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Analyzing Operator Actions During Loss of AC Power Accident with Subsequent Loss of Secondary Heat Sink

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ABSTRACT

The thermo-hydraulic analysis of plant response on total loss of AC power is very demanding and challenging job due to a number of phenomena included. Such an analysis becomes even more complicated and interesting if we include also the assumption of total loss of secondary heat sink.

In this paper we are presenting the NPP Krško specific analysis of complete loss of AC power with subsequent total loss of secondary heat sink and influence of specific operator actions. The aim of this analysis is to verify if emergency operating (EOP) or severe accident management guidelines (SAMG) procedures should be changed and if design change on pressurizer pressure relief valves (PORV) should be implemented to be able to cope with this kind of accidents better.

The analyses were performed with three different state of the art codes used at NPP Krško and IJS: RELAP5/MOD3.3, ANTHEM and MAAP4. The last two codes are used in the NPP Krško plant specific full scope simulator, one for the simulation of the design bases transients and accidents and the second for simulation of the severe accidents.

This type of analyses has been done also for the simulator validation, performed during vendor and site acceptance testing.

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ACKNOWLEDGEMENT

The RELAP5/MOD3.3 NPP Krško base input model, nodalization diagram and plant full scope simulator data are courtesy of Krško NPP.

ABBREVIATIONS

AC	Alternating Current
AFW	Auxiliary Feedwater
CC	Component Cooling
CET	Core Exit Temperature
CFR	Code of Federal Regulations
CVCS	Chemical and Volume Control System
ECCS	Emergency Core Cooling system
EOP	Emergency Operating Procedures
HPIS	High Pressure Injection System
IJS	Institut "Jožef Stefan"
LPIS	Low Pressure Injection System
MFW	Main Feedwater
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
PORV	Power Operated Relief Valve
PRZ	Pressurizer
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RCP	Reactor Coolant Pump
RHRS	Residual Heat Removal System
SAMG	Severe Accident Management Guidelines
SG	Steam Generator
TSC	Technical Support Center

1. INTRODUCTION

Station blackout is the accident where it is assumed that all plant AC power sources are lost. That means that all the offsite AC power sources are unavailable and at the same time it is also assumed all AC sources on the site are lost. Within this paper we are going to discuss concurrent total loss of all AC and loss of secondary heat sink (no SG feed available). One should know that this type of the accident is beyond the design bases of all the nuclear power plants currently operating around the word. If AC power is not recovered within certain time this scenario will lead to the severe accident sequence of events when the core damage and degradation becomes unavoidable.

This type of the accidents would also result in loss of coolant accident due to the fact that the operators would not be able to isolate the letdown and reactor coolant pump seal leakoff lines. This is true in case of NPP Krško and many other plants of same or similar design, since isolation valves on those lines are motor driven valves. As a consequence of loss of seal injection and the fact that the thermal barrier cooling is also lost, reactor coolant pumps (RCP) seals are exposed to high temperature of the primary coolant. This will cause seals degradation and seal leakoff flow will develop to a break flow. This is one of the well-known safety issues, which are known as the unresolved safety issues. This one was recognized as no.25 [1]. As a response on this issue the industry initiated a large research program, which resulted in certain seal improvements. New seals have higher temperature resistance so it is predicted that instead after approximately 30 minutes they would fail after approximately 2 hours.

If the AC power is not recovered before the core is uncovered and significantly overheated, core melt and consequently primary system failure from high pressure could not be avoided. As soon as the core exit temperature is higher than 650 °C for more than 30 minutes it is considered that accident becomes a severe accident. For this type of the accidents nuclear industry, including NPP Krško, have developed so called Severe Accident Management Guidelines (SAMG), where the main objectives are to maintain fission products barriers intact (primary system and containment). If this is not possible or in case that release of fission product is in progress then these procedures (SAMG) will suggest actions to minimize the fission products release to the environment, with the purpose to protect the public health.

2. PLANT DESCRIPTION

NPP Krško is a Westinghouse 2-loop PWR plant, in commercial operation since 1983.

2.1 Containment

The NPP Krško Nuclear Power Plant utilizes a cylindrical steel shell with a hemispherical dome and ellipsoidal bottom designed to accommodate normal operating loads, functional loads resulting from a loss-of-coolant accident, and the most severe loading predicted for seismic activity. A concrete shield building surrounds the steel shell to provide biological shielding for both normal and accident conditions and to provide collection and holdup for leakage from the containment vessel. Inside the containment structure, the reactor and other NSSS components are shielded with concrete. In addition to a containment spray system, a containment recirculation and cooling system is provided to remove post-accident heat.

2.2 Turbine Building

The turbine building contains the turbine generator and all the power conversion related accessories. The building is of closed construction. The building does not contain any safety related equipment and is designed in accordance with local and national building codes.

2.3 Auxiliary Building

The auxiliary building structures are of reinforced concrete design with shear walls and beam and slab floor systems. The portion of the auxiliary building that is below grade elevation is suitably protected with a waterproofing membrane to prevent the intrusion of groundwater. In addition, redundant safety equipment below grade is located in separate compartments to preclude simultaneous flooding due to a fluid-system rupture.

2.4 Intake Structure

The intake structure consists of two separate substructures: a non-safety category structure containing the main condenser circulating pumps and related equipment and a safety category structure containing the service water pumps and the related equipment. For cooling water intake, a dam has been built across the Sava River with the pumping station, and water intake and discharge structures. Two batteries of cooling cells are included for combined cooling in the event of low river flow rates.

2.5 Fuel Handling Building

The fuel handling building is an integral part of the auxiliary building and is a reinforced concrete structure that utilizes shear walls and beam and slab floor systems. The spent fuel pool within the fuel handling building is lined with stainless steel to prevent leakage of water.

2.6 Nuclear Steam Supply System

The power rating of the NPP Krško nuclear steam supply system (NSSS) is 2000 MWt, composed of 1994 MWt core power output plus 6 MWt of reactor coolant pump heat input. The NSSS consists of a pressurized water reactor, reactor coolant system (RCS) and associated auxiliary fluid systems. The RCS is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

2.7 Reactor Core

The reactor core is composed of 121 fuel assemblies. Square spacer grid assemblies and the upper and lower end fitting assemblies support the fuel rods in fuel assemblies. Each fuel assembly is composed of 16 x 16 rods; of these only 235 places are used by fuel rods; of the 21 remaining places, 20 places which are evenly and symmetrically distributed across the cross section of the assembly, are provided with thimble tubes which may be reserved for control rods, and one control instrumentation tube for incore thimble.

Of all fuel assemblies in the core, 33 are equipped with control rod clusters. The core is of the multi-region type. All fuel assemblies are mechanically identical, although the fuel enrichment is not the same in all assemblies. Fuel assemblies with the highest enrichments were placed in the core periphery, and the two groups of lower enrichment fuel were arranged in a selected pattern in the central region. Core design strategy depends on plant operation strategy (length of cycle) and improvements in fuel design.

2.8 Reactor Coolant System

The RCS consists of two reactor coolant loops connected in parallel to the reactor vessel, each loop containing a reactor coolant pump and a steam generator. The coolant loops are filled with high pressure water driven by the reactor coolant pumps. The water circulates through the reactor core to remove the heat from the fuel assemblies generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generator. Here it gives up its heat to the feedwater to generate steam for the turbine generator.

The reactor coolant pumps, one per coolant loop, are Westinghouse vertical, single-stage, centrifugal pumps of the shaft-seal type. The power supply system to the pumps is designed so that adequate coolant flow is maintained to cool the reactor core under all conceivable circumstances. The pump capacity is about 17,000 t/h. All pump parts in contact with the coolant are made of austenitic steel or stainless steel covered.

The steam generators, one per loop, are vertical U-tube units, recently installed Siemens-Framatome steam generators type SG 72 W/D4-2, which replaced highly degraded Westinghouse D-4 steam generators with preheater. Internal moisture separation equipment reduces the moisture content of steam to 0.1 % or less.

The reactor coolant piping and all of the pressure-containing and heat transfer surfaces in contact with reactor water are stainless steel except the steam generator tubes and fuel tubes, which are Inconel and zircaloy, respectively. Reactor core internals, including control rod drive shafts, are primarily stainless steel.

An electrically heated pressurizer connected to one reactor coolant loop maintains RCS pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions.

2.9 Auxiliary Systems

Auxiliary systems components are provided to charge the Reactor Coolant System and to add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control (CVCS – Chemical and Volume Control System), cool system components (CC – Component Cooling), remove residual heat when the reactor is shut down (RHRS - Residual Heat Removal System), change fuel and cool the spent fuel storage pool, sample the reactor coolant water, provide for safety emergency injection, and vent and drain the Reactor Coolant System.

2.10 Engineered Safety Features Systems

Engineered safety features are provided to prevent accident propagation or to limit the consequences of postulated accidents, which might otherwise lead to damage of the system and release of fission products. The principal criteria is that under the conditions of a hypothetical loss of coolant accident, the system can, even when operating with partial effectiveness, maintain the integrity of the containment, and limit the potential offsite radiation dose to less than the values of applicable US Federal Regulations (10 CFR Part 100). A number of engineered safety features are included in this plant, as follows:

$ Containment Spray System

$ Hydrogen Control System

$ Emergency Core Cooling System

$ Component Cooling Water System

$ Essential Service Water System

$ Auxiliary Feedwater System

3. ACCIDENT PROGRESSION AND OPERATOR ACTIONS

Current emergency operating procedures cover design bases accidents and also a part of the beyond design bases accidents. One should know that this type of the procedures have been enhanced and completely revised after Three Mile Island accident in beginning of eighties of the past century. Emergency operating procedures have been developed with the goal to cover accidents with initiating frequency higher than 1E-6 per year. They have been developed on a bases that the operators do not need to know what kind of accident is in progress, that's why they are also called as "symptom based" and not "event based" as before. Among these procedures one covers the case in which all AC power is lost. Within this procedure operators are instructed to try to restore the offsite or onsite AC power source, to power at least one train of emergency core cooling systems. In the meantime the operators are instructed to isolate paths through which primary coolant is lost - mainly letdown and reactor coolant pump seal leakoff lines. Due to high primary coolant temperature and no thermal barrier cooling, reactor coolant pump seal degradation is expected due to high temperature seal leakoff flow. As a consequence a relatively small leak flow will be increased to a break flow with the upper bound value of 25 gpm [2]. If the secondary heat sink is available the operators are instructed to cooldown and depressurize the primary system with secondary heat sink - steam generators. There are the following main benefits of this action. The first is to decrease the leakage - break flow of the primary coolant due to lower primary pressure and the second is that if the primary pressure will decrease low enough so the passive injection of additional borated water from the accumulators can be expected. By this we would gain some cold borated water and increase the mass of the primary coolant. As a consequence operators are gaining some additional time for AC power restoration.

If in a case of loss of all AC power we also assume the failure of the turbine driven auxiliary feedwater pump, this will cause also the loss of secondary heat sink. Per the current EOP procedures [3] operators are not instructed to initiate primary system cooldown by opening of the steam generators pressure control valve(s). They are instructed to loop in the procedure, trying to restore secondary heat sink and AC power. Due to that the primary temperature and pressure will, after all the secondary side steam generator water will boil off, start to increase (see Figure 1, 2 or 3 case with 0 SG and 0 PRZ PORVs). Available analyses [2], [4], [5], [6] indicate that the break flow through the letdown line and reactor coolant seals will not be large enough to bring the pressure below the accumulator injection pressure. Above-mentioned seal improvements support such a conclusion even further.

