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European Nuclear Society
Rue Belliard 65
1040 Brussels, Belgium
Phone + 32 2 505 30 54
Fax +32 2 502 39 02
E-mail ens@euronuclear.org
Internet www.euronuclear.org

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State-of-the-Art of the Ignalina RBMK-1500 Safety

E. Uspuras
Lithuanian Energy Institute
Breslaujos str., 3, LT-44403 Kaunas, Lithuania
uspuras@mail.lei.lt

ABSTRACT

Ignalina NPP is the only nuclear power plant in Lithuania consisting of two units, commissioned in 1983 and 1987. Unit 1 of Ignalina NPP was shutdown for decommissioning at the end of 2004 and Unit 2 is to be operated until the end of 2009. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500.

The paper summarizing the results of deterministic and probabilistic analyses, developed within 1991 – 2007 by specialists from Lithuanian Energy Institute. The main operational safety aspects, including analyses performed according the Ignalina Safety Improvement Programs, development and installation of the Second Shutdown System, Guidelines on Severe Accidents Management and results of the implementation of (SIP, SIP-2 and SIP-3) will are discussed. Also the phenomena related to the closure of the gap between fuel channel and graphite bricks, multiple fuel channel tube rupture, containment issues as well as implication of the external events to the Ignalina NPP safety, are discussed separately.

1 INTRODUCTION. HISTORICAL CONTEXT

Preparatory works of construction of the Ignalina NPP have been started in 1974, and the first unit of Ignalina NPP was commissioned in December 31, 1983. At the same time the second unit was under construction and construction of the third unit began. The second unit it was planned to start to operate in 1986 year, but because of accident in Chernobyl works on preparation to operate 2 unit have been rescheduled. Second unit was commissioned in August 31, 1987. At that time 60 % of the third unit have already been constructed, but later construction was suspended and terminated soon. Nowadays because of political reasons the first unit of Ignalina NPP is shutdown, the second unit is planned to shutdown at the end of 2009.

Ignalina NPP with RBMK-1500 reactors belongs to the second generation of RBMK type reactors (it means, that this is most advanced version of RBMK reactor design series in comparison with others RBMK type nuclear power plants). In comparison with infamous Chernobyl NPP, Ignalina NPP reactors are by a third more powerfully and already from the beginning of operation had substantially advanced emergency protection systems (e.g., emergency core cooling and accident localization systems) [1].

After 1990 Lithuania declared its independence, Ignalina NPP with two largest in the world RBMK-1500 reactors came under authority of the Lithuania Republic, however nobody in the world did not know about the real level of these reactors safety. The first Safety

Justification of Ignalina NPP has been prepared by Russian experts of Research and Development Institute of Power Energy (RDIPE), organization - designer and developer of RBMK reactors, after Chernobyl NPP accident. In this document the analysis of all design basis accidents (except partial breaks of pipes) is presented in sufficient details. The analysis is performed using at that time existing tool – quasistationary derivative approximation method, being based on conservative assumptions and existing experimental data. From the present-day viewpoint such safety justification [2] has lacks:

- it was limited only to the systems description and the analysis of design basis accidents;
- computer codes, developed in Russia, have been used for simulations, but these codes have not been verified;
- the independent expertise of safety analysis has not been performed.

Therefore, at the beginning of the 90-s of the last century reasonably there were doubts how such safety justification of Ignalina NPP, presented in the first safety justification, corresponds to the real situation. In 1992 at G7 Munich Summit the decision on closing of Soviet-design nuclear power plants, first of all the nuclear power plants with reactors of RBMK and VVER-440/230 types was accepted. In 1994 Lithuania has signed the agreement with the European Bank for Reconstruction and Development (EBRD) Account of Nuclear Safety on which has undertaken to perform in-depth safety analysis of the Ignalina NPP and to not change fuel channels in a reactor.

Right from the start, when Lithuania assumed control of the Ignalina NPP, the plant, its design and operational data has been completely open and accessible to Western experts. A large number of international and local studies have been conducted to verify the operational characteristics of the Ignalina NPP and analyze its level of risk. Ignalina NPP is unique nuclear power plant of RBMK type about which information it was collected, checked, systematized and accessible. Collected and verified data base has allowed:

- to assess present safety level of NPP,
- to compare it level with others RBMK type NPPs safety level,
- to plan improvements of plant equipment and operating procedures increasing safety of the NPP.

2 DETERMINISTIC AND PROBABILISTIC IGNALINA NPP SAFETY ANALYSES

In this chapter the main Ignalina NPP safety analyses, performed since 1991 till these days, are discussed:

- Ignalina NPP Unit 1 safety analysis report and its review,
- Modifications of activation algorithms for reactor shutdown and emergency core cooling systems,
- Second diverse reactor shutdown system development, safety justification and implementation,
- Studies of Ignalina NPP 1 and 2 levels of Probabilistic Safety Assessment (PSA),
- External events at Ignalina NPP analysis.

2.1 Deterministic Ignalina NPP safety justification

In 1995 – 1996 it has been prepared In-depth Ignalina NPP Unit 1 Safety Analysis Report, using USA and Western Europe methodology and computer codes for providing of safety analysis [3]. It was comprehensive international study sponsored by EBRD. The purpose of this international study was to provide a comprehensive overview of plant status with special emphasis placed on its safety aspects. Specialists from the Ignalina NPP, Russia (RDIPE), Canada and Sweden contributed. During implementation of the project it has been described more than 50 systems of normal operation, safety important systems and auxiliary systems. Also analysis of these systems has been performed, considering compliance of these systems to the Lithuanian standards and rules as well to practice of safety used in the West. Analysing systems the attention has been concentrated on their consistency to criterion of single failure, as well as to auxiliary safety aspects: maintenance, inspections and impact of external factors (fire, flooding by water). This analysis of systems has defined the main lacks of systems and has developed conditions for elimination of the deficiencies. The performed review on operation and safety has allowed to identify all possible malfunctions, which can potentially cause an emergency situation.

In the safety analysis report of the Ignalina NPP Unit 1 the comprehensive accident analysis, equipment assessment has been provided as well as discussed questions concerning equipment ageing, investigated topics related to operators action and power plant control, provided conclusions about safety of Ignalina NPP (NPP safety level was assessed realistically), main lacks has been defined and measures for elimination of the deficiencies has been foreseen. It is the first western type report on safety for nuclear power plants with RBMK reactors.

One of the basic conclusions in this safety analysis report was such that in this case there was no problem, which would demand immediate shutdown of the Ignalina NPP. Detail accident analysis (accidents because of different pipelines ruptures, reactivity initiating accidents, equipment failures, transients with additional failure of reactor shutdown system, fuel channel ruptures in the reactor cavity) has shown, that accident occurring because of equipment failures does not cause such condition of the plant station which would cause violation of acceptance criteria, as well as safety system ensures a safe condition of the plant even doing the assumption, that operator does not take any action for 10 minutes from the beginning of accident to mitigate an emergency situation. Because of reactivity initiating accidents (exactly such type of initiating event became the reason of accident on the Chernobyl NPP) acceptance criteria of power plant also are not violated, even postulating single failures additionally. It has been shown, that Ignalina NPP is reliably protected against loss of the coolant accidents if ruptures of pipelines do not cause local stagnation of flow. In case of one steam line rupture the acceptance criteria will not be exceeded. But there are two steam lines located in the shaft at the Ignalina NPP, thus rupture of one steam line can cause rupture of other steam line, and in this case radiological dozes can be exceeded. Being based on these results of accident analysis the recommendations for modifications of activation algorithms for reactor shutdown and emergency core cooling systems have been prepared.

It is necessary to note, that in parallel with the Ignalina NPP Unit 1 safety analysis report in 1995–1997 it was performed independent Review of the Ignalina Nuclear Power Plant Safety Analysis Report [4]. This studio was performed by experts from USA, Great Britain, France, Germany, Italy, Russia and Lithuania. Independent Review has confirmed the main conclusions of Safety Analysis Report.

In recommendations of Ignalina NPP Unit 1 safety analysis report it has been shown, that Ignalina NPP will be reliably protected from any ruptures of pipelines and steam lines after improving of activation algorithms for reactor shutdown and emergency core cooling systems. According to these algorithms the system will automatically activate on coolant flow rate decrease in single Group Distribution Header (GDH) and sharp pressure decrease in drum-separators. These modifications have been implemented in both Ignalina NPP units. Safety justification of these modifications have been performed in Lithuanian Energy Institute (LEI). Further discussed situation, when conditions for local flow stagnation because of GDH rupture in the fuel channels connected to this affected GDH are developed [5]. The reason of such flow stagnation is the leak of the certain size, and at discharge of a part of the coolant through this leak the zero gradient of pressure is developed in fuel channels (7 – see Figure 1), i. e. pressure in a bottom of the channel is close to pressure in drum - separators (1). Coolant flow rate stagnation in fuel channels can be broken only in case of early activation of emergency core cooling system (ECCS) (see Figure 2 (a)). Thus if ECCS would operate according to design algorithm (reactor cooling water started to supply only after approximately 400 seconds from the beginning of accident), acceptance criteria for both fuel rod cladding and fuel channel walls temperatures in high power channel would be exceeded (see Figure 2 (b ir c)). After implementation of ECCS activation algorithms according coolant flow rate decrease in separate group distribution headers, water from ECCS starts to supply already after 5–10 seconds from the beginning of flow stagnation. Thus stagnation is broken and fuel channels, connected to affected GDH, are reliably cooled (see Figure 2). These modifications of activation algorithms for reactor shutdown and emergency core cooling systems are installed in power plant unit 1 in 1999, and unit 2 – 2000 m.

