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OLKILUOTO 2 — RELAP5/MOD3.2.1.2 Analysis of the Reactor Scram on June 13, 1997

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ABSTRACT

The reactor scram, which occurred at Olkiluoto 2 June 13 in 1997 has been analysed with the RELAP5/MOD3.2.1.2 computer program. The RELAP5 model of the Olkiluoto reactors is based on the model prepared for analysing PSA level 1 success criteria. The purpose of the analysis was to investigate how well RELAP5 is able to calculate the behaviour of process parameters after scram and validate the RELAP5 model of the Olkiluoto reactors.

The main plant features connected with the event are briefly presented. Modelling aspects are discussed regarding the RELAP5 analysis.

The RELAP5 result agrees with the measured reactor water level at the beginning of the transient when the level drops due to the collapse of steam bubbles. Later the level rise due to supply of feed water and the warming up of the reactor water inventory is slower in the RELAP5 analysis than according to the measurements.

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FOREWORD

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EXECUTIVE SUMMARY

The reactor scram, which occurred at OL2 June 13 in 1997 has been analysed with the RELAP5/MOD3.2.1.2 computer program. The RELAP5 model of the Olkiluoto reactors is based on the model prepared for analysing PSA level 1 success criteria. The purpose of the analysis was to investigate how well RELAP5 is able to calculate the behaviour of process parameters after scram and validate the RELAP5 model of the Olkiluoto reactors.

The plant unit OL2 operated on June 13, 1997 at the old 105 % power (2272) during the test operation for the power up rating. The periodic test of main steam line radioactivity level measurements was being performed. The tests actuated the scram signal in each of the four subs A, B, C and D, one channel at the time. The reactor scram operates with 2/4 logic, which means that a scram signal in two of the four channels will lead to reactor scram. After the test of the channel D the scram signal (SS chain) was not restored. The channel B was tested after the channel D, which led to actuation of scram signal also in the channel B. Because the scram signal was actuated in two channels the reactor was scrammed automatically.

Regarding the behaviour of the process parameters after the scram the most important observation was that the reactor water level decreased below the scram level L2 (+3.1 m above top of active fuel). This led to the stop of the pumps supplying water to the reactor water clean-up system. However, because the reactor was scrammed before the L2 signal no other consequences followed. The stopped pumps started again when the water level was again above L2.

The reason for the large reactor water level drop was the void collapse of the reactor after the scram. The original steam separators and the shroud cover were replaced in the refuelling outages in 1997 with new ones in order to accommodate the uprated power level of 2500 MWth. The capacity of the old steam separators would not have achieved sufficiently low moisture content of the steam. The shroud cover height was increased by 0.4 m and the shroud cover volume was thus increased by 6.3 m^3. This increase of the reactor two-phase mixture volume further increased, in addition to the larger core void content due to the power uprating, the reactor level drop compared to the original design values. The lowest reactor water level was +2.91 m above the top of active fuel at 24 seconds from the scram signal. The lower fine level measurement connection is at the level +2.5 m above the top of active fuel and the upper at the level +6.2 m.

The reactor scram, has been analysed with the RELAP5/MOD3.2.1.2 computer program. The RELAP5 model of the Olkiluoto reactors is based on the model prepared for analysing PSA level 1 success criteria. The purpose of the analysis was to investigate how well RELAP5 is able to calculate the behaviour of process parameters after scram and validate the RELAP5 model of the Olkiluoto reactors.

Several changes were required in the Olkiluoto 1 and 2 models, which was prepared before the scram, in order to reach good agreement between the calculation and the measurement data. The changes concerned nodal volumes, node junction flow areas and pressure loss coefficients and heat structures. Default values were retained for RELAP5 thermal hydraulic parameters.

The RELAP5 result agrees with the measured level at the beginning of the transient when the level drops due to the collapse of steam bubbles. According to the analysis the fine level dropped from the initial value +3.86 m above top of active fuel to the minimum value +2.91 m in 23 s. According to the measurements the same minimum level was reached at 24 s after the scram. The level rise from 24 s to 120 s due to supply of feed water and the warming up of the reactor water inventory is slower in the RELAP5 analysis than according to the measurements. The reason can be that RELAP5 calculates the amount of steam condensing in water is larger than in reality. The amount of steam in water is thus smaller than the real void. The stronger steam condensation is also reflected by the observation that the calculated water level drops faster than the measured level. RELAP5 also calculates at the beginning of the transient a lower reactor pressure than what is measured. At this stage there is due to feed water flow a large amount of sub cooled water in the vessel to condense steam.

1. INTRODUCTION

On the west coast of Finland, in Eurajoki, Teollisuuden Voima Oy (TVO) owns and operates two 840 MWe boiling water reactors. The reactors Olkiluoto 1 (OL1) and Olkiluoto 2 (OL2) were supplied on a turnkey basis by the Swedish company AB ASEA-ATOM (currently part of Westinghouse Electric Company). The OL1 unit was first connected to the national grid in September 1978 and the OL2 unit in February 1980.

The reactor core has 500 fuel bundles. The units have been uprated twice since the commissioning. The thermal power of each reactor was increased from 2000 MWth to 2160 MWth in 1984 and to 2500 MWth in 1998. The corresponding nominal values of the net electrical output were 660 MWe, respectively.

Four-fold redundancy is used throughout for systems having safety functions. The systems, e.g. safety systems and safety related instrumentation, are divided into four sub-systems, designated A, B, C and D which are subject to specific separation rules as regards the physical location of mechanical and electrical components (including cabling), power supplies and signal transmission.

The reactor scram, which occurred at OL2 June 13 in 1997 has been analysed with the RELAP5/MOD3.2.1.2 computer program. The RELAP5 model of the Olkiluoto reactors is based on the model prepared for analysing PSA level 1 success criteria. The purpose of the analysis was to investigate how well RELAP5 is able to calculate the behaviour of process parameters after scram and validate the RELAP5 model of the Olkiluoto reactors.

RELAP5/MOD3.2.1.2. computer program simulates the thermal hydraulics of light water reactors. The code is based on best-estimate methods to solve inhomogeneous and unsteady two-phase flow in one dimension. The code development is financed by the United States Nuclear Regulatory Commission.

Several changes were required in the Olkiluoto 1 and 2 model, which was prepared before the scram, in order to reach good agreement between the calculation and the measurement data. The changes concerned nodal volumes, node junction flow areas and pressure loss coefficients and heat structures. Default values were retained for RELAP5 thermal hydraulic parameters. TVO uses RELAP5 for the analyses of PSA level success criteria and other thermal hydraulic analyses.

2. OLKILUOTO 2 REACTOR SCRAM ON JUNE 13, 1997

The plant unit OL2 operated on June 13, 1997 at the old 105 % power (2272 MWth) during the test operation for the power uprating. The periodic test of main steam line radioactivity level measurements was being performed. The tests actuated the scram signal in each of the four subs A, B, C and D, one channel at the time. The reactor scram operates with 2/4 logic, which means that a scram signal in two of the four channels will lead to reactor scram. After the test of the channel D the scram signal (SS chain) was not restored. The channel B was tested after the channel D, which led to actuation of scram signal also in the channel B. Because the scram signal was actuated in two channels the reactor was scrammed automatically.

Regarding the behaviour of the process parameters after the scram the most important observation was that the reactor water level decreased below the scram level L2 (+3.1 m above top of active fuel). This led to the stop of the pumps supplying water to the reactor water clean-up system. However, because the reactor was scrammed before the L2 signal no other consequences followed. The stopped pumps started again when the water level was again above L2.

The reason for the large reactor water level drop was the void collapse of the reactor after the scram. The original steam separators and the shroud cover were replaced in the refuelling outages in 1997 with new ones in order to accommodate the uprated power level of 2500 MWth. The capacity of the old steam separators would not have achieved sufficiently low moisture content of the steam. The shroud cover height was increased by 0.4 m and the shroud cover volume was thus increased by 6.3 m^3. This increase of the reactor two-phase mixture volume further increased, in addition to the larger core void content due to the power uprating, the reactor level drop compared to the original design values. The lowest reactor water level was +2.91 m above the top of active fuel at 24 seconds from the scram signal. The lower fine level measurement connection is at the level +2.5 m above the top of active fuel and the upper at the level +6.2 m.

The behaviour of the reactor water level and reactor pressure are presented in the report with the corresponding RELAP5 results in the figures 3 and 4. Main circulation flow, feed water flow and feed water temperature given as input for the RELAP5 calculation are presented in the figures 2, 5 and 6. Reactor water level, reactor pressure, main circulation flow and feed water flow are obtained from OL2 process computer print out of the reactor scram event. The feed water temperature is obtained from OL2 measurement computer print out.

3. OLKILUOTO 1 AND 2 RELAP5 MODEL

3.1 General

The model includes the reactor pressure vessel and the main steam lines to the main steam isolation valves. Other reactor process system components are modelled as boundary conditions utilizing the time dependent volumes and junctions of RELAP5.