Since the pressure and the temperature will increase in the primary system and the fact that pressurizer pressure relief control valves are not operable due to the fact that there is no instrument air available, primary safety valves will control the pressure in the reactor coolant system. Consequently the break flow rate will significantly increase and speed up the coolant depletion. This will cause the core uncovery and heatup of the core up and above the 650 °C. At this point the operators are instructed to leave emergency operating procedures and enter the severe accident management guidelines. Within these procedures [7] technical support center (TSC), which is responsible for making the decisions and instructions for the operators, will not be able to provide the instructions how to decrease the primary system pressure. For further development of this type of the accident it is beneficial that the primary system fails at low internal pressure (reactor vessel failure or hot leg creep failure) [5], [7].

Even with use of SAMG procedures TSC and the operators will not be able to prevent further core degradation without any AC power. Eventually core will melt and primary system will fail from high internal pressure (see Figure 3 and 4). Operators and plant technical support team will then focus on trying to prevent containment failure and to minimize fission product release to the environment.

As can bee seen from the above discussion it is essential to try to meet the following objectives during such an accident:

$ prolong as much as possible the time until the core will be overheated, melted and the primary system would fail - to gain additional time for AC power restoration and

$ decrease the primary pressure to prevent high pressure reactor vessel failure, which is causing potential threat to the containment integrity due to so call direct containment heating phenomena [4]. Small very hot particles could cause very rapid containment pressure increase and its failure. In case that the containment reactor vessel would fail at low pressure, core debris would be relocated into the reactor cavity and no immediate threat to the containment integrity is expected [5].

To get appreciation of certain key operator actions and to see if it is feasible also to consider any potential design change in the future, we decided to perform further analyses of this transient.

4. INPUT MODEL DESCRIPTION

To perform this analysis, NPP Krško has provided the base input model, so called "Master input deck". The scheme of the NPP Krško nodalization for the RELAP5/MOD3.3 [8] code is presented in Figure 1. A full two-loop plant model was developed, including the new Siemens-Framatome replacement steam generators (RSG) type SG 72 W/D4-2.

The model consists of 469 volumes, connected with 497 junctions. Plant structure is represented by 376 heat structures with 2101 mesh points, while the reactor protection and regulation systems, safety systems operational logic and plant instrumentation is represented by 401 logical conditions (trips) and 575 control variables.

4.1 Hydrodynamic component description

Components numbered from 101 to 165 represent reactor vessel in the following manner:

171, 173 and 175	- lower downcomer
101 and 103	- lower head
105	- lower plenum
107	- core inlet
111	- reactor core
115	- core baffle bypass
121	- core outlet
125, 131 and 141	- upper plenum
151 and 153	- upper head
165	- upper downcomer
113 and 145	- guide tubes

Components numbered 51, 53 and 55 represent the pressurizer surge line and volumes 61, 63, 65, 67 and 69 represent the pressurizer vessel. Pressurizer spray lines (80, 81 and 84) are connected to the top of the pressurizer vessel and include the spray valves numbered 82 and 83. Valves numbered 28 and 32 represent the two pressurizer PORVs and valves numbered 14 and 22 represent pressurizer safety valves.

Primary piping is represented by the following components:

201, 203, 205, 207, 209 and 211	- hot leg no.1
251, 253, 255, 257 and 259	- intermediate leg no.1 with cold leg no.1 loop seal
265, 271, 273, 275, 277 and 279	- cold leg no.1 with the primary coolant pump no.1
301, 303, 305, 307, 309 and 311	- hot leg no.2
351, 353, 355, 357 and 359	- intermediate leg no.2 with cold leg no.2 loop seal
365, 371, 373, 375, 377 and 379	- cold leg no.2 with the primary coolant pump no.2

Loops are symmetrical except for the pressurizer surge line and CVCS connections layout.

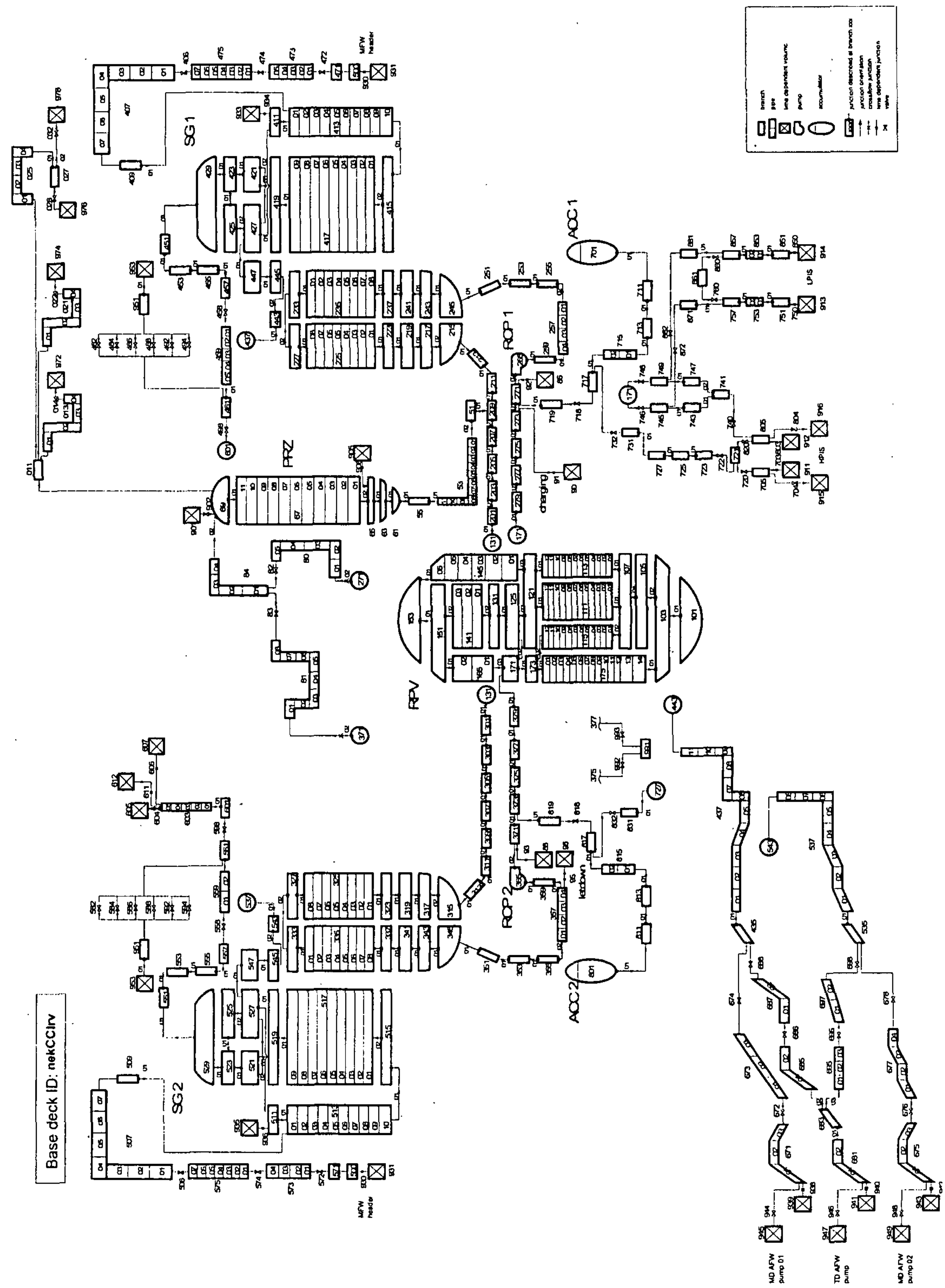

Figure 1: NPP Krško nodalization scheme

ECCS piping nodalization and connections are represented by hydrodynamic components numbered from 701 to 882. The hydrodynamic components representing HPIS pumps are time-dependent junctions 703 and 803, while time-dependent junctions 750 and 850 represent LPIS pumps. Accumulators are numbered 701 and 801 their lineup provides cold leg injection only. ECCS connects to both cold legs (junctions 719-01 and 819-01). Direct vessel ECCS injection through junction no. 746 and 748 opens simultaneously at the SI signal generation.

Primary side of the SG is represented by inlet and outlet plenum, among which a single pipe is representing the U-tube bundle:

215, 217 and 219 -	- SG 1 inlet plenum (hot side) and tube sheet inlet
223, 225, 227, 233, 235, and 237	- SG 1 U-tubes
241, 243 and 245	- SG 1 tube sheet outlet and outlet plenum (cold side)
315, 317 and 319 -	- SG 2 inlet plenum (hot side) and tube sheet inlet
323, 325, 327, 333, 335, and 337	- SG 2 U-tubes
341, 343 and 345	- SG 2 tube sheet outlet and outlet plenum (cold side)

The parts of the SG secondary side are represented by the following hydrodynamic components:

415, 417 and 419	- SG 1 riser
421 and 427	- SG 1 separator and separator pool
411 and 413	- SG 1 downcomer
423, 425 and 429	- SG 1 steam dome
515, 517 and 519	- SG 2 riser
521 and 527	- SG 2 separator and separator pool
511 and 513	- SG 2 downcomer
523, 525 and 529	- SG 2 steam dome

Main steamlines are represented by volumes 451, 453, 455, 457, 459 and 461 (SG 1) and 551, 553, 555, 557, 559 and 561 (SG 1), divided by main steam isolation valves (458 and 558). SG relief (482 and 582) and safety valves (484, 486, 488, 492, 494 and 584, 586, 588, 592, 594) are situated upstream the isolation valves. Turbine valve (604) and steam dump (611) flow is regulated by corresponding logic.

Main feedwater (MFW) piping is represented by volumes 471, 473, 475, 407, 409 (SG 1) and 471, 573, 575, 507, 509 (SG 2), branching from main feedwater header (500).

Auxiliary feedwater (AFW) is injecting above the SG riser (via volumes 437, 443, 445 and 447) and its piping is represented by volumes 671, 673 (motor driven AFW 1), 675, 677 (AFW 2), and 681, 683, 685, 687, 695, 697 (turbine driven AFW).

4.2 Regulation and protection logic

In order to accurately represent the NEK behavior, a considerable number of control variables and general tables are part of the model. They represent protection, monitoring and simplified control systems used only during steady state initialization, as well as main plant control systems:

$ rod control system,

$ PRZ pressure control system,

$ PRZ level control system,

$ SG level control system and

$ steam dump.

It must be noted that rod control system has been modeled for point kinetics. Present model can be used for transient analysis with two options:

$ with constant or predefined core power transient as function of time (including decay power calculation) or

$ with rod control system in auto or manual mode.

The following plant protection systems are defined using trip logic:

$ reactor trip,

$ SI signal,

$ turbine trip,

$ steamline isolation,

$ main feedwater isolation and

$ auxiliary feedwater start.