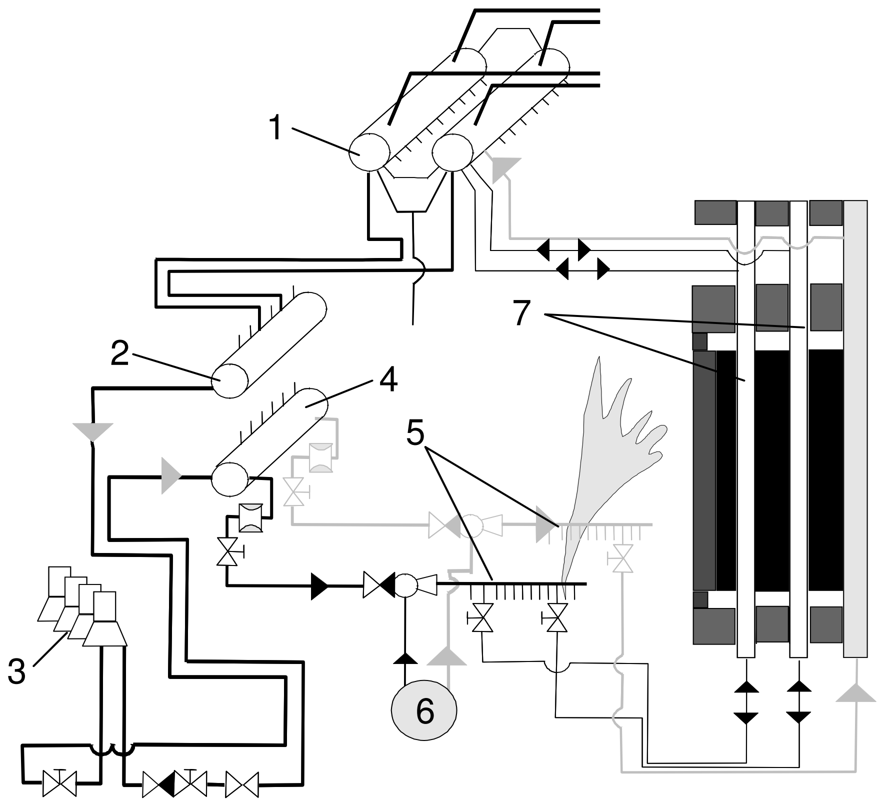


Figure 1. Ignalina NPP reactor cooling circuit (one loop) and coolant flow diagram in case of partial GDH rupture: 1 – drum-separators, 2 – suction header, 3 – main circulation pumps, 4 – pressure header, 5 – group distribution headers, 6 – water supply from emergency core cooling system, 7 – affected fuel channels

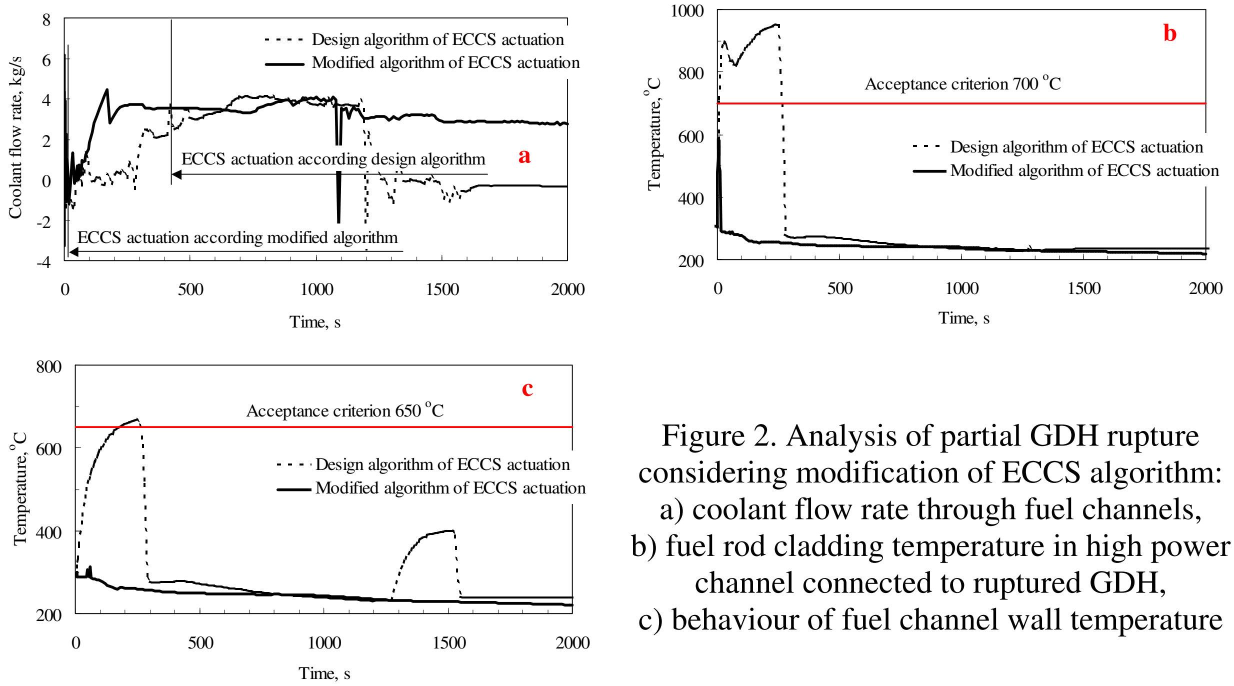
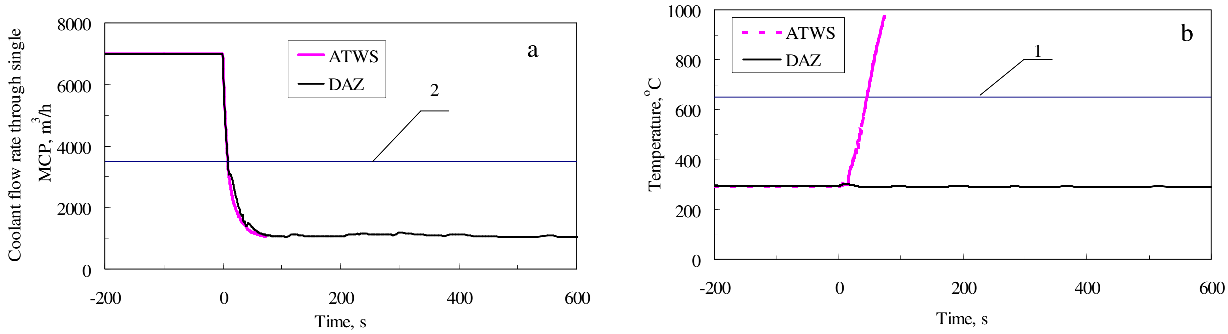


Figure 2. Analysis of partial GDH rupture considering modification of ECCS algorithm:
a) coolant flow rate through fuel channels,
b) fuel rod cladding temperature in high power channel connected to ruptured GDH,
c) behaviour of fuel channel wall temperature

In the Ignalina NPP Unit 1 safety analysis report have been investigated not only basic design accidents (discussed above), but also Anticipated Transients Without reactor Shutdown (ATWS). Investigations of such accidents are carried out at the licensing process for USA and Western Europe nuclear power plants, however for the NPPs with RBMK type reactors such analysis has been performed for the first time. Consequences of accident for RBMK-1500 reactor during which loss of preferred electrical power supply and failure of automatic reactor shutdown occurs [6] are presented in Figure 3. Due to loss of preferred electrical power supply all pumps are switched (see Figure 3 (a)) off therefore the coolant circulation through fuel channels is terminated. Because of the lost circulation fuel channels are not cooled sufficiently therefore temperature of the fuel channels walls starts to increase sharply. As it is seen from Figure 3 (b), already after 40 seconds from the beginning of the accident the peak fuel channel wall temperature in the high power channels reaches acceptance criterion 650 °C. It means that because of the further increase of temperature in fuel channels plastic deformations begin – the channels because of influence of internal pressure can be ballooned and ruptured. On the first second of accident the main electrical generators and turbines are switched off as well. Steam generated in the core is discharged through the steam discharge valves, however their capacity is not sufficient Therefore the pressure in reactor cooling circuit increases and approximately after 80 seconds from the beginning of accident reaches acceptance criterion 10.4 MPa (see Figure 3 (c)). The further increase of pressure can lead to rupture of pipelines.

Thus the analysis of anticipated transients without shutdown has shown that in some cases the consequences can be dramatic enough. Therefore the priority recommendation has been formulated: to implement the second, based on other principles of operation, diverse shutdown system. However development, designing and implementation of such system needed few years (in the Ignalina NPP unit 2 this system was installed in 2004), so the compensating means, which were used in transition period while second diverse shutdown system was developed, has been implemented. This temporary system was called according Russian abbreviation „DAZ“ („Dopolnitelnaja avarijnaja začita“ – „Additional emergency protection“). This system used the same control rods as well as design reactor shutdown system, however signals for this system control were generated independently in respect of design reactor shutdown system. In Lithuanian Energy Institute for DAZ system has been selected not only set points of activation, but also the safety justification was performed. Performed analysis has shown, that after implementation of DAZ system the reactor is shutdown in time, cooled reliably as well acceptance criteria are not violated even in case of transients when design reactor shutdown system does not functioning. In Figure 3 is shown the behaviour of the main parameters of reactor cooling circuit in case of loss of preferred electrical power supply and simultaneous failure of design reactor shutdown system. In this case two signals for activation of DAZ system (reactor shutdown) are generated: on increase of pressure in drum - separators and on decrease in the coolant flow rate through the main circulation pumps. In Unit 1 DAZ system was installed in 1999, in Unit 2 – 2000.



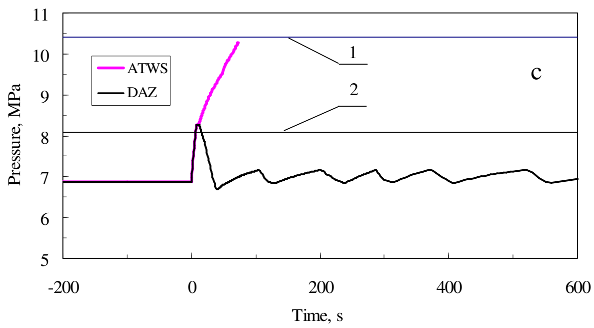


Figure 3. Analysis of loss of preferred electrical power supply and simultaneous failure of design reactor shutdown system, when DAZ system was installed: a) coolant flow rate through one main circulation pump, b) the peak fuel channel wall temperature in the high power channel, c) pressure behaviour in drum - separators, 1 – acceptance criterion, 2 – set points of DAZ system activation (reactor shutdown)

The Diverse Shutdown System (DSS) has been designed and installed in Ignalina NPP Unit 2 in 2004. In the first unit of Ignalina NPP this system has not been installed because reactor has been shutdown in 2004. Therefore, nowadays Ignalina NPP reactor emergency protection (emergency shutdown) system consists of two independent shutdown systems: first – (BSM) controls manual control rods and shortened absorber rods, which are inserted into the core from bottom. This system performs the normal reactor shutdown function and can maintain a reactor in sub-critical state. Second system (AZ) controls 24 fast acting reactor shutdown rods as well additionally 49 rods, which belong both – BSM and AZ systems. AZ system performs emergency protection function. Also the Additional Hold-down System of the reactor is installed. This system allows to prepare and inject water and neutron absorber gadolinium mixture into control rods cooling circuit. Thus, the reactor remains in sub-critical state even in the case of failure of BSM system.

DSS justification was one of the main projects increasing a level of NPP safety. Specialists from LEI together with experts from the countries of the Western Europe checked and have assessed the design documentation, carried out independent calculations, thus helping Lithuanian regulatory body (VATESI) to make the appropriating decisions concerning implementation of mentioned system at Ignalina NPP [7]. In conclusions of review it has been shown, that implementation of second, diverse reactor shutdown system protects a reactor in case of failure of design reactor shutdown system. Implementation of this system has ensured that any initiating event cannot cause accident with damage of the reactor core, as well as decreases core damage probability from $4 \cdot 10^{-4}$ up to $5 \cdot 10^{-6}$.