The essential parts of the reactor protection systems are modelled by trip and control variables. The following reactor protection system signals are included:

-reactor scram SS signals (SS1 to SS15)
-containment isolation I signals
-automatic primary system depressurization TB signals
-low reactor water level (L3 +2.0 m above top of active fuel) X1 signal
-low reactor pressure (L3 1.2 MPa) X5 signal.

The containment is not included in the RELAP5 Olkiluoto input file. That is why the following reactor protection systems are not modelled:

-reactor building room supervision Y signals (Y1 to Y16)
-main steam line supervision A signals (A1 to A23)
-feed water line supervision M signals (M1 to M8 and MT9)
-safety system room supervision HA, HB, HC, HD signals
-high radioactivity level in the reactor hall X2 signal
-high condensation pool temperature X3 signal
-low pressure difference between diesel backed normal operation second cooling system and the lower drywell space.

Neither reactor kinetics included in the RELAP5 Olkiluoto input file. That is why the following reactor protection systems are not modelled:

-reactor screw stop V signals (Y1 to Y16)
-control rod withdrawal locking at power less than 8 % (S1 to S5)
-control rod withdrawal locking at power equivalent to or greater than 8 % (E1 to E4)
-reactor scram B signals during refuelling outage

The control rod drive system provides two independent ways for insertion of the control rods, one electro-mechanical system for normal operation with fine motion and one hydraulic system for fast scram insertion.

3.2 Steady state

The operation point at the time of the reactor scram in Olkiluoto 2 June 13, 1997 is presented in the table 1. The data is obtained from process computer.

Table 1. Olkiluoto 2 steady state prior to reactor scram on June 13, 1997.

Reactor thermal power (MW)		2272
Pressure (MPa)	Steam dome pressure	7.0
Temperature	Feed water	180
Flow rate (kg/s)	Feed water	1131
	Main steam line	1158
	Downcomer	7850
Reactor fine		3.86
(m above top of active fuel)		

3.3 Transient

The scram signals SS1 (manual scram) is assumed to be released at the beginning of the transient calculation at 0 sec into the transient. Manual scram is used because the signals, which actually released the scram, are not modelled in the Olkiluoto RELAP5 model. The hydraulic scram system starts to insert control rods into the core at 0.2 sec from the scram signal. The reactor fission power is zero at 4.2 sec, when the control rods are fully in the reactor core. After that the core thermal power is assumed to follow the ANS 5.1-1979 standard decay heat curve.

The main circulation pump run-down starts 0.2 sec from the scram signal.

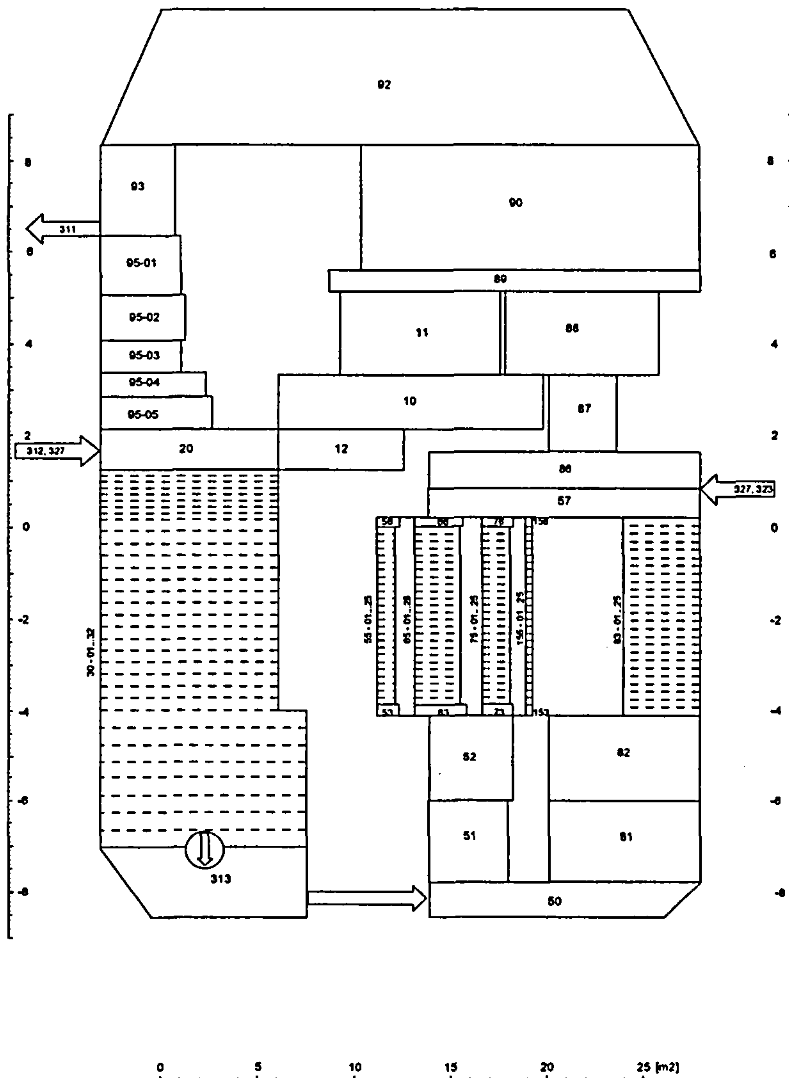
The feed water flow and temperature are set according to the data obtained from the process and measurement computers. The pressure regulator will control the reactor pressure at 7.0 MP.

4. INPUT DATA

4.1 Nodal model

The RELAP5/MOD3.2.1.2 nodal model of Olkiluoto reactors is presented in the figure 1. The nodes are drawn according to the scale regarding both height and flow area. Due to drawing technical reasons flow area of some nodes, however, is not according to scale.

RELAP5/MOD3.2.1.2. OLKILUOTO 1 AND 2 REACTOR PRESSURE VESSEL NODALIZATION.



System abbreviations

311 - Main steam lines
312 - Feed water system
313 - Main circulation pumps
323 - Core spray system
327 - Auxiliary feed water system

Figure 1. Olkiluoto 1 and 2 reactor pressure vessel. RELAP5 model nodalization

4.2 Reactor core and nuclear fuel

The data of reactor core nodal volumes, flow areas and heat structures are according to ATRIUM 10-9Q nuclear fuel.

The radial and axial power distributions of the model are based on POLCA4 calculations. POLCA4 was the incore fuel management code used by TVO at the time of the transient. Based on the radial power distribution the 500 fuel bundles of the reactor core are divided in four regions, which describe the fuel bundle groups of different power levels:

-low power region, radial power coefficient p_r=0.375, 88 fuel
-bundles
-medium power region, radial power coefficient p_r=0.998, 260 bundles
-high power region, radial power coefficient p_r=1.364, 151 fuel bundles
-hot channel, radial power coefficient p_r=1.514, 1 fuel bundle

In the axial direction the core is divided in 25 nodes.

The core by-pass includes the volume between the fuel bundles, the volume between the core boundary fuel bundles and core shroud and the water channel volume of the fuel bundles. In the axial direction the by-pass channel is divided in 25 nodes.

The RELAP5 point neutron kinetics model is not used for the reactor power calculation, but the steady state is calculated with a constant core thermal power of 2272 MWth. After the reactor scram the core power generation is assumed to follow the decay heat curve according to the ANS 5.1-1979 – standard. Two percent of the reactor power is assumed to be transferred directly to the coolant.

4.3 Heat structures

The reactor pressure walls are modelled by a heat structure of carbon steel, the thickness of which is 139 mm in the skirt, 187 mm in the bottom and 85 mm, 138 mm and 371 mm in the top head. The vessel has a 5 mm stainless steel cladding except in the bottom, which has a 5 mm Inconel cladding. The cladding is modelled as one layer and the rest of the wall is divided in five or seven layers. The thickness of one calculation layer is thus 17.0 to 52.2 mm. The default RELAP5 material properties of the materials are used for thermal conductivity and specific heat. The reactor vessel outer surface is assumed thermally insulated and the heat losses to the containment are not modelled. In reality the thermal losses at reactor operating temperature of 286 °C are ca. 700 kW.

The thickness of the reactor internal heat structures varied from 0.75 mm (fuel cladding) to 45 mm (reactor shroud and shroud head). The same material properties are used both for fuel cladding (Zircaloy-2) and fuel channels (Zircaloy-4). The material properties are presented in the table 3.

4.4 Steam separators

OL1 and OL2 have 174 GENE AS2B steam separators by General Electric. They are modelled with the mechanistic RELAP5 steam separator model for General Electric separators.

4.5 Main circulation pumps

The total core coolant flow in the reactor pressure vessel is calculated by summating the flow measurements in eight strategically located fuel assemblies.