5. RESULTS

For the reason stated above the analyses with the state of the art tools were performed to check below listed potential procedure and/or design change:

$ Potential emergency operating procedure change by which the operators would be instructed to start secondary side depressurization even in the case if there is no feed into SG.

$ If the potential design change which will assure pressurizer PORV operability under total loss of all AC power would increase available time before the core heatup starts (to open the primary system bleed path with the purpose to get feed from the accumulators) and would depressurize primary system enough after core degradation and relocation.

Since this accident is very complicated and the operator actions are very important the following tools have been used to perform required analyses:

$ RELAP5/MOD3.3 (for the analyses of available time before the core degradations starts),

$ ANTHEM (for the analyses of available time before the core degradations starts) and

$ MAAP4 (for the analyses of available time before the core degradations starts and analysis for potential primary system depressurization at the point of entrance to the SAMG procedures and no operator actions for primary system cooldown and depressurization was performed before).

For above mentioned, the analyses of the accident up to the point of fuel heatup due to low coolant level in the core was first performed with the RELAP5/MOD3.3 code [7], [9] and with the NPP Krško plant engineering simulator using ANTHEM.

ANTHEM [10], [11] is the two phase 5 equation (plus conservation equation for non-condensable mass), drift flux code build in the simulator for the simulation of the nuclear steam supply systems. With this plant engineering simulator we have capability to simulate all the design bases and beyond design bases accidents. Due to that for the nuclear steam supply systems and containment simulation we use ANTHEM for simulation of all the accidents that are covered by the emergency operating procedures and MAAP4 code if we want to simulate severe accidents including core degradation, core relocation, reactor vessel failure, molten core concrete interaction and containment failure.

5.1 Base analyses

For the first set of the analyses the assumed operator actions and sequence of main events is seen from Table 1 and Table 2.

All RELAP5/MOD3.3 runs include 1000 s of initial steady-state, while ANTHEM and MAAP4 curves start from the occurrence of Loss of all AC power.

Table 1: Loss of all AC power - assumptions common for all analyses

Assumed action - Event	Time (s) (All analyses)		
Number of PRZ PORVs -->	0	1	2
Loss of all AC/Reactor trip/loss of all SG feed	0.0	0.0	0.0
Assumed letdown isolation and start of RCP seal degradation	300	300	300
Assumed complete RCP seal degradation	2100	2100	2100
Assumed start of the first SG depressurization[1]	N/A	300	300
Assumed first PRZ PORV opening[2]	N/A	Note 2	N/A
Assumed second PRZ PORV opening[2]	N/A	Note 2	Note 2

Table 2: Comparison of available time - time until the core heatup - degradations starts

Assumed action - Event	Time (s) RELAP5/MOD3.3			E-SIM-ANTEM			E-SIM-MAAP4		
PRZ PORVs ->	0	1	2	0	1	2	0	1	2
Time before core heatup[3]	9600	7040	12100	8500	6110	12270			
RCS/Reactor vessel failure							10400	9190	15200

On Figure 2 primary pressure behavior for all the cases analyzed by RELAP5/MOD3.3 is shown. On Figure 3 the same parameter is shown calculated by plant engineering simulator - ANTHEM and on Figure 4 with MAAP4 (with the maximum seal break -300 gpm). MAAP4 code is built in the plant engineering simulator for the simulation of the severe accidents.

As it can be seen from the figures all the three codes are providing very similar prediction. The differences seen are mainly due to the fact that in RELAP5 analyses for all the breaks we defined the fix boundary conditions, while on simulator the break flow depends on the variable conditions in the system to which the break flow is directed. One can also observe that the plant engineering simulator is predicting higher rate of primary system pressurization after plant cooldown by the secondary side is terminated (no secondary heat sink). The reason for that is slightly higher decay power calculated by the core model and as already mentioned slightly different boundary conditions. It has to be pointed out that for the MAAP4 analyses we doubled the size of the assumed break on the RCP seals, since this assumption is conservative.

Based on Table 2 and Figure 2, Figure 3 and Figure 4 it can be concluded that in the case of loss of all AC and secondary heat sink the operators should try to depressurize the primary system as soon the steam generators are empty.

1 Second SG depressurization was assumed to be started by the operator when in the first SG wide range level drops below 8%

2 Pressurizer PORV opening depends on second SG wide range Level. It is assumed that operators would open one or both pressurizer PORVs after the second SG water on the secondary side is almost depleted (level < 4%)

3 Time until the fuel rod temperature is below 850 °C (RELAP) and core exit temperature below 750 °C

For better insight into the analyzed scenarios, additional parameters are shown in Figure 5 to Figure 13.

From total primary leak flow (seal leak + PRZ valves) curve, shown in Figure 5, one can recognize period of increasing RCP leak flow (300 s – 2100s). This is followed by a certain period of quasi steady leak flow. At the end of this period secondary heat sink was lost due to SG valves opening (Figure 6) and consequent depletion of the secondary inventory (Figure 7). After the secondary heat sink was completely lost, primary pressure increased to the point, when PRZ safety valves started to open. In the case labeled "0 PRZ PORVs", only PRZ safety valves were opening/closing on their set/reset pressure setpoints, while in the other two cases, labeled "1 PRZ PORV" and "2 PRZ PORVs", PRZ PORV(s) were opened by the operator to mitigate the consequences of the station blackout sequence of events. After the valves opening primary pressure was successfully decreased for a certain period of time, so RCP leak flow was decreased. Some spikes can be observed in the cases "1 PRZ PORV" and "2 PRZ PORVs". These origin from accumulator injection (Figure 8).

PRZ gradually emptied (Figure 9) after the PRZ valves opening and was not restored due to continuous primary coolant leakage. Primary inventory was also gradually lost (Figure 10), which soon caused core uncovery (Figure 11) and unavoidably led to core heatup (Figure 12 and Figure 13).

By opening this additional break in the primary system intentionally and in fact starting the bleed procedure they would be able to get feed from the accumulators which will have two long term effects. Time between beginning of the accident and core heatup will increase giving more time to the operators for the AC power restoration. This would only be true in the case when both pressurizer PORVs could be opened. If this is not the case then it is better not to perform these actions, since we only increase the coolant depletion rate but we are not able to get passive injection from the accumulators.

The second important effect, which is seen from the Figure 2 and Figure 3 as well as from the Table 1, is the primary system depressurization, which can be achieved by two pressurizer PORVs. This is one of the important objectives when we entered the area of severe accidents.