2.2 Ignalina NPP probabilistic safety assessment

The Ignalina NPP first level PSA "Barselina" project (1991–1996) was initiated in 1991 [8]. It was first PSA for nuclear power plants with RBMK type reactors. From the beginning this project was carried out by nuclear energy experts from Lithuanian, Russian and Swedish institutions, and since 1995 it was carried out by efforts of experts from Lithuania (Ignalina NPP, LEI) and Sweden. Main objective of deterministic analysis was to show, that nuclear power plant reliably copes with accidents, and basic purpose of PSA 1 level is to assess probability of reactor core damage, to create a basis for severe accident risk assessment and management. Performed Ignalina NPP PSA 1 level study is predicted by assumption, that the main radioactive source is reactor core. This PSA is performed for maximum permissible reactor operating power. Only internal initiating events have been analysed – transients, loss of the coolant accidents, common cause failure and internal hazards (fire, flooding, missiles). Results of the analysis have shown that after implementation of recommendations from BARSELINA [8], Safety analysis report and its independent review [3, 4], probability of Ignalina NPP core damage is about $6 \cdot 10^{-6}$. According to the international requirements this parameter for the operating nuclear power plants should not exceed 10^{-4} per year, and for new NPPs, which are in process of construction – 10^{-5}. Therefore Ignalina NPP fulfils this requirement. Analysis has shown that in Ignalina NPP risk topography dominates transients, instead of loss of the coolant accidents. The risk of core damage most of all increases transients with loss of long-term core cooling. It is the positive fact meaning that up to consequences of severe accidents there is enough time. Thus operators supervising reactor operation can undertake corrective measures, and it means that Ignalina NPP has great potential opportunities for implementation of the program on management of severe accidents. It is necessary to note, that procedures and means on severe accident management are already implemented at Ignalina NPP Unit 2 [9, 10].

According to the international requirements probability of the large reactivity release outside nuclear power plant should not exceed 10^{-7} per year for new NPPs, which are in process of construction, and for NPPs in operation – 10^{-6}. Scenarios and probabilities of the large reactivity release outside nuclear power plant are objects of investigations for PSA level 2. Ignalina NPP PSA level 2 project was performed in 1999–2001 [11] and it was the first project of such type for nuclear power plants with RBMK reactors. This project was carried out by efforts of experts from Lithuania (LEI) and Sweden. Performing PSA level 2 as initial data were used results of level 1. According in PSA level 1 investigated accident scenarios consequences and its similarity criteria on radioactive contamination the conditions of damage of the reactor have been developed and possibilities of accident management were assessed. Results of PSA level 2 have shown that barrier of the large reactivity release after core damage is 1.5. This barrier is smaller in comparison with modern nuclear power plants having function of containment, which reaches 10 and more. Being based conservative assumptions and estimation of parameters, in PSA level 2 was calculated that general estimation of large discharge frequency is $3.8 \cdot 10^{-6}$ per year. Therefore, Ignalina NPP according the probability of large reactivity release outside nuclear power plant is not the worst in comparison with the plants of the USA and the Western Europe, constructed in the same years.

Carrying out the complex analysis about influence on Ignalina NPP units safety [12] LEI the following external events have been investigated:

- aircraft crash;
- extreme wind and tornado;
- flooding and extreme showers;
- external fire.

Aircraft or others flying objects crash caused accidents in Ignalina NPP will have local character because of its big territory. According to the Lithuanian civil aviation data it has been assumed that average congestion - up to 50000 flights per one year within the 50 kilometres zone around NPP. Three zones have been defined by a radius up to 15, 50 and 85 meters around the reactor in the territory at Ignalina NPP (15 – according reactor dimensions, 85 – according reactor building size). Probability of air crash on a 85 metres zone around of the reactor center, assuming that aircraft weight is 5700 kg as well assuming that half of these flights carry out planes of the western manufacturers, and other half – Soviet is $2.06 \cdot 10^{-9}$ 1/year. Even doing more conservative assumptions (heavy planes falling frequency equalized to easy planes falling frequency) probability of air crash on a 85 metres zone around of the reactor center will be $1.64 \cdot 10^{-7}$ 1/year. The obtained heavy plane crash probabilities are less than the probabilities obtained in probability analyses for the majority of the West-European and American NPPs.

Tornado may cause huge damage and destruction. From all buildings of nuclear power plant the tornado is most dangerous for a technical water supply system building, because it is located on the open territory on a coast of lake. Tornado and hurricane winds do not create danger for buildings of reactor and technical systems. Contrariwise probability of tornado and hurricane winds is $5.3 \cdot 10^{-6}$ 1/year. Therefore it is possible to approve, that their influence on reactor safety is insignificant.

Rise of a water level in the lake Druksiai represents the greatest danger to pump station on the lake, since the service water system is the nearest NPP construction to the lake. Water level elevation of the lake Druksiai up to a level 144.1 m is not possible practically; therefore there is no danger on flooding of pump station. The platform of the other Ignalina NPP construction is located at a level of 148 - 149 m above the sea level. Rise of a water level in the lake Druksiai up to such mark is impossible and flooding does not represent the direct danger for Ignalina NPP.

Besides lake, other external flooding source is **extreme showers**. In territory of Ignalina NPP there is drainage system and all compartments which are located below a critical mark of a level are connected to this system, therefore the water leaks in case of internal flooding. Thus, extreme showers do not cause external flooding of the reactor building. For probabilistic external flooding analysis the mathematical model to assess peak water level elevations of the lake Druksiai has been developed. Probabilistic assessment of water level elevation in the lake has been performed. Maximum amount of precipitation (not lees than 279.7 mm in 12 hours) probability is $1 \cdot 10^{-6}$ 1/year. Such event will not have influence on reactor safety.

Probabilistic analysis of external fire. Ignalina NPP is situated in the region, where 30 % of territory is occupied by forests (40 % are grassland and 30 % are occupied by lakes and swamps). The edge of the closest forest is less than one kilometres from territory of Ignalina NPP. On the territory of the NPP there are only separate trees and grass. The global fire of a forest with a high wind to the NPP side can cause the smoke cover on the territory of Ignalina NPP. The smoke does not influence work of reactor mechanisms, but will complicate work of the personnel. Fire probability of forest, which is in 10-kilometer zone around Ignalina NPP and there are more than 2000 ha woods, is $2.7 \cdot 10^{-3}$ 1/year. It is a high probability, but any fire cannot affect safety of the reactor considerably.

3 IGNALINA NPP SAFETY ASSESSMENT IN CASE OF SPECIFIC RBMK PROBLEMS

Discussing safety of RBMK type nuclear power plants three vulnerabilities more often are mentioned generally:

- problem of gas gap closing between fuel channels and graphite blocks;
- problem of multiple fuel channel ruptures;
- containment issue.

Below specificity of RBMK-1500 in respect of these problems is discussed.

3.1 Problem of gas gap closing between fuel channels and graphite blocks

The fuel channels of RBMK type reactor are separated from the graphite bricks by gaps maintained by graphite rings. These rings are arranged next to one another in such a manner that one is in contact with the channel, and the other with the graphite stack block. (see Figure 4). As a result of exposure to neutron radiation and temperature the diameters of graphite columns gaps decreases, and fuel channel tube – expands, thus the gap between them decreases.

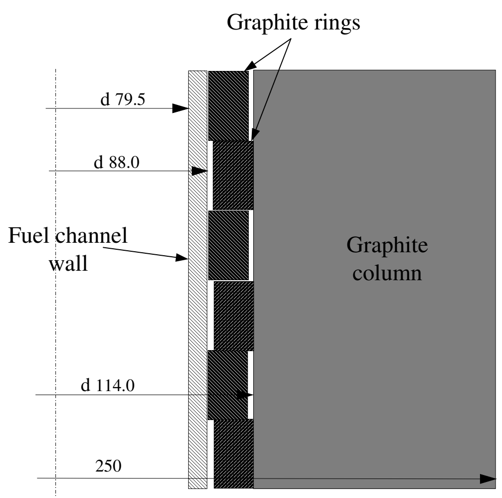


Figure 4. Fuel channel and graphite column interaction. All measurements in millimetres

The availability of the gap between graphite bricks and fuel channels is the main condition limiting the operation of RBMK type reactors. These graphite - fuel channel tubes gaps allow:

- unimpeded (axial and radial) thermal expansion and contraction of the fuel channels;
- predictable non-contacting heat transfer from graphite bricks (temperature higher than 500 °C) to fuel channels (temperature 300 - 320 °C) across the gaps;
- leakage of helium-nitrogen mixture, which provides heat transfer from graphite to coolant and protects graphite against oxidation. Furthermore helium-nitrogen mixture is part of fuel channel integrity monitoring system.

The control of gap between fuel channels and graphite blocks at Ignalina NPP Unit 1 & 2 is carried out from the beginning of its operation and now the largest database and experience of assessment of gap among all RBMK type reactors is saved. After gap closure not only some functions of the control are lost, but also worsens characteristics of the reactor. Increase probabilities of damage of the channel and deformations of graphite, withdrawing of the channel from a reactor if necessary becomes complicated, the temperature of graphite and the fuel channel changes. In Ignalina NPP Unit 1 reactor the average gap between fuel channels and graphite up to final shutdown of the reactor from an initial level (3 – 2.7 mm) has decreased three - four times. This decreasing in Unit 2 is insignificant. Estimation of such small gap is very sensitive to errors of measurements, uncertainties of used models and to strategy of selection of fuel channels for measurements.

As it is known, after signing the agreement with the EBRD Account of Nuclear Safety in 1994, Lithuania has undertaken to not change fuel channels and not operate Ignalina NPP reactor after closing even one gas gap between graphite stack and fuel channels. In Ignalina NPP in-depth safety report [3], which has been prepared by the international experts in 1996, it was predicted, that at Ignalina NPP unit 1 it happens not later than in the beginning of 1999.

In Lithuanian Energy Institute complex investigations on the problem of gap closure between fuel channels and graphite blocks at Ignalina NPP have been carried out. Assessment of the gap between graphite stack and fuel channels has very big importance because results of this problem are very important making of the decision on duration of Ignalina nuclear power plant operation. At development of a technique on assessment of gap and strategy of measurements the thermal-hydraulic, structural and probabilistic calculations have been performed. The detailed analysis [13] has shown, that in Ignalina NPP In-depth Safety Analysis Report [3] the assessment of the gap between fuel channels and graphite blocks at Ignalina NPP Unit 1 reactor has been performed using simplified deterministic calculations. Therefore obtained results were too pessimistic and conservative, predicting closure of the gap in set of channels in 1998 – 2000.

The specialists from LEI developed the integrated technique on assessment and control of risk of gas gap reduction. This allowed to develop strategy of measurement of holes diameters in graphite columns and replacement of fuel channels. This strategy has ensured existing of gap in Unit 1 reactor up to its final shutdown and by that has allowed considerably to prolong time of Ignalina NPP Unit 1 operation (until the end of 2004).

Change of a gas gap in the second unit of a reactor very much differs from the first Unit because in a reactor of Unit 2 it is used zirconium tubes of fuel channels having different hardened surfaces and the rate of their ballooning is two times slower in comparison with tubes in reactor of the Unit 1. Tendencies of change of graphite stack diameters in the second Unit are very similar to the first Unit.

3.2 Problem of multiple fuel channel ruptures

In case of fuel channel rupture a two-phase flow is discharged to gaps between graphite stack. Part of graphite blocks can be damaged cracked by coolant jet impingement, graphite columns can be displaced and coolant passes into the reactor cavity. Because graphite stack is hotter than the coolant, the pressure in tight reactor cavity increases. The leak tight Reactor Cavity (RC) performs the function of containment in the region immediately surrounding the nuclear fuel and graphite. The RC is formed by a cylindrical metal structure together with bottom and top metal plates. The reactor cavity confines the steam release in case of rupture of fuel channels. The steam-water-gas mixture from the reactor cavity is directed via Reactor Cavity Venting System (RCVS) pipelines to two steam distribution devices of the 5th (upper) condensing tray in the Accident Localisation System (Figure 5). Two pipelines d = 400 mm that come from a branch pipe d = 600 mm located above the top plate of RC are interconnected to a pipe d = 600 mm and which connects to one steam distribution device [1]. In the same way the other two pipelines d = 400 mm from the top plate of RC are connected to the second steam distribution device. On their way these pipelines have branches, which are interconnected in a leak-tight corridor and end up with three Membrane Safety Devices

(MSD). The blowdown pipes from the bottom of RC pass directly to the leak-tight corridor and also end up with three MSD.