The distribution of the core coo1ant flow throughout the core is determined by calibrated orifice plates in each fuel assembly seat. The selected seats are provided with pressure tappings for measuring the pressure drop across the orifice plate. The tappings are located immediately above the plate and approx. 500 mm under the fuel assembly support plate. The small-bore tubes from the pressure tappings are run in pairs through guide tubes to penetrations in the bottom of the pressure vessel and thence to the flow measurement differential pressure transmitters.

The instrumented orifice plates are calibrated individually during charging of the reactor using a calibration assembly incorporating a reference flow meter of the turbine type. The calibration assembly is designed to simulate an ordinary fuel assembly hydraulically, to ensure that the flow pattern will be the same as during normal operation of the reactor.

The eight flow measurement channels are sub-divided into two channels in each sub-group A, B, C and D.

The flow measurement signals from the transmitters are summated to form combined flow signals in four independent averaging units, one in each sub-group.

The flow meters have an inherent temperature dependency since the density is included as a factor in the calibration constant. The maximum error due to normal variations in coolant temperature is estimated as ±1%. The error is negative at low coolant temperatures (i.e. low main circulation pump flow).

The reactor pressure vessel core coolant flow is measured continuously and is compared with the neutron flux density in order to ensure that the thermal operating margins are maintained. The analog signals from the four flow averaging units may be read in the control room, while these and the eight individual flow signals from the fuel assemblies are also supplied to the station computer.

The core coolant temperature is measured continuously, displayed and recorded in the control room, and supplied to the station computer, as is the pressure rise signal from the main circulation pumps. Together with the measurement of core coolant flow and the main circulation pump speeds, the latter measurements are used to establish the pump operating points, and to check that the correction for total core coolant flow is correctly adjusted.

The six main circulation pumps are modelled by RELAP5 pump component. For the steady state calculation a control function was used to regulate the main circulation flow to 7850 kg/s, which flow was measured at the time of the reactor scram. In the transient calculation the controller was removed and the pump was steered down by changing directly the pump rotation speed. The pump run-down was performed as a function of time so that the calculated main circulation flow agreed as well as possible with the measured value (figure 2).

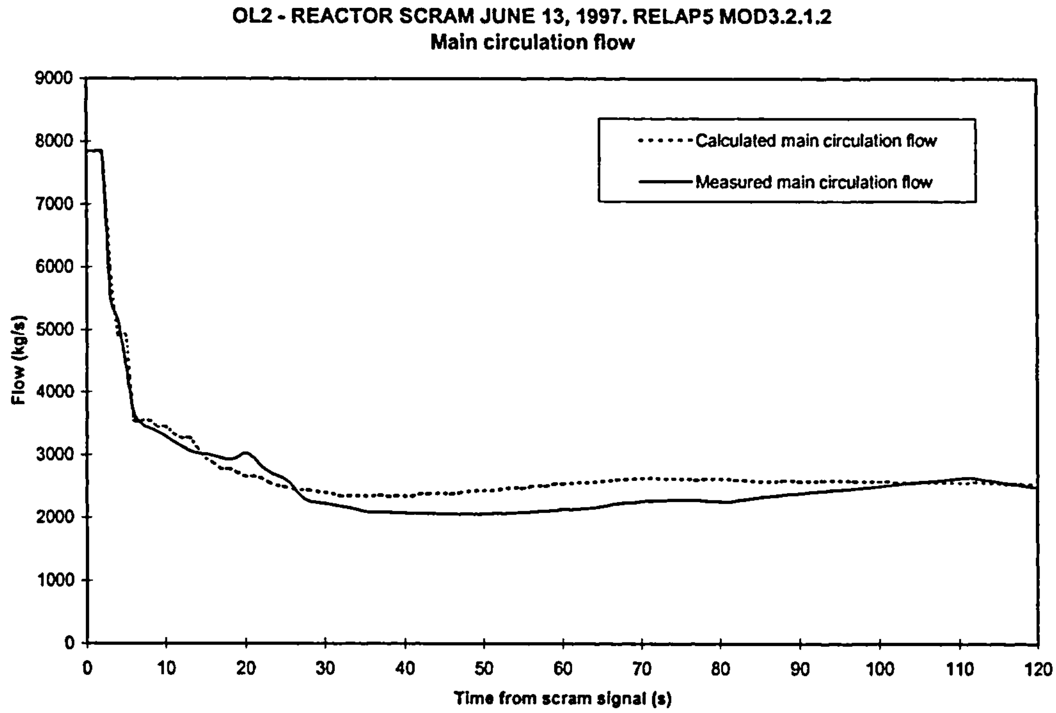


Figure 2. Olkiluoto 2 reactor scram June 13, 1997. Main circulation flow.

4.6 Main steam lines

The main steam lines are modelled to the main isolation valves (node 311). For the steady state calculation a control function was used to regulate the steam flow out the reactor pressure vessel so that a constant steam dome pressure of 7.0 MPa was attained. The controller was used also in the transient calculation. The pressure controller simulates the actual reactor pressure regulator in a very simplified way. However, the function is satisfactory for the analysis.

4.7 Reactor level measurement

4.7.1 Method of measurement

The level instrumentation is designed to measure the water level in the reactor pressure vessel downcomer. The indirect method, by which the pressure head is measured at a fixed point below the lowest level which must be indicated, is used for this purpose.

Since in this case the level is being measured in a closed vessel under pressure, with condensable vapor above the free surface of the liquid, the method used must employ differential pressure measurement. A differential pressure measuring device is arranged with the negative pressure side connected to an instrument tapping on the vessel at the lower limit of the desired range and the positive pressure side connected through a reference vessel to a tapping on the steam chamber of the vessel at or above the upper limit of the range. A specific water level is maintained in the uninsulated reference vessel by means of condensing steam, the excess water being returned to the reactor pressure vessel through an overflow.

The dp-cell is installed below the level of the negative pressure tapping. The connections to the tapping and the reference vessel are of small-bore piping sloping continuously upwards towards the vessel to allow the escape of gas.

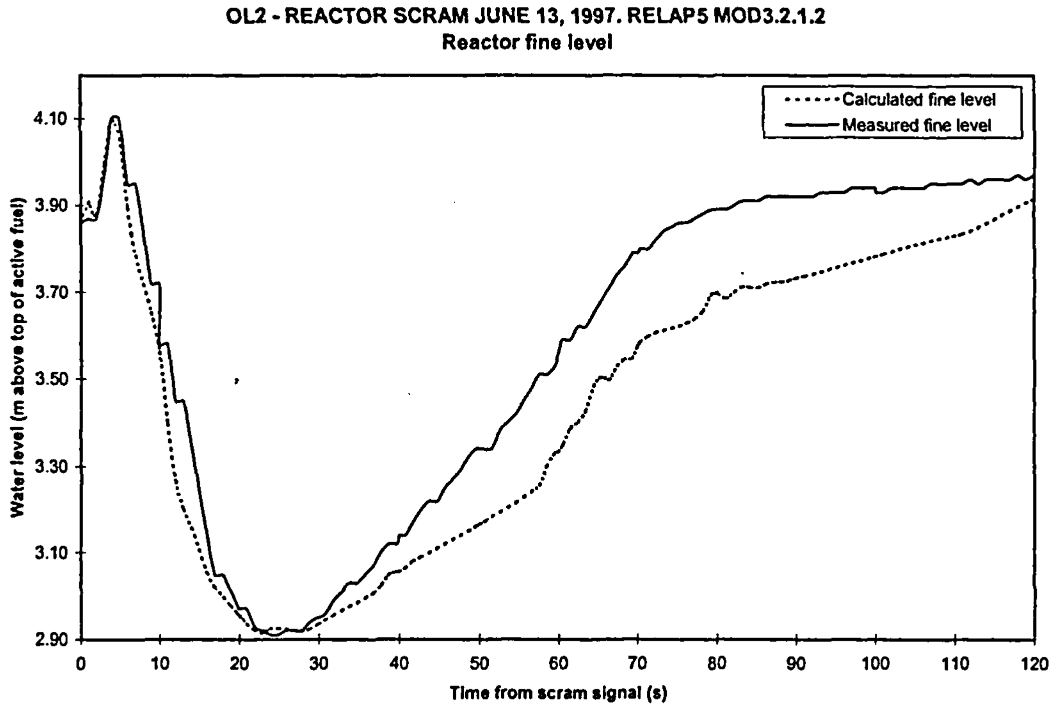


Figure 3. Olkiluoto 2 reactor scram June 13, 1997. Reactor fine level.

4.7.2 Density compensation

The major advantage afforded by the method of measurement described above is that no instruments require to be installed in or on the reactor pressure vessel, thus avoiding the considerable environmental problems which this would entail. However, there is a corresponding disadvantage in that the results of measurement are dependent on the density of the medium. The influence of variations in density may be considerable. The differential pressure corresponding to a specified water column is approximately 30% lower at the normal operating conditions of 7 MPa and 286°C than at room temperature

The results of measurement are also affected, albeit to a lesser extent, by the steam density and by the density of the reference water column i.e. the water present in the instrument line from the reference vessel to the level of the negative pressure tapping. Below this level, the heads due to the water columns in the water lines to the differential pressure transmitter cancel each other. Thus, the location of the transmitter is of minor importance.