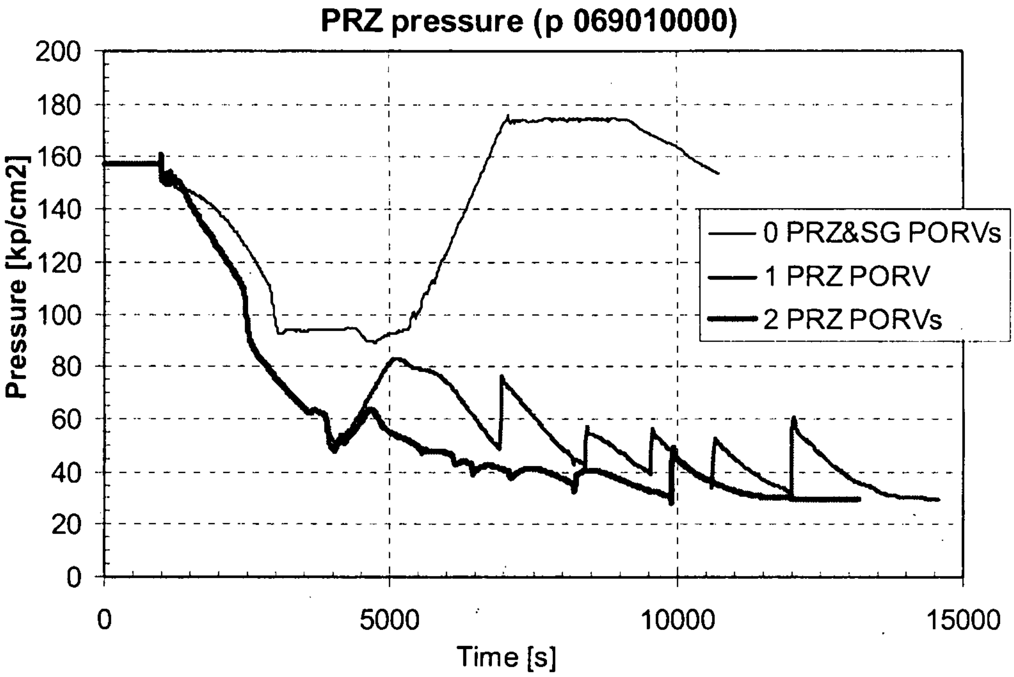

Figure 2: Primary system pressure - RELAP5/MOD3.3

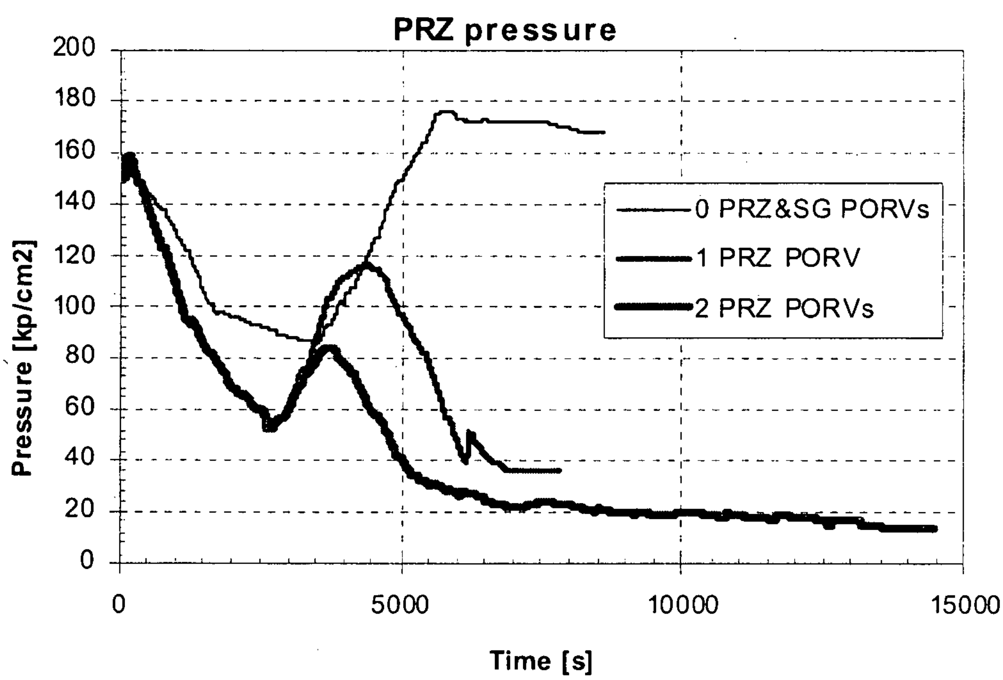

Figure 3: Primary system pressure - SIM- ANTHEM

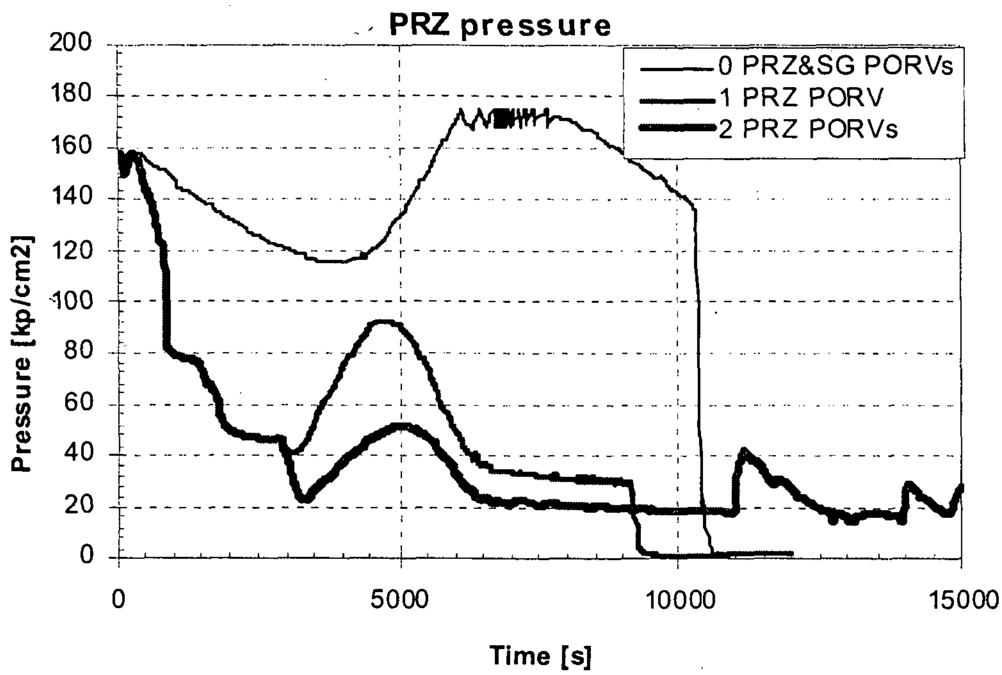

Figure 4: Primary system pressure - MAAP4

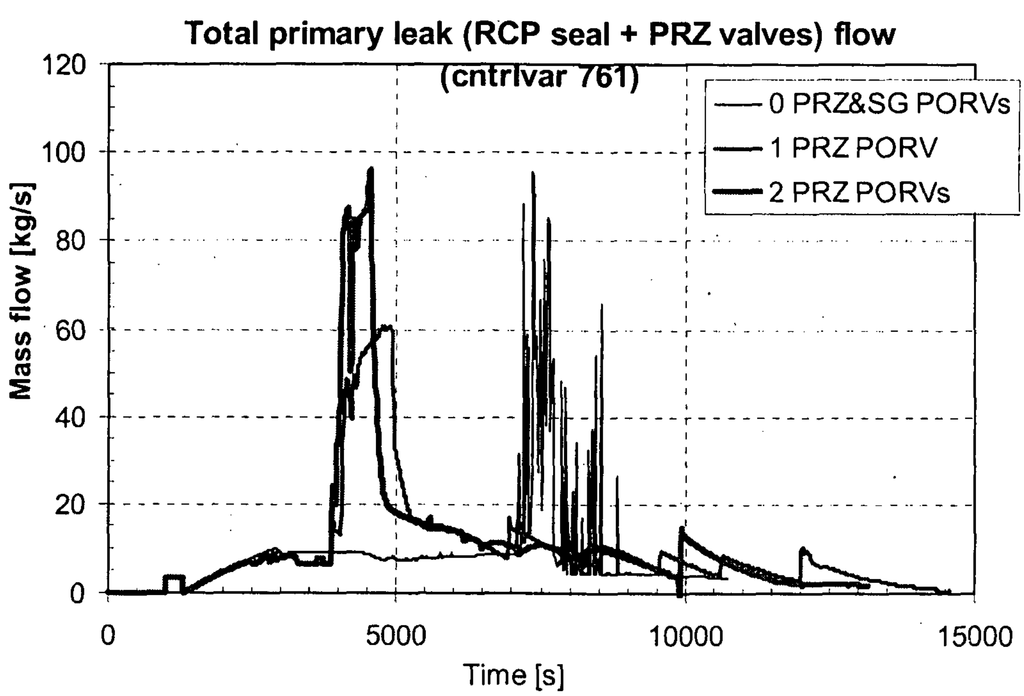

Figure 5: Total primary leak - RELAP5/MOD3.3

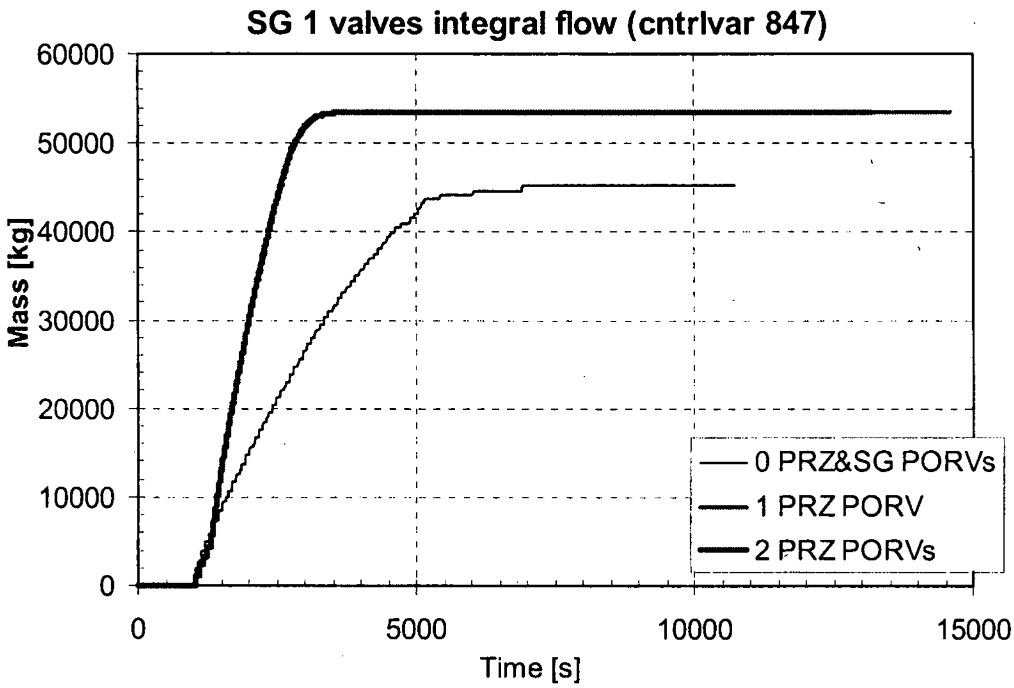

Figure 6: SG 1 valves flow - RELAP5/MOD3.3

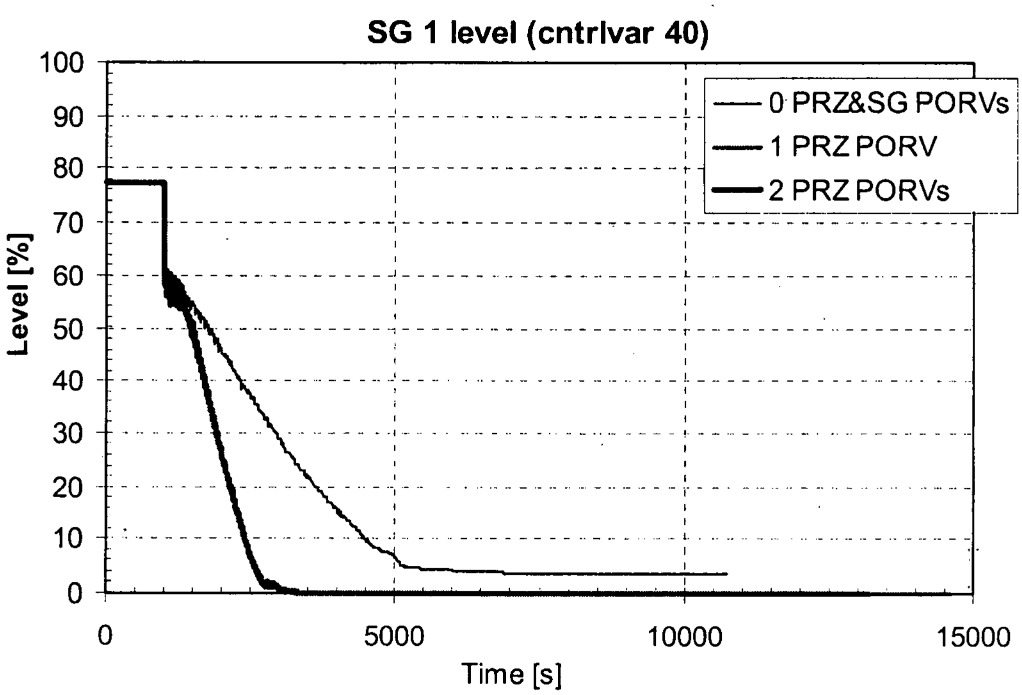

Figure 7: SG 1 WR level - RELAP5/MOD3.3

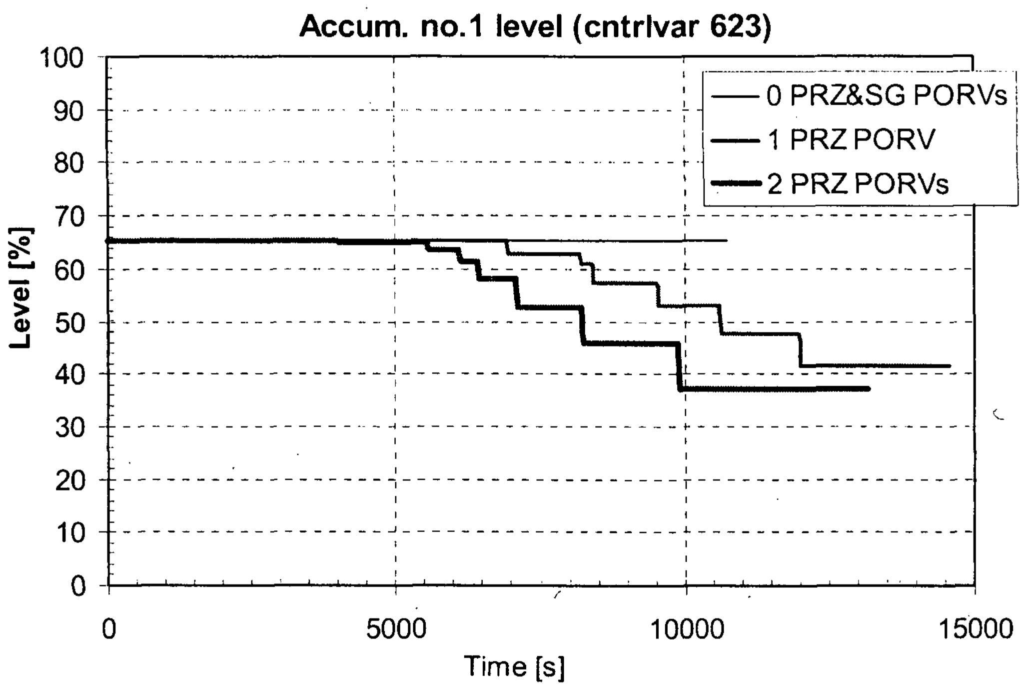

Figure 8: Accumulator no.1 level - RELAP5/MOD3.3

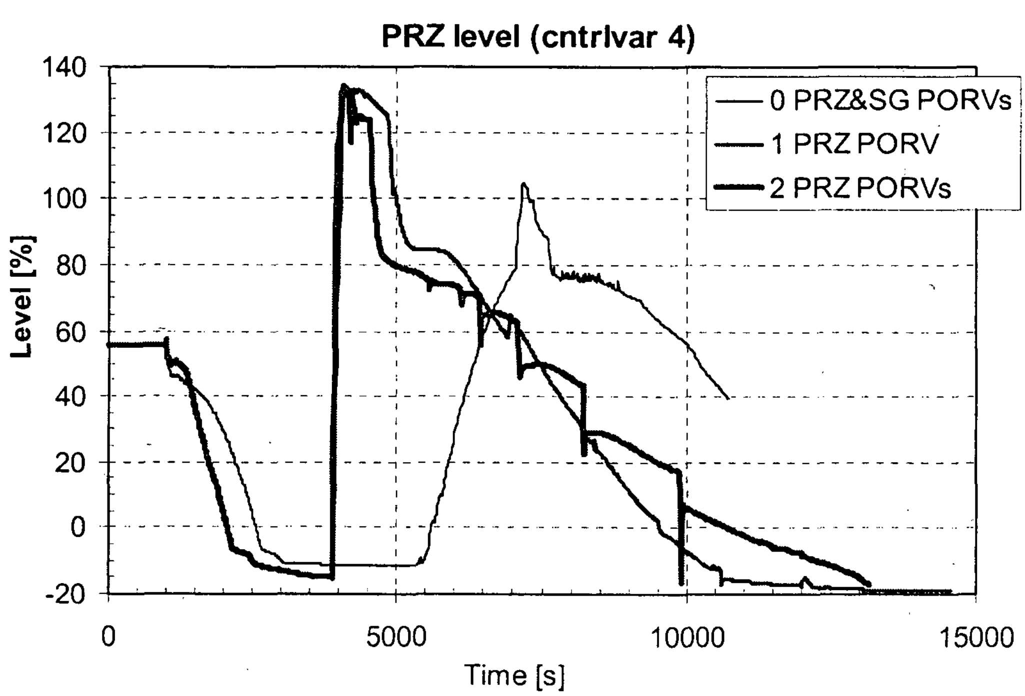

Figure 9: PRZ level - RELAP5/MOD3.3

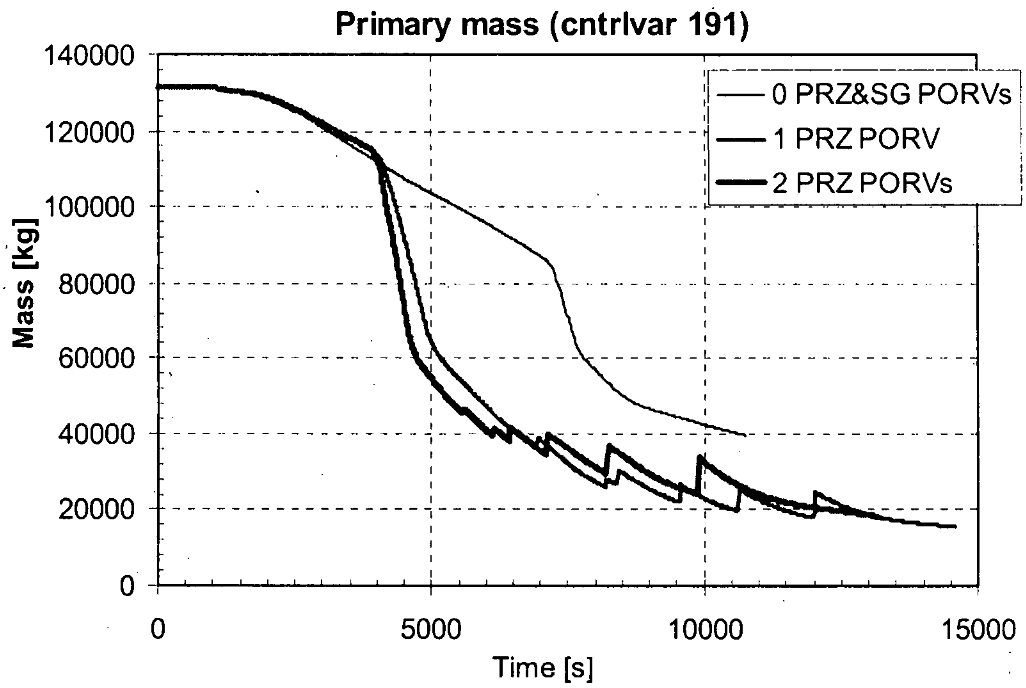

Figure 10: Primary mass - RELAP5/MOD3.3

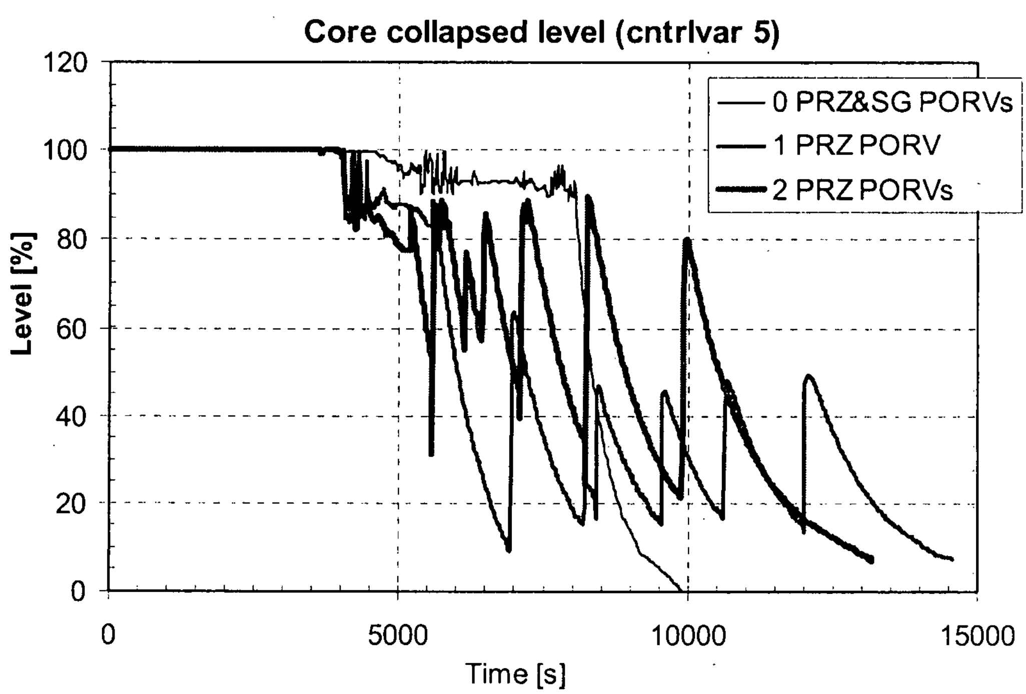

Figure 11: Core level - RELAP5/MOD3.3

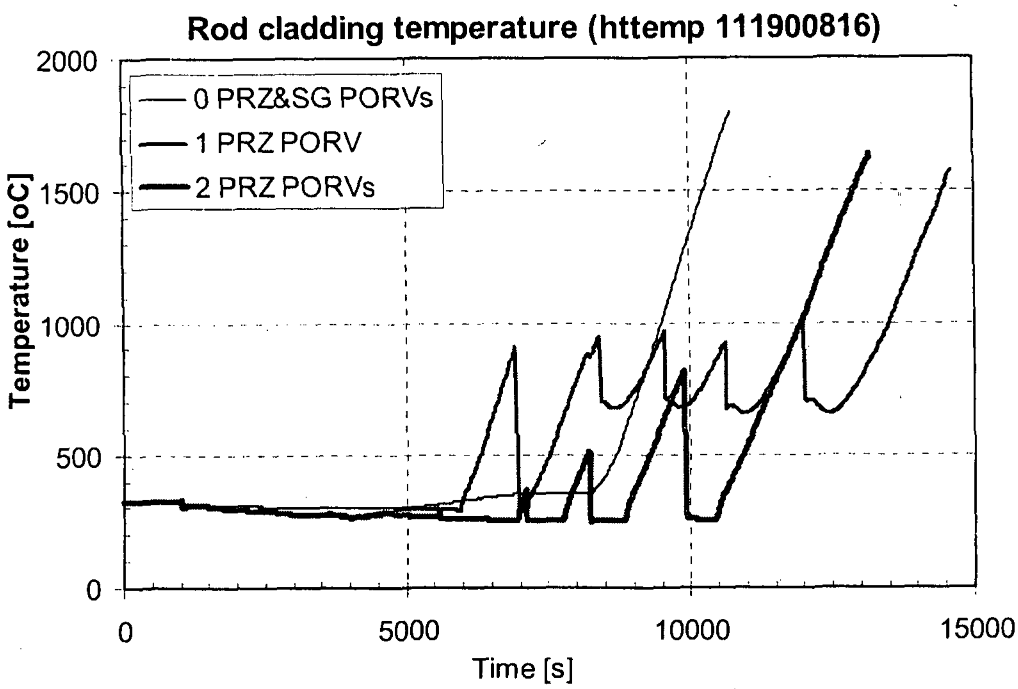

Figure 12: Rod cladding temperature - RELAP5/MOD3.3

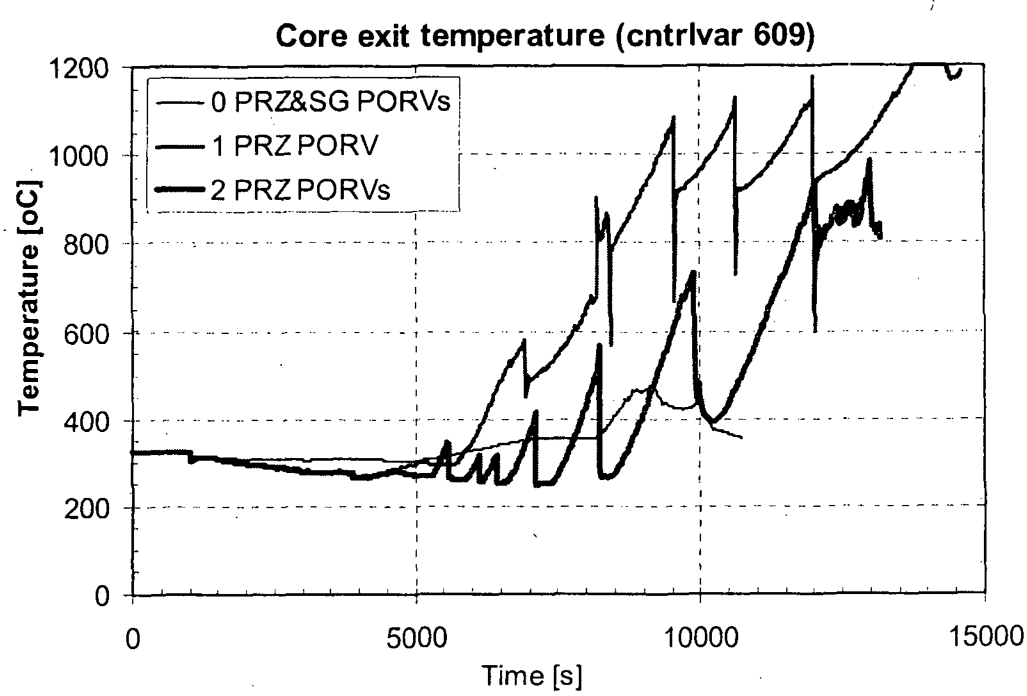

Figure 13: Core exit temperature - RELAP5/MOD3.3

5.2 Variation analyses

Second set of analyses performed only with the plant engineering simulator with MAAP4 code was done with the objective to verify if the primary system can be depressurized if we do not assume operator actions for SG depressurization. Here the objective was to check if it is feasible to expect that the operators will be able to depressurize primary system below the pressure for high-pressure debris ejection. For NPP Krško that would mean that at the time of the primary system failure internal pressure would lower than 22 kp/cm^2. For this analyses it was assumed that we would not change any of the steps