In the case of multiple fuel channel tube ruptures, if the RCVS does not assure relief of steam-water-gas mixture from RC, the pressure increase in the RC will lift top plate of the RC. Those, structural integrity of the RC and the rest fuel channels would be lost as well. Such event would cause very severe consequences similar to Chernobyl accident. Therefore it is important to maintain RC integrity, which is assured if pressure in the RC is below permissible pressure (314 kPa, abs) i.e. the pressure of upper plate of biological reactor shielding weight [14].

Rupture of one fuel channel is design basis accidents for RBMK-1500 reactors. Probability of such rupture – 10^{-2} 1/year. According design the reactor cavity venting system assured the integrity of RC in the case of up to 3 fuel channels ruptures. This system has been modernized in 1996 as shown in Figure 5.

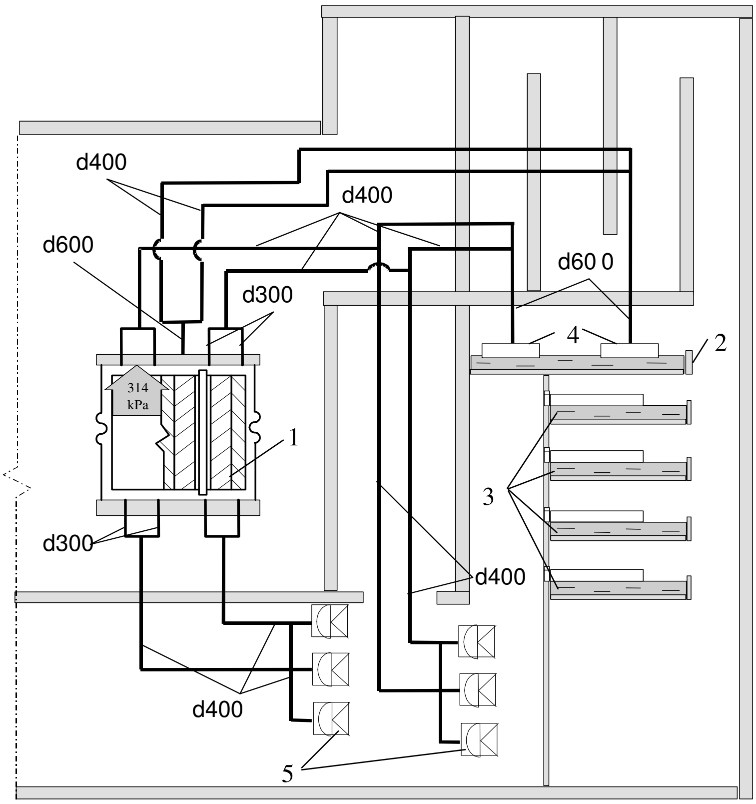


Figure 5. Simplified schematic of the reactor cavity venting system: 1 - reactor, 2- the fifth ALS suppression pool, 3 - suppression pools 1-4, 4 - steam distribution devices, 5 - membrane safety devices (350 mm diameter)

Moscow Research and Design Institute for Power Engineering (RDIPE), designer and developer of RBMK reactors, specialists in 1996 have analysed pressure behaviour in the Reactor Cavity in case of multiple fuel channel rupture [14]. Results of these calculations have shown that acceptance criterion – maximum permissible load (310 kPa) to upper reactor cavity plate will be exceeded in case of 9 fuel channels rupture (according RDIPE calculations). In RDIPE calculations the coolant discharge through the rupture conservatively was assumed equal 32 kg/s through one fuel channel. This flow rate has been selected as constant versus time. Because of such conservative assumptions amount of discharged coolant into reactor cavity is largest and number of channels, when permissible pressure in reactor cavity is not exceeded, will be minimal.

Such analysis is conservative with impact of uncertainties. The best estimate analysis of Ignalina NPP response to multiple fuel channels tubes rupture was performed at the Lithuanian Energy Institute. Sensitivity and uncertainty analysis was performed as well [15]. At performance of the analysis it has been considered, that results of calculations can be influenced by uncertainties such as the plant initial conditions, assumed at the modelling, as well as assumptions and correlations of CONTAIN code. Summarizing the results of the uncertainty and sensitivity analysis, it was concluded, that the capacity of RCVS comprises from 11 up to 19 ruptured fuel channels, i.e. 15±4 channels (Figure 7).

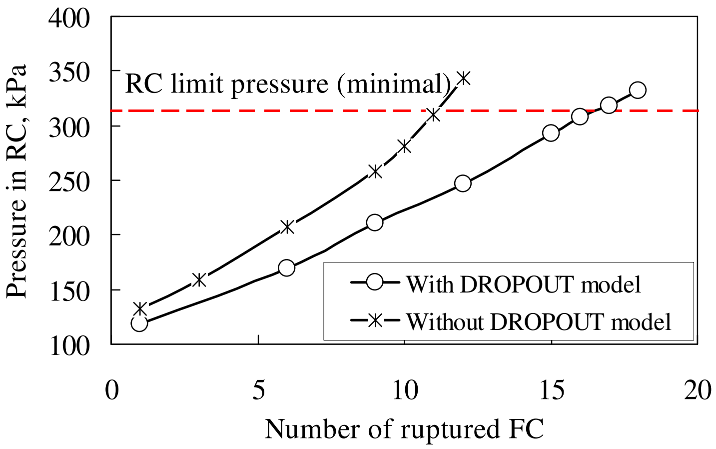


Figure 6. Pressure in the reactor cavity as a function of a number of ruptured fuel channels (FCs)

It is necessary to note, that the analysis was performed for the case, with RCS filled by coolant (the water levels in drum separators are nominal). Thus, after the fuel channels rupture the steam-water mixture is discharged into the gaps of graphite stack. If the “dropout” model is used in CONTAIN 1.1 code, it is assumed, that all the water released from the ruptured fuel channels in liquid fraction leaves from RC to the water drain. If the “dropout” model is not used in CONTAIN 1.1 code, it is assumed, that all not evaporated water remains in a dispersed condition, and it may be transferred into RC and through the pipelines into ALS. The last assumption leads to higher calculated pressure in the RC (see Figure 6).

It is necessary to note, that during operation of RBMK reactors there were only three cases of ruptures of separate fuel channels:

- at Leningrad NPP Unit 1 in 1975,
- at Chernobyl NPP Unit 1 in 1982,
- at Leningrad NPP Unit 3 in 1992.

In any of these cases adjacent channels have not been damaged. Thus, in reality there was no so-called “cascade rupture of fuel channels” when rupture of one channel causes ruptures of other channels. Experiments made on the large scale TKR-Test facility at Electrogorsk Research & Engineering center for NPP safety [16] have shown also, that cascade rupture of fuel channels are impossible.

3.3 RBMK reactor containment issue

In case of accident in nuclear power plant (rupture of reactor cooling circuit pipelines), the coolant with radioactive materials will spread into reactor and compartment enclosed reactor cooling circuit. In many (but not in all) reactors of the USA and the Western Europe function of containment carries out visible from afar, photogenic, semicircle form protection enclosure. Usually non-existence of containment is treated as deficiency of RBMK reactors. However such containment as for vessel type reactors it is technically impossible to implement for RBMK reactors. In the Ignalina NPP the function of containing accidentally released radioactive material is accomplished by an extensive system of interconnected steel lined, re-enforced concrete compartments called the Accident Localization System (ALS). The ALS uses the "pressure suppression" principle employed by G.E. designed boiling water reactors. The ALS encloses the large Ignalina NPP reactor core, the coolant pumps and all of the piping providing coolant to the core. It is not necessary to enclose the pipes above the reactor core, which carry the exiting two-phase (steam-water) mixture to the drum - separators, because if one of them is breached, coolant flow to the fuel channels (which is provided by pipes entering the core from bellow) will not be interrupted. Significant amounts of radioactive material can escape only if fuel rods are over-heated. Breaches in the exiting pipes will not reduce coolant flow, therefore the fuel rods will not overheat.

The effectiveness of the ALS has been verified by extensive international analysis and experimental programs. They all show that even if events leading to release of radioactive materials are postulated, these materials will be contained by the ALS, thus the ALS performs the function of containment [17]. The minimal amounts (due primarily to non-condensable noble gases) which would eventually reach the environment, would not exceed the amounts that would be released by Western built reactors provided with the more familiar, prominently visible "dome containments".

4 CONCLUSIONS

Requirements on nuclear power plants safety depends on the accumulated experience, a level of a technical society evolution, which always raises, and from position of the state. About safety level of Ignalina NPP it was worried after Chernobyl accident in 1986. The first modernizations of reactors have been implemented at that time. RDIPE, designer and developer of RBMK reactors, experts have prepared the first safety justification for operating power plant in 1989. When Lithuania assumed control of the Ignalina NPP in 1991, a large number of studies on safety level have been conducted. It is necessary to note Safety Analysis Reports for Ignalina NPP Units 1 & 2, Safety Justifications of Reactor Cooling System and Accident Localization System. The Ignalina nuclear power plant is distinguished from all RBMK type reactors for the matter of that many international studios to investigate design parameters as well level of its risk have been performed. Ignalina NPP, its design and operational data have been completely open and accessible to Western experts. At first the effective initial help in questions of nuclear safety has provided by Sweden, and after by other countries (Germany, United Kingdom, USA etc.), capable to perform expertises of the safety analysis.

The detailed analysis of accidents has shown, that design basis accidents do not cause such condition of the plant, which postulates violation of acceptance. As well safety systems of the plant ensures a safe condition of the plant even doing the assumption, that operator does not take any action for 30 minutes from the beginning of accident to mitigate an emergency situation.

The performed Probabilistic Safety Analysis of level 1 and 2 has allowed to compare safety level of Ignalina NPP with the reached level on other nuclear power plants and to plan, how to improve NPP safety systems and operational procedures. Investigations have shown that Ignalina NPP according the probability of large radioactivity release outside nuclear power plant is not the worst in comparison with the plants of the USA and the Western Europe, constructed in the same years.

On the basis of the performed investigations the recommendations on safety improvement were developed by efforts of local and foreign experts. These recommendations were brought into Ignalina NPP Safety Improvement Programs (SIP-1, SIP-2 ir SIP-3) which implementation strictly was checked by Lithuanian regulatory body VATESI. These means have allowed to improve safety level of the Ignalina NPP constantly. These works do not stop even on forthcoming final shutdown of the plant. In outcome of last significant project the Severe Accident Management Guide is developed. Now this guide is under implementation at Ignalina NPP. Severe Accident Management Guide will supplement Symptom-Oriented Emergency Operating Procedures and will provide safe elimination of accident consequences in all range of accidents.

