{-\lambda_i dt} + \beta_i \cdot \sum_g \phi_g \cdot \left(\nu \Sigma_f\right)_g \cdot \left(1 - e^{-\lambda_i dt}\right) / \lambda_i \tag{9}$$

The power distribution is calculated and transmitted to the thermo-hydraulics subroutines.

5.2 Thermohydraulic models in RECRIT

5.2.1 Nodalization

The RECRIT core is described by a two-dimensional model, cylindrically symmetric geometry by allowing typically 10-100 axial nodes and 3-20 radial rings, giving 30-2000 mesh points. The hydraulic nodalization in the core could be as dense as in the neutronics model, but the flexibility of the model allows also a smaller number of hydraulic nodes in both the axial and the radial directions. The radial rings may have different flow areas , in the axial direction the nodes are of equal length. The hydrodynamic nodes, junctions and heat structures are generated automatically from the input parameters defined for the major process components. The hydrodynamic sections of a BWR in RECRIT is shown in Figure 2.

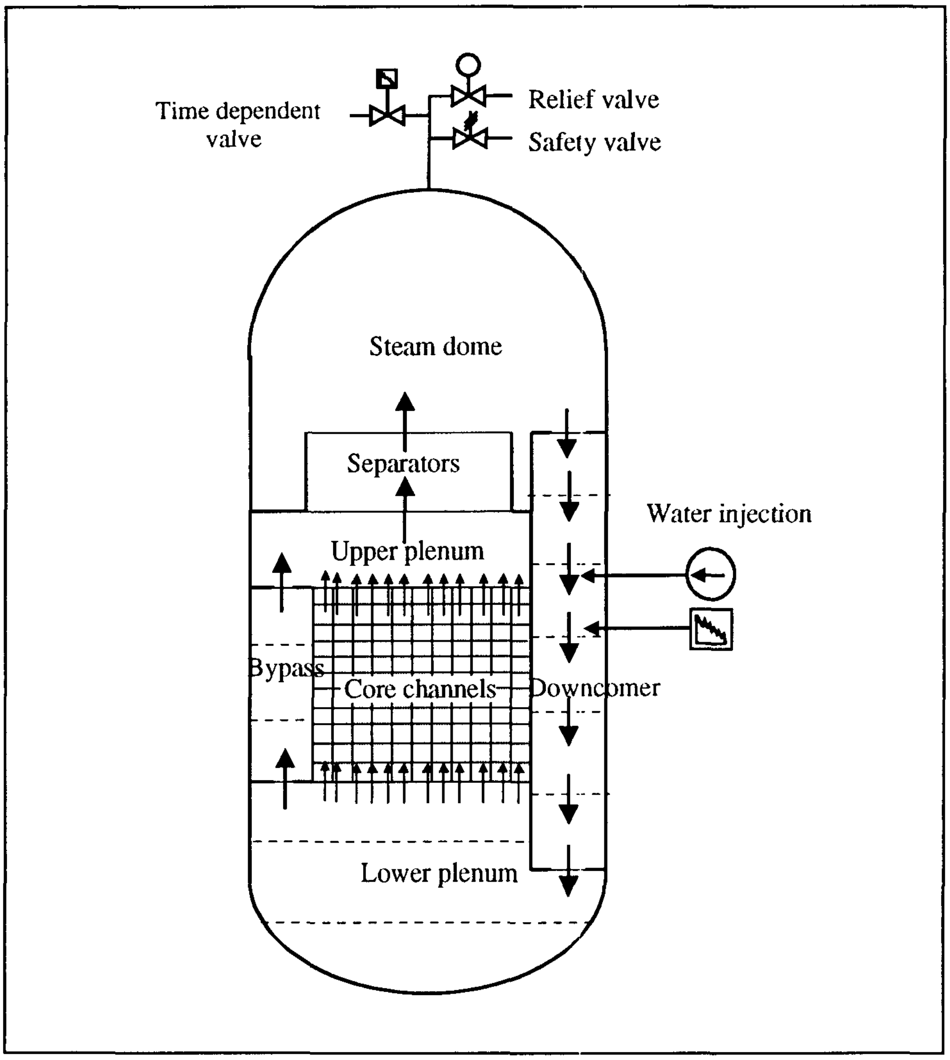


Figure 2. *Hydrodynamic nodalisation in RECRIT code.*

The sections outside the core are defined as one-dimensional components. The user defines integral data for bypass, upper plenum, steam separator, lower plenum, downcomer and steam dome. The model assumes the connections between the different sections according to the BWR plant geometry and the user has to define the additional friction and area contractions for these connections between the lower plenum and the core, the lower plenum and the bypass, the core and the upper plenum, the bypass and the upper plenum, the upper plenum and the steam separator, the steam separator and the steam dome, the lower plenum and the downcomer and the downcomer and the steam separator. The initialization system automatically creates the calculational level nodalization from the user given number of nodes for different sections. The calculation level components include the nodes, the junctions connecting nodes and heat structures with a solid wall connected to the node.

The node geometry is defined by volume, flow area, length, elevation and equivalent diameter. Node thermo-hydraulic state is determined by pressure, void fraction, (steam mass, water mass), water enthalpy, steam enthalpy, concentration of noncondensibles (optional) and concentration of boric acid (optional).

Junction geometry is defined by flow area, elevation and friction coefficient. Thermo-hydraulics in a junction covers liquid mass flow, steam mass flow and mixture volume flow, (liquid velocity, steam velocity).

The fuel heat structure is defined by cladding area, pellet area, gas gap thickness, cladding volume and pellet volume. Fuel cans and grid spacers are not included in the model. The needed fuel material properties are heat capacities, conductivity between the fuel and the cladding and conductivity in the cladding. The fuel cladding temperature and the average fuel temperature (volumetrically homogeneous heat generation assumed) are solved.

The reactor vessel internal steel structures and the reactor vessel walls are not modelled.

Water injection by a pump is defined by a quadratic polynomial curve. Additionally the water injection may be defined by using a time dependent table. Steam discharge from the reactor coolant system can take place via relief valve, safety valve or time dependent valve.

5.2.2 Conservation Equations

The conservation equations for the thermohydraulic solution are written for liquid and steam mass conservation, mixture momentum conservation and liquid and steam energy conservation. The momentum equation is solved by the integral momentum principle. Additionally the conservation equations for the boric acid concentration in the liquid and the concentration of the noncondensible gases in the steam can be integrated.

- The phase separation is solved by a drift flux model, integral correlation being valid for single phase flow, bubble flow, annular flow, droplet dispersed flow and single phase steam.
- For the pressure loss wall friction and singular pressure loss in contractions are accounted for. The two-phase multiplier is defined by the homogeneous model.
- The wall heat transfer correlations cover the whole heat transfer regime map from single phase convection into subcooled liquid to the single phase convection into superheated steam. For the quench front movement a mathematical model based on the analytical solution of the heat conduction controlled rewetting is applied. In high temperatures the radiation absorption into liquid as a function of void fraction is calculated.
- The interfacial heat transfer between liquid and steam comprises condensation, when the two-phase mixture liquid is subcooled, flashing, when liquid is superheated and energy transfer from superheated steam to droplets. The condensation on subcooled walls is calculated as well.

Equations (10)-(13) are the basic conservation laws applied in the RECRIT thermal hydraulic model. Equation (10) defines the mixture mass conservation. The equation is used in four steps: First the system pressure in the vessel is calculated based on the mass balance changes and compressibility of the mixture. After the integral momentum solution the volumetric flow distribution is defined around the circuit. After defining the phase separation by the drift flux model, the phasial liquid and steam masses are determined. Finally, the mass conservation is integrated for the individual nodes.

The momentum Equation (11) is applied in two steps: The circulation rate is solved from the integral momentum integrated over the loop. An integral momentum equation is solved for the core inlet into each radial channel and into bypass. After the volumetric flow rate in these junctions is fixed, the volumetric flow distribution around the circuit may be determined. After the local mass flow rates have been determined, the momentum equation is used for calculation of pressure distribution around the circuit.

The energy Equations (12) and (13) for liquid and for steam, respectively, are integrated after the local mass flow rates are determined. For the energy equation the donor cell discretisation is applied and to avoid matrix inversions, integration is performed from node to node following the nominal flow direction.

$$\frac{\partial}{\partial t} A\rho_m + \frac{\partial}{\partial z} w_m = S_\ell + S_g \tag{10}$$

$$\frac{\partial}{\partial t} w_m + \frac{\partial}{\partial z}\frac{1}{A}\frac{w_m^2}{\rho_m} = -A\frac{\partial p}{\partial z} - A\frac{\partial p_f}{\partial z} - A\rho_m g\cos\theta \tag{11}$$

$$\frac{\partial}{\partial t} A\rho_\ell h_\ell + \frac{\partial}{\partial z} w_\ell h_\ell - A\frac{\partial p}{\partial t} = h_{\ell,s} S_\ell + Q_\ell \tag{12}$$

$$\frac{\partial}{\partial t} A\rho_g h_g + \frac{\partial}{\partial z} w_g h_g - A\frac{\partial p}{\partial t} = h_{g,s} S_g + Q_g \tag{13}$$

The phase separation is based on the drift flux formulation by Zuber & Findlay and the model can be expressed as an equation for steam velocity u_g as a function of the average mixture velocity as:

$$u_g = C_0 u_m + \tilde{v}_{Gj} \tag{14}$$

$C_0 = f(\alpha, D_e) =$ distribution parameter (1.10 ... 1.20)
$\tilde{v}_{Gj} = f(p, \alpha, D_e)$ = drift velocity (0.2 ... 1.0 m / s)

Expressions for C_0 and $\tilde{v}_{Gj}$ for steam-water misture in high pressure are given as:

$C_0 = 1.13$ is the default variation range for Zuber drift flux (1.10 ... 1.20) for other flow patterns

$$\tilde{v}_{Gj} = 1.41\left[\frac{\sigma \cdot g \cdot \left(\rho_\ell - \rho_g\right)}{\rho_\ell^2}\right]^{0.25} \cdot F(D_e) \tag{15}$$

$F(D_e) = 1.0 \ldots 5.0$ depending on geometry, default for Zuber drift flux velocity $F(D_e) = 1$

5.2.3 Heat Transfer From Fuel Structure to Coolant

The RECRIT thermohydraulics model takes into account single phase convection into liquid and steam and boiling of subcooled and saturated water. The quenching front movement and the cooling effect is described by an analytical correlation, where the effective heat transfer in the wetted part and the Leidenfrost temperature are used as fitting parameters. The post-dryout heat transfer is calculated by a droplet-dispersed heat transfer model. Radiation absorption into droplets is accounted for. Critical heat flux is calculated by a dryout correlation. Precursory cooling effect by transient boiling is modeled by a simple heat transfer decay law between the critical heat flux and the minimum film boiling points. The liquid continuous part is assumed to end at the quench front. The postulated heat transfer regimes during reflooding are depicted in Figure 3.

When the wall temperature is below the saturation temperature, heat transfer takes place by forced convection to sub-cooled liquid. In the case of laminar flow ($Re < 2000$) a constant Nusselt number is assumed:

$$\mathrm{Nu} = 4.36 \tag{16}$$

If the flow is turbulent ($Re > 2000$) the Dittus-Boelter correlation is used:

$$\mathrm{Re}_\ell = G \cdot (1 - x) \cdot D_h / \mu_\ell \tag{17}$$

$$\mathrm{Nu}_\ell = 0.023 \cdot \mathrm{Re}_\ell^{0.8} \cdot \mathrm{Pr}_\ell^{0.4} \tag{18}$$

Chen's correlation is used to calculate heat transfer coefficient in the nucleate boiling regime:

$$h = h_{NB} + h_{conv} \tag{19}$$

$$h_{NB} = 0.00122 \frac{K_l^{0.79} C_{p_l}^{0.45} \rho_l^{0.49} \Delta T^{0.24} \Delta p^{0.75}}{\sigma^{0.5} \mu_l^{0.29} h_{lg}^{0.24} \rho_g^{0.24}} \cdot S \tag{20}$$

$$h_{conv} = Nu_{\ell} \cdot k_{\ell} / D_h \cdot F \tag{21}$$

$$q_{NB} = h_{NB} \cdot \left(T_w - T_{sat}\right) \tag{22}$$

$$q_{conv} = h_{conv} \cdot \left(T_w - T_{\ell}\right) \tag{23}$$

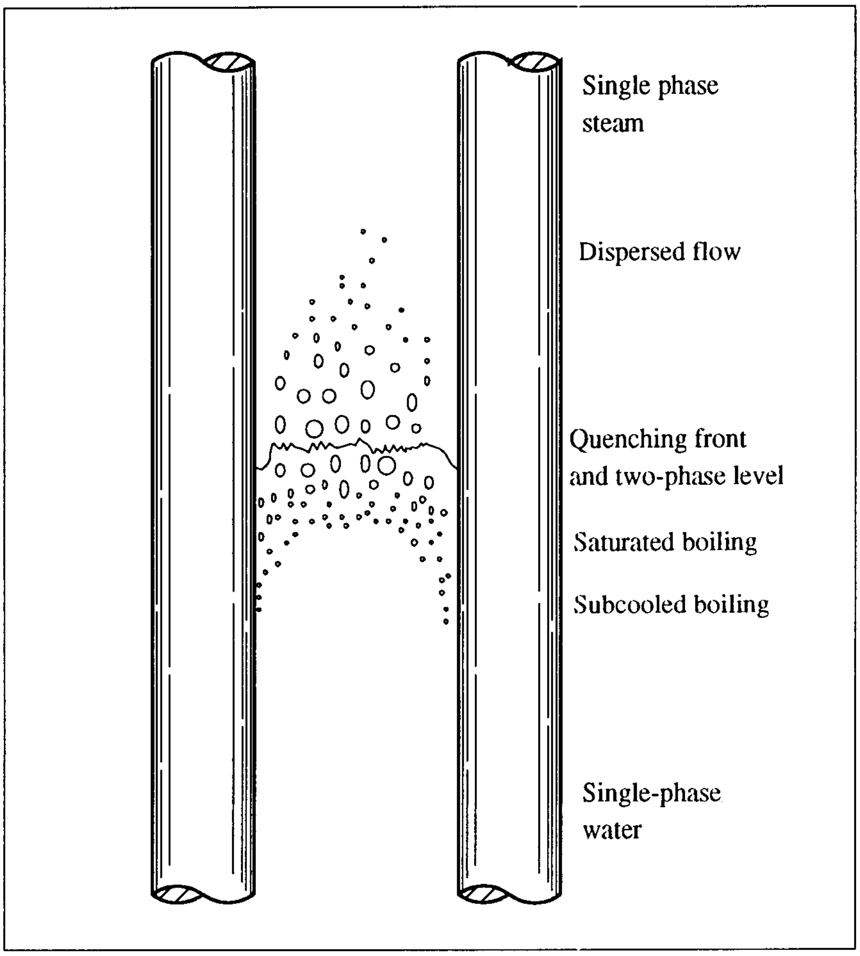


Figure 3. *Typical heat transfer regimes of reflooding. No inverted annular film boiling region assumed in RECRIT.*

The heat transfer coefficient decreases in the quenching front by more than 2 orders of magnitude from that of nucleate boiling to that of film boiling. An analytical rewetting correlation for Peclet's number is used (Eq. (26)). The correlation is valid for conduction controlled rewetting in cylindrical geometry and is a combination of the thin and the thick wall rewetting velocity. The Peclet's number is defined by Eq. (27), the Biot's number by Eq. (24) and the dimensionless temperature by Eq. (25).

$$Bi = h_{fr} \cdot s_w / k_w \tag{24}$$

$$\theta = \left[\frac{(T_w - T_s)(T_w - T_0)}{(T_0 - T_s)^2} \right]^{1/2} \tag{25}$$

$$Pe = \left\{ \left(\sqrt{Bi}/\theta\right)^3 + 2^{-3/4\sqrt{\pi}} \left(Bi/\theta^{1+\sqrt{\pi}/2}\right)^3 \right\}^{1/3} \tag{26}$$

$$V_{fr} = \frac{Pe \cdot k_w}{\rho_w \cdot c_w \cdot S_w} \tag{27}$$

The correlation expresses the rewetting front velocity when zero heat transfer in the dry region and heat transfer coefficient h_{fr} in the wetted region are assumed. The quench front velocity is depicted in Figure 4 as a function of temperature difference *(T_{wall} - T_{sat})* with various Leidenfrost temperatures *(T_{LEID}=T_0)* and heat transfer coefficients *(HTC=h_{fr})*.

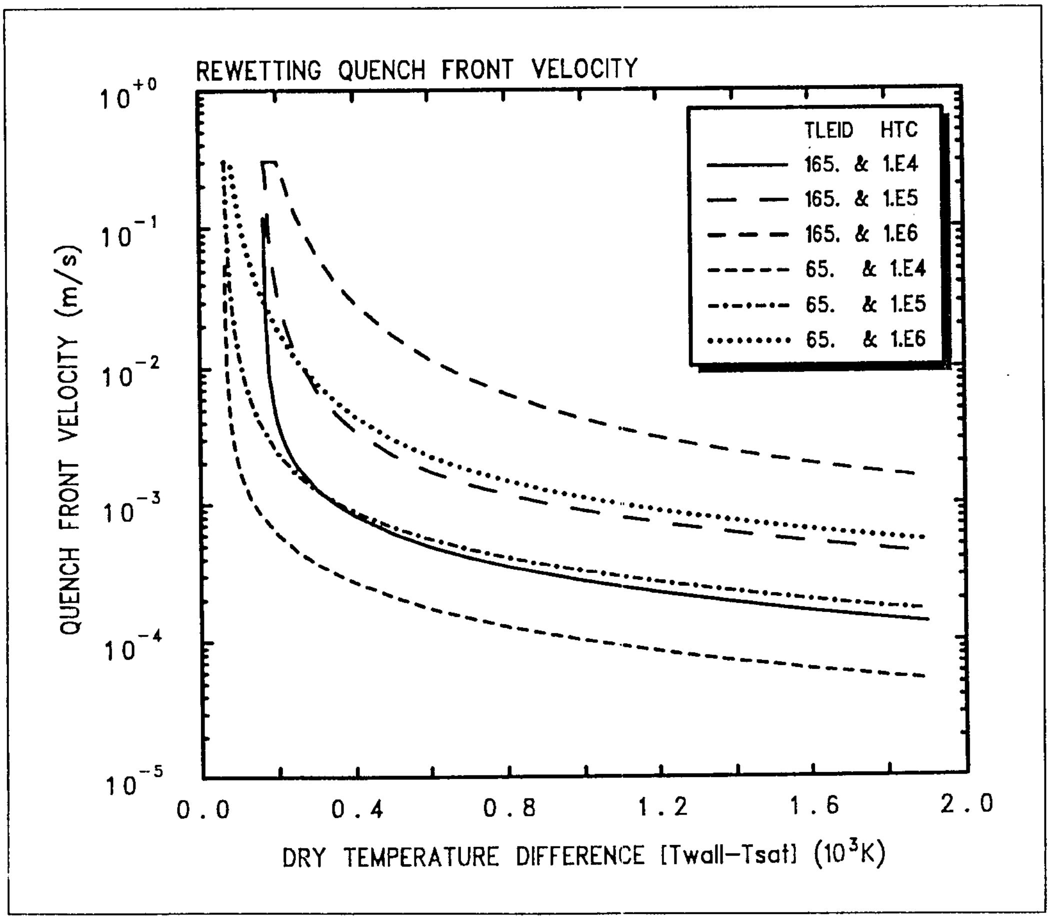


Figure 4. *Typical rewetting quench front velocity as a function of temperature difference (T_{wall} -T_{sat}) with various Leidenfrost temperature (T_{LEID}) and effective quenching heat transfer coefficient (HTC) values.*

Radiation heat transfer is calculated assuming only absorption into liquid:

$$h_R = \sigma \cdot \varepsilon \cdot \frac{T_w^4 - T_s^4}{T_w - T_s} \tag{28}$$

$$\varepsilon = \frac{1}{\frac{1}{\varepsilon_w} + \frac{1}{\varepsilon_l} - 1} \tag{29}$$

$$q_{rad} = h_R \cdot (T_w - T_{sat}) \tag{30}$$

Forced convection from the fuel wall to steam and interfacial heat transfer between the droplets and steam is calculated in the liquid dispersion and single phase steam regions. Forced convection heat transfer coefficient in the steam region is analogical to Eq. (21):

$$h_{conv} = Nu_g \cdot k_g / D_h \tag{31}$$

$$q_{conv} = h_{conv} \cdot (T_w - T_g) \tag{32}$$

The radiation heat transfer is solved between two-dimensional core heat structure elements by assuming the effective radiation heat transfer area to be equal to the horizontal flow area between radial rings in vertical direction and the vertical walls between the radial rings in the horizonatal direction. The radiative heat flux per unit radiation heat transfer area is expressed by

$$q_R = \sigma \cdot \varepsilon \cdot (T_{w1}^4 - T_{w2}^4) \tag{33}$$

The interphasial heat transfer as flashing, condensation and droplet evaporation are formulated with convection analogy with temperature differences for liquid superheat, liquid subcooling and steam superheating, respectively.

5.2.4 Heat Generation

The fission power generation and the decay power generation are calculated in the neutronics part of RECRIT.

A cladding oxidation model is optional and activated by a user input parameter. The zirconium oxidation is calculated only in the cladding and the kinetics is assumed to obey a parabolic rate law with Urbanic-Heidrick correlation:

$$[W(t+\Delta t)]^2 = [W(t)]^2 + K(T_{clad})\cdot \Delta t \qquad (34)$$

where W = mass of Zr oxidized per unit area
t = time
Δt = time step
T_{clad} = cladding temperature

$$K(T_{clad}) = \begin{cases} 29.6\cdot e^{-16820/T_{clad}} & ; \quad T_{clad} \le 1853K \\ 87.9\cdot e^{-16610/T_{clad}} & ; \quad T_{clad} \ge 1853K \end{cases} \qquad (35)$$

Oxidation stops if:

1. Cladding temperature > 2973 K (melting point of ZrO_2) or
2. No steam is available

5.2.5 Control rod melting

The model predicting the control rod melting is based on the assumption of a thermal balance between control rods and fuel rods. As soon as the fuel temperature exceeds control rod melting temperature (1473 K), a part of the heat generated in fuel is used for the control rod melting process. In reality, the fuel temperature is about 50 - 100 K higher than the control rod temperature close to the melting point of the control rods. For RECRIT applications the model simplification in material temperatures may be considered acceptable.

With the principle of thermal equilibrium an artificial time constant is used for converting the excess enthalpy in the fuel to the heat flux for control rod melting

$$Q_{melt} = V_{rod}\cdot \rho_{cr}\, H_{melt}\,(T_{fuel} - \mathrm{T}_{melt})\ /\ \Delta \mathrm{t}_{melt} \qquad (36)$$

where the right side terms are the control rod volume related to the heat structure, material density of control rods (500 kg/m^3), melting heat ($3.0\cdot 10^5$ J/kg) and relaxation time for melting.

5.2.6 Auxiliary system models

The code models the operation of different valves from the steam line into the containment in the following way:

- Relief valve opening as a function of pressure
- Safety valve opening by a pressure hysteresis characteristics

- Time dependent valve position as a function time
- Injection pump feed curve as a function of pressure
- Injection feed rate as a function of time

Valve mass flow rate is calculated from a model based on the quadratic pressure loss in the valve as a function of mixture mass flow rate. The friction coefficient is determined from the design flow rate at the nominal pressure. The formula for the valve mass flow rate is given by the Equations (37) and (38):

$$w_m = \sqrt{\frac{\Delta p \cdot \rho_m}{X_k}} \tag{37}$$

$$X_k = x_k / A_v^2 \tag{38}$$

where X_k is pressure loss coefficient defining the mass flow as a function of the pressure loss and the mixture density and including the valve flow area.

The mass injection pumps are modelled assuming the pump head pressure to be a second-order polynomial of mass flow rate *w*:

$$\Delta p = a_0 + a_1 w + a_2 w^2 \tag{39}$$

where coefficients a_0, a_1 and a_2 are calculated by the input proceesor from the three given data pairs for the injection mass flow rate and pump head.

5.3 Reference plant input

TVO plant was chosen to be the reference plant for the first calculations with RECRIT. The key input quantities for the RECRIT TVO model are indicated in Table 1.

Tables of two-group diffusion cross sections, i.e. diffusion coefficients, diffusion matrix, $\nu\Sigma_f$, Σ_f, and fission spectrum are given for the following ranges of fuel and water temperatures, coolant void fraction and control rod status:

- Fuel temperatures: 373 K, 1272 K, 2272 K
- Liquid temperatures: 293 K, 372 K, 433 K
- Void fractions: 0.0, 0.4, 0.7, 0.99
- Control rod status: rodded/unrodded

The diffusion cross section values are determined as functions of the local fuel temperature, liquid temperature, void fraction and control rod status (0.0=unrodded, 1.0=rodded) using a four dimensional interpolation procedure. The procedure extrapolates the fuel temperature if it is above 2272 K.

The data set calculated by the CASMO code included data point variations for five burn-up values and three void fraction histories during the plant operation:
- Average fuel burn-up values 0, 5, 10, 20 and 30 MWD/kgU

- Void fraction history during the reactor operation 0.0. (fuel pin located at bottom), 0.4 (fuel pin located in the middle), 0.75 (fuel pin located at top).