in the emergency operating procedures, but rather in the SAMG procedures. It was assumed in the analyses that the operators would deliberately open both PRZ PORVs as soon as the core would be overheated - core exit temperature would be higher than 750 °C. During such an accident, assuming station blackout, loss of secondary heat sink, no injection and seal failure, core damage can not be avoided any more. Then the primary objective is to decrease the primary system pressure. Assumed operator actions and sequence of main events listed in Table 3

As it can be seen from the Figure 14, if the operators would be able to open both pressurizer PORVs, primary pressure would drop below 22 kp/cm^2. It is then expected that primary system would fail from low internal pressure. Even if they would open only one PORV this would not prevent the vessel failure however will fail from significantly lower pressure.

The same sorts of analyses have been performed by RELAP5/MOD3.3, to compare with MAAP4 results, of course only up to the point of significant core overheat. The only difference from the assumptions given in Table 3 was, that the second PRZ PORV opening (one or both PORVs) was performed when rod cladding temperature in core node no.8 exceeded 850 °C (instead core exit temperature > 750 °C).

Pressure development for the base case and the two variation cases is shown in Figure 15. It can be observed that even quantitatively both RELAP5/MOD3.3 and MAAP4 predicted the course of the transient very similarly.

Table 3: Loss of all AC power - assumptions for MAAP analyses

Assumed action – Event	Time (s) (All analyses)
Number of PRZ PORVs	2