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TOOLS TO SUPPORT IMPORTANT TECHNICAL DECISIONS DURING ACCIDENT CONDITIONS

Authors:
J. Tenschert,
Nordostschweizerische Kraftwerke (NOK) AG,
Kernkraftwerk Beznau,
5312 Döttingen, Switzerland
jens.tenschert@nok.ch

C. Bergiers,
Westinghouse Electric Belgium SA,
1400 Nivelles, Belgium
bergiec@notes.westinghouse.com

ABSTRACT

To handle design basis and beyond design basis accidents with an intact reactor core, Nuclear Power Plants are using Emergency Operating Procedures (EOP) that they may have developed based on the generic Westinghouse Emergency Response Guidelines.

Even though the EOPs are very directive, some questions are left to external support. In many Western NPP, a so-called Technical Support Center (TSC) is responsible for the engineering support. The Pressurized Water Reactor Owner Group (PWROG, previously Westinghouse Owner Group, WOG) has developed a generic TSC manual to support the TSC in their decision making processes.

Due to the specific and particular design of the Beznau NPP (KKB) Safety Systems, the development of a plant-specific TSC manual required a lot of additional issues compared to the generic PWROG material. The majority of considered issues are relevant for beyond design basis accidents and external events.

The new developed plant-specific TSC manual covers all technical issues the shift supervisor could ask the SED to address during a technical emergency. The background material for the SED's decisions regarding issues of the EOPs is documented in fifty separate evaluations; one for each issue.

The plant specific TSC manual is a helpful tool for the SED of the KKB to better evaluate issues and potential concerns arising while executing the EOPs. Moreover, the manual can be used for the training of Pikett engineers and other technical specialists involved in technical emergencies.

This article gives an overview of the structure of the TSC manual and the necessary steps for the development of a plant-specific TSC manual on basis of the generic manual.

1 INTRODUCTION

NOK operates the Beznau NPP (KKB) with two 380 MW electrical power PWR units. First operation was in 1969 for unit 1 and in 1971 for unit 2. The nuclear island (NSSS) was designed and constructed by Westinghouse, the secondary systems by BBC. During 38 years of operation a lot of upgrades were carried out.

Following the TMI accident, the Westinghouse Owner Group (now PWROG) developed the Emergency Response Guidelines (ERGs) [1] as guidance for the control room operators to handle Emergency Conditions.

Within the ERGs, operators are sometimes instructed to consult the Technical Support Centre (TSC), a group of persons mobilized at the time an Emergency Situation is declared on the plant, and that remains outside of the Control Room to technically support the operators during their recovery actions. In the Westinghouse ERGs, there are seventeen issues for which the advise of the TSC is being asked. For some time, the TSC members had no specific tool to support their decision. To fill this gap, a TSC Manual [2] was developed to enhance the guidance available to the TSC for performing these 17 evaluations and making recommendations during implementation of the EOPs. This manual can be adapted into a plant-specific document in whatever format suits the utility the best (for example, as a job aid or as a training aid).

Like many other plants, NOK developed its plant-specific Emergency Operating Procedures (EOPs) based on the Westinghouse ERGs. The first KKB EOP package in German language was implemented in 1985 [3]. Because of the complex safety architecture of the KKB units, the actual EOPs [4] consists of two sets, one usable from the Main control Room and a second one usable for the Notstand Control Room when the plant safety is ensured by the Notstand system (which is described later in this document). However, as in the ERGs, there are several places in the KKB EOPs where the TSC guidance is requested.

In 2004, NOK decided to develop a plant-specific TSC manual in cooperation with Westinghouse.

2 PARTICULAR SAFETY FEATURES OF THE KKB PLANT

The procedures and TSC manual developed by the WOG are based on two reference PWR designs. Both units of the Beznau plant differ significantly from the two reference plants, especially in terms of safeguards systems which include lots of redundancies. Of course, these particular features had to be taken in account while developing the plant specific EOPs and the TSC Manual.

2.1 Notstand system

The Notstand system is a bunkered building designed to ensure plant safety following external events. The main safety features include:

- One feedwater train which is able to cool down the plant. The Notstand feedwater tank is fed by the Notstand well water system. During low pressure conditions in the SGs, the Notstand well water can also feed the steam generators.
- One Emergency Seal Injection train to ensure seal # 1 cooling for the Main Coolant Pumps
- One high pressure safety injection train with throttling capability.
- One low pressure safety injection train with its own recirculation capability (dedicated jet pump).
- An Emergency Cold Shutdown System, which fulfills the same function as the residual heat removal system.
- A containment spray function ensured by the diversion of part of the Notstand safety injection to the containment spray nozzles.

- An independent power supply by Diesel generator. The 6 kV Notstand bus system may also be fed by the Notstand Diesel generator of the other unit or by the electrical grid.
- An independent well water system for component cooling and residual heat removal. The well water system may also be fed by the Notstand well water system of the other unit.
- An Emergency control room (ECR) that allows taking control of all Notstand systems and some systems out of the Notstand building, as for example the main steam relief valves. The controls from the ECR have priority over those from the Main Control Room once the ECR has been activated.

2.2 Emergency Feedwater System

An additional bunkered feedwater system (independent from the feedwater system available with the Notstand system) is installed on each unit. Each emergency feedwater system can feed the other unit.

2.3 Specific Containment features

The containment consists of a steel liner and an armed concrete structure. The free volume between the liner and the concrete wall is referred to as the annulus space. To ensure the containment isolation function, a back-up containment seal water system is installed.

In addition, the containment is equipped with passive autocatalytic hydrogen recombiners and a filtered containment venting system which are designed for Severe Accident conditions.

2.4 Emergency power supply

In addition to both external grids (50 kV and 220 kV), one emergency power bus of each unit is continuously supplied by the Beznau hydro power station. In case of loss of external power, one additional emergency power bus of each unit can be supplied by another hydro power supply.

3 PROCEDURES AND DOCUMENTS FOR EMERGENCY RESPONSE IN CASE OF DESIGN BASIS ACCIDENTS AND BEYOND DESIGN BASIS ACCIDENTS

The KKB has established a set of procedures in order to give the control room operators the best information for normal operation, anticipated operational transients, design basis accidents and beyond design basis accidents. **Figure 1** gives an overview on procedure type applicability with respect to the different operation modes from normal operation to severe accidents.

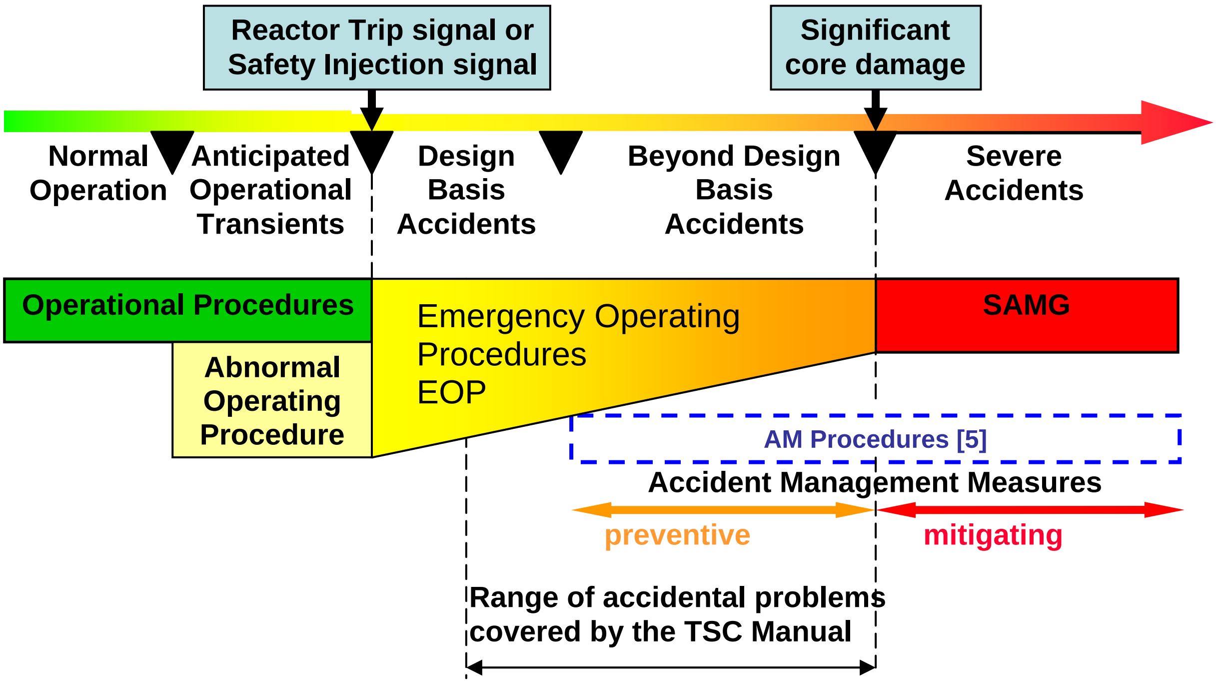

Figure 1: Overall system of procedures for the Beznau plant operation

After a Reactor Trip or a Safety Injection (SI) signal occurs, control room operators have to enter the EOPs, starting with E-0 ("Reactor Trip or Safety injection") procedure. The main goals of the E-0 procedure are to verify the status of automatic actions resulting from the reactor protection and safeguards system actuation and to identify the appropriate optimal recovery procedure to handle the on-going event. If the Reactor has successfully shut down and at least one emergency electrical power bus is available, the control room operators move to the appropriate procedure, which is normally an Optimal Recovery Procedure (ORP) to recover from one of the following events:

- "No break" type transients
- Loss of Coolant Accidents (LOCA)
- Secondary Line Breaks (SLB)
- Steam Generator Tube Rupture (SGTR).

Within each optimal recovery procedure, a continuous diagnostic takes place (looking at symptoms) making it possible to address time-evolving events, including combination of events (beyond design basis).

In parallel to the implementation of the ORPs, the control room operators have to monitor the critical safety functions (CSF) with the following priority:

- Subcriticality
- Core Cooling

- Heat sink
- Integrity (of the reactor vessel)
- Containment
- Primary Inventory.

The critical safety functions are monitored through their corresponding status trees. Based on the decision criteria (plant symptoms), at any moment, the possible status of each one of the CSF is one of the following:

Table 2: CSF status and requested operator actions

CSF Status	**Example: Core Cooling CSF**	**Color Coding**	**Operator Action**
Satisfied	Subcooling is guaranteed		Remain in procedure in use
Not Satisfied	Subcooling is lost		Optional transition to the specified Function Restoration (FR) procedure
Severe Challenge	650 ºC > Core exit temperature > 376 ºC		Prompt transition to the specified FR procedure
Extreme Challenge	Core exit temperature > 650 ºC		Immediate transition to the specified FR procedure

4 KKB EMERGENCY ORGANISATION

4.1 Normal Operation and anticipated operational transients

During normal operation and anticipated operational transients, the Beznau plant is controlled by the operating crew directed by the shift supervisor in the MCR. In case of operational problems, the shift supervisor has to inform the head of operations during normal working hours or the Pikett engineer outside working hours. Depending on the situation, he has to ask for technical advice or a decision.