The burnup values 0 and 5 MWD/kgU are possible only for the new core. The burnup value of 10 MWD/kgU with the void history 0.4 was selected for a fuel representative for beginning-of-cycle (BOC) conditions of the old core. The burnup value 20 MWD/kgU with the void history 0.4 was selected as a representative case for middle-of-cycle (MOC) conditions. The burnup value of 30 MWD with void history 0.75 was selected for a representative case for end-of-cycle (EOC) conditions.

Table 1. *Key input data for RECRIT calculations, (varied rates in paranthesis).*

INPUT ITEM	VALUE
Number of core nodes : neutronics calculation, axial radial thermohydraulic, axial radial	 40 (20) 10 40 (5, 10, 20) 10 (5)
Decay heat power density	ANSI curve
Start of reflooding or or	at 4010 s from beginning of the accident at fuel maximum temperature of 2060 K, at 30 % control rods left
Start of ADS depressurization	1700 s, initial capacity 550 kg/s
Reflooding rate [kg/s]	22.5 , 45.0, 160.0 and 540.0
Neutronics data set	BOC fuel at present (fresh, MOC,EOC)
Prompt neutron lifetime	$0.5 \cdot 10^{-3}$ s
Number of fuel rods	31 500
Number of control volumes:	Bypass: 4 Upper plenum: 3 Steam separators: 3 Lower plenum: 6 Downcomer: 10

6 SIMULATE-3K Code

SIMULATE-3K is a transient version of the steady-state core analysis computer code SIMULATE-3. The latter is part of the Studsvik Core Management System (CMS) code package for BWR and PWR applications which consists of two principal codes; CASMO and SIMULATE. The relation between the different codes is illustrated by Figure 5. CASMO-4 is the latest version of the CASMO transport theory assembly, spectrum and depletion code, which is widely used by utilities in many countries. SIMULATE-3 is a recently developed three-dimensional two-group reactor analysis code used for incore fuel management studies, core design calculations, and calculations of safety parameters. One of the original goals in the development of SIMULATE was to introduce an accurate advanced nodal method which required no normalisation to fine-mesh calculations or to measured data

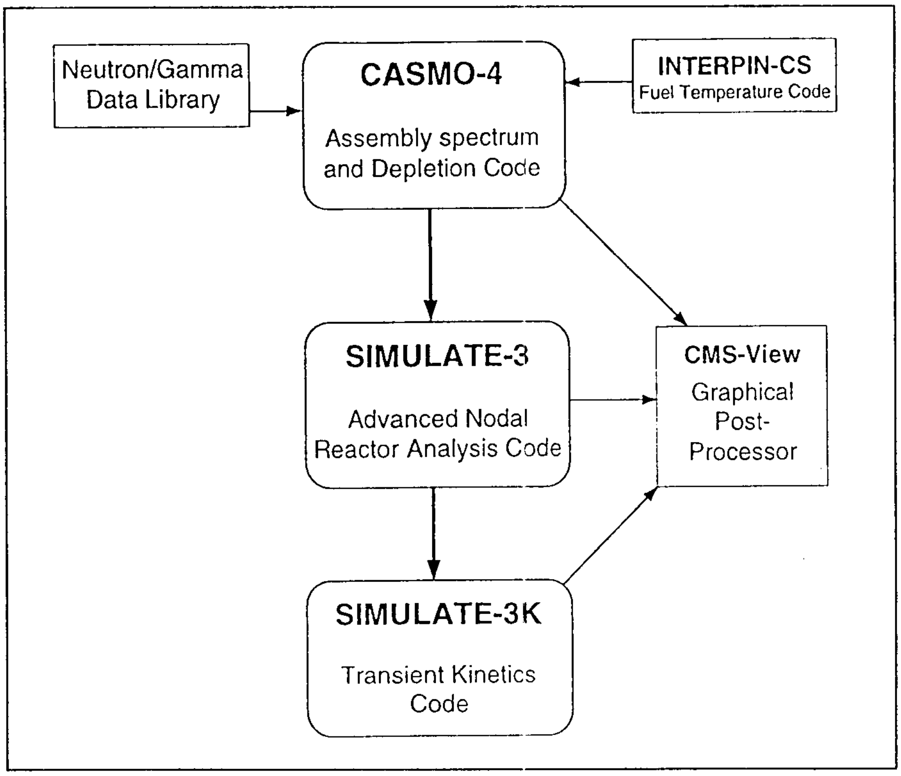


Figure 5. *Studsvik Core Management System (CMS).*

With SIMULATE-3K (S-3K) Studsvik has developed and incorporated transient analysis capabilities into SIMULATE-3 (S-3). The work has resulted in the addition of a neutron kinetics model and a transient pin model coupled to a hydraulic channel model. This has required incorporation of extra numerical features to ensure the stability and accuracy of the numerical solution.

S-3K has been successfully applied on and validated against various full-plant transients such as control rod drop and ejection events, and BWR stability measurements. However, the code is still under development in order to extend its range of applicability. To be able to simulate the extreme conditions prevailing under reflooding and fast recooling transients additional development and upgrading was needed, especially of the thermal-hydraulic models in the code. The code development for RAK-2 applications has been and is being made by Studsvik of America (SOA) in Idaho Falls, USA. There have been several diffuculties with the present 3-equation (for the thermal-hydraulic solution) code version to apply it for full plant simulations within the time schedule of RAK-2.1. A parallel development at SOA of a 5-equation version seem to be more promising, and that version will be the basis for continued recriticality studies, which will continue within the EU project, SARA.

6.1 Neutronics model in SIMULATE-3K

The base neutronics model in S-3K is essentially the same as in the steady-state version S-3, which comprises the following three-dimensional neutron physics models:

- Two-group nodal diffusion model
- Assembly homogenisation model
- Baffle/reflector model
- Cross section/depletion model
- Pin power reconstruction model

Two-group neutron diffusion theory models in advanced nodal codes are generally accepted to give satisfactory accuracy for predictions of the spatial behaviour of nuclear reactor cores. The nodal model "QPANDA" used in SIMULATE-3 implies a substantial improvement in accuracy over most other 3-D homogenised nodal models, not far from that of a detailed Fine Mesh Diffusion theory codes, such as "PDQ", but is much faster.

The equations which constitute the QPANDA model are formally derived by subdividing the spatial domain of the reactor into a set of rectangular parallelepipeds, referred to as nodes. Each node will typically represent a full assembly, or a quarter assembly in the radial plane and a 15-30 cm axial region of an assembly. Intra-nodal flux is calculated involving node-averaged fluxes and transverse leakages by polynomial representations for each two groups and treating two nodes simultaneously.

The assembly homogenisation calculations generate appropriate cell-averaged values of reactor physics parameters. These are based on CASMO results for single or a symmetry fraction of one assembly with reflective boundary conditions which are prepared for every unique combination of fuel pin enrichment/burnable absorber loading and layout. CASMO state points comprise multi-group (40, or 70) pin-by-pin fluxes, reaction rates, and isotopic number densities. The assembly-averaged parameters in S-3 are derived by integrating reaction rates and fluxes over the volume of the assembly and summing over all the CASMO groups contained in each S-3 group for every assembly type. However, homogenised fluxes within each assembly will probably give rise to discontinuity in the flux between assembly interfaces, which is in conflict with the assumption of continuity in nodal methods. S-3 therefore introduces "Assembly Discontinuity Factors, ADFs" which allows the homogeneous flux to be discontinuous at a nodal interface. These are calculated as the ratio of CASMO surface flux to the CASMO assembly averaged flux.

Baffle/reflector effects, which in many other codes are treated by user-adjusted albedos, are in CMS/S-3K taken care of directly by the CASMO-4 reflector model. For the calculation of the net leakage of neutrons through reflector regions special discontinuity factors for these components are employed.

The cross-section depletion model in S-3 is based on CASMO cross section data generated for each type of assembly in the core. Multi-dimensional, interpolation, table sets for the complicated relationship between various parameters and the cross-sections are generated by the interface code TABLES-3. Such parameters are e.g. node-wise isotopic number densities, instantaneous moderator and fuel temperatures, moderator density (void), boron concentration, control rods, etc. CASMO assembly calculations have therefore to be made for the necessary parameter range valid in the S-3 simulation. Depletion calculations are made in CASMO for core average conditions. The nuclide concentration also depends on the conditions that might have changed during the core depletion, e.g. variation in water density and the insertion of control rods. This history dependence is treated by a special method in S-3 utilising weight factors, which takes into account the instantaneous values and the fact that cross-sections are more sensitive to recent conditions than to those of the far distant past.

The S-3 **pin power reconstruction** method is based on the assumption of separability of global flux (homogeneous intranodal flux) and local flux shapes (heterogeneous form functions). If only spatially homogenised cross-sections were used, as in normal nodal models, only smoothly-varying flux distributions within each assembly could be calculated. With a separability approximation detailed pin-by-pin distributions of heterogeneous fluxes and power within an assembly can be approximated by the product of a global homogenised distribution and a local heterogeneous form function. The S-3 method for pin power reconstruction is based on single assembly form functions, so that accurate pin powers can be obtained without the use of multi-assembly spectrum/depletion calculations.

In SIMULATE-3K the 3-D transient neutronics model uses the same high order nodal model as in the steady-state S-3 version, coupled with a fully implicit time integration scheme. The neutron spatial model in S-3K extends the QPANDA two-group nodal spatial model. The fission source due to delayed neutrons accounts for the spatial dependence of the delayed neutron precursor concentrations which are integrated assuming linear, or exponential variation in nodal power over a time step. Control rod motions can be simulated either for a single rod or groups (banks) of control rods. There is also an automatic control rod scram module and rod velocities can be specified by input.

6.2 Thermal-Hydraulic models in SIMULATE-3K

The coupling between reactor power, coolant density and fuel temperatures is simulated by the neutronics/thermal-hydraulics iterative process in SIMULATE. In the steady-state calculations the reactor power distribution can be considered known, and the coupled problem is reduced temporarily to a problem of determining the coolant density and fuel temperature distributions for a fixed power distribution. For transient conditions modelled in S-3K a number of time-dependent phenomena have to be taken into account, such as transient heat transmission with stored heat, heat capacity in materials, heat conduction in fuel pins with heat resistance, etc.

As well as for the neutron physics, the Thermal-Hydraulic (T-H) part of the transient S-3K version contains most of the models from the steady-state code. The models for BWRs are there more comprehensive than those for PWRs, because of the complex core geometry and the two-phase flow regimes present in a BWR. The T-H modelling comprises the following features:

- Node-centered quality, void fraction and density determined by nodal heat addition and local mass flow.
- Two-dimensional assembly active flow distribution, either fixed by input, or calculated by the code.
- Heat balance calculation. This can be turned on, or off by input.

Figure 6 from [Ref 7] illustrates the modelling of a BWR core for the flow balance calculation, which is based on loss coefficients and geometrical data specified in the input. The T-H modelling is not strictly three-dimensional. Each fuel assembly and any water rod inside the channel box and the bypass outside the boxes are treated as parallel flow channels. The whole bypass area is treated as one single channel and no cross flow is considered.

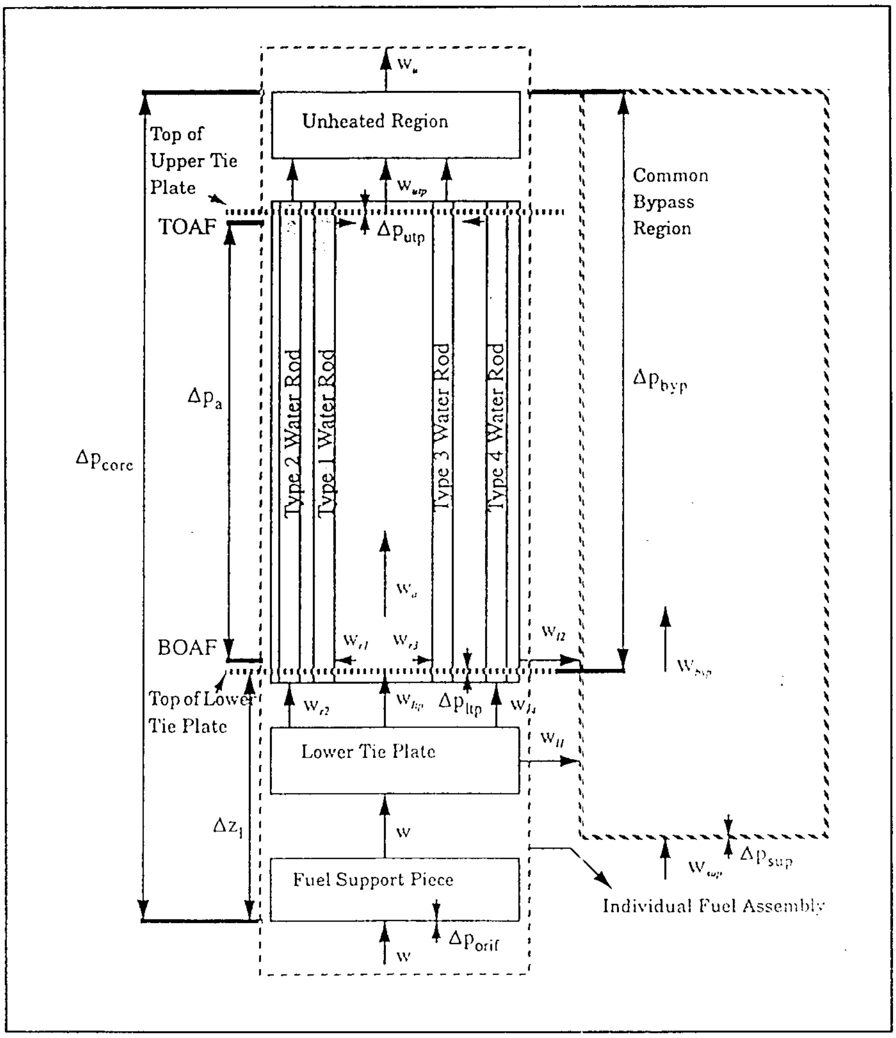


Figure 6. *Individual assembly and common bypass flows and pressure drops in SIMULATE3.*

The pressure drop calculation is coupled to the heat balance calculation which takes into account the energy generated in the fuel and deposited in the coolant by convective and conductive heat flow from fuel pins to the water and steam in the channels, and through the box walls into the bypass water. Radiative heat transfer is not taken into account. Boiling in channels and in bypass is considered, but not in the water rods, which are assumed to contain only liquid water. A fraction, specified by input, of the nuclear power can be deposited directly into the bypass water. The fraction of power for heat-up of solid compounds is taken into account to determine the temperature profiles there, but in the steady-state solution there is no net power accumulation in these materials. Figure 7 shows a schematic of major heat balance control volumes.

The EPRI void model is used to calculate in-channel void fractions for both subcooled and bulk boiling regimes. This incorporates a modified Zuber-Findlay relationship with a drift-flux model. For the bypass region, however, a simple homogeneous equilibrium void/quality model is used which gives no subcooled void. Pressure drop calculation includes effects of gravitation, acceleration, friction and local pressure drops due to area changes. Pressure and slip dependent two-phase multipliers are used. The two-phase calculations have been benchmarked against various test data, among others from the FRIGG tests.

In the transient S-3K code various transient boundary conditions can be specified by input, such as feedwater temperature variations, flow transients and control rod movements. Under transient conditions heat transfer calculations have to consider the problem of unsteady-state heat generation and heat conduction in solid bodies and heat transport to coolant with respect to thermal diffusivity and its variation with temperature. In S-3K the spatial pin conduction equation is solved on a multi-region mesh in cylindrical coordinates in the radial direction only, neglecting axial and azimuthal conduction.

6.3 Oskarshamn 3 input

It was originally planned to perform the SIMULATE-3K calculations for Forsmark 3 BWR, since that plant was modelled for the reflooding calculations in the previous task of project RAK-2.1. It was also thought that existing SIMULATE-3 models for the similar, smaller Forsmark units, F-1 and F-2 could be obtained and modified, which showed not to be the case. A preliminary F-3 input for the CMS codes was then set up and some test calculations started. However, in order to adapt the input model to realistic F-3 plant data it was necessary to have the preliminary values verified and to get access to information about real fuel cycle data at Forsmark, which turned out to be a rather tedious work.

However, SIMULATE -3 runs had recently been made also for the twin plant, Oskarshamn 3. By courtesy of the utility, OKG AB, input data for the first four cycles were made available to us. The primary systems and the in-vessel designs are identical in F-3 and O-3. It was then decided to to perform the recriticality simulations with O-3 instead of F-3.

In the SCDAP/RELAP model all fuel bundles were assumed to be of one single assembly type, viz. SVEA-64. This is a hypothetical assumption since there are a number of different assembly types in most real core loadings, especially in later cycles which often include some new fuel types for testing. The first two cycles are, however, not representative either, having still many unenriched assemblies from the initial cycle. Cycle 3 in O-3, however, was considered to be a good choice as a base for present recriticality studies. It is closer to an equilibrium state and has only two main fuel types, as shown on the fuel map in Figure 8. Cycle 3 comprises ABB Atom fuel only, with 516 8x8 rod bundles (with one water rod) plus 176 SVEA-64 bundles (without, or having one water rod), the remaining 8 assemblies are of two similar SVEA-100 types.

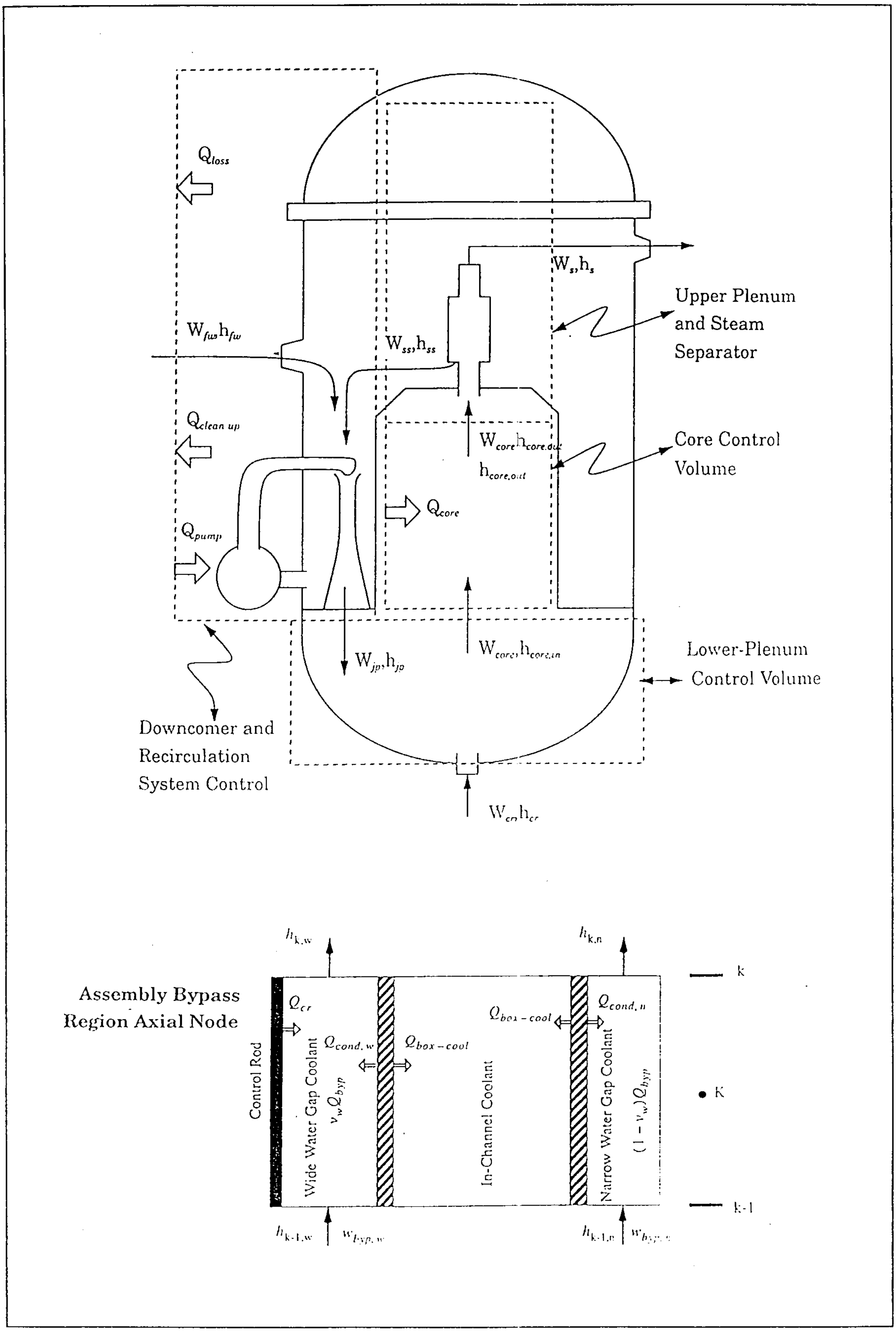
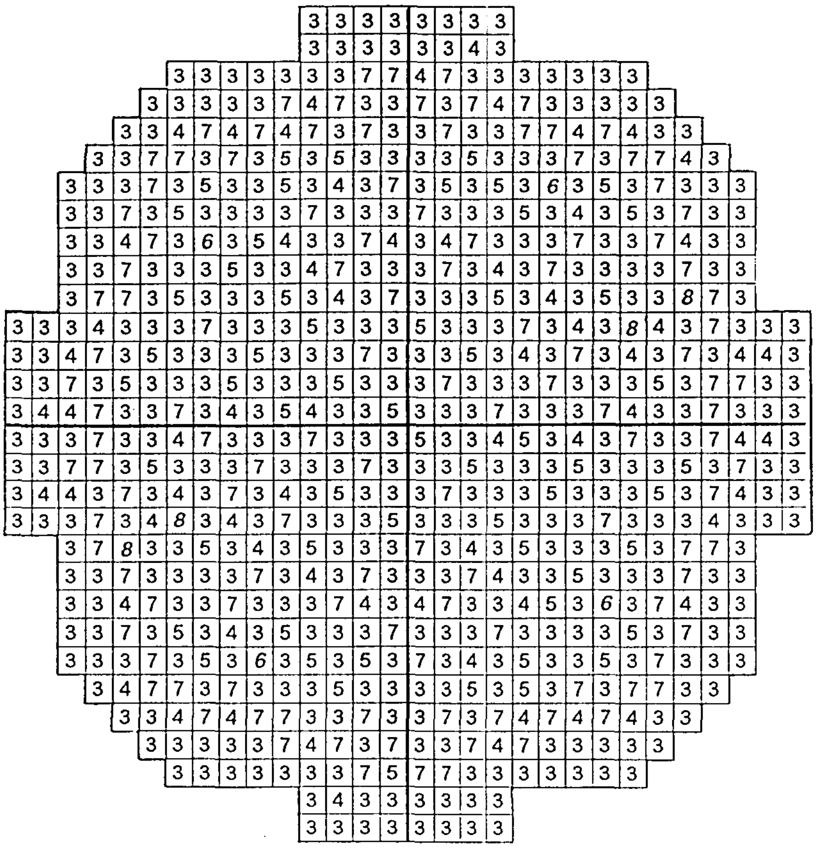


Figure 7. *Heat balance control volumes in SIMULATE-3 and assembly bypass region axial node heat flow.*



Fuel assembly types:

No.	Type	No. of assemblies	No. (from cycle No.)	No. of heated rods
3	ABB81 (8* 8) with Gd	448	442(2)+6(1)	63
4	-"- no Gd	68	68(2)	63
5	SVEA-63 with Gd	56	4(2)+52(3)	63
6	SVEA-100 with Gd	4	4(2)	100
7	SVEA-64 with Gd	120	128(3)	64
8	SVEA-100 with Gd	4	4(3)	100

Figure 8. *Assembly loading pattern in Oskarshamn 3 core, cycle 3.*

The input obtained from OKG AB was set up for steady-state core follow calculations and included some details, like in-core measurement data, which were stripped off to facilitate the recriticality calculations. For the same reason some modifications had to be done, since all steady-state input data are either not needed or not accepted by the transient S-3K code. The eight SVEA-100 assemblies with 375 cm heated length were replaced by SVEA-64 types so that a consistent 368 cm active fuel height was retained throughout the core. With this modification a quarter core symmetry was obtained, as well. All input data for the core geometry, fuel loading and operating conditions are specified in the S-3 calculations, and the main data are shown in Tables 2 and 3. The transient S-3K calculations are then restarted from a specified point in time, or burnup state, and the input needed in S-3K is limited to data for the transient boundary conditions that are to be specified for e.g. flow rates, inlet temperature, control rod positions, etc. as functions of time.

Table 2. *Main core and fuel assembly data for the Oskarshamn 3 plant model*

CORE ACTIVE HEIGHT	3.68 M
Number of fuel assemblies	700
Number of assembly types	6
Number of control rods	169
No. of axial core nodes in S-3K	25

Table 3. *Fuel assembly type data in cycle 3 in the SIMULATE-3 model (See Fig 8)*

NO.	TYPE	NO. OF ASSEMBLIES	NO. (FROM CYCLE NO.)	NO. OF HEATED RODS
3	ABB81 (8* 8) with Gd	448	442(2)+6(1)	63
4	-"- no Gd	68	68(2)	63
5	SVEA-63 with Gd	56	4(2)+52(3)	63
7	SVEA-64 with Gd	128	128(3)	64

Initial conditions, such as burn-up, core flow and power, etc. are described in section 8.1.