Loss of all AC/Reactor trip/loss of all SG feed	0.0
Assumed letdown isolation and start of RCP seal degradation	300
Assumed complete RCP seal degradation	2100
Assumed start of first SG depressurization	N/A
Assumed first PRZ PORV opening	N/A
Assumed second PRZ PORV opening	Core exit temperature > 750 DEGC

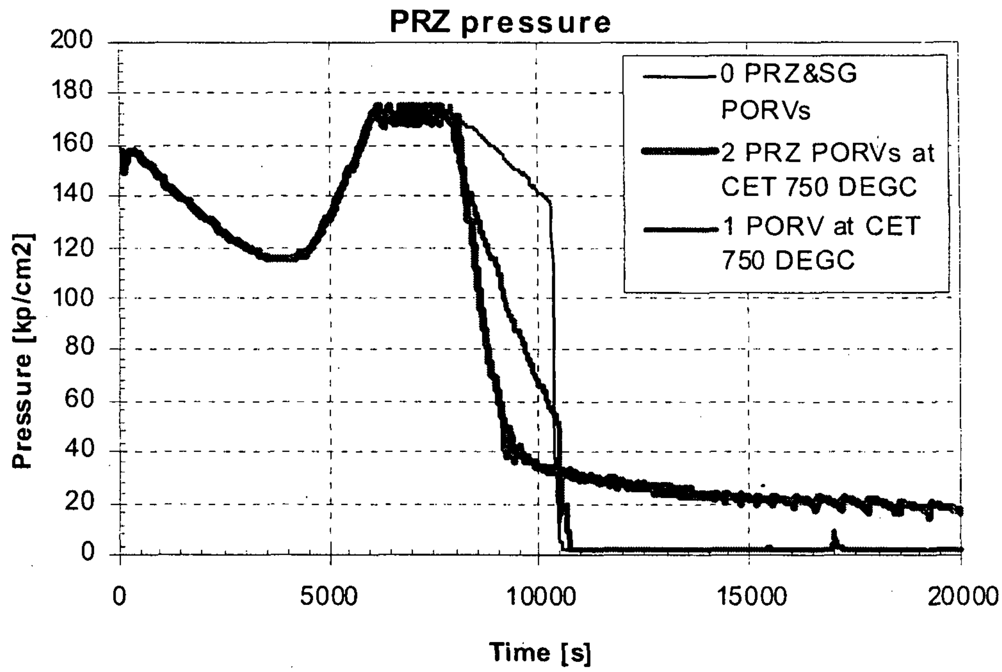

Figure 14: Primary system pressure without and with depressurization using PRZ relief valves – MAAP4

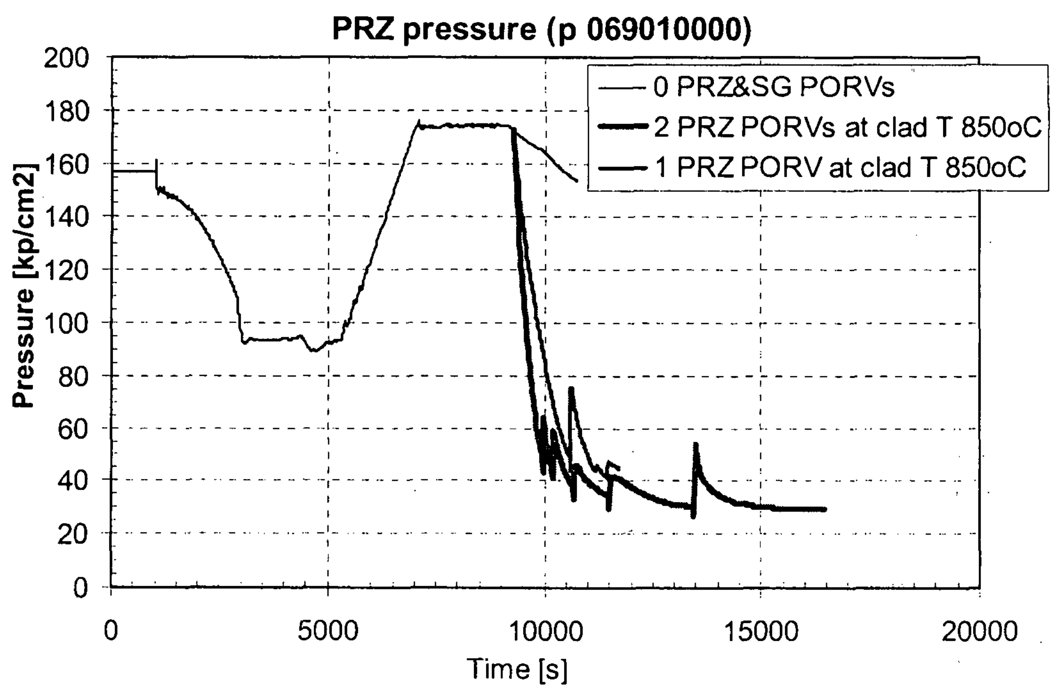

Figure 15: Primary system pressure without and with depressurization using PRZ relief valves – RELAP5/MOD3.3

6. RELAP5/MOD3.3 RUN STATISTICS

MOD3.3 Calculations were performed on SUN FIRE V880 server with 4 UltraSPARC III 750 MHz processors, with 16 GB main RAM, running under SOLARIS 9 operating System.

Table 4 shows run-time statistics for all the analyzed cases.

Table 4: Run time statistics

Analyzed case	Computer CPU time [s]	Total number of time steps (NT)	Total number of volumes (N)	Grind time CPU/(NT*N)
all cases - 1000 s of steady state	1669.90	26172	469	1.36E-04
0 PRZ&SG PORVs	7293.84	98268	469	1.58E-04
1 PRZ PORV	45931.01	688048	469	1.42E-04
2 PRZ PORVs	55307.01	850588	469	1.39E-04
1 PRZ PORV at clad T 850°C	11408.40	169260	469	1.44E-04
2 PRZ PORVs at clad T 850°C	42654.50	659750	469	1.38E-04

Consumed CPU time for all base and variation analysis cases is shown in Figure 16.

Mass error, time step and Courant Δt for the 3 base analyses cases and the 3 variation analyses cases are shown in Figure 17 to Figure 19 and Figure 20 to Figure 22, respectively.

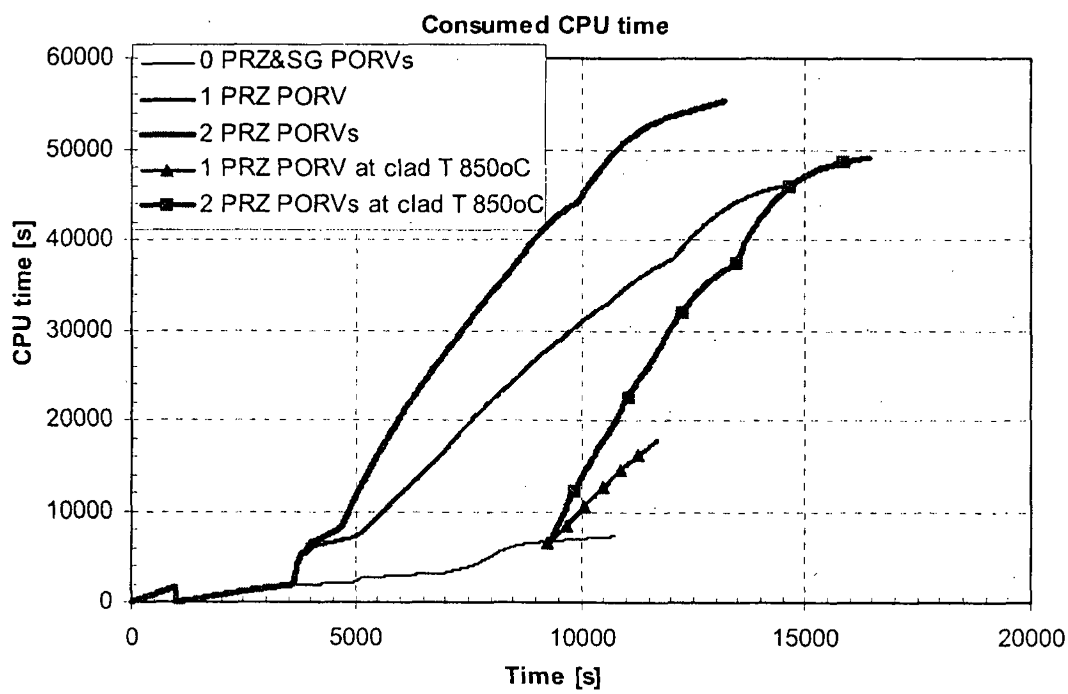

Figure 16: Consumed CPU time for all base and variation analyses cases– RELAP5/MOD3.3