The Pikett engineer (Pikett is a Swiss German term for an individual or organisation on duty) remains on the plant area during his duty time such that he can reach the MCR it in a reasonable time. In case of operational disturbances he serves as an advisor for the shift crew and the involved maintenance staff. The licensing process of a Pikett engineer requires an engineering degree, qualification as a shift supervisor and additional specific qualification.

4.2 Design basis accidents and beyond design basis accidents

After a reactor trip has occurred, the shift supervisor must call for the Pikett engineer into the control room The Pikett engineer must evaluate whether a technical emergency has occurred or not based on the following decision criteria:

- Safety Injection or Containment Spray actuation (automatic or manual)
- Shut down and/or the following cooldown of the plant might be difficult or complicated
- Station Blackout
- Non explicable reactivity transient
- Long term spent fuel pit cooling is jeopardized.

In contrast to plants in other countries, the intervention criteria for a technical emergency in KKB are determined in a very early stage of an accident.

If one of the above conditions is satisfied, the Pikett engineer declares the technical emergency and calls for the activation of the plant emergency organization. The structure of this emergency organization is shown on **Figure 2**.

As long as the Site Emergency Director (SED) has not overtaken his responsibility by formal declaration, the Pikett engineer acts as the SED. In this function, the Pikett engineer is responsible for control of the emergency and directs first actions on the site.

As soon at least 3 members of the Emergency Staff including one individual for the position of SED are on the plant, the leadership for the emergency may be handed over. Outside of working hours this step is expected one hour after the begin of the event, during working hours within 30 minutes.

The Emergency Staff consists of the plant department managers and some technical experts. The designated persons for the SED position are the plant manager and his deputy. If both persons are not available, one suitable member of the Emergency Staff may function as the SED.

The SED is responsible for all important technical and organizational decisions in emergencies. The Emergency Staff support the SED in terms of preparation and realisation of the decisions. If the SED is asked for a decision on a complex issue, the member of Emergency Staff will involve additional experts from the engineering support teams. The SED decides on basis of a proposal of the Emergency Staff within the next Emergency Staff meeting.

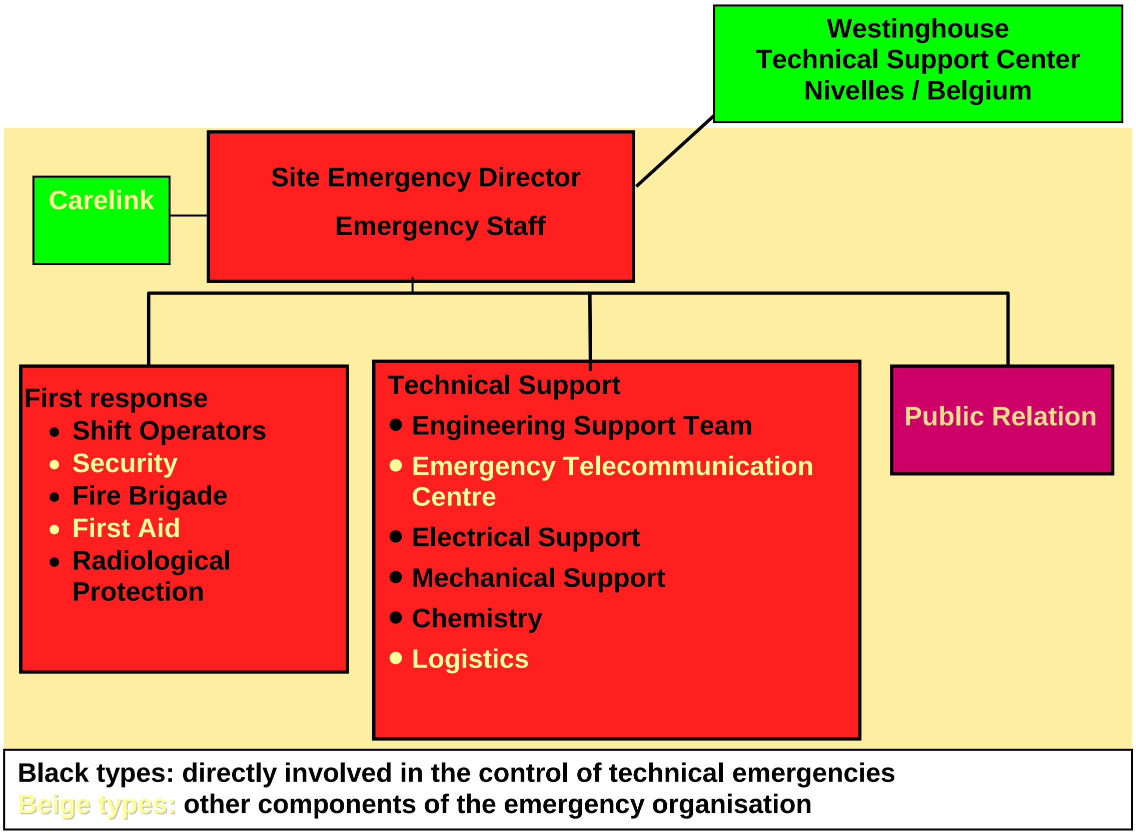

Figure 2: Structure of the emergency organisation in KKB

5 PROBLEMS TO BE DEALT WITH IN THE PLANT SPECIFIC TSC MANUAL

The crew on shift operates the plant according to the EOPs independently to a large extent. However the EOPs in the KKB include many instruction steps where the shift supervisor has to inform the SED or refer to him for a decision.

The plant specific “TSC manual” [6] is a decision making tool with the objective to cover all technical issues the shift supervisor could ask the SED to address during a technical emergency. The same technical issue can appear in several EOPs. In the TSC manual, each technical issue is the subject of a separate Evaluation.

5.1 Example

Part of Westinghouse approach to recover from SGTR and SLOCA accidents is to continue RCP operation, if possible. If RCP seal injection was interrupted due to station blackout or other reasons, restart of RCP is permitted only if the seals are in good condition. The objective of the subsequent evaluation is to find out whether the RCP shall be restarted or not. This particular technical issue appears in 13 steps of 11 different EOPs.

In the Beznau specific TSC manual, 50 different evaluations are documented. The following presents the split of the evaluations as it applies to Beznau.

5.2 Classification of the problems with respect to the different EOP packages

The generic TSC manual contains only 17 evaluations. Due to the particular safety features of the Beznau units, 33 additional evaluations have been added to its plant specific TSC manual. **Figure 3** illustrates that a large amount of evaluations are applicable for the ECR. The reason is that a lot of systems located outside of the Notstand building cannot be controlled from the ECR, whereas the EOP for ECR requires taking actions on components located outside of the Notstand building, which requires assessing their accessibility.

Problems related to operation from the MCR are in minority. Furthermore, some generic technical issues are considered not relevant for Beznau SED decision.

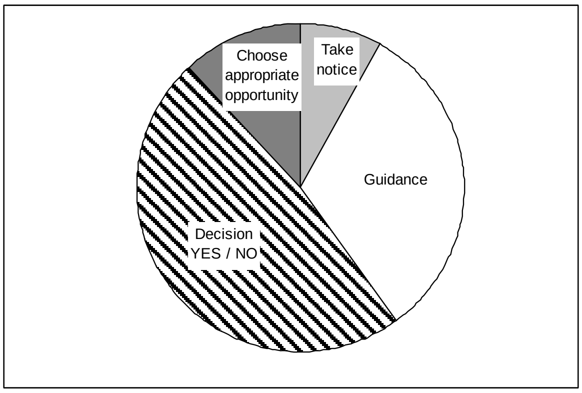

Figure 3: Distribution of evaluations for the different EOP packages

5.3 Types of required actions by the SED

The required actions of SED ranges from "take notice of information" up to complex decisions between two or more alternate solutions (see **Figure 4**). "Take notice of information" means that hard criteria are satisfied and the SED has to be informed. "Guidance" will be given if an EOP step was not successful and the control room operators need instructions to solve that problem with the help of a substitute.

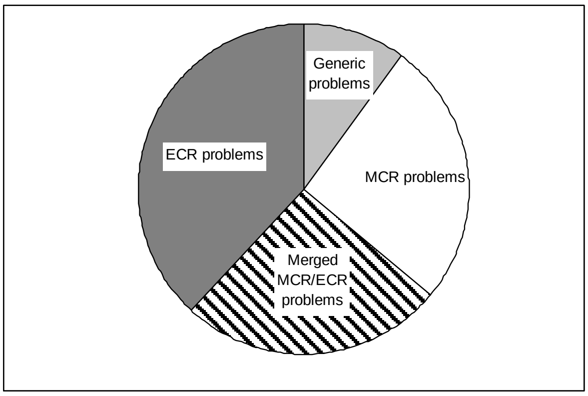

Figure 4: Range of SED actions requested by the shift supervisor

Half of the evaluations are developed to decide YES/NO, e.g. "Should the RPV be vented or not?" In some evaluations, the SED has to decide which method to apply, i.e. which cooldown method to use following a SGTR (backfill, blow down or steam dump).

5.4 Severity of considered accident situations

The majority of the evaluations in the Beznau TSC manual are developed to handle problems arising in beyond design basis situations. Even in the ORPs, some evaluations are addressing conditions outside of the expected plant behavior during design basis accidents.

Just 10 of 50 evaluations are developed to decide on measures in the range of design basis accidents, e.g. decision on the long term plant status after an accident and the already mentioned decision on cooldown method following a SGTR.

Figure 5 gives an overview of the different sources of problems handled in the evaluations. Some particular evaluations (6) handle problems arising in the whole bandwidth of ORP, ECA and FRP procedures.