7. APROS CODE

The APROS simulation software can be used for the simulation of fossil and nuclear power plants and for the process dynamics of a chemical plant [Ref. 9]. The sofware has been developed by the Technical Research Centre of Finland (VTT) and IVO Power Engineering Ltd. (IVO PE). The key features of the software are the grouping of the physical models into general and application specific packages, on-line simulation, inter-activity with possibility to make modifications both in the physical parameters and in the model structure, and the use of the model with a graphical user interface. In many applications a real-time or faster than real time simulation can be achieved. The APROS concept allows building of detailed models of various plant types for design, analysis and training purposes. In the nuclear field APROS has been used primarily in plant analyser applications, safety analysis and compact training simulator applications [Ref.10].

7.1 Neutronics Model in APROS

APROS code has one-dimensional and three-dimensional core neutronics models. Both models have two energy groups and six delayed neutron groups. In the one-dimensional model the discretized base equations are solved using a block-tridiagonal matrix solver. In the three-dimensional model the neutron flux equations are integrated over the node volumes, a few approximations are made and the fast and the thermal equations are solved applying Gauss-Seidel iteration [Ref.11]. The finite-difference type three-dimensional neutronics model is able to describe both the hexagonal and the quadrilateral fuel assembly geometry. The concentrations of the six delayed neutron groups are calculated. Iodine, xenon, samarium and promethium calculations are included, too, with a user -selected speed-up factor. The reactivity feedback effects due to fuel temperature, coolant density and temperature, coolant void fraction, coolant boron content and control and scram rods are taken into account in the model.

The calculation of fuel rod temperatures, coolant conditions and boron concentration are performed in the thermal hydraulic part of APROS. In the radial direction the fuel rod consists of possible centre hole, fuel, gap and cladding. The one-dimensional heat conduction solution in a fuel rod is usually calculated using ten radial nodes: seven in the fuel, one at the outer rim of the fuel pellet, one at the cladding inner rim and one at the cladding outer rim. At present, the axial heat conduction in the fuel rod can also be taken into account, if desired.

The neutronics nodes are combined with the adjacent thermal hydraulics nodes that have been described with the five- or the six-equation thermal hydraulic models. In the three-dimensional core model the division of the fuel assemblies into one-dimensional thermal hydraulic channels is very flexible. In the case of a symmetric core identical assemblies have often been grouped into a same thermal hydraulic channel. In most detailed core models there is only one assembly per thermal hydraulic channel.

Hot channel analysis is possible either simultaneously with the actual calculation, or afterwards, using special correlations for the critical heat flux. The decay heat calculation in APROS is usually performed with a decay power algorithm based on ANSI/ANS-5.1-1979. However, the user can also give the desired decay power versus time.

The reactor core in the three-dimensional model consists of fuel assemblies, reflector assemblies and control assemblies and thermal hydraulic channels. Each assembly has a name and coordinate position in the reactor and for each assembly axially varying composition can be given. The assemblies are placed into one-dimensional thermal hydraulic channels. Each channel has its own thermal hydraulic properties. The same axial division, typically 10-30 axial segments, is used for the thermal hydraulic channels and for the fuel, reflector and control assemblies.

7.2 T/H Model in APROS

For the one- and three-dimensional core models the user can select the homogeneous, the 5- or the 6-equation thermal hydraulic model. The homogeneous two-phase model is based on the mass, momentum and energy conservation equations of the mixture. Water and steam are assumed to have equal velocities and temperatures. In large vertical volumes a special node, where water and steam are fully separated, can be used [Ref.12].

The thermal hydraulic models presently used in the APROS reactor core are the five-. and the six-equation approaches.

The five-equation model is based on the conservation equations of mass and energy for liquid and gas phases and momentum equation for mixture of gas and liquid. In the five-equation model the gas and the liquid interface friction is not calculated, but the differential phase velocities are obtained through the drift flux correlations. A separate drift flux model calculates the mass flow rates of the phases. The quantities to be solved in the model are pressure, volumetric flows, void fractions and phasial enthalpies. The drift flux correlations used in the five-equation model are based on the EPRI full range, Ishii and Adron correlations. The wall heat transfer correlations are selected on the basis of wall, fluid, saturation and Leidenfrost temperatures as well as with the aid of the critical heat flux. The forced convection is calculated with the Dittus-Boelter correlation. The Chen's correlation is used for the calculation of boiling heat transfer. The Bromley correlation computes the heat transfer in the film boiling regime and the critical heat flux is calculated with Zuber-Griffith, Biasi or W3 correlations.

The six-equation model describes the behaviour of one-dimensional two-phase flow. The model is based on the conservation equations of mass, momentum and energy for the gas and the liquid phases. The equations are coupled with empirical correlations describing various two-phase phenomena. The pressures, the velocities, the volume fractions and the enthalpies of each phase are solved from the discretised equations using an iterative procedure. Special correlations are provided for reflooding phenomena. A moving mesh model for axial heat conduction can be used, if a great accuracy is needed for the heat flows.

The heat transfer modules connect all thermal hydraulic models with their own heat conduction solutions. Each thermal hydraulic model contains a boron concentration solution, too. The thermal hydraulic part of APROS calculates the loss of fuel enthalpy, the oxide layer thickness on cladding surface and the energy release from the cladding oxidation according to Baker-Just model.

The calculation of fuel temperatures, coolant densities and temperatures, void fractions and boron concentrations is performed in the thermal hydraulic model package. The information is transmitted to the neutronics model for use in the calculation of reactivity feedback effects. The information on the positions of control rods in the core is obtained usually from the automation system package. The neutronics model calculates the power production consisting of the fission power and the decay power. The power produced is transmitted back to the heat structures of the reactor channels that are a part of the thermal hydraulic model. It is possible to direct part of the power produced directly to coolant when five- or six- equation thermal hydraulic model is used.

7.3 TVO Input

The three-dimensional BWR core model in APROS describes the TVO I plant of 2500 MW_{th} with Atrium10 beginning of cycle core loading. The input data for the core were obtained on proprietary basis from TVO. A two-group cross section data set according to VTT Energy's HEXBU-3D/Mod5 code formulation was calculated at VTT Energy. This data set formulation enabled the dynamic modelling of various BWR transients with the APROS code in the similar manner as earlier for VVER-440 and VVER-91 (VVER-1000).

The TVO core was modelled with 500 fuel assemblies, that were basically of two fuel types and consisted of axially varying composition. The core description had half-core symmetry and thus two identical fuel assemblies were placed in each one-dimensional thermal hydraulic channel. The total number of thermal hydraulic channels in the core was 250. Axially each fuel assembly and thermal hydraulic channel were divided into 25 sections. In the equilibrium state there was a control rod pattern consisting of 17 BWR control rods. This pattern was described in the model using 68 APROS control assemblies (4 assemblies per each control rod). For the recriticality studies all the 121 control rods were modelled using 484 APROS control assemblies. The 3-D core nodalization overview is depicted in Figures 8 and 9 with the statistics of the calculation model shown in Table 4.

In the model the core input flow is first divided into five parallel pipes. The flow from each pipe is further divided into fifty pipes each of which is connected to one thermal hydraulic core channel at the core inlet. The corresponding piping is also included at the reactor outlet, as indicated in Figure 9.

Table 4. *Statistics of APROS BWR 3-D core model.*

Component type	Number of components
Nuclear element (fuel assembly)	500
Nuclear control element (control rod)	500
Reactor channel	250
Pipe	511
Thermal hydraulic nodes (core)	6250
Heat structure nodes	62500
Neutronics nodes	12500

The three-dimensional two energy group neutronics model and the five-equation thermal hydraulics model were used in the studies. The principal reason for selecting the five-equation thermal hydraulics model instead of the 6-equation model was the considerably faster calculation speed obtained with the five-equation model.

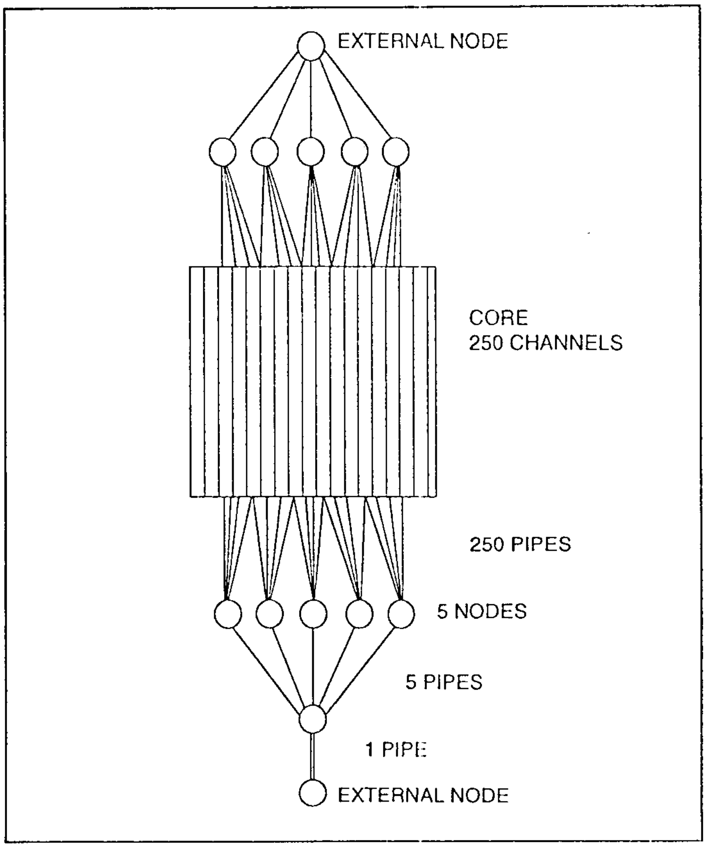


Figure 9. *Core and piping in the APROS model used for recriticality analysis.*

8. INITIAL STATE OF CORE DEGRADATION FOR EVALUATION OF POSSIBILITY OF RECRITICALITY

8.1 Oskarshamn 3

In order to simulate realistic conditions in the transient S-3K calculations, operating data for Oskarshamn 3 were taken from SIMULATE-3 steady-state core follow calculation results provided by the utility. As mentioned in section 6.3 data for cycle 3 were used, and burn-up step No 11 was selected corresponding to an average cycle burn-up of 3.59 MWd/kgU. At this point the reactivity is quite high since much of the burnable poison is gone. The power level was 100%, with a recirculation flow reduced to 79 % of the nominal value. Average burn-ups for the foregoing operation were 7.5 MWd/kgU and 8.2 MWd/kgU for cycle 1 and 2, respectively, which is valid for fuel retained from those cycles. Since minor modifications were made, among others replacing the 8 SVEA-100 assemblies by the same number of type 7 SVEA-64 bundles, new steady-state SIMULATE-3 runs were made up to, and including cycle 3. The quarter core map in Figure 10 shows the steady-state radial power factors, exit node quality and the average assembly burnup for cycle 3, burn-up step No. 11.

0.436 0.090 19.986	0.409 0.082 19.617	0.360 0.069 19.778	0.281 0.048 19.712	2RPF -REL POW FRAC 2XE3 -QUALITY 2EXP -EXPOSURE										
0.698 0.138 14.951	0.644 0.124 17.677	0.501 0.089 11.988	0.531 0.116 16.865											
0.911 0.152 16.157	1.130 0.212 3.698	0.874 0.144 15.027	0.760 0.154 18.554	0.686 0.160 20.952	0.607 0.138 20.194	0.535 0.117 20.219	0.472 0.100 19.887	0.417 0.085 17.772						
1.222 0.235 3.924	0.956 0.162 22.003	1.289 0.255 4.204	1.014 0.176 14.726	1.182 0.265 3.772	0.847 0.177 20.899	0.788 0.161 21.657	0.738 0.149 19.775	0.642 0.125 19.038	0.511 0.111 19.774					
0.730 0.111 21.534	1.307 0.257 4.286	1.031 0.179 22.191	1.035 0.180 22.187	1.395 0.284 4.357	1.428 0.296 4.506	1.125 0.202 14.484	1.327 0.269 4.092	0.964 0.166 14.744	0.733 0.148 20.092	0.578 0.130 20.304				
0.747 0.114 21.487	0.993 0.169 22.239	1.421 0.290 4.779	1.096 0.193 21.768	1.135 0.203 20.998	1.161 0.210 22.436	1.568 0.339 4.846	1.153 0.209 22.366	1.450 0.305 4.291	1.264 0.252 3.708	0.733 0.148 20.092	0.511 0.111 19.774			
1.327 0.262 4.683	1.055 0.183 20.887	1.090 0.192 22.261	1.132 0.202 22.607	1.576 0.337 5.221	1.326 0.250 14.548	1.223 0.225 22.555	1.629 0.358 4.741	1.188 0.218 22.007	1.450 0.305 4.291	0.964 0.166 14.744	0.642 0.125 19.038	0.417 0.085 17.772		
1.065 0.185 20.600	1.056 0.183 22.975	1.095 0.193 22.420	1.534 0.324 5.328	1.180 0.213 22.836	1.219 0.223 22.651	1.255 0.232 20.019	1.229 0.226 22.119	1.629 0.358 4.741	1.153 0.209 22.366	1.327 0.269 4.092	0.738 0.149 19.775	0.472 0.100 19.887		
1.077 0.188 17.759	1.384 0.279 5.273	1.089 0.191 20.528	1.151 0.206 20.485	1.234 0.226 17.722	1.613 0.349 5.230	1.244 0.229 20.027	1.255 0.232 20.019	1.223 0.225 22.555	1.568 0.339 4.846	1.125 0.202 14.484	0.788 0.161 21.657	0.535 0.117 20.219		
0.964 0.162 22.322	0.949 0.159 21.928	1.304 0.256 4.989	1.115 0.197 17.508	1.157 0.207 20.559	1.169 0.211 23.022	1.613 0.349 5.230	1.219 0.223 22.651	1.326 0.250 14.548	1.161 0.210 22.436	1.428 0.296 4.506	0.847 0.177 20.899	0.607 0.138 20.194		
1.152 0.215 5.039	0.630 0.090 20.518	0.648 0.094 20.483	0.963 0.162 20.340	1.409 0.286 5.447	1.157 0.207 20.559	1.234 0.226 17.722	1.180 0.213 22.836	1.576 0.337 5.221	1.135 0.203 20.998	1.395 0.284 4.357	1.182 0.265 3.772	0.686 0.160 20.952		
0.850 0.137 20.006	0.594 0.083 20.306	0.587 0.082 20.309	1.116 0.205 5.097	0.963 0.162 20.340	1.115 0.197 17.508	1.151 0.206 20.485	1.534 0.324 5.328	1.132 0.202 22.607	1.096 0.193 21.768	1.035 0.180 22.187	1.014 0.176 14.726	0.760 0.154 18.554	0.531 0.116 16.865	0.281 0.048 19.712
0.880 0.144 21.566	1.097 0.201 5.139	0.771 0.120 21.829	0.587 0.082 20.309	0.648 0.094 20.483	1.304 0.256 4.989	1.089 0.191 20.528	1.095 0.193 22.420	1.090 0.192 22.261	1.421 0.290 4.779	1.031 0.179 22.191	1.289 0.255 4.204	0.874 0.144 15.027	0.501 0.089 11.988	0.360 0.069 19.778
0.962 0.163 21.829	0.901 0.149 22.251	1.097 0.201 5.139	0.594 0.083 20.306	0.630 0.090 20.518	0.949 0.159 21.928	1.384 0.279 5.273	1.056 0.183 22.975	1.055 0.183 20.887	0.993 0.169 22.239	1.307 0.257 4.286	0.956 0.162 22.003	1.130 0.212 3.698	0.644 0.124 17.677	0.409 0.082 19.617
1.336 0.268 6.167	0.962 0.163 21.829	0.880 0.144 21.566	0.850 0.137 20.006	1.152 0.215 5.039	0.964 0.162 22.322	1.077 0.188 17.759	1.065 0.185 20.600	1.327 0.262 4.683	0.747 0.114 21.487	0.730 0.111 21.534	1.222 0.235 3.924	0.911 0.152 16.157	0.698 0.138 14.951	0.436 0.090 19.986

Figure 10. *Initial power factor, upper node steam quality and average burn-up for the assemblies in the O-3 core from SIMULATE-3 steady-state calculation.*

The core damage conditions to be simulated in the transient S-3K calculations were chosen from the SCDAP/ RELAP5 results produced for Forsmark 3 in the foregoing task of RAK-2.1. The layout of the safety system, i. e. safety valves, auxiliary feedwater and Emergency Core Cooling Systems (ECCS) are the same in O-3 as in F-3, so the core degradation states can be assumed to be equal for the same initiating events. There is a deviation, however, in the assumption of the fuel types as mentioned before. The power level was also lower, 3000 MW_{th} in cycle 3 in O-3, not 3300 MW_{th} as in the F-3 calculation. (The power in O-3 was raised to 3300 MW_{th} after cycle 4).

The original goal was to perform the recriticality analysis for cases which corresponded to realistic scenarios and boundary conditions. This should include core damage states from the SCDAP/ RELAP5 calculations obtained with late initiation of reflooding. This means start of reflooding at peak cladding temperatures of 1600 K and 1800 K and with fractions of control rod melted of 5 to 42 %, according to [Ref 8]. Figure 11 shows the histories of the core maximum temperature and collapsed water level for case No. 4 in [Ref 8]. The calculated core end state is illustrated by Figure 12. Recriticality calculations were also planned to be made for some new cases with extended range of reflooding mass flow rates, which can be obtained from the various injection systems. These are the high pressure auxiliary feed water system, 327, the low pressure ECCS, system 323, and the normal feed water system 312, which all, or in various combinations can be assumed to be activated at return of electric power supply.

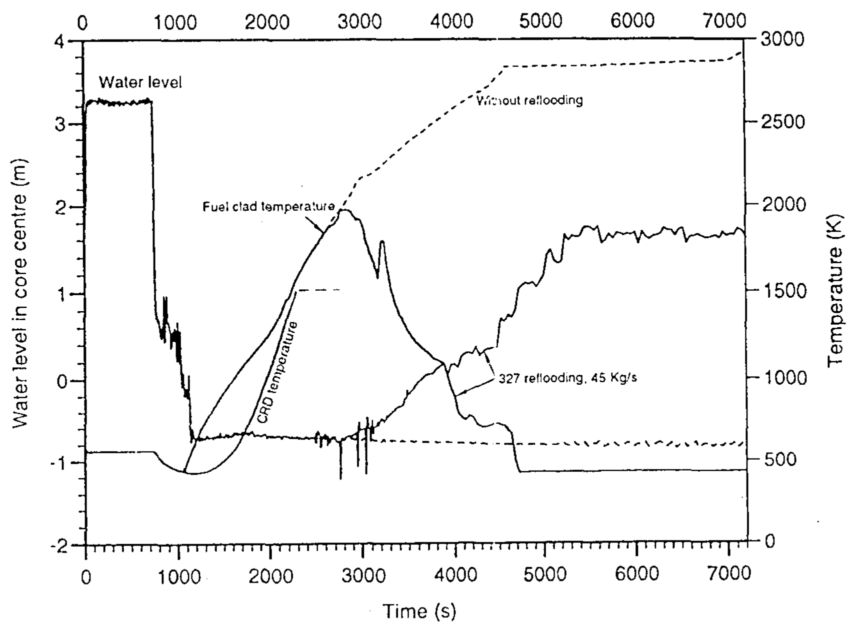


Figure 11. *Results from core damage and reflooding calculations for Forsmark 3 with SCDAP/RELAP5 in RAK-2.1 [ref 8]. Water level and core temperatures to form basis for recriticality analyses.*

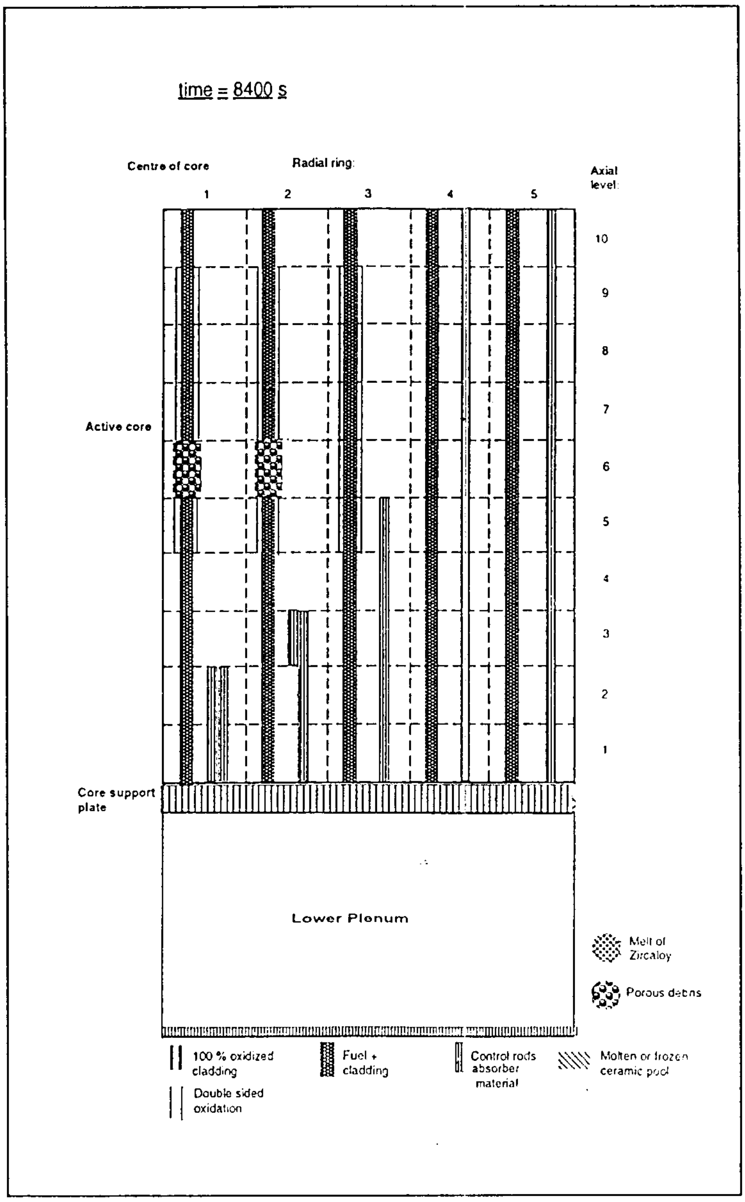


Figure 12. *Core end state in F-3 for case 4 from SCDAP/RELAP5 calculations in RAK-2.1 [ref 8].*

Since there were problems to simulate these realistic conditions with the current version of S-3K, the original goal for the transient calculations had to be abandoned in favour of more simplified approaches.

8.2 TVO

In the low pressure scenarios MELCOR typically predicts that the control rod melting starts about 0.5 h after depressurization of reactor coolant system. The eutectic melting temperature of B_4C/stainless steel is 1523 K and when the control rod node reaches this temperature, B_4C mixture starts to candle down. All boron carbide from the inner part of the core has relocated onto the lower core support structures in about 15 minutes after the melting has started. The periphery of the core, however, remains less damaged due to lower heat generation.

The time window between the melting of control rods and the onset of fuel damage is narrow, only 1-2 minutes, if the default options for fuel degradation model are used. The default assumption in MELCOR is that, when the cladding oxide layer thickness becomes large, the fuel rod collapses and forms particulate debris. If the oxide layer thickness criteria is turned off by selection of sensitivity coefficient, the code allows the formation of bare pellet stacks in the core.

The maximum cladding temperature is around 1800 K, when the control rods have melted and from the recriticality point of view, the most critical time for start of reflooding occurs (Fig. 13). If the bare fuel pellet stacks are allowed to form, MELCOR predicts that 66 - 73 % of the core could be without absorber material at the start of reflooding (Fig. 14). If the particulate debris formation is allowed, the control rod-free region in the core would be significantly smaller and the control rods in the outer boundary of the core would be still intact, when substantial fuel damage in the inner parts would begin.