The level measuring units are designed to provide the most accurate possible indication of the level of the boundary surface between the steam and water phases (the latter of which may consist of a mixture of water and small steam bubbles or voids) during normal operation, start-up and shut-down. Thus, the units are provided with compensators, which automatically correct the differential pressure signals for variations in density.

The density of the water is a function of the temperature, pressure and void fraction, while the steam density is a function of the pressure and temperature. However, since the saturated steam in the reactor is in thermal equilibrium with the water in the downcomer, the temperature is governed by the pressure. Therefore, the reactor pressure and the temperature of the reference water column are used as the controlling input parameters for the density compensator.

4.7.3 Scope of instrumentation

Each of the instrument sub-groups (A, B, C and D) includes a fine and a coarse measuring unit, both of which are equipped with density compensators. The positive sides of both units are connected to the same reference vessel (+6.4 m above the top of active fuel), while the negative sides are connected at different levels to individual branches on the pressure vessel (fine level +2.5 m, coarse level +0.2 m). Each sub-group is assigned its own reference vessel and instrument tappings. In addition, each group is provided with a reference temperature sensor in the form of a resistance surface thermometer mounted on the line to the reference vessel and supplying a control signal to the density compensators in the subgroup through a transmitter.

The instrumentation also includes two core level measuring units (sub-groups C and D, positive side +6.4 m)) with the negative sides connected to tappings on the moderator tank (-6.2 m).

In addition, the unit includes a single high level measuring unit. The individual reference vessel for this unit is mounted below the pressure vessel flange (+8.2 m) while the negative pressure side is connected to one of the fine level measurement tappings (-6.2 m). This unit is not provided with density compensation.

The level signals at the time the analyzed OL1 scram were initiated at the levels presented in the table 2.

Table 2. OL1 and OL2 reactor water level signals at the time of the OL2 scram June 13, 1997.

H	+4.7 m ATAF	
2	+4.2 m ATAF	
H	+3.9 m ATAF	Normal level
1	+3.6 m ATAF	
L0	+3.1 m ATAF	
L1	+2.0 m ATAF	
L2	+0.7 m ATAF	
L3		
L4		

After the OL2 scram the reactor water level was increased due to the large level drop and the current level signals are presented in the table 3.

Table 3. OL1 and OL2 current reactor water level signals.

H	+5.0 m ATAF	
2	+4.3 m ATAF	
H	+4.2 m ATAF	Normal level
1	+3.6 m ATAF	
L0	+3.1 m ATAF	
L1	+2.0 m ATAF	
L2	+0.7 m ATAF	
L3		
L4		

4.7.4 Normal system operating conditions

The signals from the differential pressure transmitters used for level measurement are supplied to density compensators, which are also supplied with signals from the coarse pressure measuring unit and the reference temperature sensor in the corresponding sub-group. The outputs from the density compensators are in the form of analog signals proportional to the water level in the reactor pressure vessel downcomer.

These signals are fed to limit switches which produce limit value signals and also to one or more isolation amplifiers. The control instrumentation (i.e. controllers, indicating instruments, recorders and station computer inputs) is connected to these amplifiers. The provision of sufficient level information is ensured by this means, while galvanic isolation between the sub-groups is also maintained.

The foregoing does not apply to the high level measuring unit which is not equipped with a density compensator. The transmitter for this measuring unit is calibrated for correct level indication when the reactor is cold (30-60°C). The level signal is transmitted to an indicating instrument in the control room.

A number of systematic error sources are present under normal operating conditions. These are due partly to the flow of steam and water through the reactor vessel and partly to the inflow of feed water.

The pressure sensed at the level measurement tappings and transmitted to the differential pressure transmitters is the static pressure of the medium, steam or water. However, the application of Bernoulli's equation demonstrates that the level measurement method requires that the total pressure, i.e. the sum of the static and dynamic pressures, be measured. Thus, a measuring error equivalent to the difference between the dynamic pressure components at the positive and negative tappings, is present. This error is greatest in the case of the fine level measuring units, since the water at the negative tappings of these is more or less static. The error is also dependent on the reactor power since the flow velocity is proportional to the quantity of steam generated. At full steam flow, the maximum error is estimated as +0.05 m, which is negligible with respect to the safety margins of the level limits.

A measuring system error may also be caused by voids or undercooling in the downcomer, since the density compensators are designed to operate with void-free water at saturation temperature.

The level reading supplied by the level measurement units represents in reality the equivalent level of void-free water above the negative instrument tappings. The true level of the boundary surface between the water and steam phases may differ from the indicated level. In the downcomer, the deviation is determined mainly by the void fraction while the drop in steam pressure in the steam dryers and the radial pressure drop due to the main circulation flow between the steam separator branches, are additional factors effecting the deviation inside the steam dryer casing. The latter factor results in convexity of the boundary surface among the steam separators. The steam dryer pressure drop is about 600 Pa (0.09 m water height at saturation temperature). It is estimated that the convexity water level gradient is about 0.16 m.

At normal full power operation, the sum of these effects results in the level among the central steam separator groups being slightly higher (about 0.07 m) than the indicated level. This fact leads to a systematic negative indicating error. At half power, the error is of the order of 0.10-0.14 m.

Under normal station operating conditions, the fine level measuring units supply the feed water controller with information regarding the water level in the reactor pressure vessel. The feed water controller regulates the speed of the feed pumps (and thereby, the supply of feed water to the reactor pressure vessel) so that the water level in the downcomer is maintained at the desired value within permissible limits.

In the event of operating upsets, the alarm system will alert the operator if the indicated level should exceed the H1 limit setting or fall below the L1 setting.

The level indicated by the coarse and fine level measuring units is displayed on indicating instruments and recorders in the control room, and is also supplied to the station computer. Under normal operating conditions, the only function of the coarse unit is to provide back-up indication for the fine unit.

The indication of core level is also displayed in the control room, although this does not fulfill a function under normal operating conditions. This indication is relatively inaccurate due to the wide measuring range and the low situation of the negative instrument tappings. With the reactor on load, undercooling of the water in the downcomer gives a systematic deviation of the order of +0.25-0.30 m in relation to the fine measurement.

Analog signals from the pressure measuring channels are used under normal operating conditions for indicating and recording purposes in the control room, and for supplying measured value signals to the station pressure control system. The analog signals are also supplied to the station computer. An alarm is initiated if the pressure falls below L1 or exceeds H1 in the event of an operating disturbance. The output signals from the two pressure vessel temperature measuring points may be recorded on a recording instrument. Pressure and temperature relationships will be unbalanced during start-up and shut-down. Since the operation of the density compensators is based on the assumptions that the reactor water is at saturation temperature and that the water columns are of equal temperature, the indication error in the level measurement channels may be somewhat greater than during normal operation.

During cold shut-down (e.g. during refueling) the level is measured up to the level of the pressure vessel flange by means of the high level measuring unit.

The core coolant measuring units are calibrated for correct indication at normal reactor water temperature. The flow indicated during the heating-up period will be too low. At reactor temperatures of 50°C and 180°C, the indication errors will be approx. 15% and approx. 8% respectively.

The temperatures of two selected points on the reactor pressure vessel cover and cover flange are recorded during start-up and shut-down.

The rate of temperature rise of the coolant flow, which is a measure of the power, is measured and displayed during nuclear heating using the computer. The prevailing rate of temperature rise is displayed on the screen with a time lay of approx. 1 minute.

4.7.5 Abnormal plant operating conditions

Level limit signals to other systems are supplied through the logic circuits of the reactor protection. Level H2 initiates reactor scram high reactor water level condition SS5, level L2 reactor scram low reactor water level condition SS4, level L3 condition X1, and level L4 containment isolation condition I2 and automatic reactor pressurization conditionTB1. An alarm is initiated at level H1 and reactor scram is initiated at H2 in the event of an abnormal rise in reactor pressure vessel level due, for example, to a fault in the feed water control system. The H2 setting is specified to ensure that reactor scram takes place under all operating conditions before the moisture content of the outlet steam becomes excessive.

In analogous manner, an alarm is initiated at L1 and reactor scram occurs at L2 as the level falls. In addition, partial operation of the auxiliary feed water system is initiated by L2. System 327 is subsequently subject to on-off control by L1 and H1. Level L2 is specified to ensure that reactor scram occurs without fail before steam can flow through the steam separator water outlets. L1 also reduces the speeds of the main circulation pumps, thus reducing reactor power.

If the level should continue to fall, due for example to loss of water caused by pipe failure, the coarse measurement level signals L3 and finally L4 will be initiated. Containment I isolation together with low pressure X5 initiates water injection by the core spray system. Under these conditions, it must be expected that the water in the downcomer may be heavily undercooled, which will result in a systematic positive error indication by the coarse measurement units. However, at L4, the error will not exceed 0.1 m even at the maximum conceivable degree of undercooling (100 °C). This value is negligible.