Next is the excerpt from base case simulation with 0 PRZ&SG PORVs (time step statistics):

```
0 advancement       total   between edits|
  attempted:        26302       26302   |  min.dt=          0.        sec   last dt=   3.823475E-02 sec     emass=   8.279464E-03 kg
   repeated:            0           0   |  max.dt=          0.        sec   crnt.dt=   3:823475E-02 sec     tmass=   1.040078E+06 kg
 successful:        26302       26302   |  avg.dt=          0.        sec  merr.est=   2.385202E-10         em/tm=   7.960427E-09
  requested:        26304       26304   |  req.dt=   0.100000         sec       cpu=    2.15907     sec      time=    999.990      sec

  attempted:        28550        2248   |  min.dt=   3.293432E-02 sec   last dt=   5.839124E-02 sec     emass=    10.4555      kg
   repeated:            0           0   |  max.dt=   0.100000     sec   crnt.dt=   0.220598     sec     tmass=   1.034003E+06 kg
 successful:        28550        2248   |  avg.dt=   8.897246E-02 sec  merr.est=   3.884262E-07         em/tm=   1.011170E-05
  requested:        28552        2248   |  req.dt=   0.100000     sec       cpu=    179.494     sec      time=    1200.00      sec

  attempted:        36886        8336   |  min.dt=   6.002535E-02 sec   last dt=   6.672375E-02 sec     emass=    12.3051      kg
   repeated:            0           0   |  max.dt=   0.100000     sec   crnt.dt=   0.414794     sec     tmass=   1.011846E+06 kg
 successful:        36886        8336   |  avg.dt=   9.596929E-02 sec  merr.est=   1.421074E-07         em/tm=   1.216107E-05
  requested:        36888        8336   |  req.dt=   0.100000     sec       cpu=    844.417     sec      time=    2000.00      sec

  attempted:        45161        8275   |  min.dt=   5.408423E-02 sec   last dt=   5.408423E-02 sec     emass=    14.0035      kg
   repeated:            0           0   |  max.dt=   0.100000     sec   crnt.dt=   0.333887     sec     tmass=    991881.      kg
 successful:        45161        8275   |  avg.dt=   8.459215E-02 sec  merr.est=   9.242832E-08         em/tm=   1.411813E-05
  requested:        45163        8275   |  req.dt=   0.100000     sec       cpu=    1499.03     sec      time=    2700.00      sec

  attempted:        46394        1233   |  min.dt=   5.949265E-02 sec   last dt=   0.216676     sec     emass=    17.9101      kg
   repeated:            0           0   |  max.dt=   0.250000     sec   crnt.dt=   0.288100     sec     tmass=    983268.      kg
 successful:        46394        1233   |  avg.dt=   0.243309     sec  merr.est=   1.825119E-06         em/tm=   1.821492E-05
  requested:        46396        1233   |  req.dt=   0.250000     sec       cpu=    1597.83     sec      time=    3000.00      sec

  attempted:        65445       19051   |  min.dt=   1.984276E-03 sec   last dt=   0.132841     sec     emass=    44.3164      kg
   repeated:          730         730   |  max.dt=   0.250000     sec   crnt.dt=   0.328195     sec     tmass=    922300.      kg
 successful:        64715       18321   |  avg.dt=   0.163747     sec  merr.est=   2.378124E-07         em/tm=   4.804983E-05
  requested:        64717       18321   |  req.dt=   0.250000     sec       cpu=    2956.55     sec      time=    6000.00      sec

  attempted:        69494        4049   |  min.dt=   0.125000     sec   last dt=   0.191908     sec     emass=    86.9009      kg
   repeated:          732           2   |  max.dt=   0.250000     sec   crnt.dt=   0.487488     sec     tmass=    912773.      kg
 successful:        68762        4047   |  avg.dt=   0.247097     sec  merr.est=   2.395330E-06         em/tm=   9.520533E-05
  requested:        68764        4047   |  req.dt=   0.250000     sec       cpu=    3260.92     sec      time=    7000.00      sec

  attempted:       113784       44290   |  min.dt=   5.171839E-03 sec   last dt=   6.922516E-02 sec     emass=    92.1338      kg
   repeated:          739           7   |  max.dt=   0.100000     sec   crnt.dt=   0.224675     sec     tmass=    872500.      kg
 successful:       113045       44283   |  avg.dt=   4.064765E-02 sec  merr.est=   3.580817E-07         em/tm=   1.055975E-04
  requested:       113047       44283   |  req.dt=   0.100000     sec       cpu=    6474.08     sec      time=    8800.00      sec

  attempted:       124566       10782   |  min.dt=   7.079435E-03 sec   last dt=   0.200000     sec     emass=    90.2556      kg
   repeated:          740           1   |  max.dt=   0.200000     sec   crnt.dt=   0.240902     sec     tmass=    864452.      kg