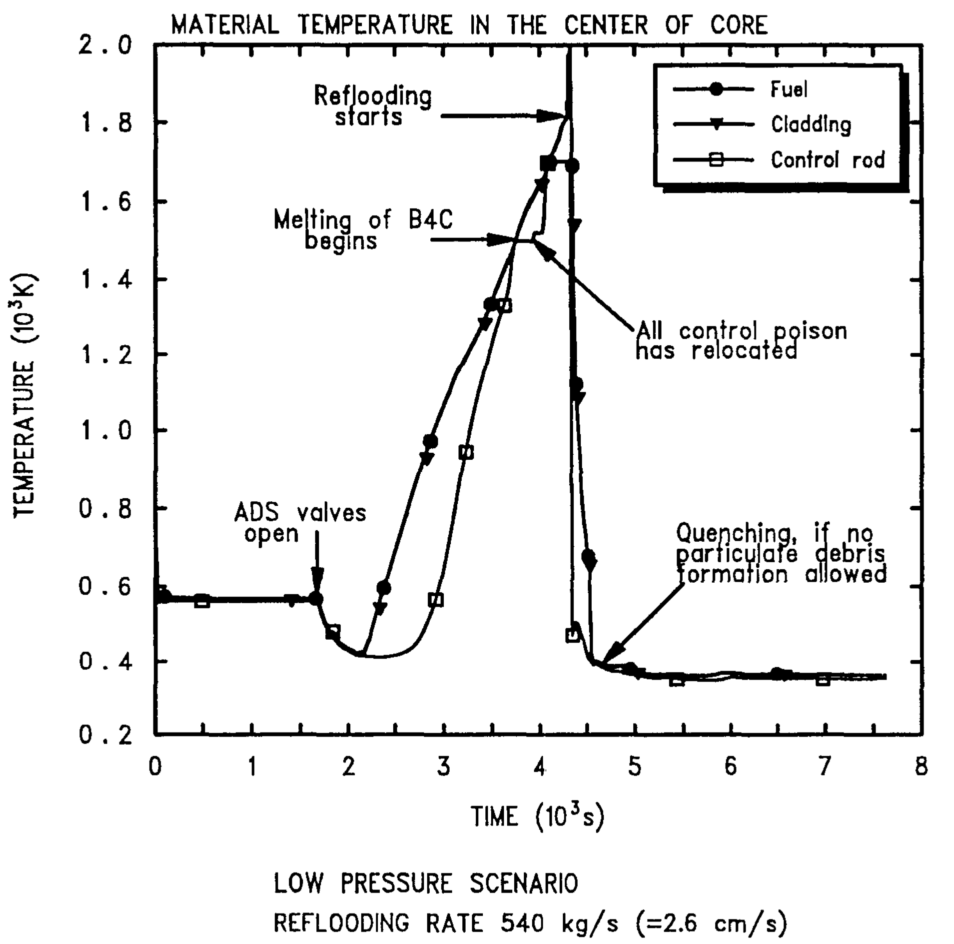
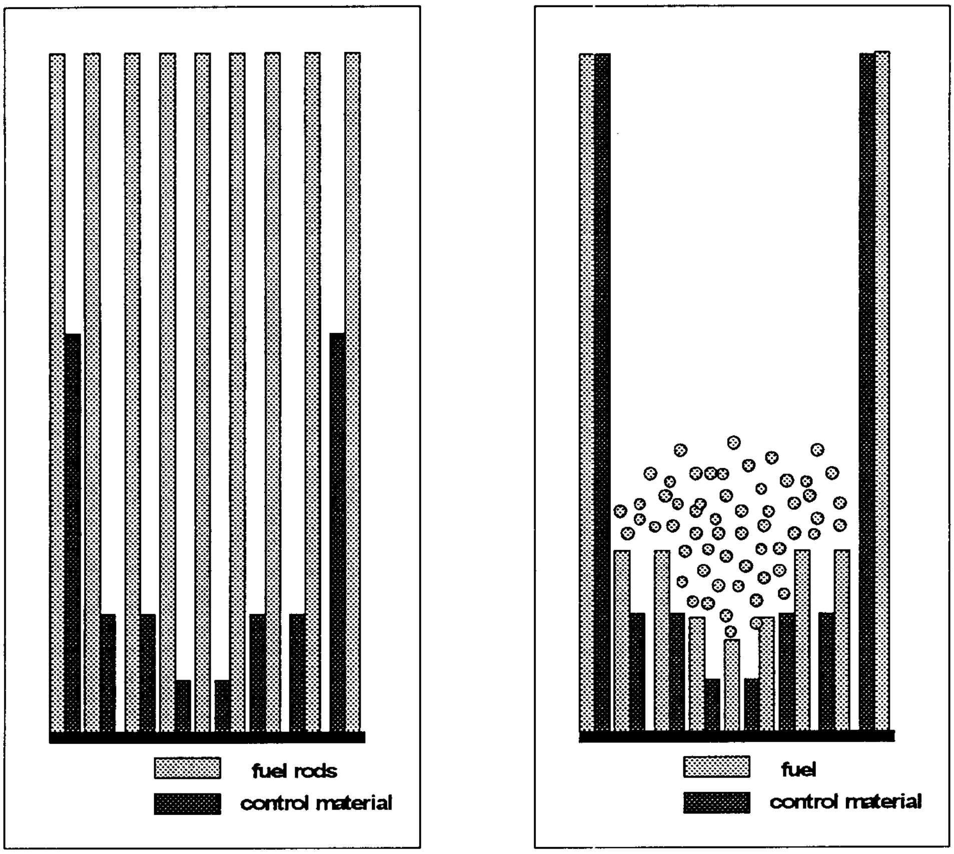


Figure 13. *Core material temperatures in the center node of the core predicted by MELCOR. Low pressure scenario with reflooding at 4200 s with 4 x 323 and 4 x 327 injection.*



A) B)

Figure 14. *Damaged core state before start of reflooding predicted by MELCOR.*
A) 66-73 % of control rods have melted, no particulate debris,
***B)** particulate debris formation allowed, when high cladding oxidation.*

According to MAAP4 predictions the time window for possible recriticality in the station blackout with depressurization at 1 h is circa 40 min from 1 h 20 min to 2 h into the accident. Reflooding may accelerate the melting and relocation of control rods due to oxidation heat but does not widen the time window. About 15 minutes after start of melting 60 % of absorbing material has relocated, about 5 min later 80 % of control blades had melted and almost all control materials had relocated in 40 minutes.

In the feedwater LOCA the time window for possible recriticality caused by reflooding is circa 20 min from 20 min to 40 min into accident.

9. RECRITICALITY ANALYSES WITH RECRIT

The reference plant chosen for the analyses with RECRIT code is a TVO reactor, which is a BWR of Swedish design. Its main data (of special relevance for the present analysis) are :

- Core height: 368 cm
- Core radius: 194 cm
- Nominal power density: 50.5 KW/l
- Thermal power: 2200 MW

9.1 General Process Behaviour

Before initiation of the accident the power plant is operating on full power. The accident is initiated with total loss of electiricity, which trips the reactor and the turbine and switches of the circulation pumps. In RECRIT no circulation pumps are modelled and the initial state is defined by the initial liquid temperature in the beginning of the accident.

The heat generated in the beginning by decay power and quite soon a natural circulation pattern is stabilized corresponding to the real plant state after blackout. The energy from the core is removed by slow boiling through the back-up steam relief valves controlling the vessel pressure.

When the water level in the vessel drops below 0.9 m from the top of the core (L4 signal) and 900 s has passed from the L4 signal, the vessel depressurization is initiated by opening the ADS dump system. The blowdown capacity just after valve opening is 550 kg/s in the beginning . The ADS system is designed for dropping the vessel pressure below the low pressure injection point (7 bar). The core is uncovered during the ADS blowdown and the core heatup commences. As a consequence of the heatup the control rods are melting and premises for the criticality are created.

After the low pressure pumps have started, the reflooding water fills the core. The reflooding start signal was defined to be the maximum fuel temperature of 1800 °C or the remaining fraction of control rods of 30 %. The control rod concentration criteria was selected in order to get a reasonable height for the location of the recriticality. In the analyses the low control rod fraction caused the reflooding to start at 1690 °C. In the hot core a significant fraction of the incoming water is entrained above the quenching front forming a droplet-dispersed flow regime. The upward movement of the quenching front defines the point, where neutron flux peak can be formed. Criticality may be expected when the quenching front reaches the unrodded zone of the core. In these calculations the transition zone from fully rodded core to unrodded core is around 0.5 m.

In the calculated cases the injection rate was varied from 22.5 kg/s representing capacity of a single high pressure injection pump, 45 kg/ s representing two high pressure injection pumps, 160 kg/s representing one high pressure and one low pressure injection pumps as well as 540 kg/s being the capacity of all four low pressure injection pumps . If all decay heat were removed by boiling of the reflooding water, the required heating rates w ould be 40 MW, 80 MW, 300 MW and 950 MW. Considering the decay heat, the 22,5 kg/s reflooding rate can be considered insufficient injection capacity.

The general process behaviour pictures are displayed for the injection rate of 160 kg/s. In all other cases the the behaviour is the same until the start of reflooding. The RPV pressure history is depicted in Figure 15. The periods of full pressure, controlled by safety relief valves, ADS depressurization at 1700 s and reflooding (at 4020 s) can be seen from the result.

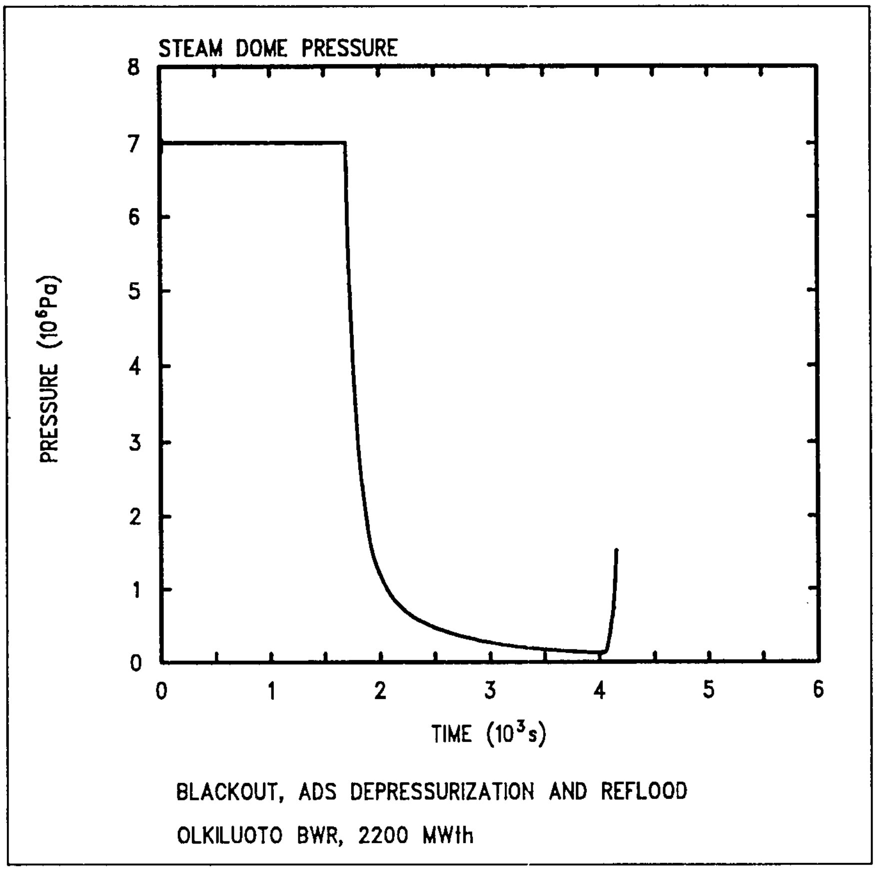


Figure 15. *Vessel pressure behaviour during the incident.*

In Figure 16 the energy balance during decay heat generation of the core is illustrated. The power peak after the reflooding start exceeds the scale. The decay power drops according to the ANSI curve. The depressurization period can be recognized as the core wall heat flux exceeds the core power. The core heatup period can be seen as power generation exceeding the wall heat flux. The injection of coolant into the core causes an increase in the total wall heat flux.

The maximum and average temperatures of the core and cladding are depicted in Figure 17. With the wall heat flux being in the range of the decay heat generation, the fuel and cladding temperatures are close to each other. The maximum difference between the average and the maximum temperature is 500 K.The oxidation model was excluded from these calculations, but it will be included into the future results.

In Figure 18 the core average void fraction, the core average control rod status (1. = 100 % rodded, 0. = 0 % control rod concentration) and the effective multiplication factor are depicted for the whole incident. The control rod melting begins at 3000 s. The sensitivity of the reactivity with respect to the void fraction can be seen clearly.

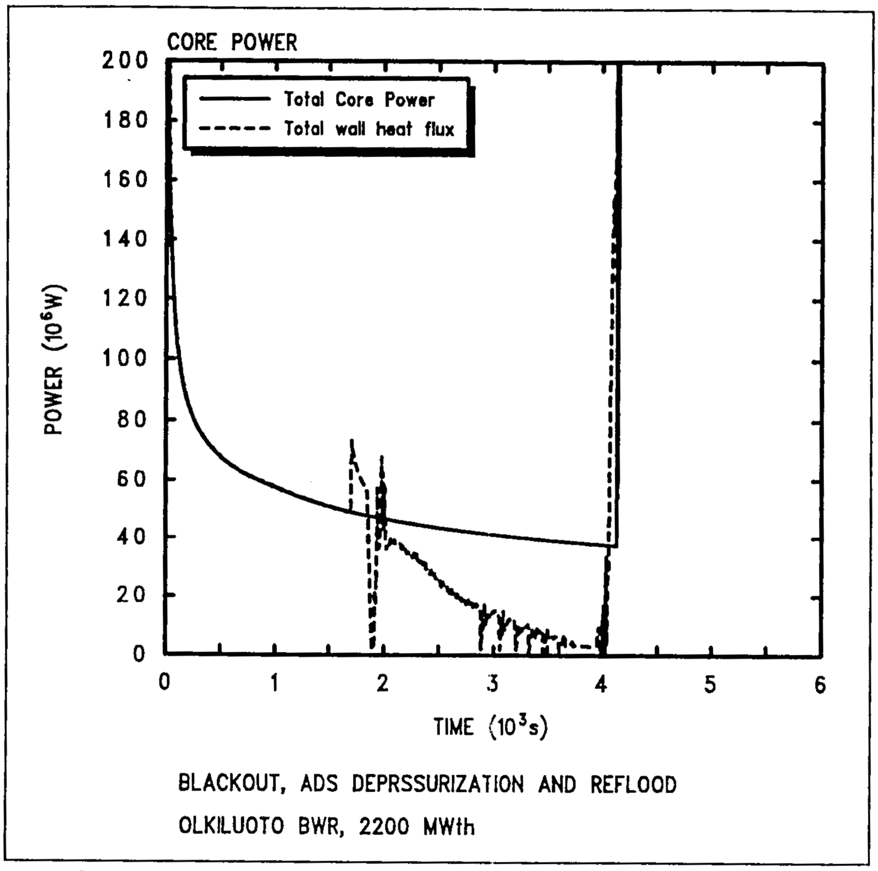


Figure 16. *Decay heat power and heat removed from the fuel. The effect of ADS blowdown and core heatup can be seen as a detachment of total wall heat flux from the heat generation.*

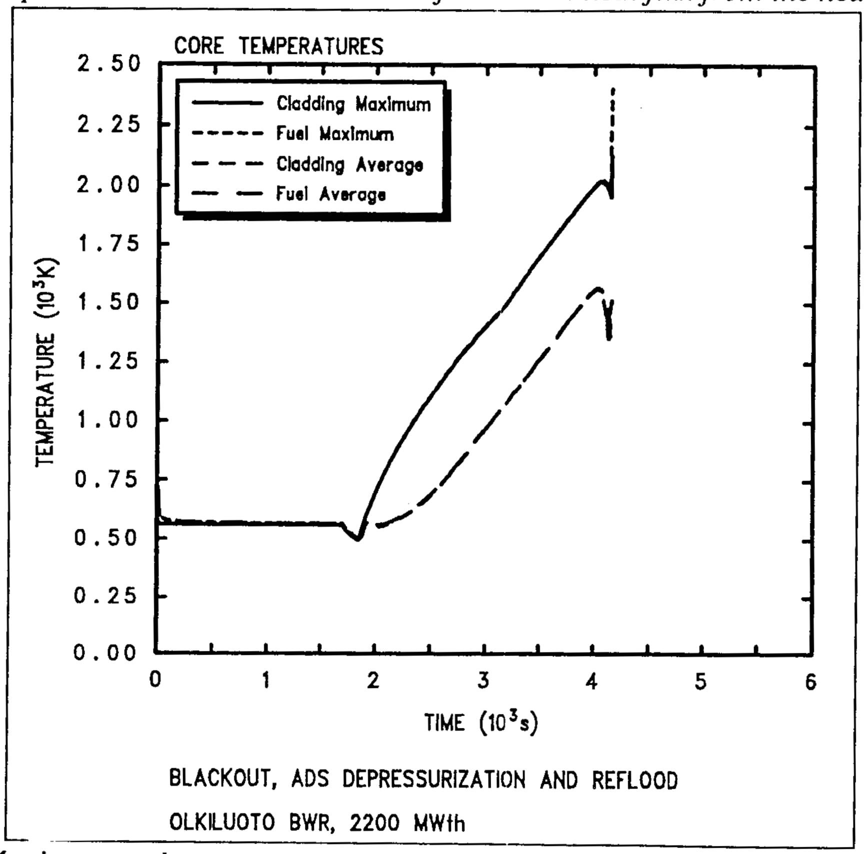


Figure 17. *Maximum and average temperatures of cladding and fuel. Due to low power the differences between the fuel and the cladding temperatures are very small before criticality.*

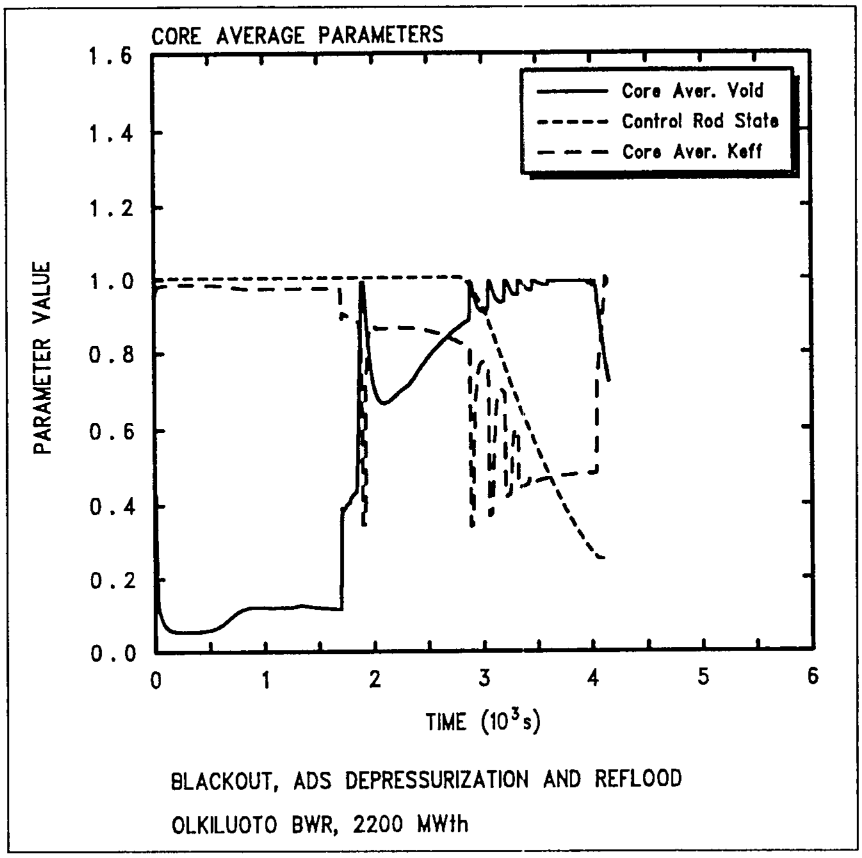


Figure 18. *Core average void fraction, core average control rod status (1. = fully rodded, 0. = fully unrodded) and effective multiplication factor.*

Figure 19 illustrates water levels in the different vessel sections. For the core and the bypass the levels are measured from the bottom of the core region. The downcomer and the lower plenum levels are measured from the bottom of the lower plenum. In the level scale from the lower plenum bottom the core is located between 4.5 m and 8.2 m. The uppermost sections in the vessel are emptied first, the lowest sections last. The core water levels in different radial rings are about the same. That is why the level for a single radial ring is displayed. The ADS was asumed to be started at 1700 s, at same time as in MELCOR analyses. At this time the downcomer level of RECRIT prediction is 9.2 m, 1.0 m above the core top. The designed ADS start takes place at level 0.9 m above the core top. The core is totally uncovered at 3500 s. In this result only minor U-tube oscillation can be seen between the downcomer and the core during blowdown period. When the reflooding starts (160 kg/s), the bypass level remains lower than the core level. This is a consequence of the bypass inlet restrictions. The relationship between the core level and the bypass level confirms the selected option for neutronics data calculation to be realistic, if not slightly conservative. For neutronics data it was assumed, that the void fraction in the channel equals the void fraction in the bypass.

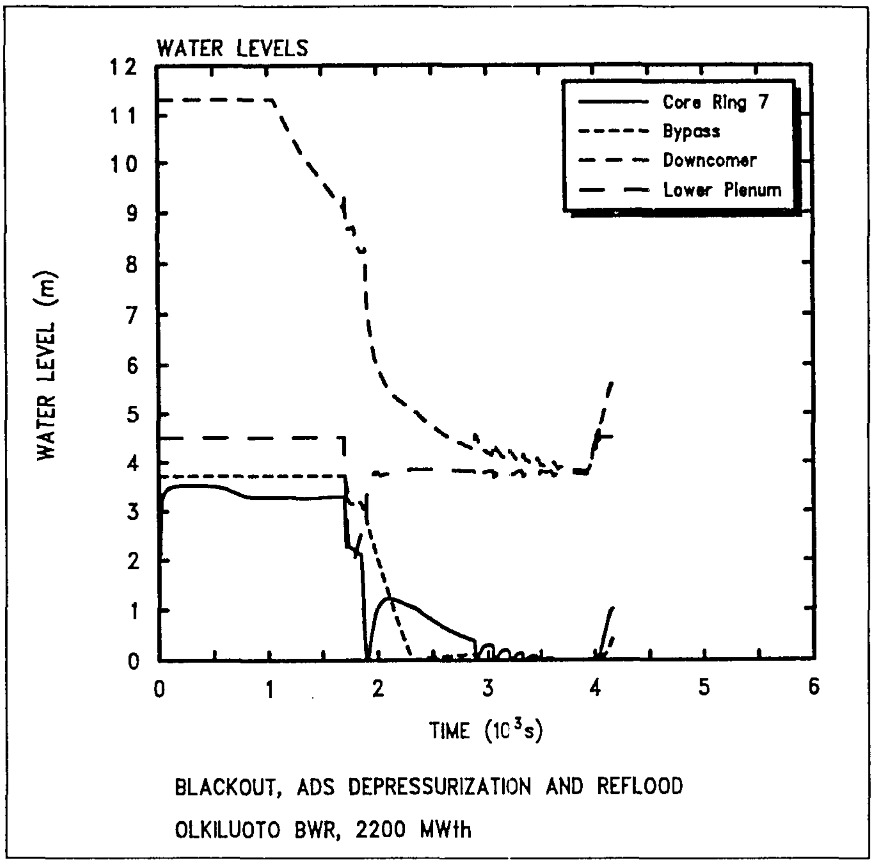


Figure 19. *Water levels in different sections of the reactor vessel measured from the bottom of the section. The water level in all core radial rings is about the same.*

9.2 The characteristics of the criticality behaviour

Prior to the critivcality the temperatures of the fuel, water and steam in the nodes as well as the steam content (void) and the control rod state are calculated by the thermo-hydraulics models in small time steps (0.1 sec.). The heat source is the decay heat plus the heat from fissions. The reactor is still subcritical, but, as k_{eff} is approaching unity, an increasing neutron flux will be created, due to the "amplification" of the neutron source, which is always present from the spontaneous fissions in U^{238}.

When, eventually the effective multiplication factor k_{eff} becomes greater than unity, the "recriticality" phase begins.

In this phase, the time steps have to be equal to the prompt neutron lifetime for reasons explained in section 5.1.

Otherwise, the procedure is, as before, to make alternate calls of the thermo-hydraulic subroutine and the neutron flux subroutine, now always in the "dynamic source-mode" (see section 5.1).

With convenient intervals, also a 'k_{eff}-value call' is made, but only to get the k_{eff}-value for output purposes. At the same time, any other "interesting" results (fuel temperature, void fraction, power, etc.) are printed out.

The core power for the three variation cases has been shown in Figure 20.

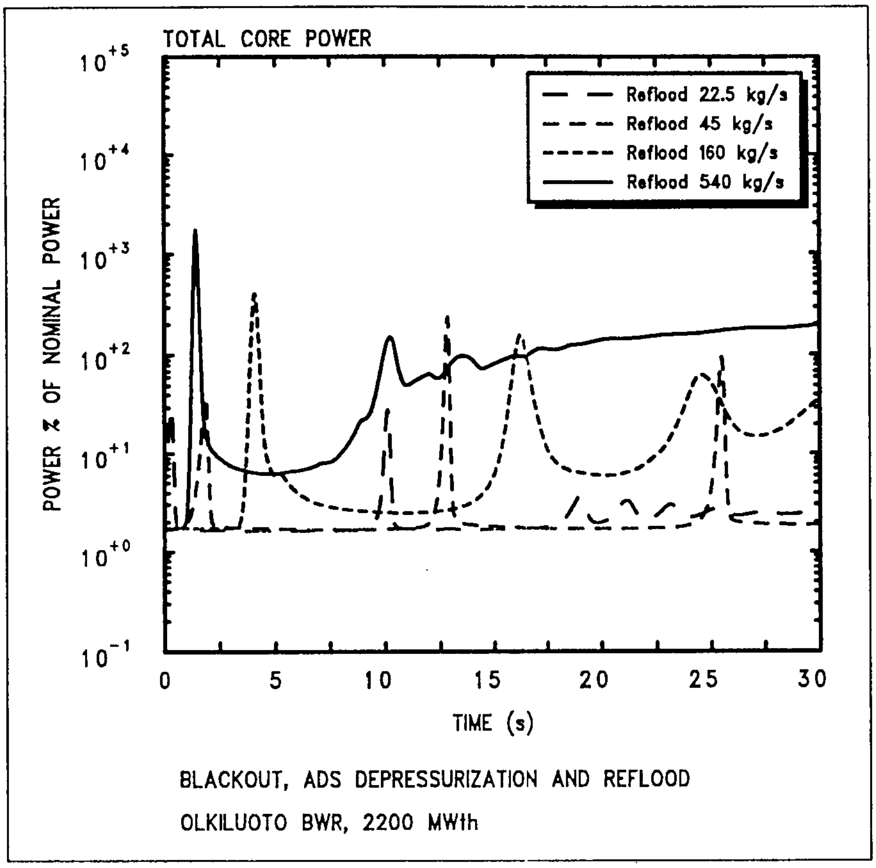


Figure 20. *Total core power generation after reaching the criticality.*

Highest power peak of 1730 % of the nominal power can be seen for the 540 kg/s case. Another peak occurs at 10 s and after that the power remains around twice the nominal power. The first power peak in 160 kg/s case is about 400 % of the nominal power, the second is 100 % and the average power level after 15 s is around 25 % of th nominal power. In the 45 kg/s case the second power peak is the highest, around 220 % of the nominal power and three peaks were encountered during the 30 s of criticality. In the case of 22.5 kg/s two peaks of 30% of the nominal power can be seen.