Conditions in the reactor pressure vessel may be monitored with the aid of the core level measuring units if extremely low levels are reached.

Rapid reductions in pressure in the reactor pressure vessel may be caused by conditions such as a major pipe failure or certain types of malfunction in the turbine system. This will result in the generation of voids in those sections of the reactor pressure vessel in which the water is at or close to saturation temperature (i.e. flashing will occur). The voids will give rise to an increase in the volume of the water phase, resulting in a rapid surge in water level in the downcomer. It is essential that the containment isolation valves in the steam lines be closed before the water level reaches the steam outlets, since the valves may otherwise be damaged by water hammer.

The sequence of events in circumstances of this type are dependent both on the nature of the incident and on the operating status of the reactor when the incident occurs. In most cases, closure of the containment isolation valves will be initiated from other systems such as isolation monitoring system, although there are other cases in which the impulse must be supplied by level limit signal H2 or the limit switches after the high pass filters connected to the coarse pressure measuring units.

In this instance, one of the problems is the fact that voids are also generated in the downcomer and the negative error in the level indication thereby increases during the incident. If the total void fraction in the downcomer exceeds a certain limit, the downcomer level will reach the level reference vessel tappings before the indicated level has reached H2. Since the signal obviously cannot increase following this, H2 will never be initiated. However, there is a counter-effect in the form of a pressure increase at the negative tapping due to the flow of water in the downcomer which is interpreted as a level increase by the level measuring system. Due to this effect, the indicated level is increased by approx 0.2 m at a surge rate of 0.5 m/s.

The above factors are considered in specifying level setting H2, to ensure that H2 is initiated under all relevant operating conditions, in sufficient time before the water level reaches the steam outlets. Both the closing times of the containment isolation valves and the time delay in the level instrumentation are taken into account. Certain improbable turbine failures may cause so rapid pressure reductions in the reactor pressure vessel that there is a risk that these cannot be handled by the level instrumentation. To obtain closure of the main steam line isolation valves at these failures, rapid pressure reduction is detected by high pass filtering of the signals from the coarse pressure measuring units. If the pressure in the reactor pressure vessel is reduced faster than can be tolerated during a certain minimum of time, signals are obtained from limit switches for main steam line A isolation.

Coarse level measurement is usually unimportant in cases in which flashing occurs. However, there are cases in which operation of the low level trips may conceivably occur, such as failure of a feed water line during hot shut-down. Steam will then be generated in the whole reactor pressure vessel, even below the feed water distributors and the coarse level measuring units will show a negative error over entire range. This, however, is an advantage under such operating conditions since the critical parameter is the total amount of water in the reactor pressure vessel and the indicated level is a better measure of this parameter than the actual level of the water surface.

4.7.6 Water level indication in the Olkiluoto RELAP5 model

The Olkiluoto RELAP5 model includes fine, coarse and core level measurements. The level is indicated as the collapsed water level height between the positive and negative side of the measurement tappings because the level reading supplied by the level measurement units represents in reality the equivalent level of void-free water above the negative instrument tappings,

-fine level (negative side +2.5 m, positive side +6.4 m)
-coarse level (negative side +0.2 m, positive side +6.4 m)
-core level (negative side -6.2 m, positive side +6.4 m).

4.8 Reactor pressure measurement

The level reference vessels are connected to a number of pressure transmitters. One fine and one coarse pressure measuring unit are included in each sub-group (A, B, C and D) and one extreme coarse pressure measuring unit in sub B. By high pass filtering of the signals from the coarse pressure measuring units fast pressure reduction in reactor pressure vessel is supervised.

4.8.1 Normal system operating conditions

The static head due to the water column between the level reference vessel and the pressure transmitters is included in the transmitter calibrations. Thus, transmitter outputs are proportional to the pressure in the steam space of the reactor pressure vessel. The signals are supplied to limit switches which produce pressure limit signals and to isolation amplifiers. The latter supply analog signals to the instrumentation and to other systems requiring information on pressure.

The signals from the coarse level measuring units are also supplied to the density compensators in the same sub-group. Further, these signals are high pass filtered. The outputs from the filters are supervised by limit switches.

The dynamic pressure component due to the steam flow is negligible in comparison with the reactor pressure and is completely unimportant to this measurement, nor would there appear to be any other systematic error sources of importance to the measurement.

4.8.2 Abnormal system operating conditions

The above discussion regarding possible malfunctions in the level measuring channels, and their consequences, applies also to the other pressure and differential pressure measuring systems, with obvious modifications.

Signals are supplied to the reactor pressure relief system at pressure limits H4, H3, H2, H1 and L2. In addition, alarms are initiated at pressure limits H1 and L1.

Monitoring of extremely high reactor pressure has been added to recorder in main control room.

4.8.3 Reactor pressure indication in the Olkiluoto RELAP5 model

The reactor pressure in the Olkiluoto RELAP5 model is indicated as the steam dome pressure.

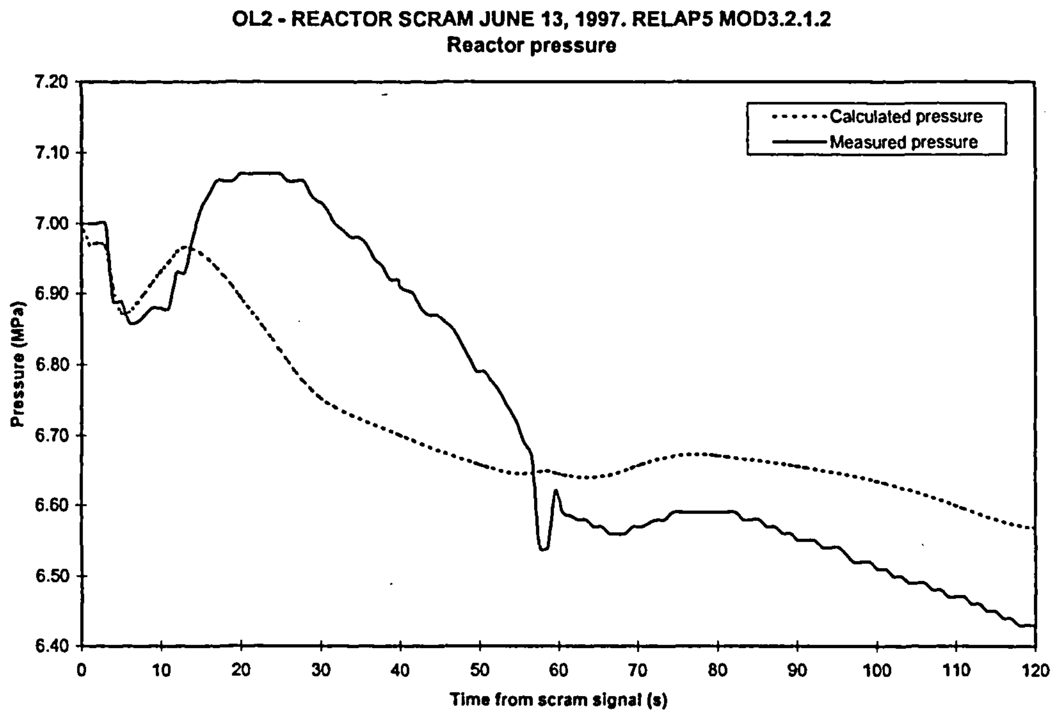


Figure 4. Olkiluoto 2 reactor scram June 13, 1997. Reactor pressure.

4.9 Feed water

In the Olkiluoto RELAP5 model feed water flow is regulated by a controller so that the steady state reactor water level was the same as the measured value 3.86 m above the top of active fuel.

The controller was removed in the transient calculation. The feed water flow rate and temperature were instead set as a function of time according to the measured values (figures 5 and 6).