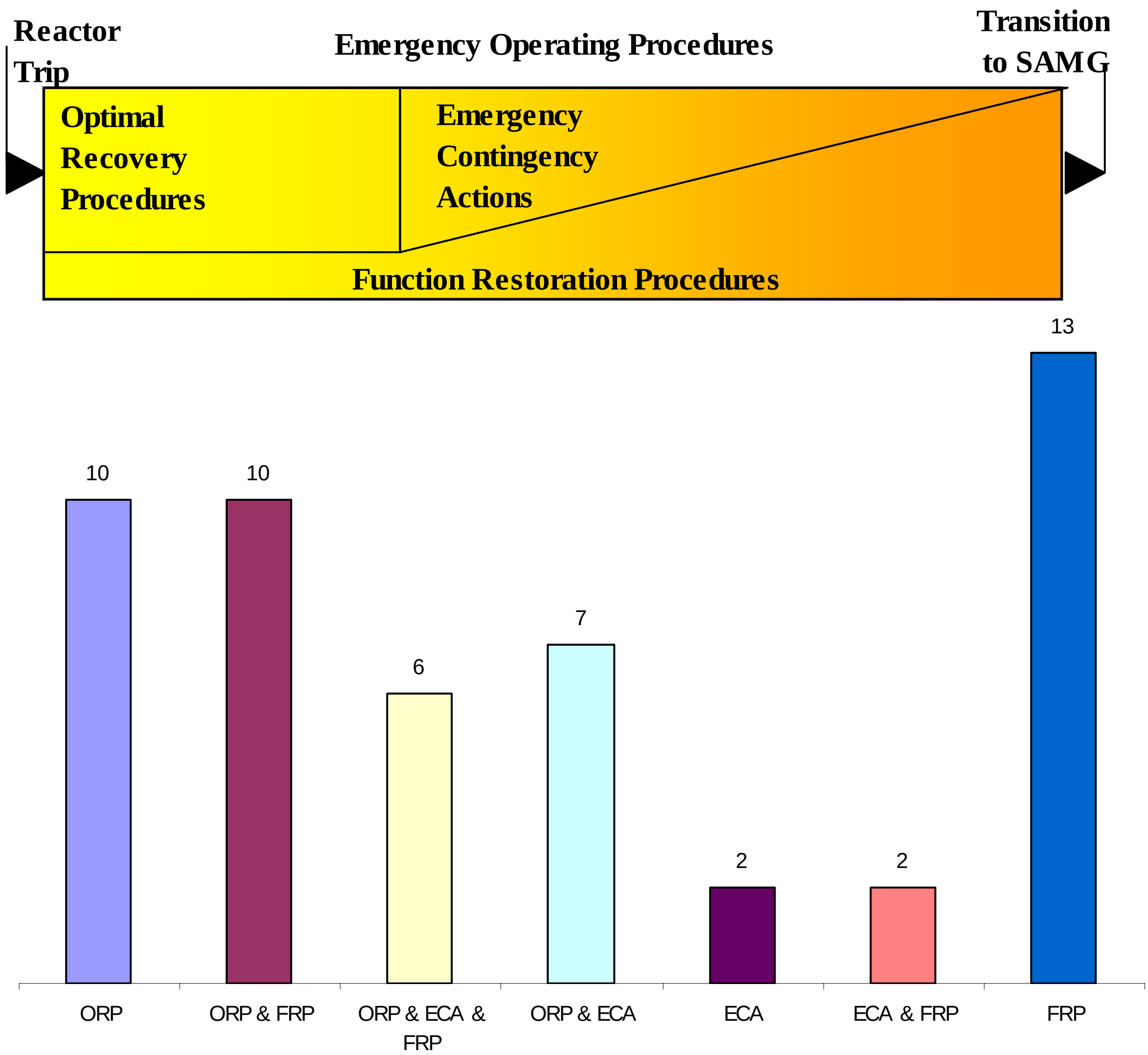

Figure 5: Distribution of accident severity covered by the evaluations of the TSC manual

6 STRUCTURE OF THE TSC MANUAL

The first part of the manual informs how to use the manual and gives an overview where in the EOP packages are the technical issues applicable.

Fifty separate evaluations are documented. The structure of each evaluation is similar to each other:

1. Concern: This is a brief statement of the underlying concerns prompting the operators to consult with the SED or to obtain a recommendation from the SED.
2. Applicable EOPs: This is a list of EOPs during which the SED might be called upon to perform this evaluation.
3. Plant specific Considerations: These are plant-specific issues and/or design characteristics (in particular compared to the Westinghouse ERGs "reference plant") pertaining to this particular evaluation.
4. Plant conditions: This is a list of the most likely plant conditions anticipated to exist at the time the SED is called upon to perform this evaluation.
5. Prevailing cautions and notes: This is a list of the EOP cautions/ notes anticipated to be in effect/ applicable at the time the SED is called upon to perform this evaluation, and that are in some way directly related to the evaluation.
6. Guidance
   a. Evaluation Objectives: Essential question to be answered or determination to be made by the NFS in this evaluation. In other words, it is the expected output of the evaluation or decision-making process.
   b. Points to Consider: This is a comprehensive survey of points the SED should consider in performing this evaluation. The documented facts inform the SED about the available options and their advantages, disadvantages and risks.
   c. Review of background documentation: This is the documentation of all information related to the evaluation. This documentation can be found in the Background Information Documents that were developed together with the EOPs. Each EOP has it own BID that includes information about analyses that were realized to develop the strategy of the procedure, some information about the physics of the accident the procedure is supposed to deal with, as well as detailed explanation of each step of the dedicated procedure.
   d. References: Any known document that brings additional information about topics covered by the evaluation, but was too big to be included directly into the Manual.

7 EVALUATION EXAMPLE: DECISION ON REFILL OF ONE DRY STEAM GENERATOR

The above explained content of an evaluation is demonstrated using an evaluation with simple structure.

Applicable EOPS: The issue is integrated in two procedures concerning countermeasures at low SG narrow level (YELLOW CSF condition) in the MCR EOP package as well as in the ECR EOP package.

Plant conditions: Although the Beznau plant is equipped with four trains of feedwater, the plant specific EOPs consider the total loss of feedwater. This may be a result of a station blackout or an external event. Consequently, the critical safety function “Heat Sink” will eventually not be satisfied. One of the strategies applied in this event is to remove residual heat with one SG and isolate the other SG. So, the following situation may occur: One SG is dry and the level of the other SG is yet within the Narrow Range. The feedwater supply has now been reestablished.

In order to reestablish the secondary heat sink completely, the SED has to decide whether to refill the dry SG or not.

The **objective** of the evaluation is to decide whether the dry SG should be fed with feedwater after re-establishing of any feedwater system.

The **points to consider** contain the necessary information for the decision process; it is summarized here below:

- At least one SG is active and has a level within the narrow range. A YELLOW PATH condition means no urgent need for dry SG refill.
- The refill process has to be delayed in order to cool down the structures of the dry SG by heat losses.
- Thermal shock phenomenon must be considered, a critical temperature for the SG shell is given
- The preferred method is to use one of both auxiliary feedwater trains.

8 WORK FLOW AND STATUS OF PROJECT

The plant specific TSC manual has been developed in co-operation between Westinghouse and NOK, stepwise:

1. Identify of all steps in the plant specific EOPs requiring a decision of the SED
2. Summarize the amount of steps and definition of the problems respective evaluations
3. Establish a draft version of the manual (Westinghouse)
4. Check the draft material by a group of NOK experts
5. Establish the final version (Westinghouse)
6. Final overall check by a NOK expert not involved before
7. Implementation of TSC manual.

A total of 15 experts in both organizations were involved in the project.

At the time the present article was written, step 7 was on-going. The objective is to implement the manual by mid 2008.

9 FUTURE WORK

After release of the manual, training measures are needed.

1. Brief training for the Emergency Staff and the Pikett engineers in the field of covered problems and methods of the manual
2. Detailed training of the Pikett engineers and some experts of the Emergency Staff. This training is to provide the practical skills on how to use the manual.

Some evaluations of the manual are very detailed and difficult to apply in urgent situations. For these evaluations, development of simplified decision schemes is undergoing.

Thanks to the new full scope simulator on the plant site, the TSC manual will often be used during the emergency trainings of Pikett engineers and other experts. It is to be expected, that these trainings will provide a verification/validation of the manual and give opportunities to improve it on a continuous basis.

10 CONCLUSIONS

In some extreme situations, the EOPs rely on the TSC to provide advices of “what to do” and “how to do”. So far, the Beznau Emergency Staff had no specific tool to provide the operating crew the requested advice. With the Beznau TSC manual which has now been developed, the Emergency Staff have the needed tool to provide that advice taking all relevant information into account.

Moreover, the manual can be used for the training of Pikett engineers and other technical specialists, because both categories “Know how” and “Know why” are contained in the manual.

In the framework of the manual's development, plant specific design and perceptions led to changes and supplements in the considered 17 issues of the original (generic) TSC manual developed for Westighouse PWRs. Before implementing the original TSC manual, the individual NPP should evaluate the confidence of the guidance in the issues with plant specific design and accident analysis.

11 LIST OF ACRONYMS

Acronym	**Signification**
CSF	Critical Safety Functions
ECA	Emergency Contingency Actions
ECR	Emergency Control Room
EOP	Emergency Operating Procedure
ERG	Emergency Response Guideline
LOCA	Loss Of Coolant Accident
MCR	Main Control Room
NOK	Nordostschweizerische Kraftwerke
NPP	Nuclear Power Plant
ORP	Optimal Recovery Procedure
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
RCP	Reactor Coolant Pump
RPV	Reactor Pressure Vessel
SED	Site Emergency Director
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SLB	Steam Line Break
TMI	Three Mile Island NPP
TSC	Technical Support Center
WOG	Westinghouse Owners Group

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An Estimation of Operator Action Success Criteria Time Windows with Best Estimate Code

Andrej Prošek, Borut Mavko
Jožef Stefan Institute
Jamova cesta 39, 1000 Ljubljana, Slovenia
Andrej.Prosek@ijs.si, Borut.Mavko@ijs.si

ABSTRACT

To estimate the success criteria time windows of operator actions the conservative approach was used in the conventional probabilistic safety assessment (PSA). The current PSA standard recommends the use of best-estimate codes. The aim of this study was to estimate the operator action success criteria time windows, which were needed for updated human reliability analysis (HRA). The RELAP5/MOD3.3 best estimate code calculations were performed for the three selected initiating events: small or medium loss of coolant accident (LOCA) requiring manual auxiliary feedwater (AFW) start, loss of normal feedwater requiring AFW start, and LOCA requiring manual actuation of safety injection (SI) signal. In these events human actions are supplement to safety systems actuations. For calculations the qualified RELAP5 input model representing a two-loop pressurized water reactor, Westinghouse type, was used. The results of deterministic safety analysis were examined what is the latest time to perform the operator action and still satisfy the safety criteria. The results showed that the time available to perform operator action is larger than the time needed to perform operator action. The results showed that uncertainty analysis of realistic calculation in general is not needed for human reliability analysis when additional time is available and/or the event is not significant contributor to the risk.

1 INTRODUCTION

To estimate the success criteria time windows of operator actions the results of a severe accident code such the MAAP has been used in the conventional probabilistic safety assessment (PSA). However, information from these is often too conservative to perform a realistic PSA for a risk-informed application. On the other hand, the PSA standard [1] recommends the use of best-estimate code to improve the quality of a PSA. Therefore the aim of this study was to estimate the operator action success criteria time windows needed for updated human reliability analysis by using RELAP5/MOD3.3 best-estimate computer code [2]. The specified time windows are important for human reliability analysis (HRA) to determine the likelihood of operator actions. The human error probability of certain action is lower if operators have more time available. In the control room of a nuclear power plant there is a team of operators, which is supervised by a shift supervisor. If operators have 10 or more minutes of additional time for action, it can be expected that colleagues or shift supervisor can observe and correct a possible error of their colleague. Consideration of recovery causes lower human error probability and may cause a different impact of human error to the overall probabilistic safety assessment results. The actual times needed for performing the action were assessed based on real simulator scenarios, while the time windows were determined by deterministic safety analysis. In the present study RELAP5/MOD3.3 best estimate code calculations were performed for the three selected initiating events: establishing auxiliary feedwater in case of small or medium loss of coolant accident (LOCA), establishing auxiliary feedwater in case of transients, and manual actuation of safety injection (SI) signal at LOCA. In these events human actions are supplement to safety systems actuations. For calculations the qualified RELAP5 input model representing a two-loop pressurized water reactor, Westinghouse type, was used [3].

2 SAFETY ANALYSIS METHODOLOGY

The success criteria time windows are described first. Then the input model for the RELAP5/MOD3.3 is described. Finally, for each of the three selected events the scenario is described. The realistic code calculations were performed by RELAP5/MOD3.3 P03 computer code [2].