In Figure 21 the fuel average temperature and in the Figure 22 the fuel maximum temperature are displayed. With 540 kg/s injection rate the average core temperature increases by about 240 K and with 160 kg/s injection by about 105 K. In the case with 40 kg/s injection rate the average core temperature increases by around 20 K and in the case of 22.5 kg/s injection rate by about 8 K during the maximum power peaks. The effect of power peaks on fuel maximum temperatures can be seen only for two highest injection rates. This is due to the fact, that the criticality takes place in the zone, where the fuel temperatures are decreasing in the vicinity of the quenching front. For the 540 kg/s injection case the Doppler effect is not sufficient to cut the temperature rise after the power peaks.

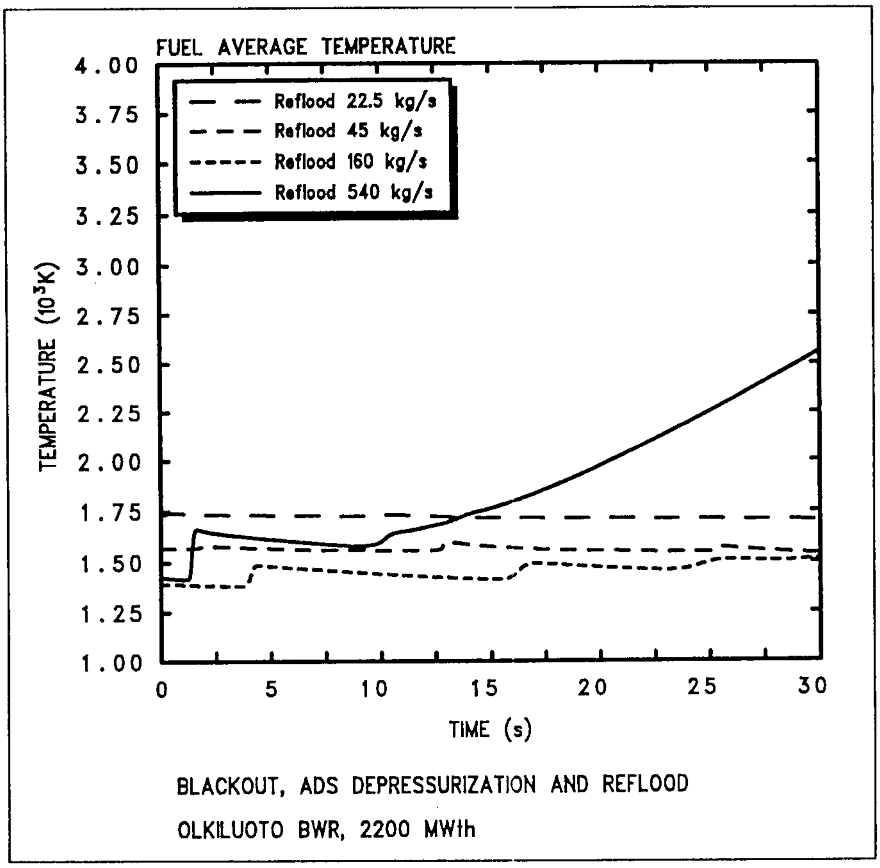


Figure 21. *Fuel average temperature behaviour after reaching the criticality.*

The history of the effective multiplication factor from incipience of reflooding to critical conditions is depicted in figure 23. Just after first reactivity period the multiplication factor drops down, but later history indicates that a new criticality is possible. Prior to reaching criticality the low injection cases show strong reactivity oscillations. These oscillations are due to sensitivity of reactivity to void fraction oscillations.

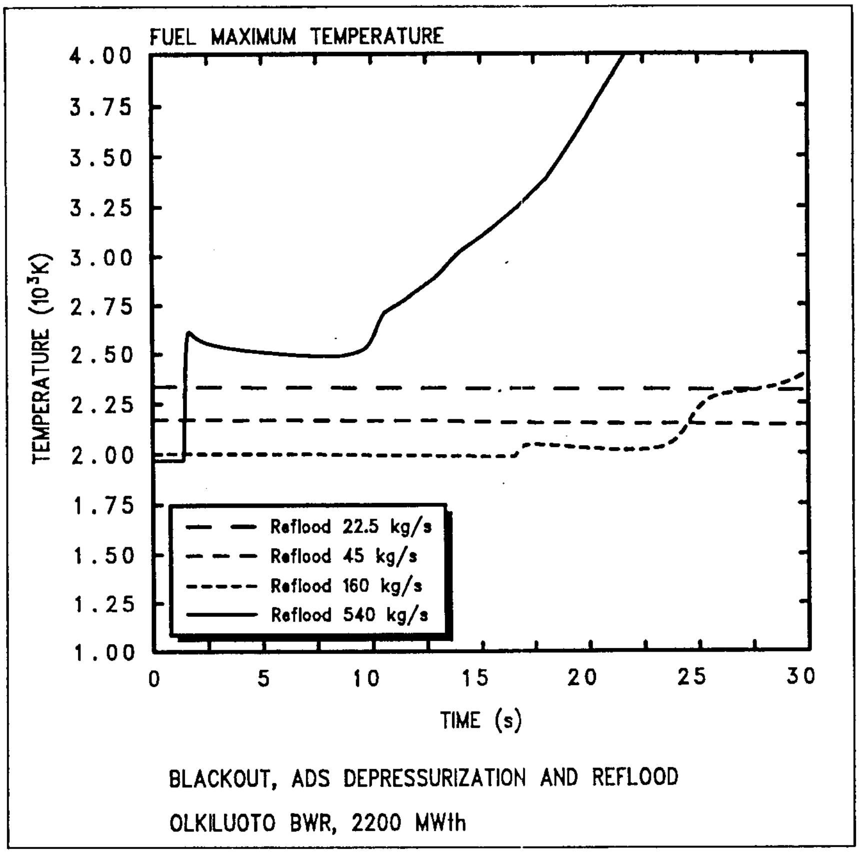


Figure 22. *Fuel maximum temperature behaviour after reaching the criticality*

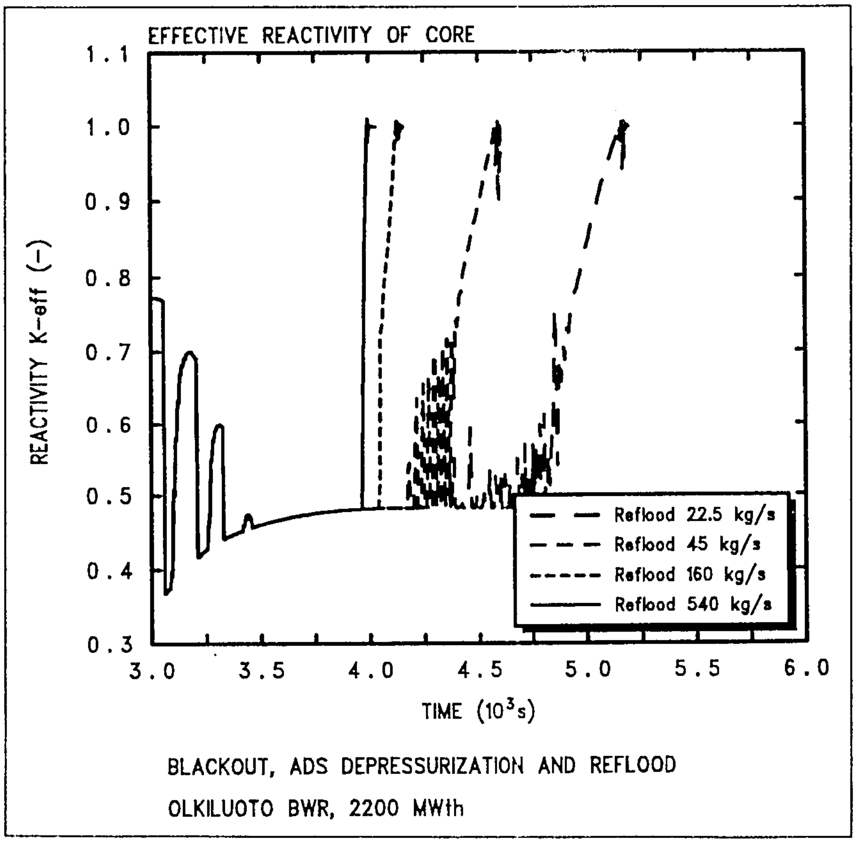


Figure 23. *Development of the effective multiplication factor from reflooding incipience to critical conditions.*

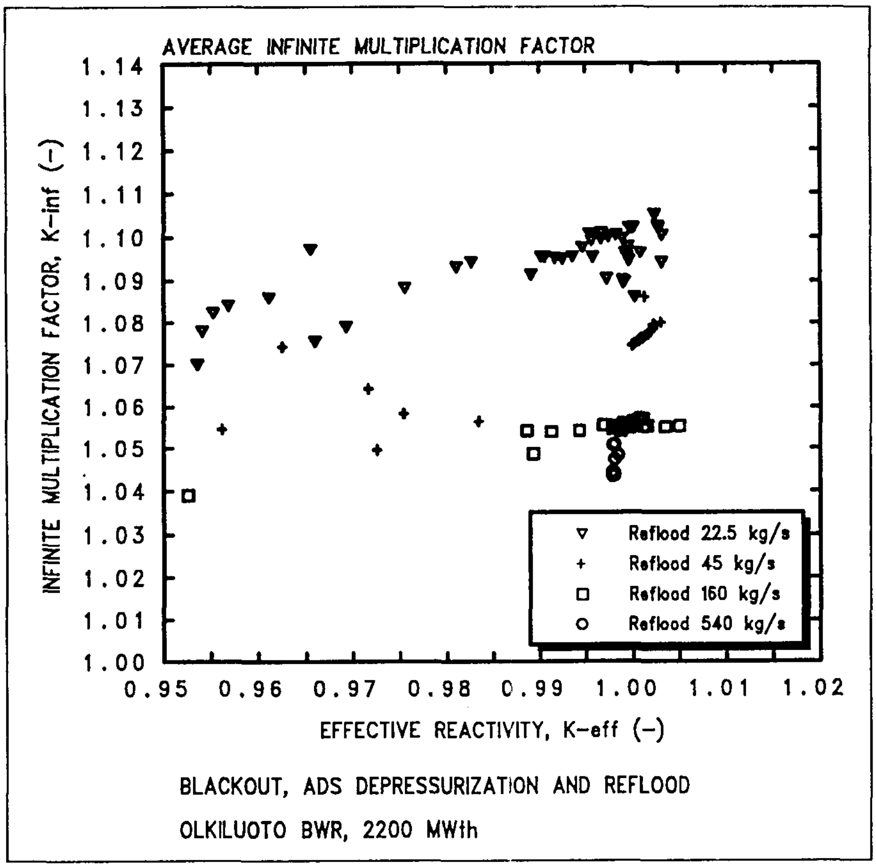


Figure 24. *Infinite multiplication factor as a function of effective multiplication factor for sections with kinf >1.*

From the reactor physical parameters two special quantities have been determined, which characterize the reactor physical behaviour. In Figure 24 the average of the infinite multiplication factor is depicted for the sections with $k_{inf} > 1.0$ as a function of the core average effective multiplication factor. The result indicates that before reaching the global core criticality the local infinite multiplication factor about 1.04 ... 1.10 is possible. Highest values are reached with low injection rates and the lowest for high injection rates.

The results of the core cross volume becoming critical is shown in Figure 25. The whole core volume in the cross scale is 40 m^3. With small injection rates the critical volume might be only 3 m^3, while in the cases with high reflooding rates the core critical volume may be half of the core size, around 20 m^3.

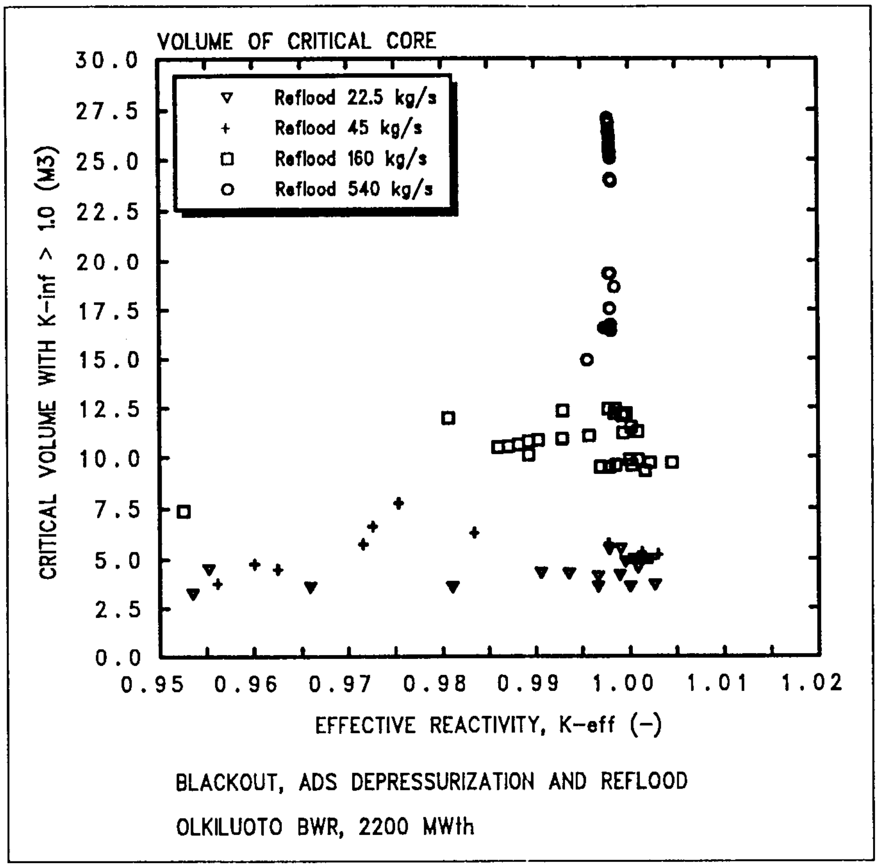


Figure 25. *Volume with local infinite multiplication factor kinf >1.*

The parameters for the different cases are summarized in Table 5. The results indicate a strong dependence of the maximum power on the injection rate. In all cases the average fuel temperature rise during the first peak is quite low. In the maximum temperature a clear effect from the power peak can be seen only for the 540 kg/s injection case. In the other cases the temperature rise in the flooded zone does not exceed the maximum temperature in the upper part of the core. At least the 160 kg/s and 540 kg/s injection rate cases indicate that further temperature increases can be expected if reflooding continues.

Table 5. *Summary of recriticality calculations with RECRIT code for TVO NPP*

CASE	INJECTION RATE [KG/S]	MAXIMUM POWER FIRST PEAK [%]	MAXIMUM FUEL TEMPERATURE [K]	AVERAGE FUEL TEMPERATURE RISE DURING HIGHEST PEAK [K]
1	22.5	30 %	2330 K	8 K
2	45.0	220 %	2170 K	20 K
3	160.0	400 %	2073 K	105 K
4	540.0	1730 %	2610 K	240 K

10 RECRITICALITY ANALYSES WITH SIMULATE-3K

10.1 Results from feasibility calculations with a small core model

In order to investigate the capability of S-3K to simulate the reflooding scenarios with a full plant model some feasibility calculations were performed with a small 4x4 assembly core model. The assumptions for the boundary conditions and the prescribed transients were chosen with regard to the fact that the available code version still lacked some features desirable for complete full scale simulations. Pressure transients could then not be specified in the input, and the stability of the calculations at low pressures had to be improved.

In spite of these limitations a few calculations were carried out which showed some interesting results that might be applicable also on a full plant. The following conditions were assumed:

- A core comprising 16 SVEA-64 full length fuel assemblies and 9 BWR control rods. Full node boundary option was used, i.e. no reflector data and no neutron leakage were assumed.
- Starting with fresh fuel the core was operated at full power (scaled from 3300 MW_{th} in F-3) to various burnup levels. Total blackout was simulated by stopping the core flow and inserting all control rods.
- Core uncovery and heat-up was achieved by gradual water boil-off due to decay power.
- When the core was fully uncovered and the maximum fuel temperature had reached a high enough temperature to melt central control rods, partial CRD melt down was simulated by withdrawal of the middle 4 of the 9 control rods. Reflooding was simulated by restarting the core flow with flow rates corresponding to scaled-down reactor reflooding rates. This scenario is depicted in Figure 26.

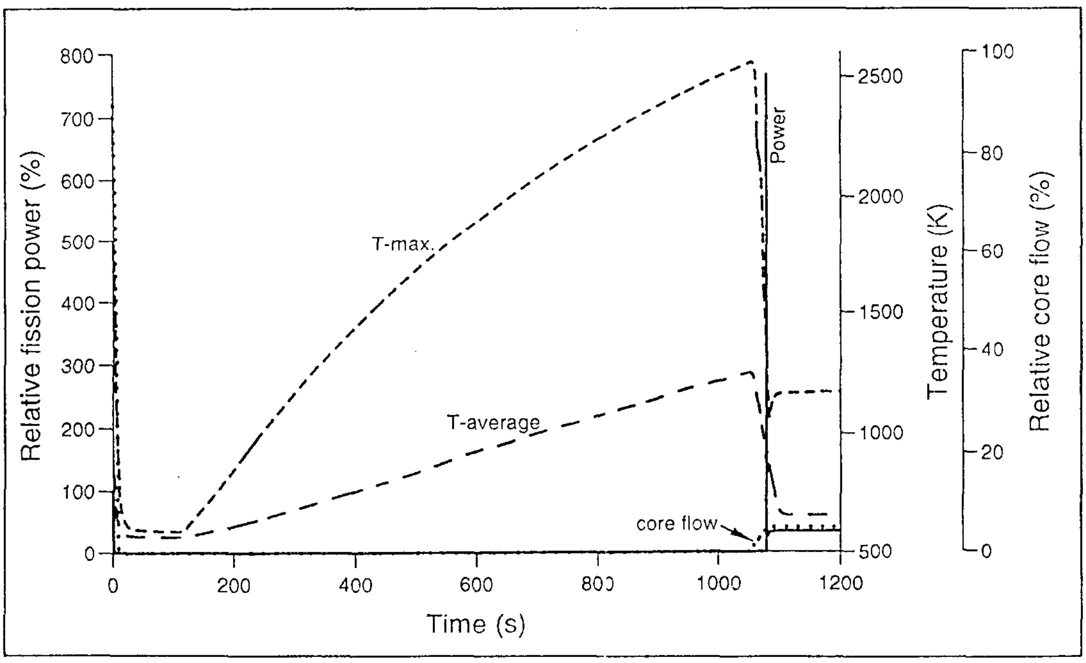


Figure 26. *SIMULATE-3K results for a 4x4 assembly core calculation. Start of reflooding at 1050 s with 5 % of nominal RC flow. Average fuel burn-up 10 MWd/kgU, 25 axial nodes.*

- Parameter studies were made for variations in core mass flow rate (Corresponding to 90, 625 and 2500 kg/s in F-3), fuel burnup (0.5 to 30 MWd/kgU) and number of axial nodes in the core model (12, 25 and 96).

Although the calculations were performed at full pressure, 7 MPa, the results give indications of the course of a recriticality event that can be expected to occur even in a full-size plant, at least for the high-pressure cases studied in the earlier subtask. Plotted results are shown in Figures 27 - 29:

The SIMULATE-3K results can be summarised as follows:

- A recriticality power peak, as highest 22 times nominal power, was obtained in all S-3K calculations within the investigated parameter range.
- The Doppler feedback stopped the power increase rapidly and limited the energy of the power pulse to less than 1 full-power-second in most cases.
- After the power peak, fission power could stabilise at an elevated level, which increased with reflooding rate (Figure 27).
- The results from S-3K were sensitive to axial nodalization (Figure 29). Since recriticality is a very local phenomenon a dense nodalization has to be used.

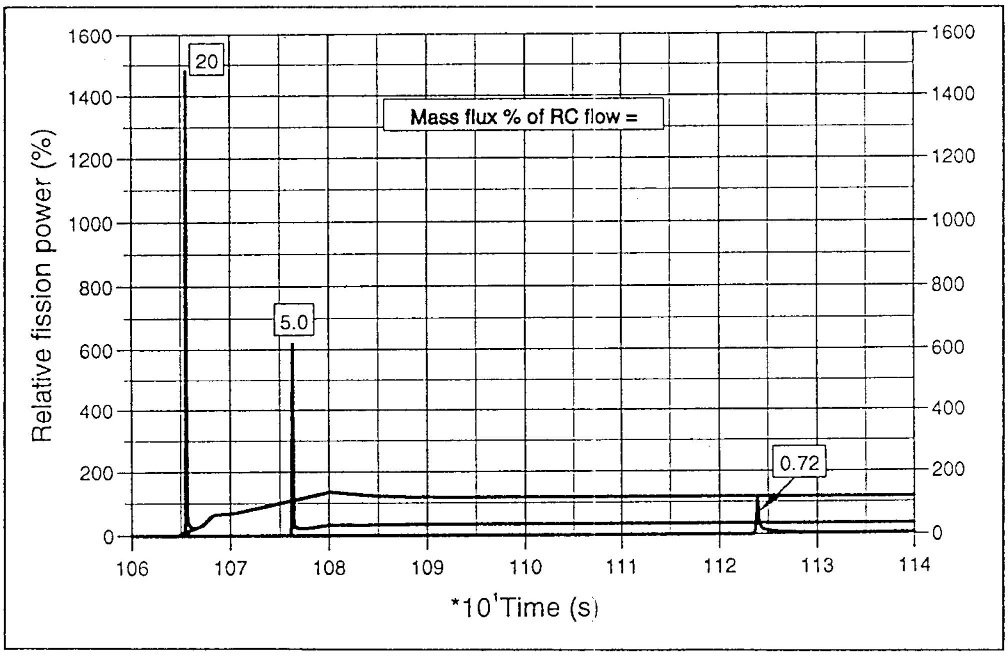


Figure 27. *SIMULATE-3K results for a 4x4 assembly core calculation. Effect of reflooding mass flow rate. Average fuel burn-up 10 MWd/kgU, 25 axial nodes.*

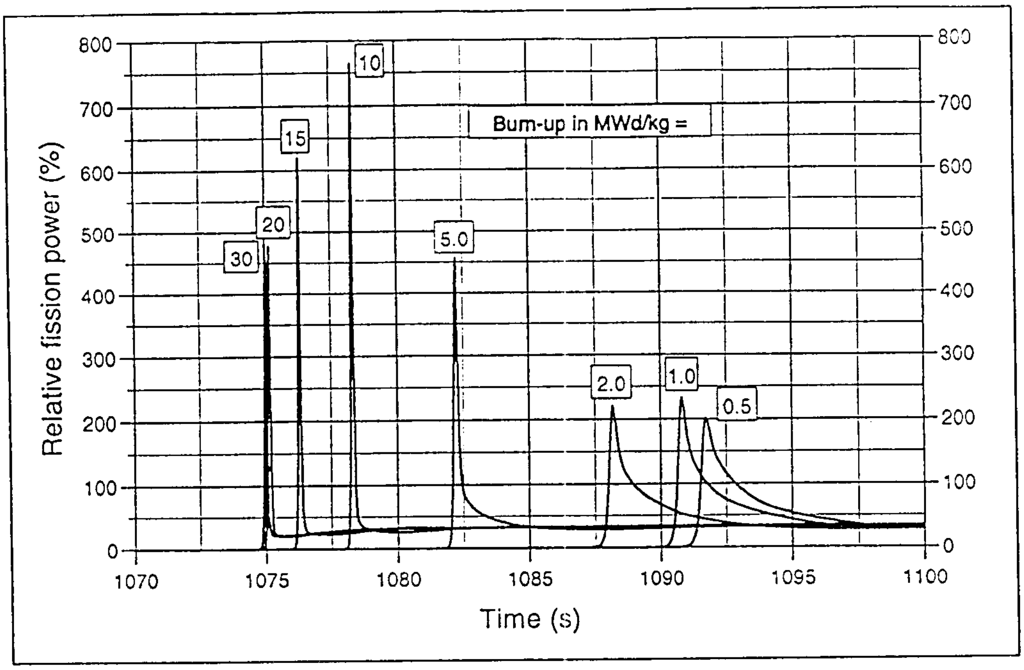


Figure 28. *SIMULATE-3K results for a 4x4 assembly core calculation. Effect of fuel burn-up. Reflooding mass flow rate = 5 % of nominal RC flow, 25 axial nodes.*

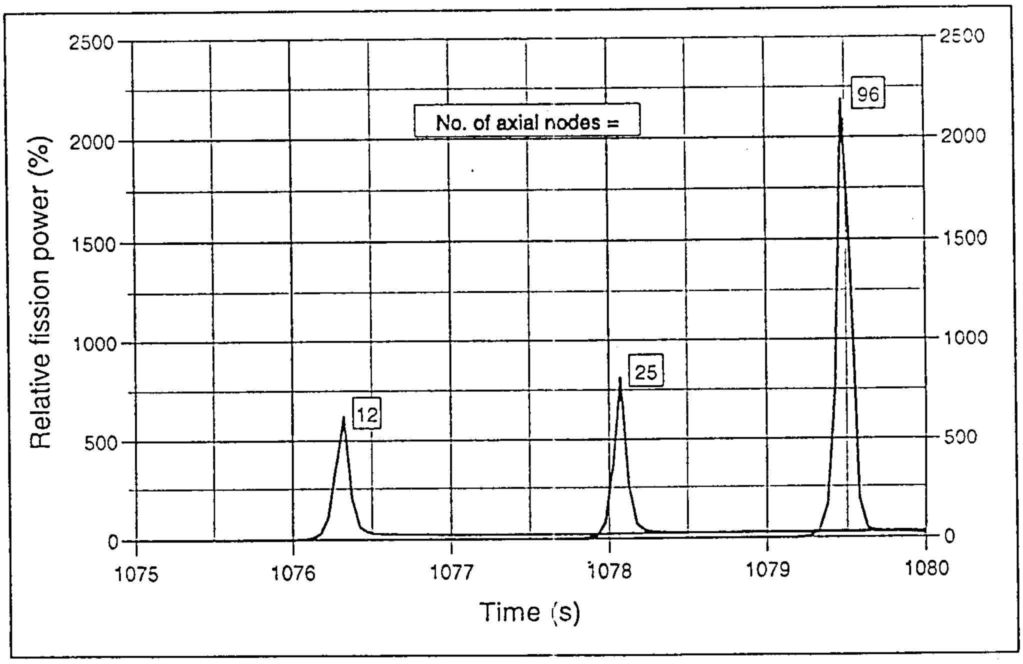


Figure 29. *SIMULATE-3K results for a 4x4 assembly core calculation. Effect of axial nodalisation. Reflooding mass flow rate = 5 % of nominal RC flow, 25 axial core nodes.*

10.2 Results from full BWR core calculations

Several problems were encountered when the transient SIMULATE-3K version was applied on the full plant Oskarshamn 3 model and that has delayed the recriticality studies considerably. The code problems were related mainly to flow and pressure balance instabilities with the 700 parallel flow channels, but also due to other difficulties with the unnormal thermal-hydraulic conditions occurring at the studied reflooding transients. Despite of extensive development work by SOA, the more capable 5-equation was not completed in time to be used in this project. Instead the simpler 3-equation version was improved and tuned to facilitate at least some calculations with simplified assumptions but still with the full plant model.