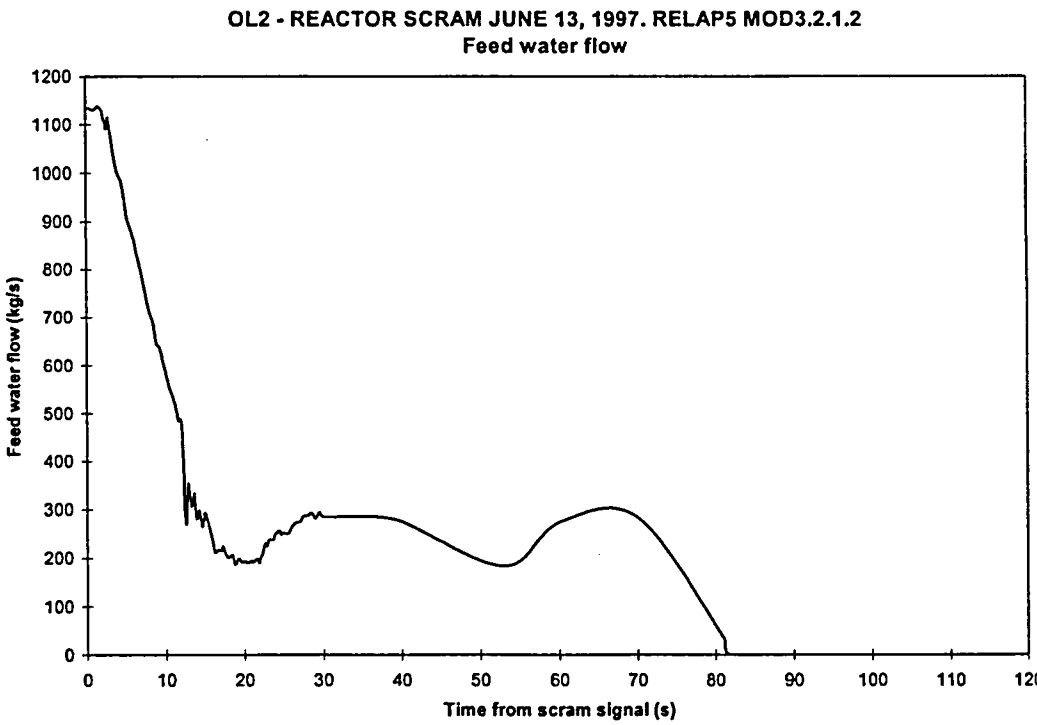


Figure 5. Olkiluoto 2 reactor scram June 13, 1997. Feed water flow.

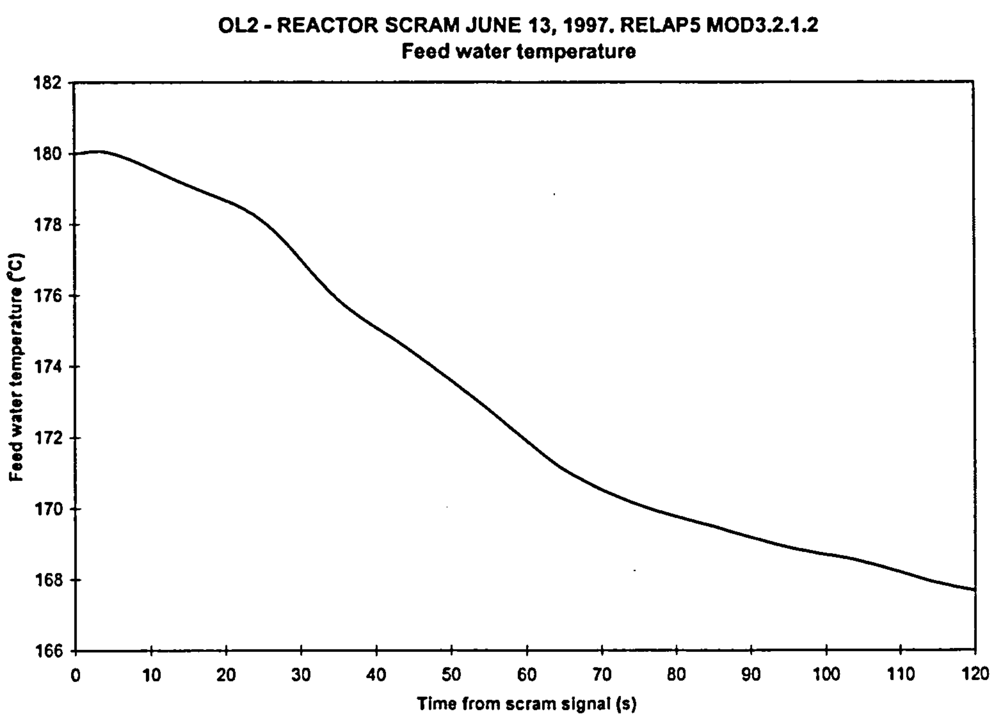


Figure 6. Olkiluoto 2 reactor scram June 13, 1997. Feed water temperature.

5. RESULTS

5.1 Steady state

The operation point of the OL2 reactor scram of June 13, 1997 is presented in table 4. The table gives both the values of the main parameters calculated with RELAP5 and also the corresponding values obtained from the process computer.

Table 4. Olkiluoto 2 steady state prior to reactor scram on June 13, 1997. Comparison between measurements and the RELAP 5 calculation.

Process parameter		Measurement	Calculation
Reactor Thermal power (MW)		2272	2272
Pressure (MPa)	Steam dome Pressure difference Between core inlet and outlet	7.0	7.0 0.147
Temperature (°C)	Feed water Core inlet Core outlet	180	180 275.5 286.3
Flow rate (kg/s)	Feed water Main steam line Downcomer Core fuel Core by-pass	1131 1158 7850	1131 7850 6731 1119
Reactor fine level (m above top of active fuel) Mass of water in reactor pressure Vessel (10^3 kg)		3.86	3.86 178.2

5.2 Transient

Figure 3 presents the calculated fine level from 0 s to 120 s after the reactor scram together with the measured fine level. The RELAP5 result agrees with the measured level at the beginning of the transient when the level drops due to the collapse of steam bubbles. According to the analysis the fine level dropped from the initial value +3.86 m above top of active fuel to the minimum value +2.91 m in 23 s. According to the measurements the same minimum level was reached at 24 s after the scram. The level rise from 24 s to 120 s due to supply of feed water and the warming up of the reactor water inventory is slower in the RELAP5 analysis than according to the measurements.

The figure 4 presents the calculated and measured pressure from 0 s to 120 s from the scram. The calculated pressure is at 28 s 0.28 MPa lower than the measured pressure. At 120 s the calculated pressure is 0.14 MPa higher than the measured pressure.

Figures 7 to 11 present the calculated void in the core low, medium and high power region fuel bundles, in the upper plenum (volume between the core and shroud head), in the steam separator stand pipes, in the volume between the steam separator standpipes and in the downcomer at the level of the shroud head. It can be pointed out that in the steady state the void fraction is 0.15 in the volume between the steam separator standpipes and 0.11 in the downcomer at the level of the shroud head.

Figure 12 presents the amount of water in the reactor pressure vessel. In the steady state the mass of water is 178 metric tons.

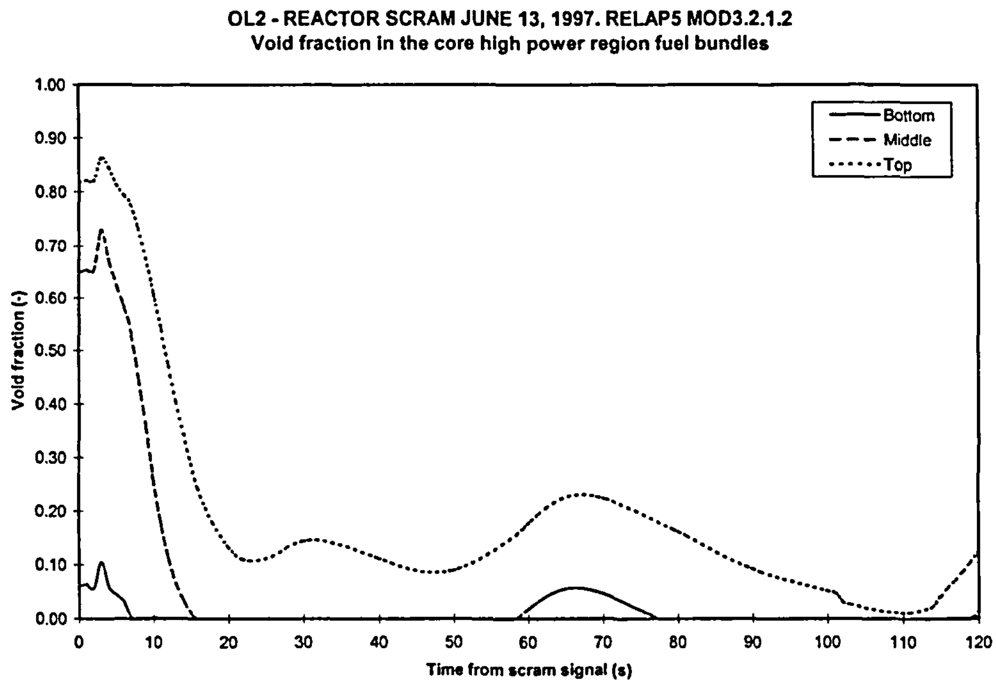


Figure 7. Olkiluoto 2 reactor scram June 13, 1997. Void fraction in the core high power region fuel bundles.