 successful:       123826       10781   |  avg.dt=   0.179975     sec  merr.est=   2.737707E-05         em/tm=   1.044079E-04
  requested:       123827       10780   |  req.dt=   0.200000     sec       cpu=    7295.71     sec      time=    10740.5      sec
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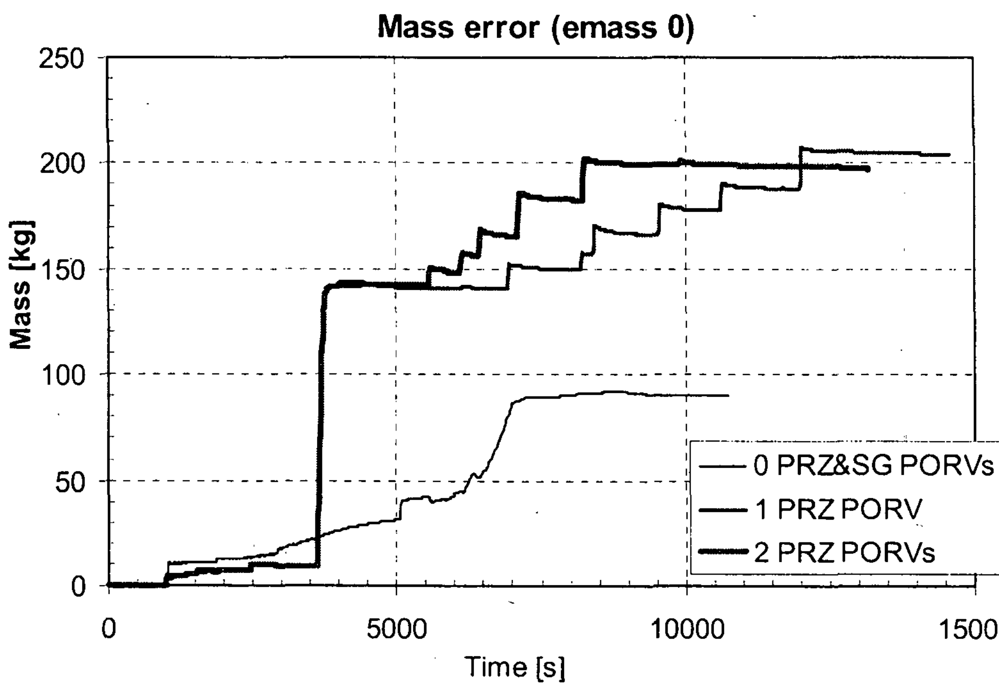

Figure 17: Mass error for 3 base analyses cases– RELAP5/MOD3.3

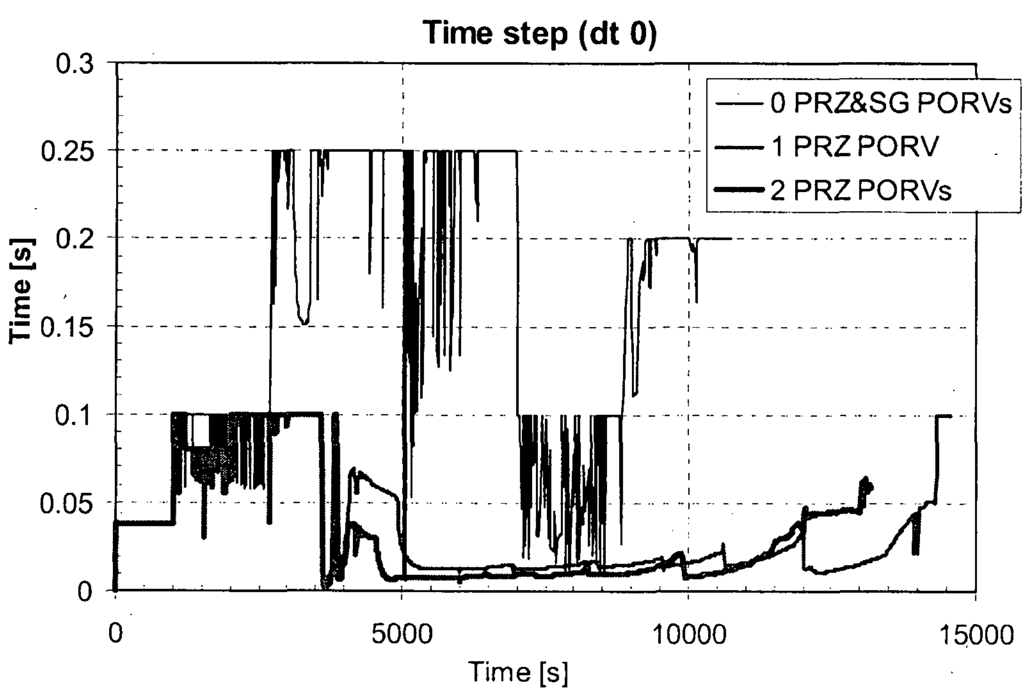

Figure 18: Time step for 3 base analyses cases– RELAP5/MOD3.3

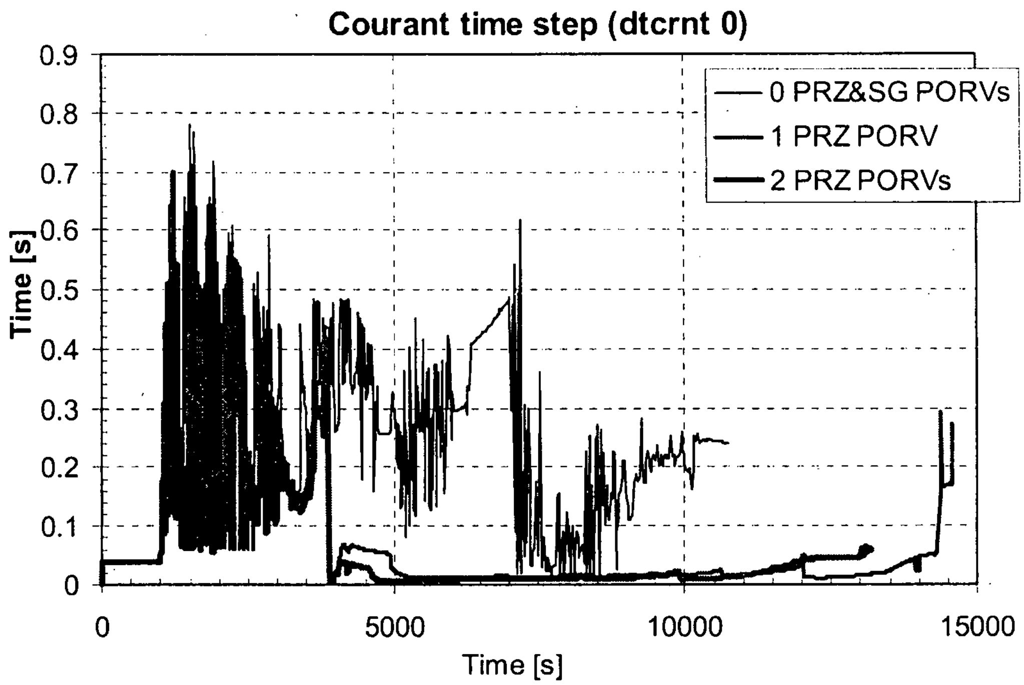

Figure 19: Courant Δt for 3 base analyses cases– RELAP5/MOD3.3

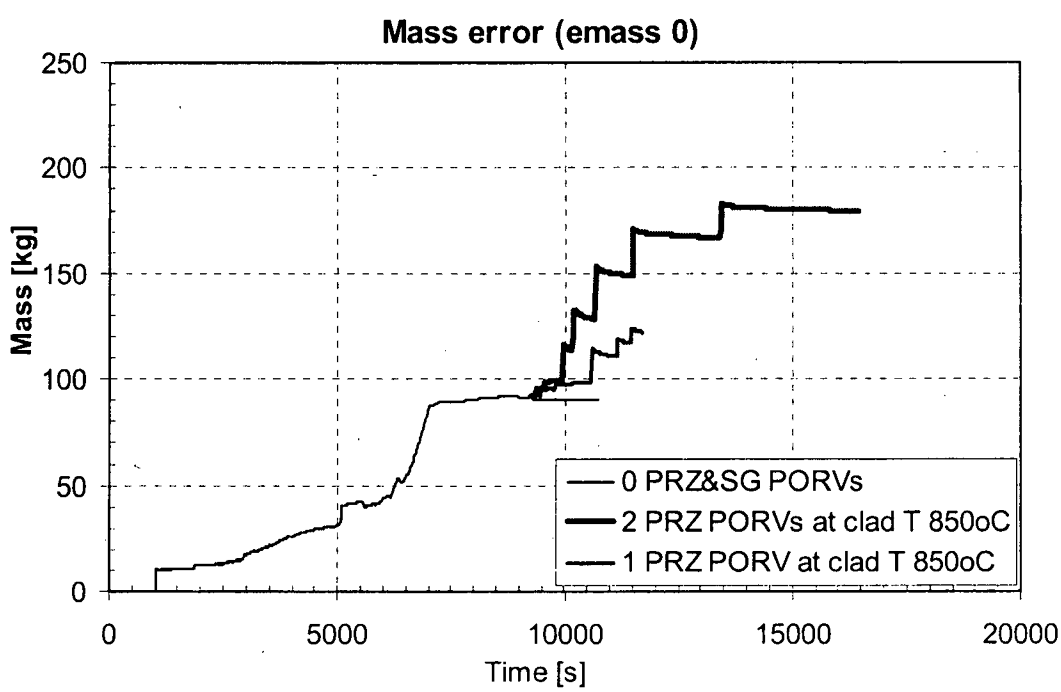

Figure 20: Mass error for 3 variation analyses cases– RELAP5/MOD3.3

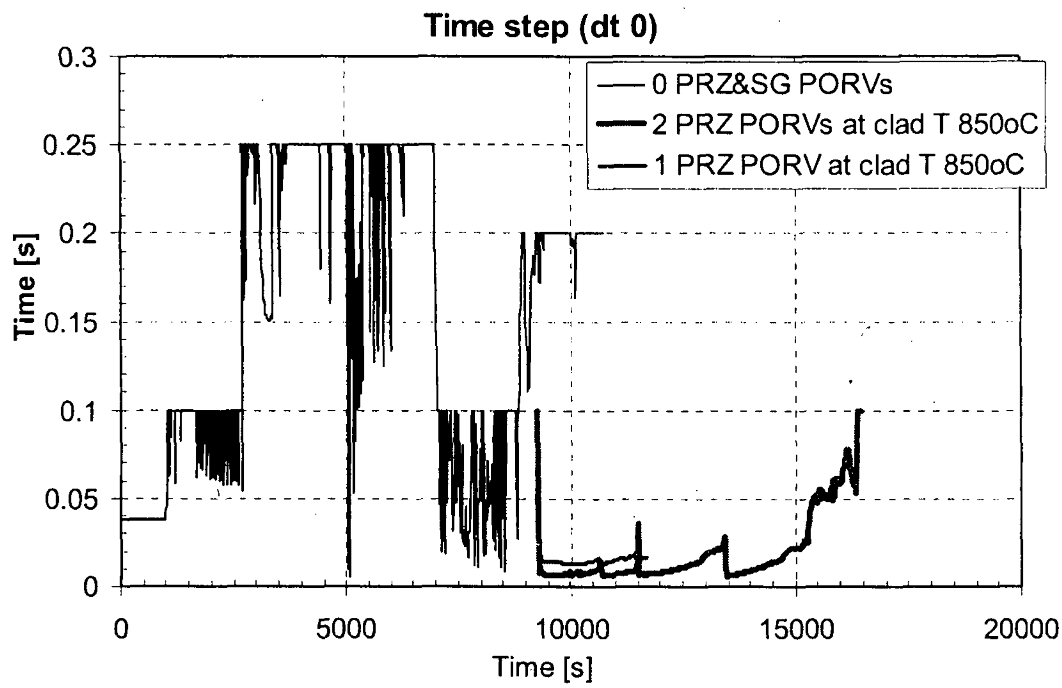

Figure 21: Time step for 3 variation analyses cases– RELAP5/MOD3.3

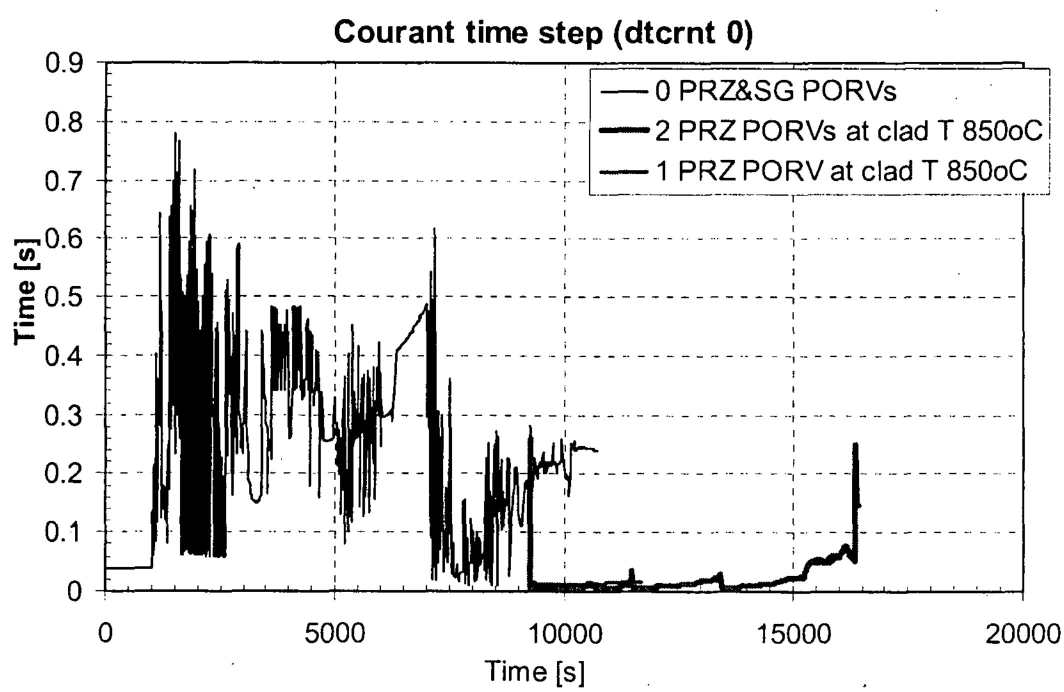

Figure 22: Courant Δt for 3 variation analyses cases– RELAP5/MOD3.3

7. CONCLUSIONS

Loss of all AC power is one of the most challenging plant transients. If the power is not restored within certain time, and if the capability to feed the steam generators is also lost this accident will progress to a severe accident. The aim of the presented analysis was to verify if the emergency operating or SAMG procedures should be changed and if the design change on pressurizer PORV should be implemented to be able to cope with this kind of accidents better.

The analyses were performed with three different state of the art codes used at NPP Krško and IJS: RELAP5/MOD3.3, ANTHEM and MAAP4. It can be concluded that no change to the existing emergency operating procedures can be recommended. However it can be concluded that if the operators would be able to open both pressurizer relief valves after the core heatup starts, this would have positive effect on further progression of the severe accident. As it can be concluded from the presented analyses by performing this action within SAMG procedures, primary pressure will be at the time of the primary system vessel failure significantly lower than in the case that there will be no operator actions for primary system depressurization. If in such situation operators would be able to open only one pressurizer PORV would this be beneficial for later accident progression.

Comparing RELAP5/MOD3.3, ANTHEM and MAAP4, it can be concluded that all the three codes predicted very similar transient course in all the analyzed scenarios.

It has to be pointed out that before making any specific conclusions related to design change and SAMG procedure change, further analyses including cost benefit analyses shall be performed, since we are dealing with very low probability events.

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10. SUPPLEMENTARY NOTES
A. Calvo, NRC Project Manager

11. ABSTRACT *(200 words or less)*

The thermo-hydraulic analysis of plant response on total loss of AC power is very demanding and challenging job due to a number of phenomena included. Such an analysis becomes even more complicated and interesting if we include also the assumption of total loss of secondary heat sink.

In this paper we are presenting the NPP Krško specific analysis of complete loss of AC power with subsequent total loss of secondary heat sink and influence of specific operator actions. The aim of this analysis is to verify if emergency operating (EOP) or severe accident management guidelines (SAMG) procedures should be changed and if design change on pressurizer pressure relief valves (PORV) should be implemented to be able to cope with this kind of accidents better.
The analyses were performed with three different state of the art codes used at NPP Krško and IJS: RELAP5/MOD3.3, ANTHEM and MAAP4. The last two codes are used in the NPP Krško plant specific full scope simulator, one for the simulation of the design bases transients and accidents and the second for simulation of the severe accidents.

This type of analyses has been done also for the simulator validation, performed during vendor and site acceptance testing.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*	13. AVAILABILITY STATEMENT
Thermo-hydraulic	unlimited
MAAP4	14. SECURITY CLASSIFICATION
RELAP5/MOD3.3	*(This Page)*
Krško NPP	unclassified
Steam generators (SGs)	*(This Report)*
ANTHEM	unclassified
Westinghouse 2-loop PWR plant	15. NUMBER OF PAGES
Jožef Stefan Institute	
Republic of Slovenia	16. PRICE
Nuclear steam supply system	

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