In the calculation case 1 the following conditions were given:

- transient started from burnup step 11 in cycle 3 with 100% power and a reduced core mass flow 79% of nominal.
- Station blackout was simulated by scramming at 3 s and reducing the core inlet flow to about 1 kg/s during 100 s. The flow reduction was obtained by decrease of the core pressure drop. A rather slow flow ramp had to be used for stability reasons.
- After 175 s control rod melting was simulated by total withdrawal of 59 of the control rods in the centre of the core.
- Reflooding was started by increase of the inlet driving pressure with a slow ramp from 170 to 230 s which increased the mass flow rate to about 175 kg/s.

The flow boundary conditions are depicted in Figure 30. No depressurisation was simulated, nor was there any noticeable core heat-up before reflooding. The reason why the normal way of specifying the flow transient was not used, i. e. by a time-versus-flow table, was that this gives an unrealistic flow distribution. The flow is then distributed in the same way as in the steady-state case, which introduces an excess amount of water into central core channels and causes flow inbalance in the continued calculations.

The increased water level and reduction of core void fraction lead, however, to an immediate increase of the reactivity resulting in increased power and fuel temperatures (Figure 31). The initial power peak reached only about 220 MW before the Doppler feedback brought the total power back to 100 MW, just a little more than the decay power. With continued reflooding and a constant mass flow rate of 175 kg/s the power started to increase again in an oscillatory way until it stabilised at about 450 MW, or 15 % of full nominal power.

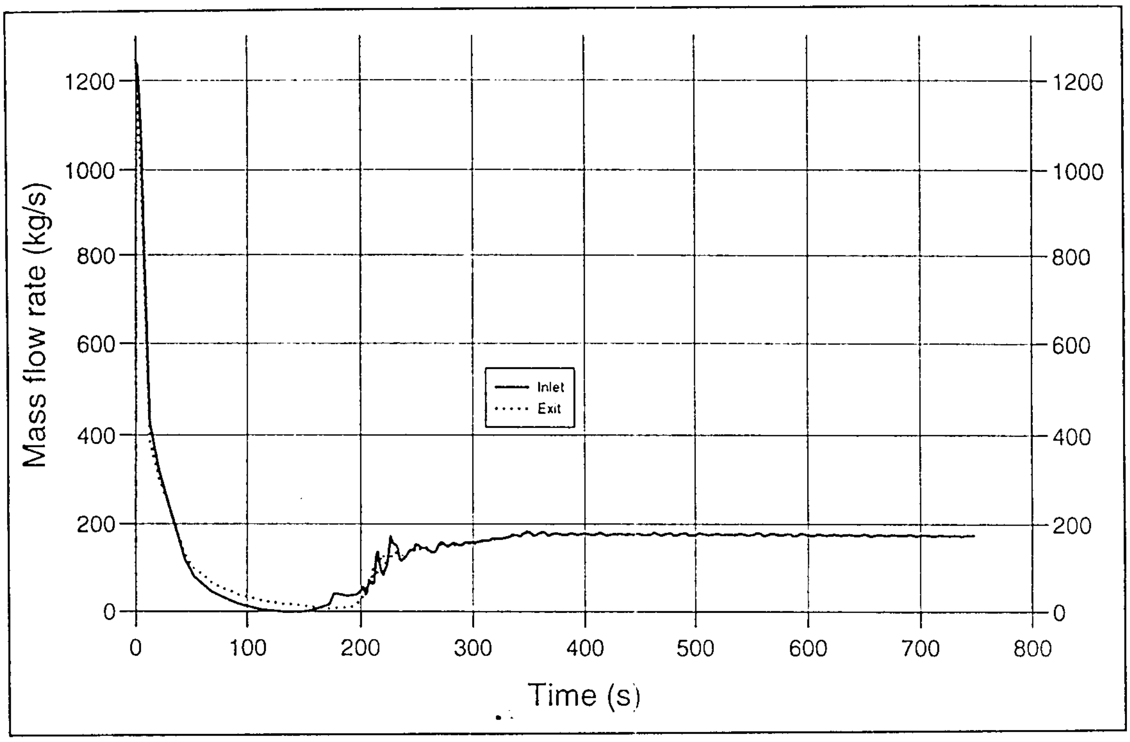


Figure 30. *SIMULATE-3K calculation case 1 for Oskarshamn 3. In- and outlet mass flow transients.*

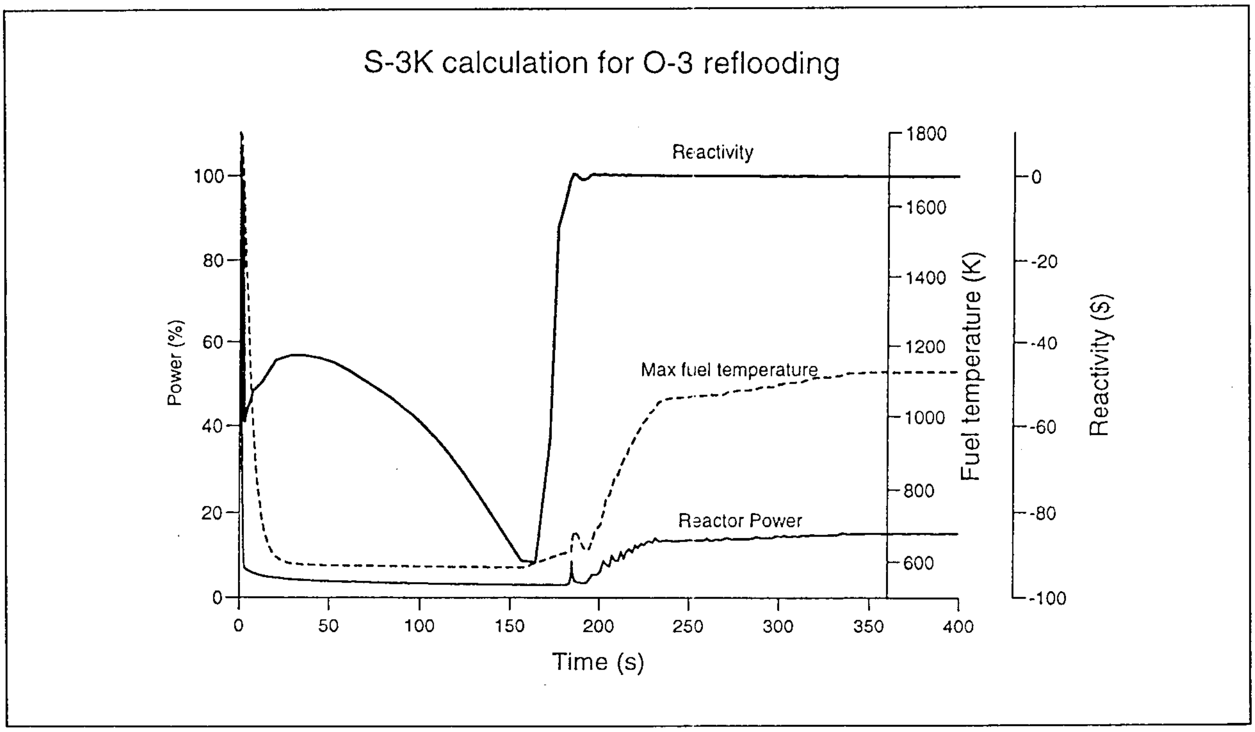


Figure 31. *SIMULATE-3K calculation case 1 for Oskarshamn 3. Reactivity, power and temperatures in the core.*

In next calculation case, No. 2, the same initial condition were assumed as in case 1. The transient boundary conditions and the resulting power response are shown in Figures 32 and 33, respectively.

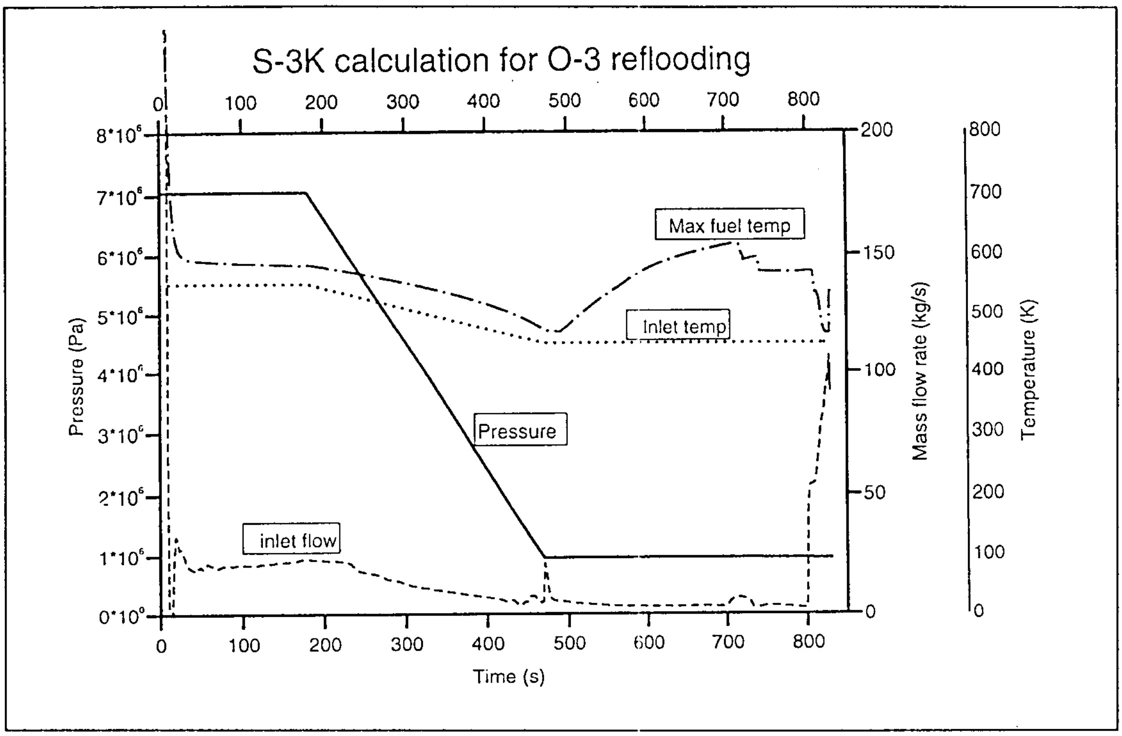


Figure 32. *Simulated transients for pressure, core inlet flow and inlet water temperature in Oskarshamn 3 calculation case 2.*

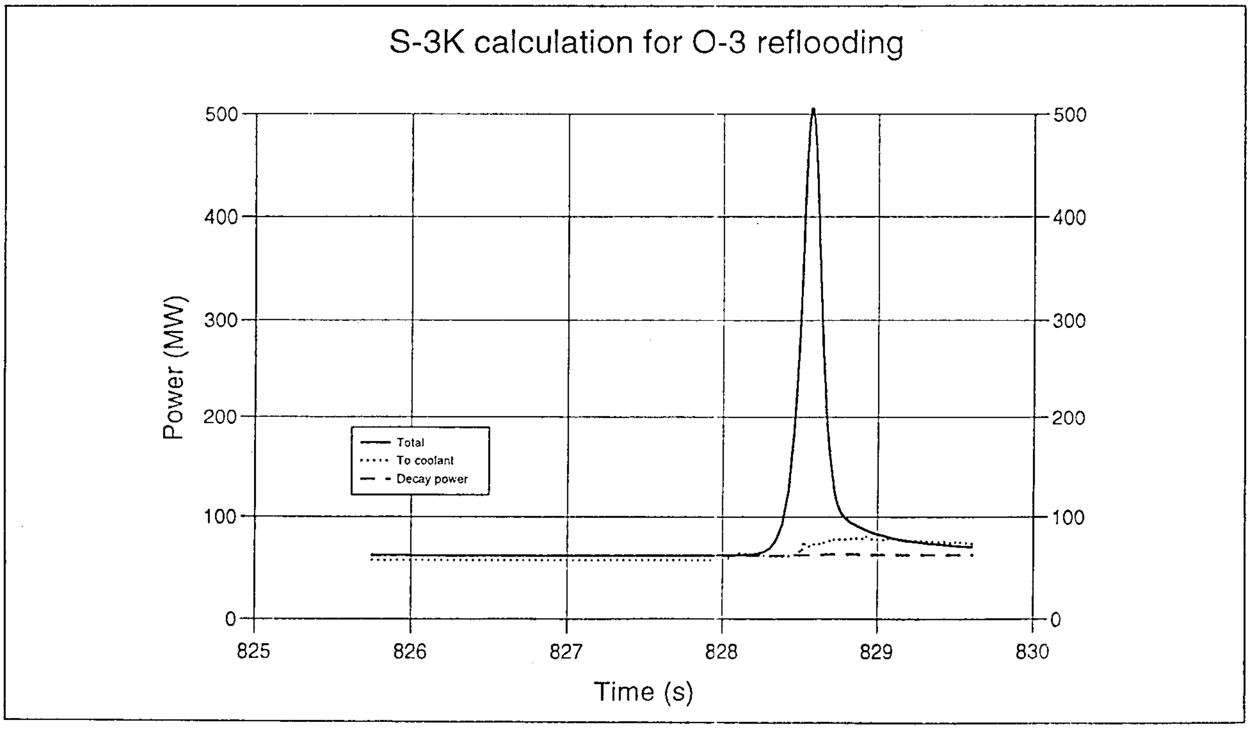


Figure 33. *Power transient with recriticality peak obtained with SIMULATE-3K for Oskarshamn 3 calculation case 2.*

Depressurisation to about 10 bar was simulated with a slow ramp after the reduction of the core inlet flow. A decrease of the inlet temperature was prescribed, as well, when the reflooding was initiated, in order to model the lower temperature of the auxiliary feed water. More realistic reflooding conditions, e. g. including larger core heat-up, were not possible to obtain due to difficulties to get a stable solution with current code version. The calculations could, however, include the first recriticality power peak, and even the turn-around due to Doppler feed-back.

11. RECRITICALITY ANALYSES WITH APROS

11.1 Scoping Studies

As the initial phase of the scoping studies the APROS 3-D BWR core model was brought to full power and the axial and radial power profiles were compared to those obtained with another code at TVO. The main purpose of this comparison was to check that there were no obvious input errors in the APROS input and that the neutronics and thermal hydraulics models of APROS were able to calculate the full BWR core at steady state. The comparison indicated quite a good agreement with the APROS results and the proprietary results of TVO. At the power level of 104 % the fuel average temperature was 517 °C, coolant average temperature 285 °C and average void fraction 47 %. In order to test the transient calculation capability of the APROS model and the principal applicability of the control rod 'fading' that was proposed as a first approximation to describe the control rod melting a test transient was performed in full power. In this transient most of the control rods present at full power at steady state were simply driven out of core in one time step. The transient resulted, as expected, into a high power peak and stabilization of power into a level above the initial power level.

The calculation speed of the TVO core model was studied with full neutronics and either one or 250 thermal hydraulics channels. The results in Table 6 indicate that the description of the core with 250 thermal hydraulics channels is still feasible for the recriticality studies from the viewpoint of needed CPU time.

Table 6. *APROS Olkiluoto BWR 3-D core model. Calculation speed with HP-9000/780 (HP C-180). Time step 0.1 seconds. Steady state. Full 3-D core with 500 assemblies and 12500 neutronics nodes.*

Case	Simulation time/CPU time
250 thermal hydraulic channels 6250 5-eq. thermal hydraulic nodes	0.06
1 thermal hydraulic channel 25 5-eq. thermal hydraulic nodes	0.15

In the scoping studies the core coolant flow was gradually reduced to zero. The reactor operated then at decay power for about 1000 seconds. The cladding temperatures reached the melting temperature already after 600 seconds in the initial model. The largest deficiency of the model was that it had only 14 control rods in the original steady state configuration.

In the scoping study most of the control rods were rapidly faded away to describe the control rod melting according to Figure 34. After fading of the control rods, reflooding to the core with various flow rates was started. Reflooding with flow rates of 10, 50 and 100 kg/s and with inlet water temperature of 50 °C was studied. With the flow rate of 10 kg/s and 50 kg/s a very high power peak was produced. The power increase was ceased by the Doppler effect. The power peaks due to recriticality are shown in Figures 35 and 36. Figures 37 and 38 show the fuel temperatures and the coolant density at the location of original maximum power, respectively.

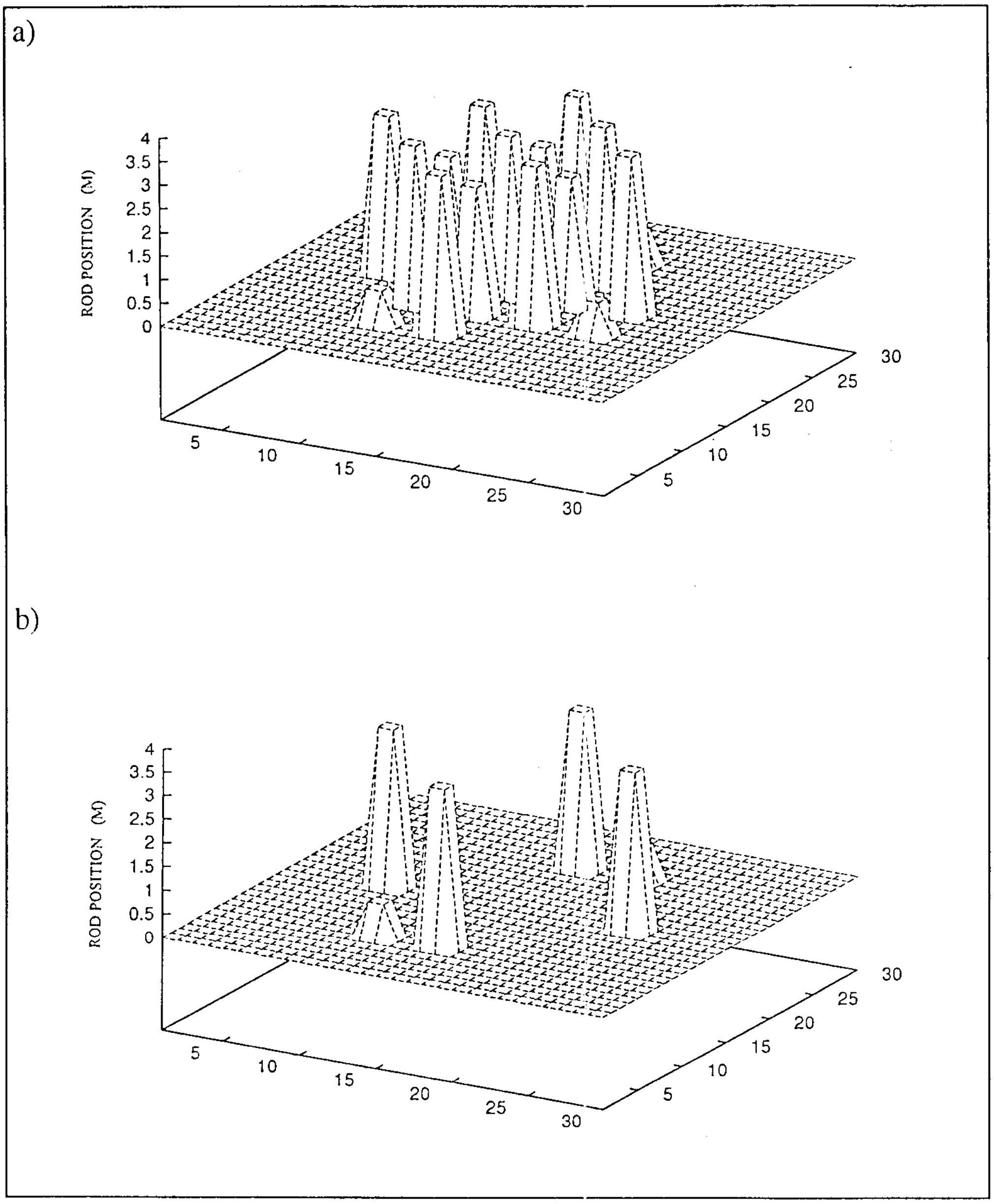


Figure 34. *Control rod melting in scoping studies. a) rod pattern at initial state b) rod pattern after 'melting'.*