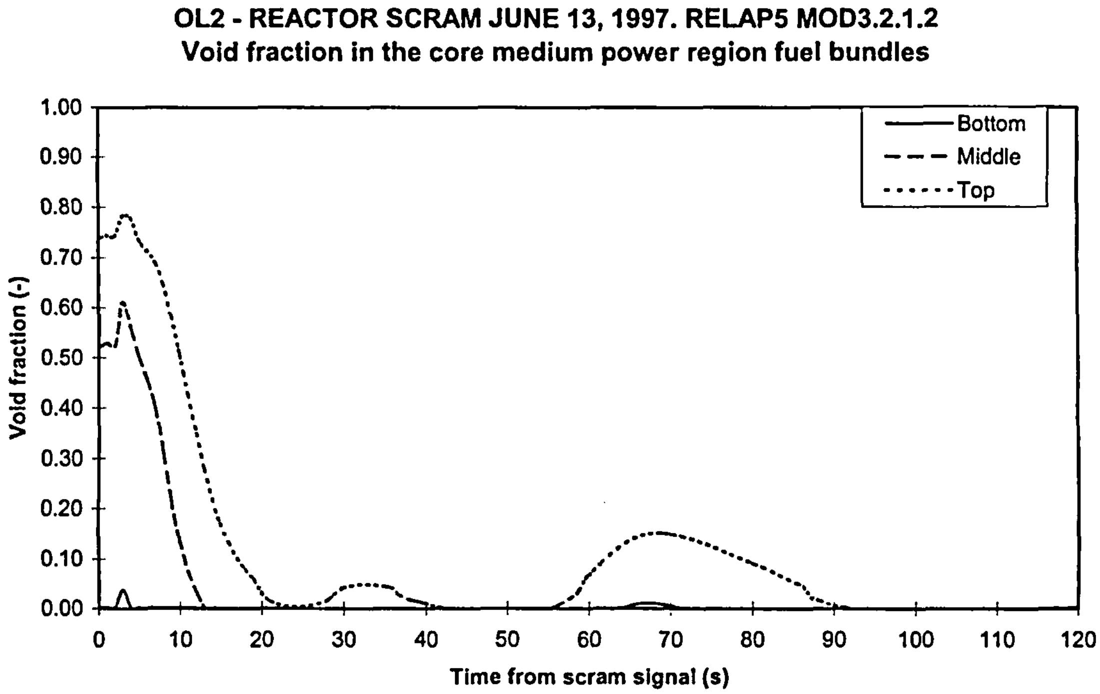


Figure 8. Olkiluoto 2 reactor scram June 13, 1997. Void fraction in the core medium power region fuel bundles.

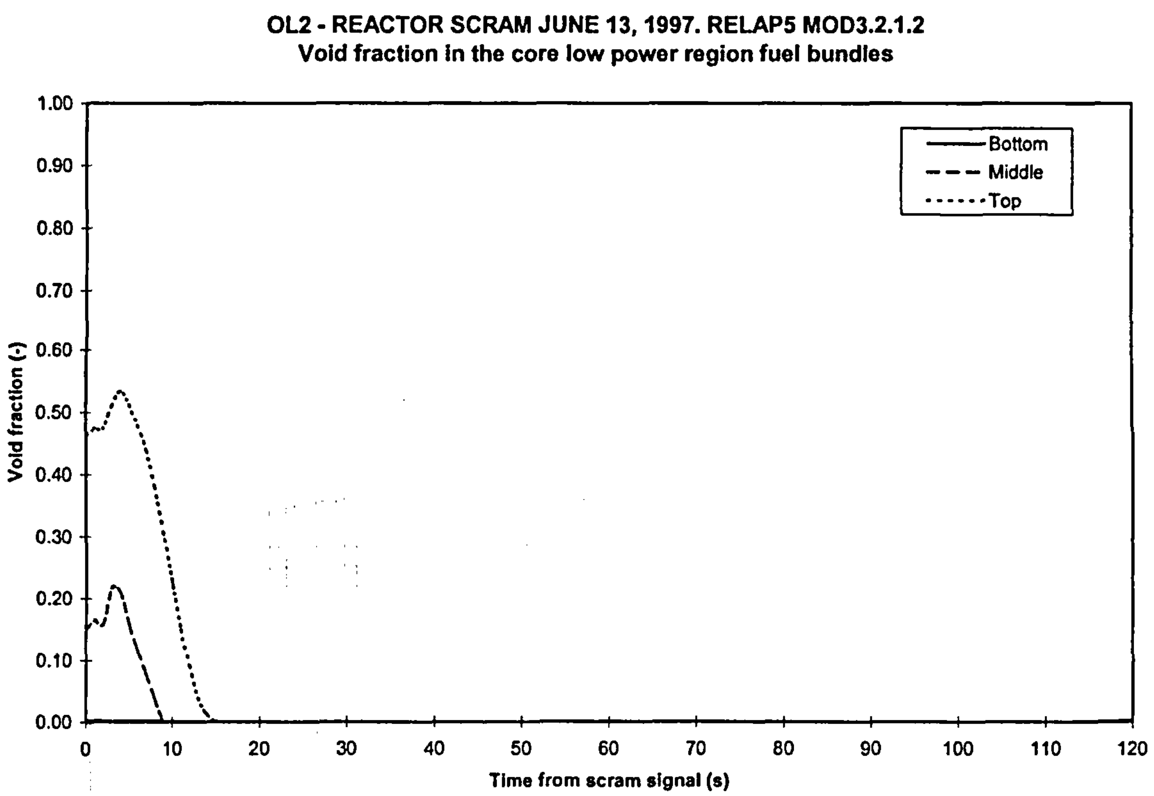


Figure 9.Olkiluoto 2 reactor scram June 13, 1997. Void fraction in the core low power region fuel bundles.

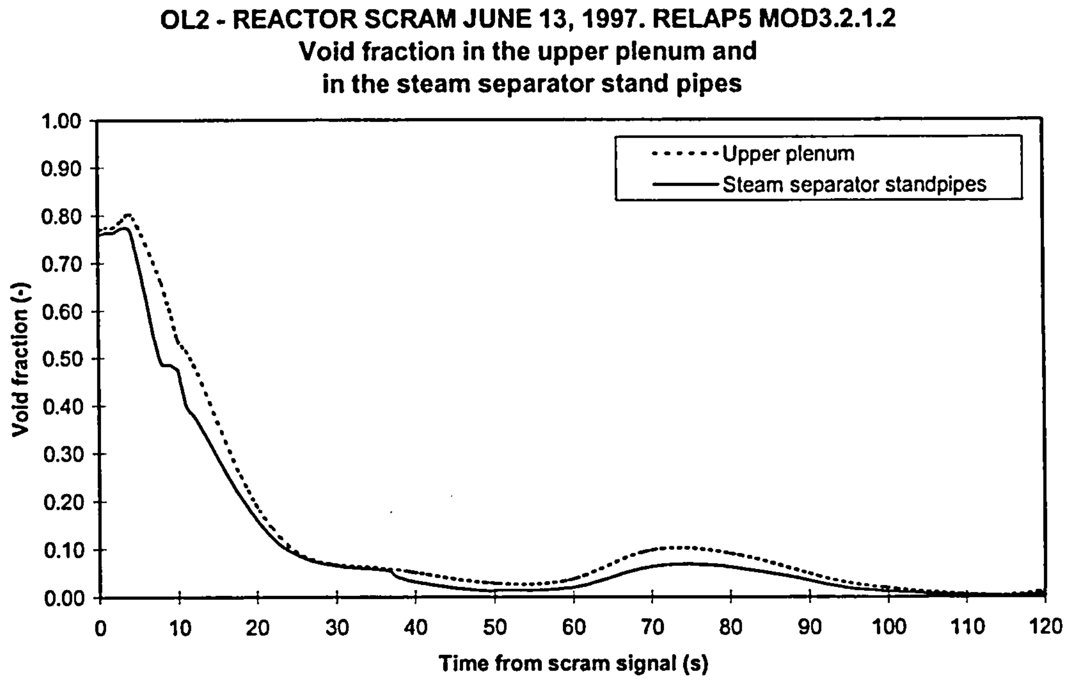


Figure 10. Olkiluoto 2 reactor scram June 13, 1997. Void fraction in the upper plenum and in the steam separator standpipes.