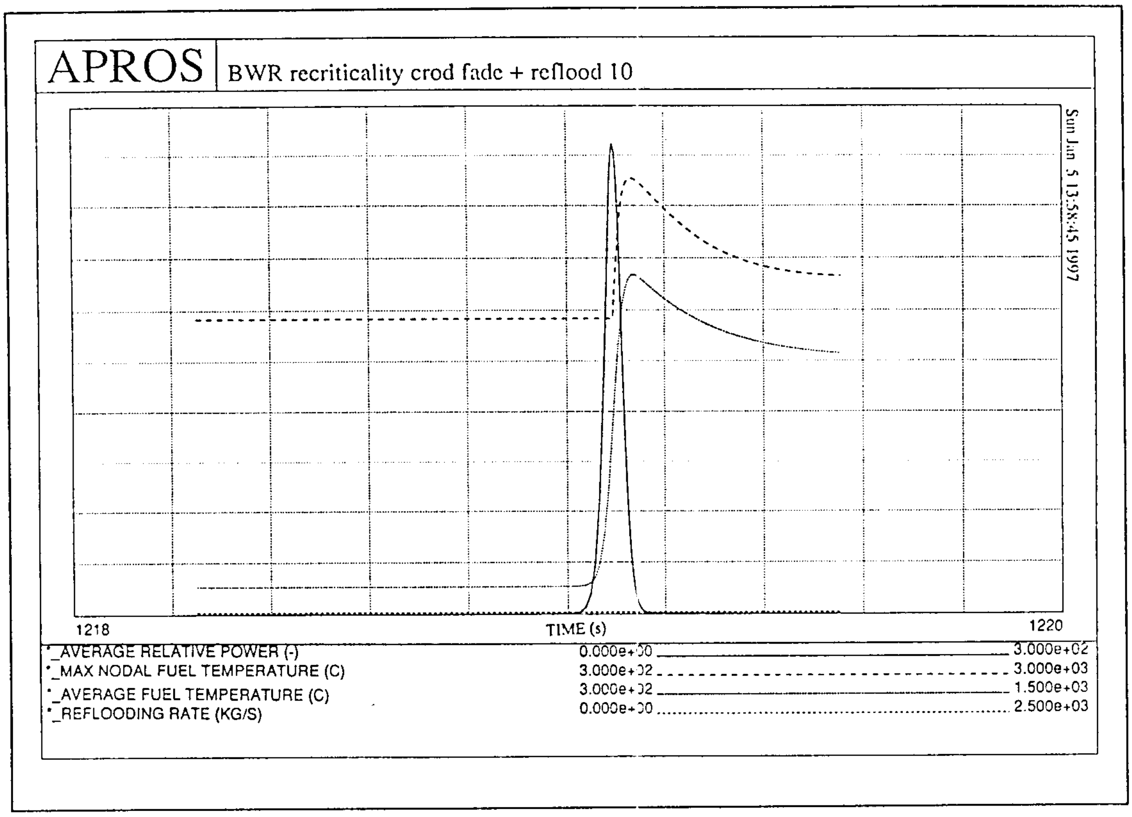


Figure 35. *Power peak produced at scoping study with reflooding 10 kg/s.*

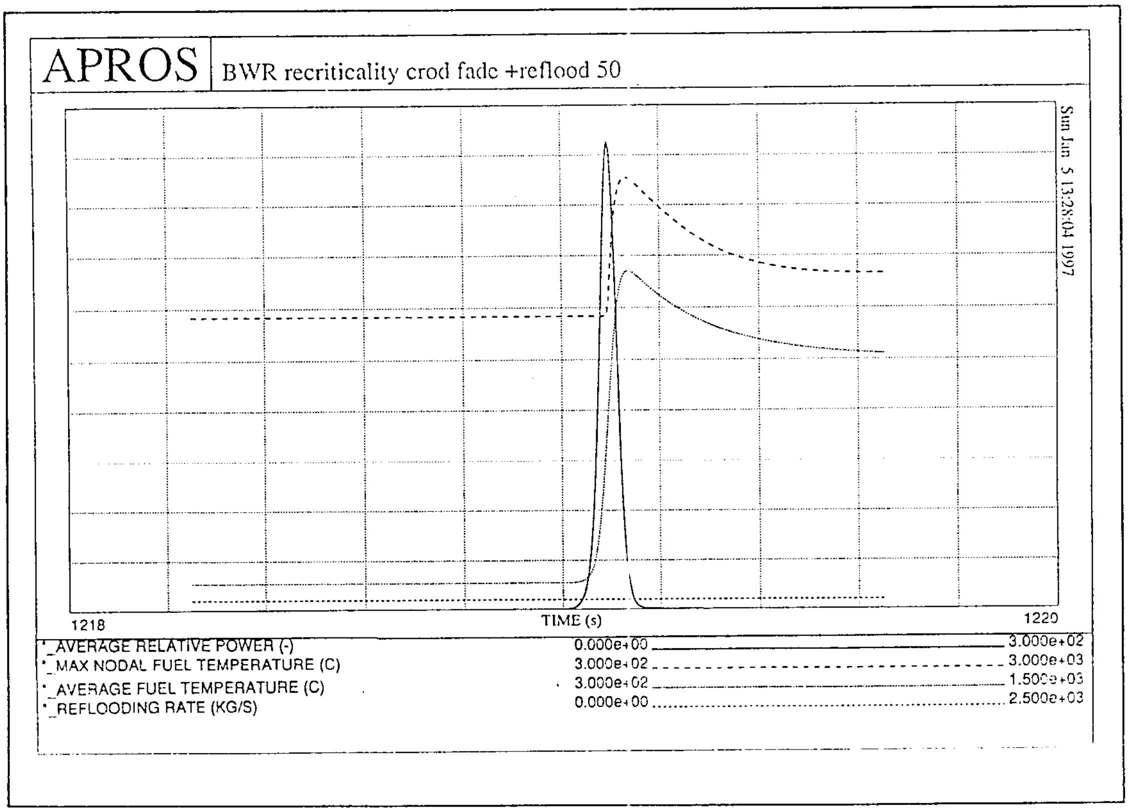


Figure 36. *Power peak produced at scoping study with reflooding 50 kg/s.*

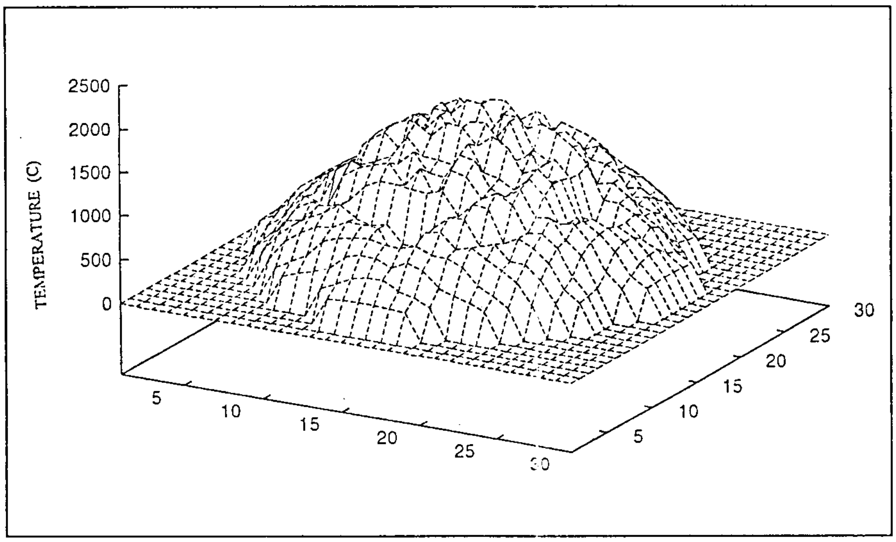


Figure 37. *Fuel temperature at the position of original maximum power during recriticality transient.*

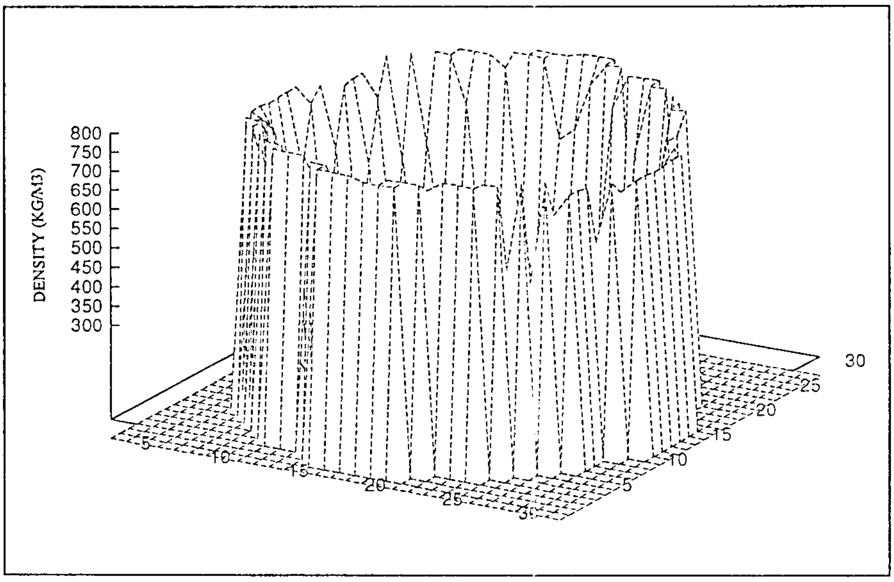


Figure 38. *Coolant density at the position of original maximum power during recriticality transient.*

When the reflooding rate was increased to 100 kg/s, the code was again able to calculate the power peak, but soon thereafter numerical problems were met. The initial conditions with the assumption of the control rod pattern without scram was not very physical, since practically no control rod material was left in the core after the 'melting', and this explains the very high power peak reached in the scoping study.

The most important result of these initial studies was that the neutronics and thermal hydraulics of APROS code were able to calculate abrupt power excursions.

11.2 Initial state for recriticality

In the actual recriticality study the reactor was first scrammed. Then the flow into the core was reduced to zero and the reactor was allowed to operate at decay power for 1000 seconds. During the decay power operation the fuel and cladding temperatures increased due to lack of coolant flow. Figure 38 shows the increase in the void fraction, in the average fuel temperature and in the maximum nodal fuel temperature and the decrease in the average core coolant density during the heatup period of the core after the coolant flow was turned off. APROS predicted uniform conditions in all channels during the core heatup, as can be seen from Figure 39. Figure 40 illustrates the nodal fuel temperature in the sixth axial node from the bottom of the core at the end of the heatup period.

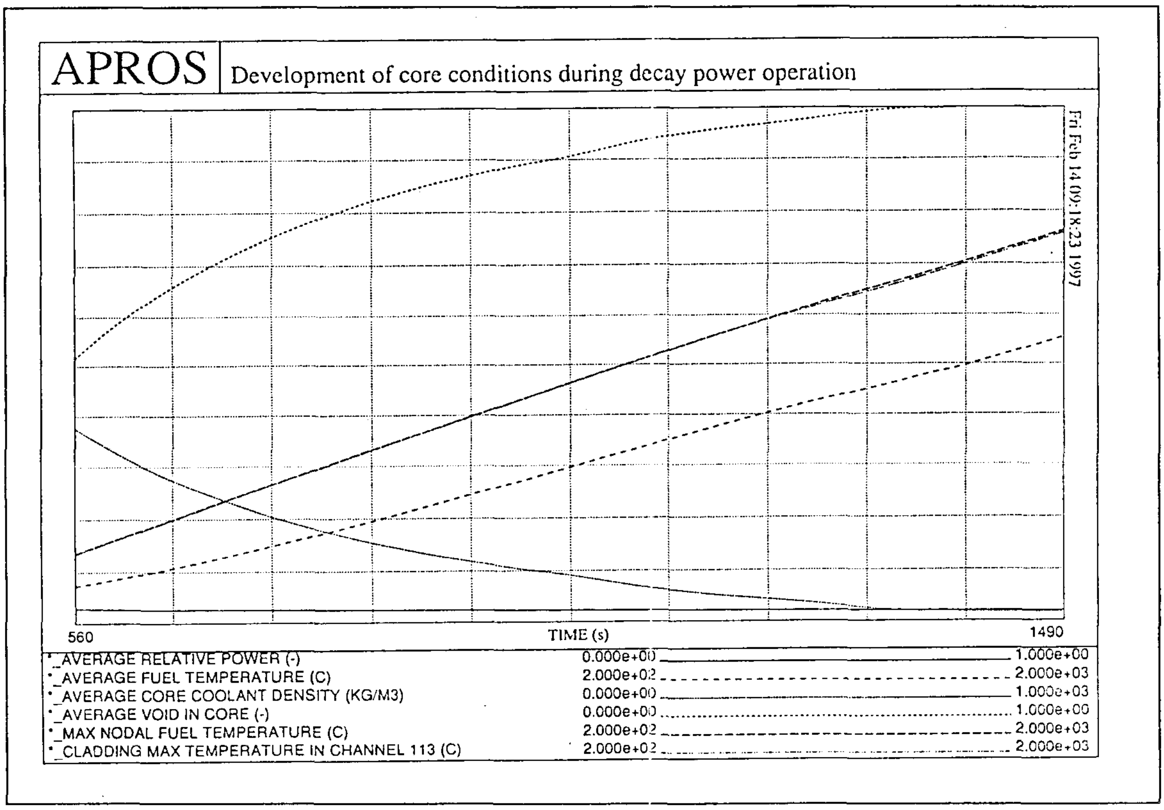


Figure 39. *Initial state for recriticality.*

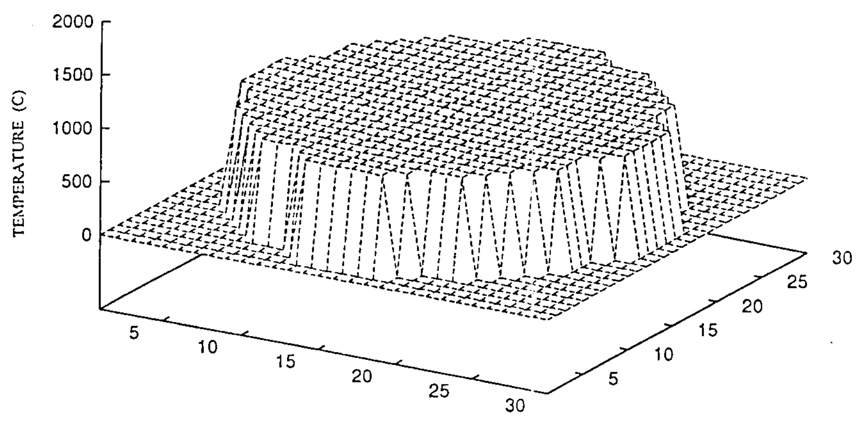


Figure 40. *Nodal core temperature in the sixth axial node from the core bottom.*

11.3 Recriticality analysis

In the recriticality analysis the core was first at decay power and the hot conditions prevailed in the fuel and cladding. The normal operational pressure was maintained in the core inlet and outlet boundary condition nodes. The control rods were suddenly faded away from the central part of the core. Two control rod fading patterns, indicated in Figures 41b and 41c and corresponding approximately 30 % and 65 % of melting of the control rods were studied. Two reflooding rates, a small capacity of 50 kg/s and a large one gradually increasing to 1350 kg/s with the inlet water temperature of 40 °C were studied.

The basic alternatives of coolant entering the core only from below or as combined injection from below and through the core top spray were investigated. In both cases numerical problems with the thermal hydraulic model were encountered and the code calculations stopped soon after the power peak occured. At the beginning of the reflooding period a timestep size of 0.1 seconds was used. The timestep was reduced down to 0.001 s at the time of power peak, but in spite of the small time step, numerical problems were encountered.

Due to the preliminary nature of the calculations, all APROS runs were performed using the neutronics data set prepared for the steady state operation. The application of this data set can be considered valid only up to scram in these studies and considering the validity range of the data set, a high pressure case was selected for the study. However, the application of this data set for the recriticality studies in hot, dry core, where part of the control rods have been removed and the control rod positions are filled with steam instead of water, adds numerical problems and uncertainty of results.

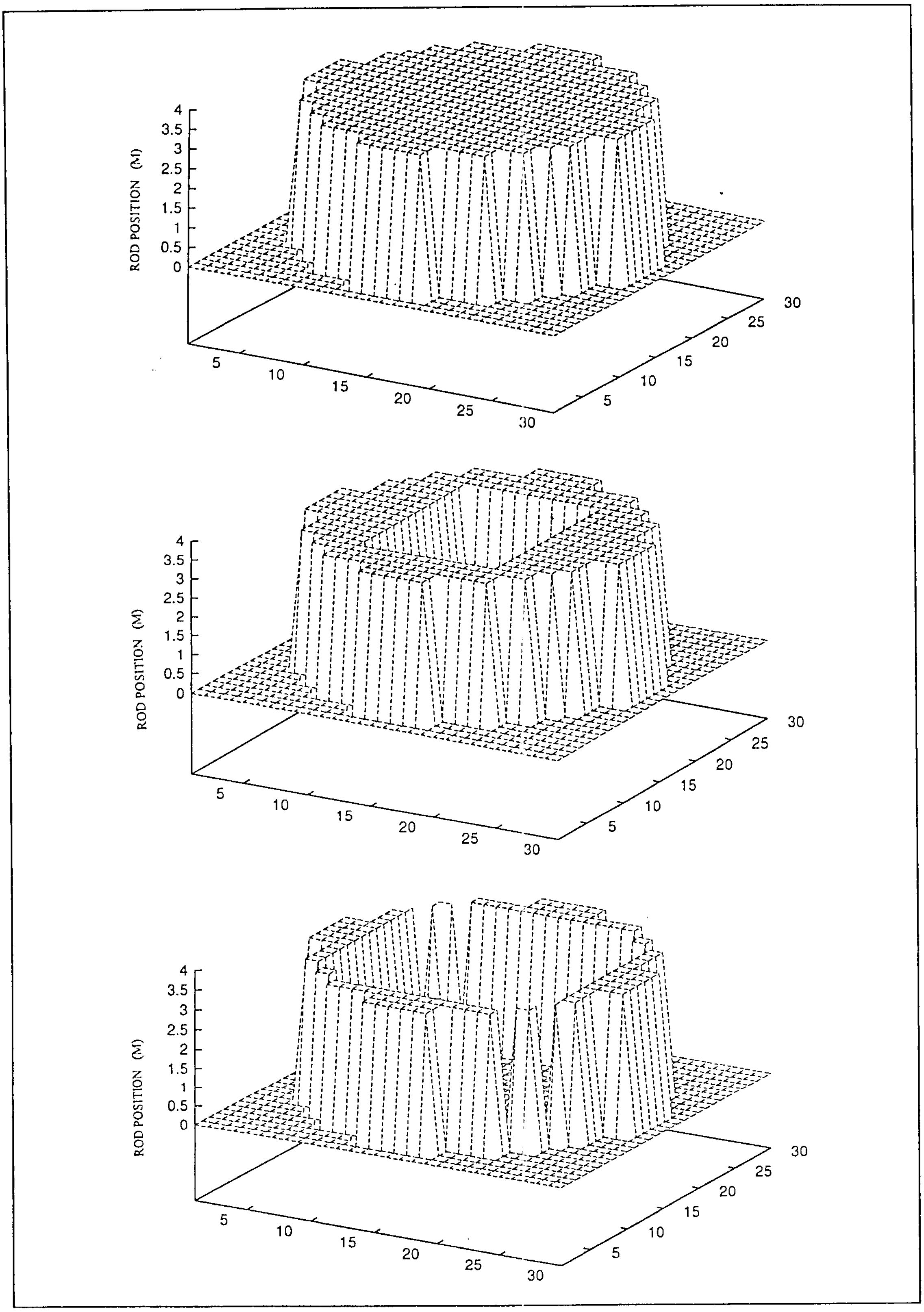


Figure 41. *Control rod fading patterns in recriticality study. a) No melted rods, b) 30 % melting, c) 65 % melting.*

The main results of the recriticality study with reflood rate of 50 kg/s and 65 % of control rods melted is shown in Figures 42, 43, 44. Figure 42 shows the decrease of the core average fuel temperature and the core average coolant temperature during the period from start of reflooding up to the time of power peak. The power peak caused by reflooding is also shown in Figure 43.

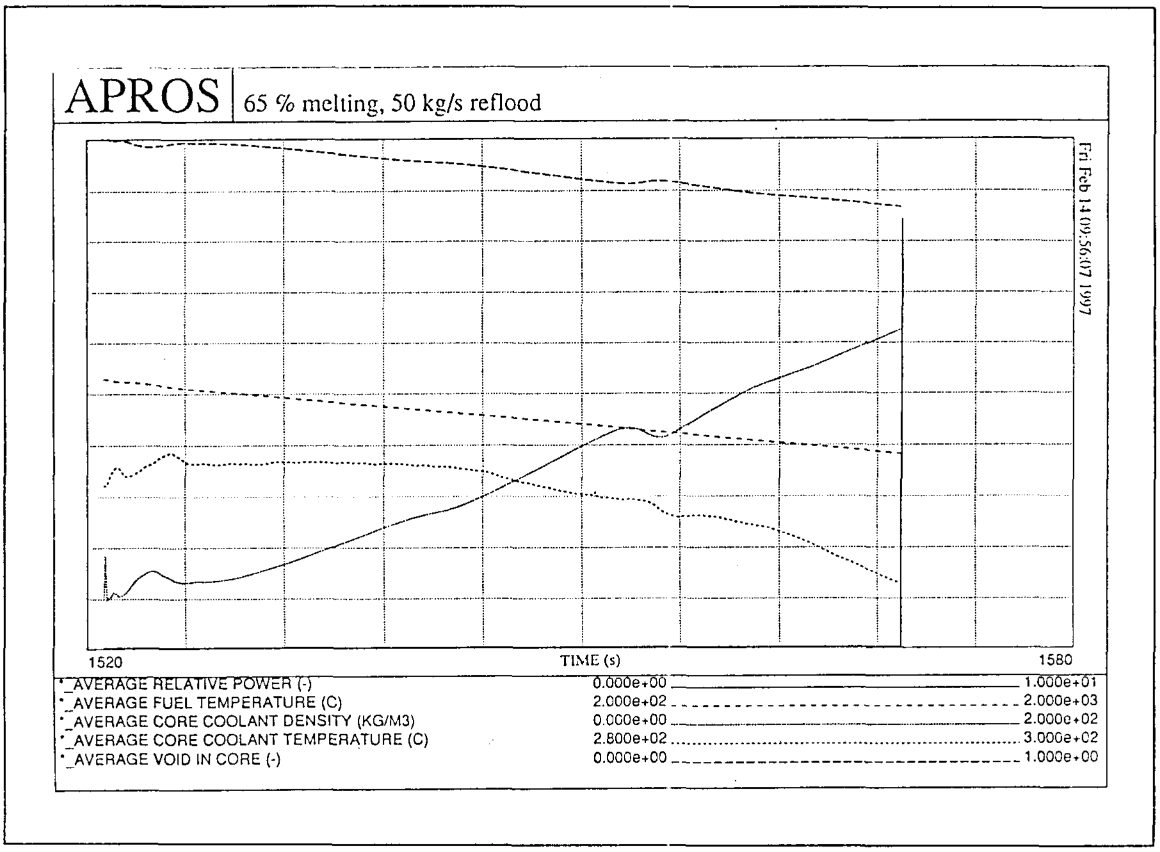


Figure 42. *Relative power, average fuel temperature, core coolant density , temperature and void fraction predicted by APROS.*

Figure 43 shows the behaviour of the nodal fast fission flux peaking factor, nodal maximum fuel temperature and effective multiplication factor. The effective multiplication factor increases in a stepwise manner, when reflooding is turned on. Thereafter a small, rather constant increase will take place until recriticality. The maximum nodal average temperature decreases during the period from start of reflooding till the time of power peak. At the time of recriticality the code predicts an extremely high peaking factor in nodal fission power.

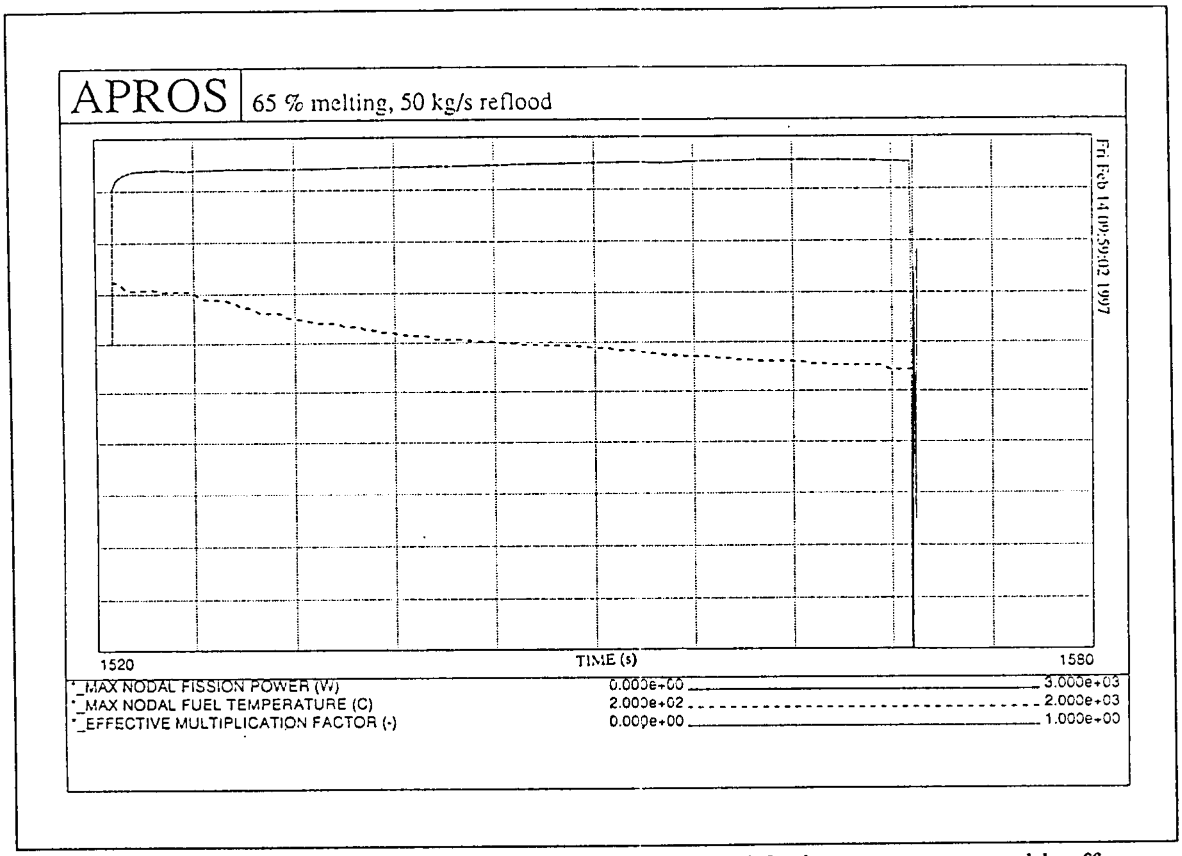


Figure 43. *Maximum nodal fission power, maximum nodal fuel temperature and k-eff predicted by APROS.*

Figure 44 shows the behaviour of the cladding maximum temperature in two channels of the core model. The channel 1 is located at the outer rim of the core, where the control rods are assumed to remain intact and channel 113 is located in the central part of the core, where control rods are assumed to have completely melted. At the beginning of the analysis the control rods have just been taken away and the reflooding is starting. Thus the cladding temperature is similar in both channels. It can be noticed, that in both channels the cladding temperature exceeded the melting temperature of control rods and thus the assumption of control rod melting only in the center of the core was not in agreement with the temperature field predicted by the code. Furthermore, on the basis of the predicted cladding temperatures, the cladding oxidation should have been considered in the analysis. The cladding maximum temperatures decrease during reflooding in both channels. The temperature decrease in the channel 113 without control rods is smaller than in the outer rim channel 1, where the control rods are assumed to remain intact.

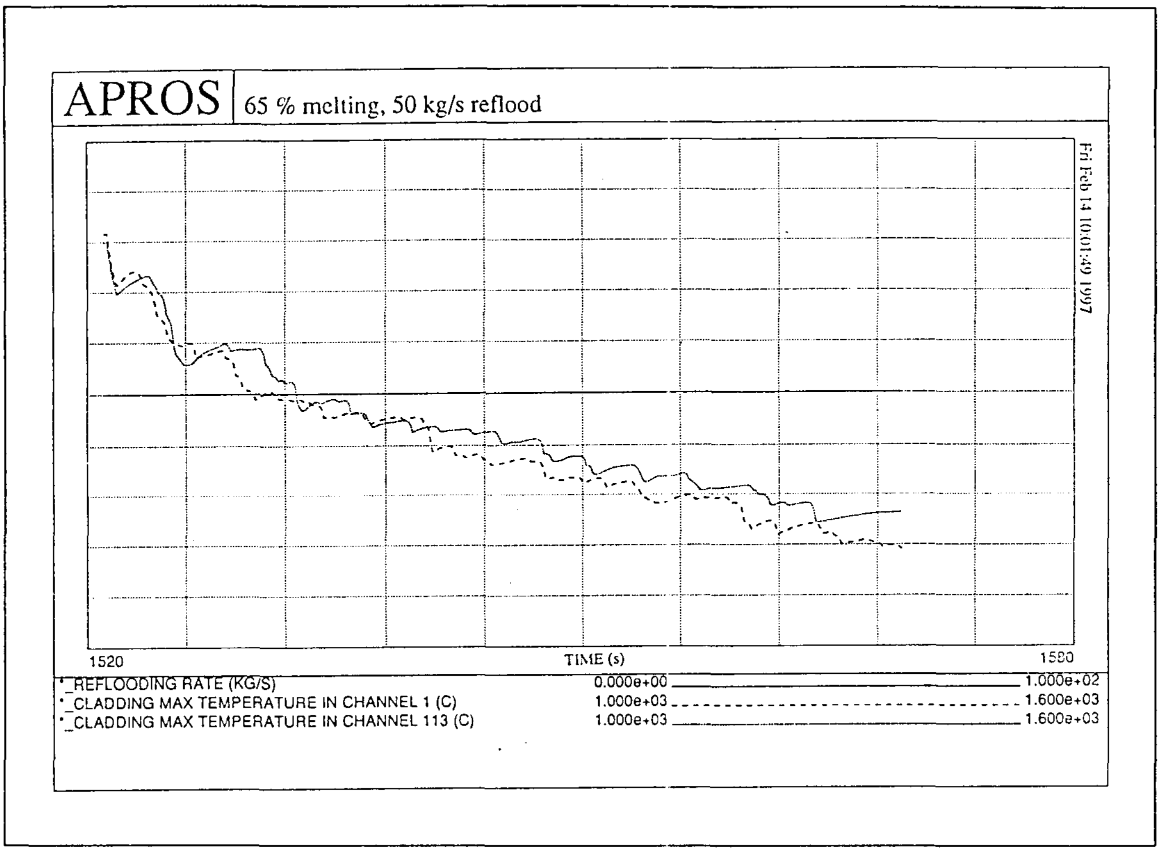


Figure 44. *Cladding maximum temperatures in channels 1 and 113 and reflooding rate predicted by APROS.*

With the larger reflooding rate numerical problems were met already at the start of the reflooding. The maximum reflooding rates reached without numerical problems were of the order of 900 kg/s.

12. SCOPING STUDY ON CONTAINMENT RESPONSE TO PRESUMED RECRITICALITY

A scoping study on the consequences of presumed recriticality was performed with the MAAP4 code. The reactor was presumed to stabilize at the power level of 20 % from full power after reaching rectiticality during reflooding with all circuits of the systems 323 and 327 at 2 h, when almost all the absorbing material had relocated, but fuel geometry was still intact. The fuel started to disintegrate during reflooding, when two-phase level was 2.1 m above the bottom of active fuel and reactor power was 73 MW (3.4 % of full power). The core collapsed totally about 15 min later, when the core was already covered by two-phase level. The reactor power was 270 MW (12.5 % of full power). This result suggests that the core fuel geometry cannot be maintained with the used assumptions of core recriticality power when the core is reflooded late with close to 100 % of control rods melted.

In the following calculations a degraded core was reflooded when circa 60 - 80 % of absorbing material had relocated but fuel geometry was intact. Reflooding took place with all circuits of the systems 323 and 327. Simultaneously all the circuits of the containment vessel spray and cooling system 322 started. After 5 minutes from the start of reflooding three feedwater pumps were started. After the initial water inventory in condenser had been consumed the water supply to reactor continued with 327 and 323 pumps. The assumed thermal power of the reactor is illustrated in Figure 45.

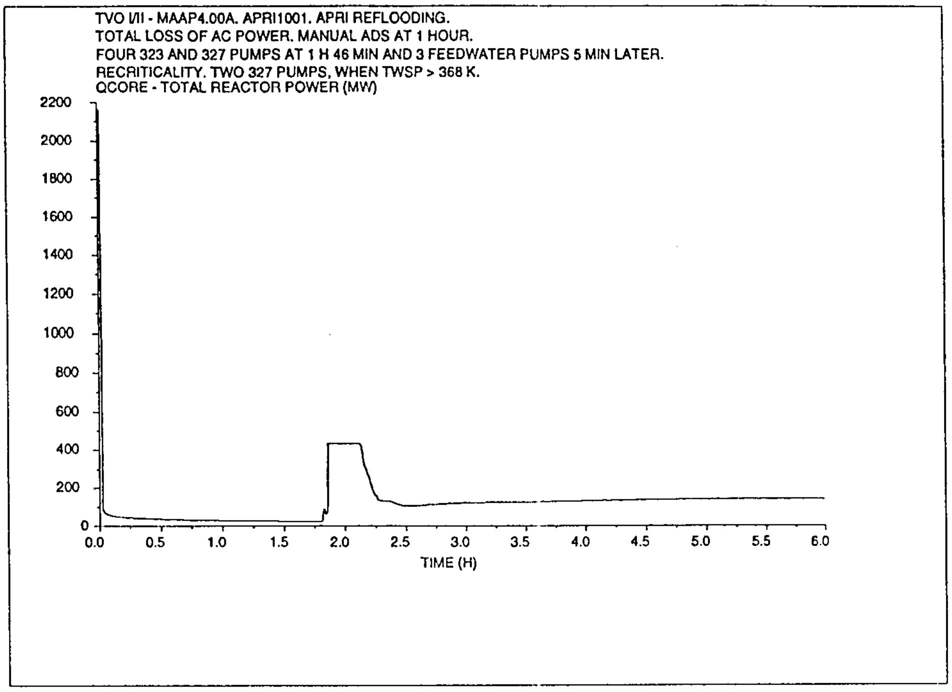


Figure 45. *Postulated reactor power history in a reflooding/recriticality case. Reflooding is started at 1h 46 min.*

The 322 and 323 were assumed to fail at condensation pool temperature of 368 K. This temperature was reached in 12 minutes. After that water supply to reactor continued with two 327 pumps.

The drywell pressure of 6 bar was reached in 2 h 30 min from the start of reflooding and the filtered venting line from wetwell was supposed to open. The capacity of the venting line was too small to prevent containment overpressurization. Containment pressure of 9 bar was reached in about 3 h from the start of venting. According to the containment analyses drywell pressure should be kept below 7.5 bar to have sufficient safety margins. This pressure was exceeded in 2 h 40 minutes from the start of reflooding. The containment pressure is shown in Figure 46.

When water was supplied to reactor with only one 327 pump after 323 had failed the containment pressurization took place slower. Containment pressure of 9 bar was reached in about 6 h from the start of reflooding.

These results suggest that there is enough time for the emergency boration to prevent containment failure. Two boron system pumps require 45 min and one pump 90 min to ensure subcriticality.

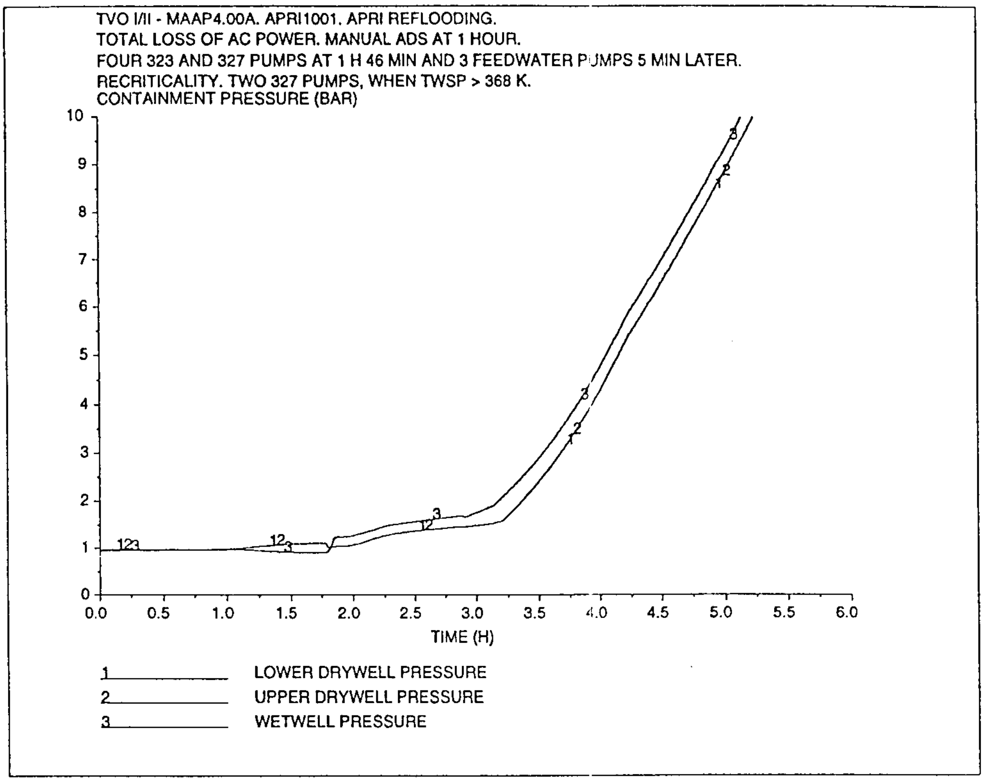


Figure 46. *Containment pressure history in case of presumed recriticality and elevated power level of 20 % of nominal power. MAAP4 prediction of TVO I/II. Reflooding with 4X 323 and 4 x 327 + feedwater 322 5 min after initiation of reflooding.*

If the core spray system 323 is the only system available for reflooding the critical factor for continuous operation of the pumps will be the condensation pool temperature. The coolant flow injected by the system 323 to the reactor oscillated due to the oscillations in the reactor power and pressure. In parallel to reflooding with 323 four 322 pumps and heat exchangers began to cool the containment. Suppression pool temperature reached the value of 368 K in about 0.5 h and was saturated 6 min later. The maximum containment pressure during this time is 1.2 bar.

When only one 323 pump was in operation after suppression pool temperature exceeds 333 K the condensation pool temperature increases a little slower.

The presumed recriticality leads to high suppression pool temperature and loss of core spray system 323, if boron is not injected to reactor from system 351 to shut down the reactor. In case of reflooding with 323 and subsequent recriticality the available time for boron injection is at least 30 min before the suppression pool temperature exceeds 95 ^{0}C and the risk for pump cavitation and loss of the system 323 increases. The capacity of the boron system 351 is sufficient to reduce reactor power fast enough to keep the core spray system in operation. The containment integrity is not threatened due to recriticality, if reflooding takes place only by core spray system 323.

If the initiating event is a feedwater LOCA and loss of reactor cooling, the control rods start to melt early, in about 20 minutes from the beginng of the accident. If the core is reflooded with all safety systems at 37 min with the augmentation of three feedwater pumps 5 min after and the recriticality power (20 % of full power) is taken into account, the containment is pressurized to 6 bar and the wetwell venting line is started in 3 h. The capacity is insufficient and containment pressure of 7.5 bar is reached at 30 min later.

In the feedwater LOCA case the boron will be diluted in the whole containment water inventory because the feedwater nozzle is 1.8 m above the top of the active fuel and thus the automatic control of the safety system would continue pumping water through the break because the water level would never be high enough. To prevent this, the pumps would have to operated manually. If the boron is diluted in the whole containment water inventory, the boron concentration would not be sufficient to ensure subcriticality in core.

13. DISCUSSION OF RESULTS

The presented analyses with all three codes should be considered tentative. However, the trends in the results seem reasonable and encourage to continue the development and analyses with all three codes. The three codes differ quite much in respect of their scope but their parallel use enhances the general understanding of the recriticality phenomenon. SIMULATE-3K has in the background extensive reactor core applications around the world and currently new transient characteristics of the core are being developed. APROS is a full range nuclear plant analyzer for various reactor incidents. RECRIT is a code specialized on the BWR LOCA phenomena.

The obtained results should be considered in light of other recent publications [1], [2], [3], [4], [5]. When comparing the core power predictions obtained in this study one recognizes that the present version of RECRIT code with TVO fuel data results in higher stabilised power levels than Scott et al. suggest [1]. SIMULATE-3K, however, predicted power levels in Oskarshamn 3 case that are in quite a good agreement with the work of Shamoun et al.

The investigation of the TVO fuel data (Fig. 1) with RECRIT suggests that recriticality would be possible if the coolant void fraction is below 60 % and very little (< 5%) of control rods have remained in the core. This void fraction limit is much higher that what Shamoun et al concluded in their work [2]. On the other hand, according to Fig 1 one could deduce that recriticality is impossible if at least 50 % of control rods are left in the core. This renders the result by Okkonen et al. conservative, since they propose that recriticality is possible if at least 1 m height of core (corresponds to 27 % of control rods melted in TVO core) is without absorber material.

The results of the SIMULATE-3K calculations of the postulated reflooding transients should be regarded as tentative only, since full plant calculations had to be carried out with much simplified assumptions. The calculations on a reduced core with only 4x4 fuel assemblies did, however, give some interesting indications, which probably are applicable also for a full size core.

The preliminary results indicate that an initial and rather abrupt reactivity increase is obtained soon after initiation of reflooding as the void fraction is reduced. The results from the mini-core calculations show that the magnitude of the power peak can be several factors of ten times the total nominal power. Since the recriticality is very local, the peaking factor of an axial node can reach still higher levels. The duration of the power peak was, however, short so that the added energy had only minor effect on fuel heat-up. The same results were indicated from the few calculations that could be carried through with the full-scale O-3 model.

What is of greater concern is the indication that continued reflooding can give rise to a more, or less, constant, elevated fission power up to 50 per cent of the nominal power. This gives a steam production, which can be a threat to the integrity of the containment, and which has to be further evaluated.

The RECRIT code results presented in this report are only for the BOC core. The full range parameter variation with different nodalization for the core and with different fu el burnup states should be analysed, before any further conclusions can be drawn. The most important parameters are the reflooding rate and the core burnup. The results suggest, that with all reflooding flow rates the first power peak increases the average fuel temperature by 0 - 240 K and the maximum temperature by 0 -650 K repectively. In addition to this, later power generation should be considered as well.

The RECRIT experiences show, that the further development of the TWODIM -code from its stationary version to the dynamic version has been useful. The code can handle short duration and spatially very local power peaks. The thermohydraulic development carried out at VTT based on earlier NORCOOL-1 and SMABRE experiences has been successful in producing an emergency core cooling calculation model for BWR plants with a reasonable accuracy in the physical description and with realtively easy-to-use user interface.

In the coupling of neutronics and thermohydraulics models a number of unexpected problems were encountered. One reason for this was that the developer of the thermohydraulics model had not initially realized the complex physical mechasnisms related to the recriticality, although the physics of the reflooding was well known. In addition, the merging of two different codes into one was itself quite an extensive task. But the final code package may be considered as a success and there are good chances to get full range, good quality results in the near future.

The recriticality analyses performed with APROS code were of a preliminary nature as well. The advantage of APROS was that it was possible to perform the studies with the full core of a BWR plant using realistic and detailed input data. The recriticality studies were all performed with the alone-standing core model. The core model operated well in the steady state. The studies performed indicated that it is possible to bring the APROS BWR core model into plausible initial conditions of recriticality studies.

In the scoping studies large, but narrow power pulses were obtained with various reflooding rates. The power increase was ceased by the Doppler feedback, and the code was able to continue the analysis with small reflooding rates.

In the actual recriticality analyses numerical problems were met at the time of power peak. In both the scoping studies and in the actual recriticality analyses the neutronics data set and the thermal hydraulic models and correlations were those valid for the normal operational transients. Thus the numerical problems were not quite unexpected. The qualitatively reasonable results obtained in the scoping studies can be largely explained by the fact that due to different procedures used to reach the initial state of the recriticality analysis, the core was not completely dry, and the core temperatures were somewhat smaller and thus both the thermal hydraulic models and correlations as well as the neutronics data set were not as far from their proper range of validity as in the actual recriticality analyses.

The work started within the NKS will be continued with the EU's SARA (Severe Accident Recriticality Analysis) Project in 1997-1998. In the coming APROS analysis a proper data set, valid for the conditions of recriticality should be used. More detailed core conditions and control rod melting patterns according to MAAP4 and MELCOR results should be used as initial conditions. The validity and limits of both the neutronics and the thermal hydraulics models should also be checked, and correspondence of the situation created by boundary conditions in the stand-alone model and the physical situation predicted by MAAP and MELCOR should be studied.

13.1 Comparison of results

Comparison of results obtained with the three different codes, i. e. APROS, SIMULATE-3K and RECRIT shows considerable differences. RECRIT seems to give higher recriticality power peaks and larger sustainable core power at high reflooding flows than APROS and SIMULATE-3K. The thermo-hydraulic models for APROS and RECRIT have some similarities, but especially the lack of quenching front model in five equation model of APROS limit currently its capabilities for BWR reflooding related phenomena. Also, differences in implementation and interaction with the reactor kinetics might be a reason for the deviating results, which have to be further investigated and explained. But as mentioned earlier, the reactor physical data and the water injection rate have a strong effect on the results and stronger conlusions can be drawn only after the whole range parameter studies are completed. Current SIMULATE-3K, on the other hand, has got less detailed T-H models for reflooding and is using another approach for the reactor kinetics. These modelling differences and their effects should be further investigated and additional modelling improvements are certainly necessary.

13.2 Uncertainties

No quantification of the uncertainties has been done. As stated above, that is difficult due to the preliminary nature of present study. The difficulties encountered made it also out of reach to perform the parameter studies necessary for an uncertainty analysis. A proposal for forthcoming parameter studies could, however, include the following issues:

- Nodalisation, especially in axial direction of the core. A considerable increase in peak power with increased number of axial nodes was noticed in SIMULATE-3K, while there was almost no such effect in RECRIT.
- The reactor physical data has so strong an impact on the results that a comparison between different models in a numerical form is needed before any comparison of power and temperature peak amplitudes.
- Fuel loading, burnup and initial core power distribution. It can be expected that the axial power distribution has an effect on the temperature distribution during core uncovery and heat-up, and that the axial distribution of the burnup will have an effect on the recriticality process.

- Heat transfer modelling. Current SIMULATE-version does not take into account axial heat conduction in fuel rods. This might be necessary to include in future code versions. Other more detailed reflooding and quench heat transfer models have also to be considered.
- The validity of the applied neutronics data set in APROS is guarateed only for normal steady state operation and 'classical' transients.
- Thermal hydraulic models and correlations were used as they appear in the APROS code and they were developed for normal operation and operational transients.
- In the APROS code version used for this study a code error in the neutronics was found contributing to the numerical problem at the time of power peak. The code error has been corrected after finishing of this study.
- Zirconium oxidation was not accounted for in the presented calculations.

14. CONCLUSIONS AND RECOMMENDATIONS

The recriticality studies in RAK-2.1 have produced interesting results, although the full goal of extensive analysis of recriticality was not achieved within this project. The most notable positive outcome of the project was the development of a major upgraded version of the RECRIT code having models for neutronics and thermal hydraulics specially tailored to reflooding situations.

The RECRIT code calculations resulted in reasonable trends in power behaviour during recriticality. The power peaks turned down rapidly after reaching criticality and then stabilises or oscillates around an elevated power level. Since the coolant void fraction and the fuel temperature are the principal factors affecting criticality, variations on coolant flow rate were carried out. Both the power peak and the stabilised power level were strongly dependent of the coolant flow rate. In the case of the lowest reflood rate (22.5 kg/s) two peaks of 30 % of nominal power could be seen. With the flow rate of 45 kg/s three power peaks could be seen during 30 s of criticality with the second peak being the highest, approximately 220 % of nominal power. With the reflooding rate of 160 kg/s the maximum power peak reached the value 400 % of the nominal power followed by two smaller peaks of sizes 100 % and 60 % respectively, with the core power stabilising on a level of 25 %. With the highest reflood rate of 540 kg/s the initial power peak was about 1730 % of the nominal power followed by another power peak of about 200 % about 15 seconds later. The power stabilised in this case to a level of 200 % of the nominal power.

The scoping studies with APROS resulted in a high initial power peak of about 260 % of nominal power, but the power stabilised in low level, close to decay heat, after the initial peak. The reflooding rate in the scoping studies was low 10 - 50 kg/s. The calculations with higher reflooding rates terminated in numerical problems soon after start of power peaking. The code will be further developed to overcome these numerical problems in the future work.

Recriticality studies with SIMULATE-3K were started with tentative calculations on a simplified 4 by 4 assembly BWR core, and on a full scale model of Oskarshamn 3 3300 MW_{th} plant. Studied cases started with total station blackout after 3 s with full power followed by scram and stopping of the core flow. The water boil-off and core heat up started and when the core temperatures had increased to 1500 K, half the number of the control rods were withdrawn and reflooding was started. The tentative SIMULATE-3K results showed that the Doppler feedback limited the power pulse to less than one full-power second, although high power peaks may occur (100 - 1500 % of the nominal power). After the peak, the fission power stabilized at an elevated level , the magnitude of which increased with reflooding flow rate. In case of water injection of 625 kg/s corresponding to 5 % nominal recirculation flow rate, the power stabilized at about 35 % of full power.

In the full core calculations for O-3 in the high pressure case the power peak after reaching recriticality was low, but the reactor stabilised at a power level corresponding to about 15 % of the nominal power with 175 kg/s reflooding flow rate. In the low pressure case SIMULATE-3K predicted an initial power peak of about 15 % of the nominal power after which, the power settled down close to decay power. SIMULATE-3K is still under development and needs additional updating and modifications the full-scale calculations could, therefore, only be carried through with some simplified boundary conditions.

A scoping study on TVO BWR containment response to a presumed recriticality accident with a long-term power level being 20 % of the nominal power was performed. The results indicated that containment venting system would not be sufficient to prevent containment overpressurization and containment failure would occur about 3-4 h after start of core reflooding. In the case of station blackout with operating ADS the present boron system would be sufficient to terminate the criticality event prior to containment failure, but in case of feedwater LOCA and boron dilution to the whole containment water pool, the present boron concentration would not be sufficient to ensure subcriticality in the core.

The reasons for the discrepancies in the results from the various codes should be further investigated and explained in the following project. Emphasis will be laid on improving the models and on using the best, and if possible, similar models in all codes. Model development will be focused on thermal-hydraulic/heat transfer mechanisms prevailing under reflooding conditions with high core temperatures. The different methodologies in APROS, SIMULATE-3K and RECRIT should still be retained to facilitate recriticality calculations with three different approaches. The goal is to obtain more consistent results with the three codes, to minimise the uncertainties and to reach a consensus for the reflooding/recriticality issue.

It can be recommended that a full range parameter sensitivity analysis until the core is completely covered should be performed with all codes. A methodology is developed for comparing the contents of the reactor physical data used in different codes, All used analysis methods have their own advantages: SIMULATE and APROS can offer their wide scope of application and experiences gained world wide, RECRIT code can offer the physical description adjusted to BWR reflooding specific phenomena.

The project succesfully initiated development of adequate analytical tools for recriticality studies and thus laid a foundation for the continued work in the field in the framework of the EU SARA project in 1997-1998.

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