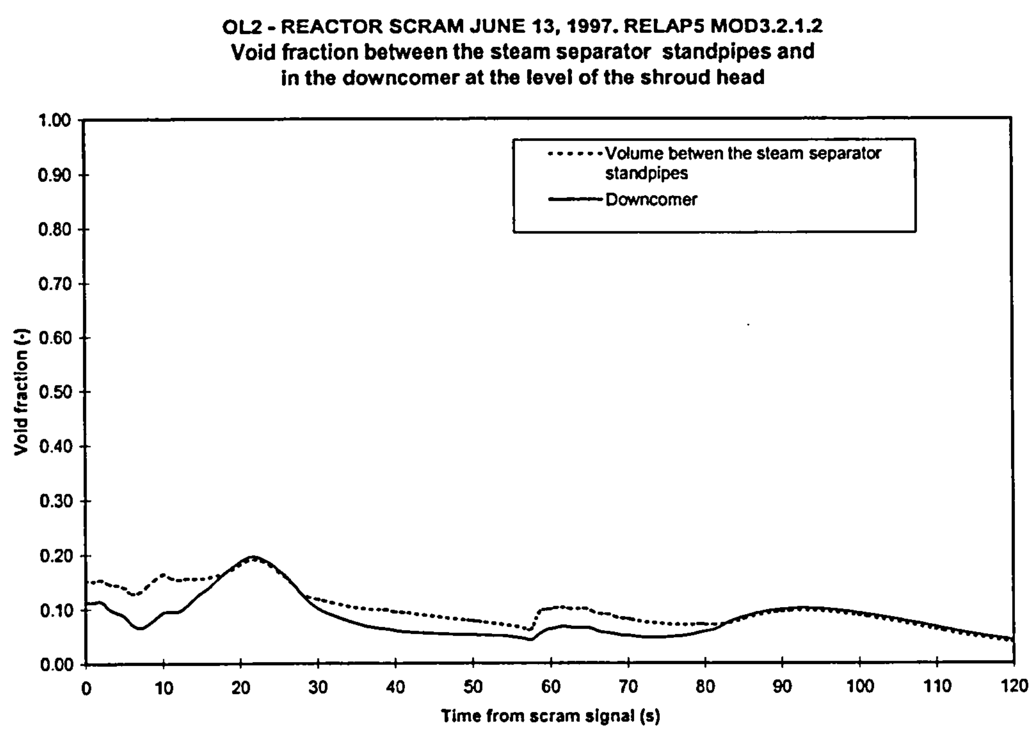


Figure 11. Olkiluoto 2 reactor scram June 13, 1997. Void fraction between the steam separator standpipes and in the downcomer at the level of the shroud head

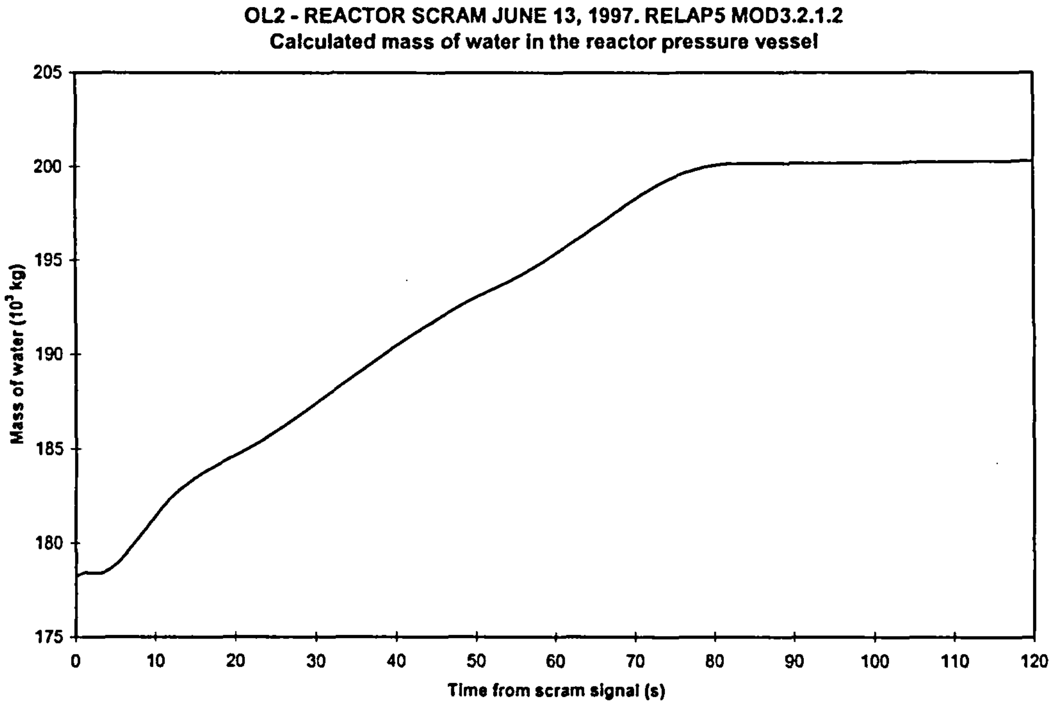


Figure 12. Olkiluoto 2 reactor scram June 13, 1997. Calculated mass of water in the reactor pressure vessel.

6. DIFFERENCES BETWEEN CALCULATIONS AND MEASUREMENTS

RELAP5 calculations agreed with the measurements at the beginning of the accident when the void in the reactor pressure vessel decreased rapidly because of the reactor scram. After the reactor water level minimum the void in the reactor core and in the water volumes above core started to increase again. At this point the difference between the calculated and the measured level increased. The reason is be that RELAP5 calculates the amount of steam condensing in water is larger than in reality. The amount of steam in water is thus smaller than the real void. The stronger steam condensation is also reflected by the observation that the calculated water level drops faster than the measured level. RELAP5 also calculates at the beginning of the transient a lower reactor pressure than what is measured. At this stage there is due to feed water flow a large amount of subcooled water in the vessel to condense steam.

The calculated water temperature is at the end of the transient higher than in reality because of the stronger steam condensation. Because the calculated pressure is essentially the same as the saturated steam pressure corresponding to the water temperature of the volumes above the core, the calculated pressure is higher than the measured value. The stronger steam condensation explains also the slower water level rise, because the amount of steam in water is smaller. Towards the end of the calculation the calculated and measured get closer.

At the end of the RELAP5 calculation the level increase is caused by the thermal expansion of water. At the end of the calculation the void fraction is about 0.1 in the upper part of the high power fuel bundles. The steam condenses almost completely in the upper plenum volume between the core and the shroud head. In the upper part of the down comer and between the steam separators the void fraction is about 0.04.

7. CONCLUSIONS

The reactor scram, which occurred at the Olkiluoto 2 BWR plant on June 13 in 1997 has been analysed with the RELAP5/MOD3.2.1.2 computer program. The RELAP5 model of the Olkiluoto reactors is based on the model prepared for analysing PSA level 1 success criteria. The purpose of the analysis was to investigate how well RELAP5 is able to calculate the behaviour of process parameters after scram and validate the RELAP5 model of the Olkiluoto reactors.

Several changes were required in the Olkiluoto 1 and 2 model, which was prepared before the scram, in order to reach good agreement between the calculation and the measurement data. The changes concerned nodal volumes, node junction flow areas and pressure loss coefficients and heat structures. Default values were retained for RELAP5 thermal hydraulic parameters.

The RELAP5 result agrees with the measured level at the beginning of the transient when the level drops due to the collapse of steam bubbles. According to the analysis the fine level dropped from the initial value +3.86 m above top of active fuel to the minimum value +2.91 m in 23 s. According to the measurements the same minimum level was reached at 24 s after the scram. The level rise from 24 s to 120 s due to supply of feed water and the warming up of the reactor water inventory is slower in the RELAP5 analysis than according to the measurements. The reason can be that RELAP5 calculates the amount of steam condensing in water is larger than in reality. The amount of steam in water is thus smaller than the real void. The stronger steam condensation is also reflected by the observation that the calculated water level drops faster than the measured level. RELAP5 also calculates at the beginning of the transient a lower reactor pressure than what is measured. At this stage there is due to feed water flow a large amount of subcooled water in the vessel to condense steam.

8. REFERENCES

1. 2-KK-R6-2/97, OL2 – Reactor scram 13.6.1997

2. NUREG/CR-5535, INEL-95/0174, Vol.1, RELAP5/MOD3 Code Manual, Code Structure, System Models, and Solution Methods

3. NUREG/CR-5535, INEL-95/0174, Vol.2, RELAP5/MOD3 Code Manual, User's Guide and Input Requirements

4. NUREG/CR-5535, INEL-95/0174, Vol.3, RELAP5/MOD3 Code Manual, Models and Correlations

5. NUREG/CR-5535, INEL-95/0174, Vol.4, RELAP5/MOD3 Code Manual, User's Guidelines

6. ENE4/22/96, OL1/OL2 - UPSA – Level 1 PSA success criteria for 2500 MWth reactor power, S14-08. Feed water system 312 pipe guillotine break, research report, VTT Energy, May 2, 1996

7. S21-TR-M-143/95, TVO I/II - Code input data to account for GENE AS2B steam separator, shroud cover and core flooder distributors

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11. ABSTRACT *(200 words or less)*

The reactor scram, which occured at Olkiluoto 2 June 13 in 1997 has been analysed with the RELAP5/MOD3.2.1.2 computer program. The RELAP5 model of the Olkiluoto reactors is based on the model prepared for analysing PSA level 1 success criteria. The purpose of the analysis was to investigate how well RELAP5 is able to calculate the behaviour of process parameters after scarm and validate the RELAP5 model of the Olkiluoto reactors.

The main plant features connected with the event are briefly presented. Modelling aspects are discussed regarding the RELAP5 analysis.

The RELAP5 result agrees with the measured reactor water level at the beginning of the transient when the level drops due to the collapse of steam bubbles. Later the level rise due to supply of feed water and the warming up of the reactor water inventory is slower in the RELAP5 analysis than according to the measurements.

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