2.1 Description of success criteria time windows

The idea of the HRA method [4] was to use those deterministic safety analyses to perform sensitivity studies of human actions, which are supplement to safety systems actuations. Sensitivity studies include variations of timing of human action to determine the latest time, when operators have to perform the needed action in order that the main plant parameters are not exceeded their limits. The core cooling success criteria as defined in [5] were used. It is assumed if the hottest core fuel/clad node temperature in the reactor core exceeds 923 K for more than 30 minutes or if temperature of the core exceeds 1348 K, the core damage may occur, which may lead to accident state. Based on the core temperature the time windows were determined.

2.2 RELAP5 Input Model Description

To perform this analysis, Krško nuclear power plant (NPP) has provided the base RELAP5 input model, so called “Master input deck”, which have been used for several analyses, including reference calculations for Krško full scope simulator verification [3], [6]. A full two-loop plant model has been used for the analysis. It includes the new Siemens-Framatome (now Areva) replacement steam generators type SG 72 W/D4-2. The model consists of 469 control volumes, 497 junctions and 378 heat structures with 2107 radial mesh points. Besides, 574 control variables and 405 logical conditions (trips) represent the instrumentation, regulation isolation, safety injection (SI) and auxiliary feedwater (AFW) triggering logic, steamline isolation, and so on.

2.3 Scenarios Description

Three scenarios are described, which were needed for updated human reliability analysis. In these scenarios the human actions are supplement to safety systems actuations. In the first scenario the human action was establishing AFW in case of small or medium LOCA assuming that high pressure safety injection (HPSI) system fails. In the second scenario the human action was establishing AFW in case of loss of feedwater (LOFW) transient. In the third scenario the human action was actuation of SI signal for the most limiting accident (excluding large break LOCA), i.e. small and medium LOCA.

In the case of small or medium LOCA in a nuclear power plant with the assumption that HPSI system fails, one of the means to cool the reactor is through the secondary side depressurization providing that AFW system is operating. Normally, AFW system is automatically put into operation when main feedwater is lost. If the AFW pumps would not start automatically, operators should intervene. The success criterion requires operation of one of three AFW pumps to maintain the flow in order to depressurize the primary system below the accumulator injection setpoint at 4.9 MPa. Besides passive accumulators it was assumed that low pressure safety injection (LPSI) is available too. The parameter to indicate depressurization was primary pressure and the parameter to indicate core cooling was rod cladding temperature. As larger breaks can depressurize through the break in any case below accumulator injection setpoint pressure after some time, AFW is not needed for depressurization. Therefore analysis was performed for a spectrum of break sizes from 1.27 cm (0.5 inch) to 15.24 cm (6 inch) to determine, for which break sizes is needed operation of one AFW pump and for them the time available to start AFW was determined based on the parametric study varying delay of AFW start. The break was located in the cold leg between the reactor coolant pump and the reactor vessel.

The most limiting transient requiring operation of AFW is LOFW. The success criterion is that capacity of one train of AFW is adequate to remove decay heat, to prevent overpressurization of primary system, and to prevent uncovering of the core resulting in core heatup. The time when the operator succeeds to start AFW pump was varied. When the AFW pump started to inject into the secondary side, cooling of the secondary side caused the pressurizer pressure to drop below the pressurizer PORV closure setpoint and then below the maximum pressure capacity of HPSI pump. The HPSI injection efficiently prevents further core uncovery.

The third considered initiating event was LOCA without automatic SI signal actuation. This means that none of the safety systems including HPSI system, LPSI system and AFW system was assumed available. The whole spectrum of LOCAs from 1.91 cm (0.75”) to 15.24 cm (case 6”) break size was evaluated and for the most critical break regarding the time available to the operator it was shown that with establishing safety injection with 20 minutes delay the core heatup could be prevented.

3 RESULTS

Results are shown in Figs. 1 through 5 showing the most important variables, based on which the time available to perform operator action was determined.

3.1 LOCA calculations with manual actuation of AFW

The spectrum of break sizes was analyzed. For the most limiting break regarding time available it was shown that operation of AFW is not enough if not supported by manual opening of steam generator (SG) power operated relief valve (PORV). These two actions were assumed to be performed with the same time delay. The results for a spectrum of break sizes are shown in Fig. 1. From Fig. 1(a) it can be seen that 5.08 cm (2 inch) and larger breaks depressurize (through the break) in any case below accumulator injection setpoint pressure at 4.9 MPa after some time and therefore AFW is not needed for depressurization. On the other hand, 2.54 cm (1 inch) equivalent diameter break size and smaller need depressurization. As reactor coolant system (RCS) mass depletion (see Fig. 1(c)) and core heatup (see Fig. 1(b)) are earlier for 2.54 cm (1 inch) break than for 1.91 cm (0.75 inch) and 1.27 cm (0.5 inch) break, the 2.54 cm break was identified as the most critical regarding the time available to start AFW. Fig. 1(d) shows that for break 2.54 cm (and smaller), the steam generators start to dry out as their inventory is lost through SG PORVs, what caused core heatup.

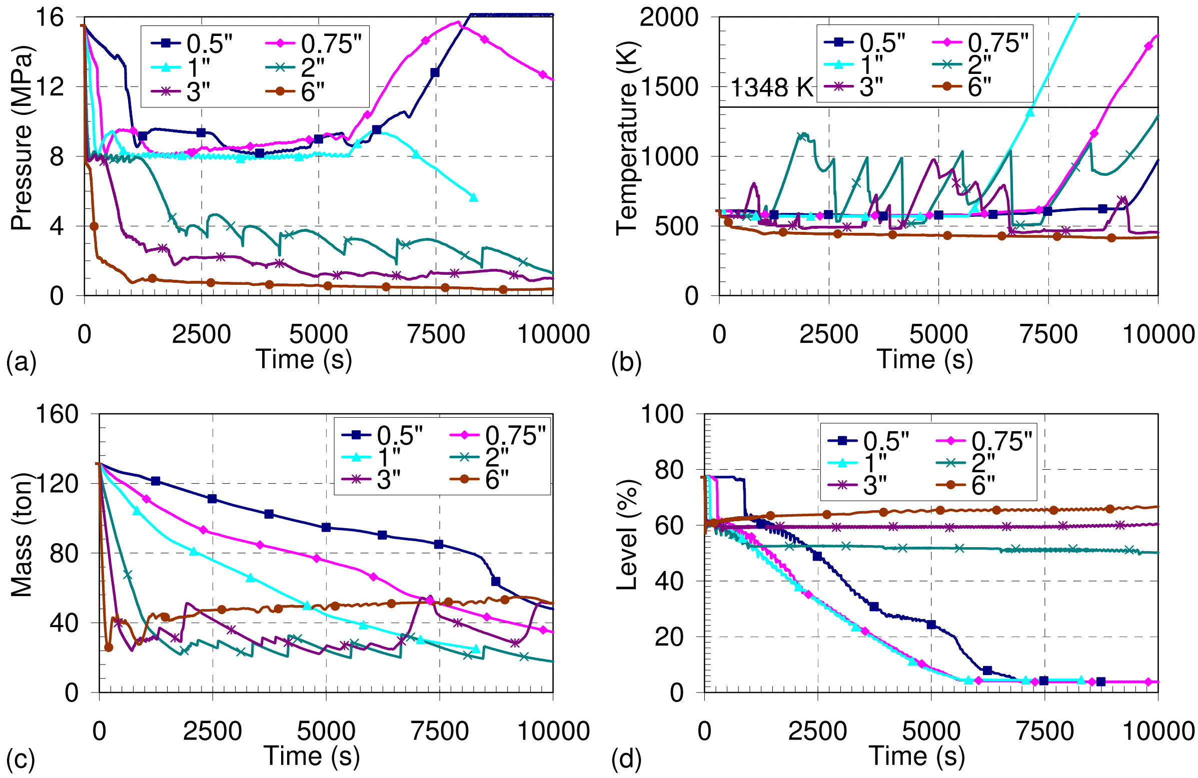

Figure 1: Calculated results for spectrum of break sizes: (a) RCS pressure, (b) Core cladding temperature, (c) RCS mass inventory, (d) SG no.1 wide range level

To establish the depressurization by cooling through the secondary side, AFW is needed. However, as shown in Fig. 2(a), just by operating AFW the RCS pressure could not be depressurized and the core heated up (see Fig. 2(b)). The reason is that the SG PORV is cycling. Once SG is filled to normal level, the AFW injected intermittently following cycling of the PORVs. Depressurization could be efficiently achieved by manual opening of SG PORV providing that SG level is maintained above the minimum level by AFW.

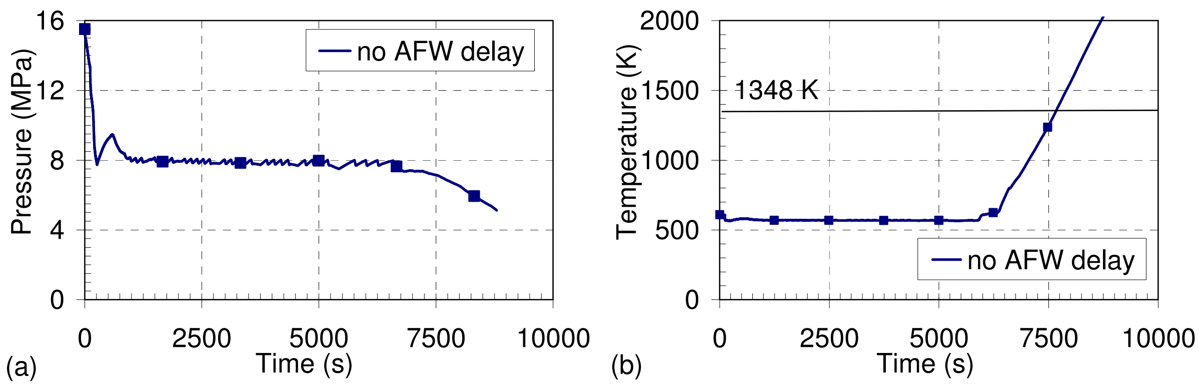

Figure 2: Scenario with AFW available: (a) RCS pressure, (b) Core cladding temperature

To achieve the depressurization for the 2.54 cm break two operator actions were assumed to be performed, manual opening of SG PORV and manual AFW start as shown in Table 1. To determine the time window available to perform these two actions, scenarios with different delays of performing operator actions were analysed.

Table 1: Operator actions delay

Case	Operator action	
	SG PORV full opening delay (min.)	AFW start delay (min.)
A	0	Not available
B	30	30
C	50	50
D	80	80
E	100	100
F	120	120

Fig. 3(a) shows that RCS depressurization with SG PORV is efficient in preventing core heatup (see Fig. 3(b)), when delay is not too large. After the RCS pressure depressurizes below the accumulator injection setpoint, the RCS systems starts to fill as shown in Fig. 3(c). For case A with immediate depressurization of RCS with one SG PORV without AFW available the SG emptied in 40 minutes and core started to heat up 25 minutes later. From Fig. 3(d) it can be seen that for cases B to D the SG level drops approximately linearly and that cooling is sufficient until SG is not emptied. In cases A, E and F the SGs emptied and the core heatup was therefore unavoidable. From case E it can be seen if operator actions are performed immediately after SGs emptying the further heat up could still be prevented. Based on the set criteria 100 minutes are available to the operators.

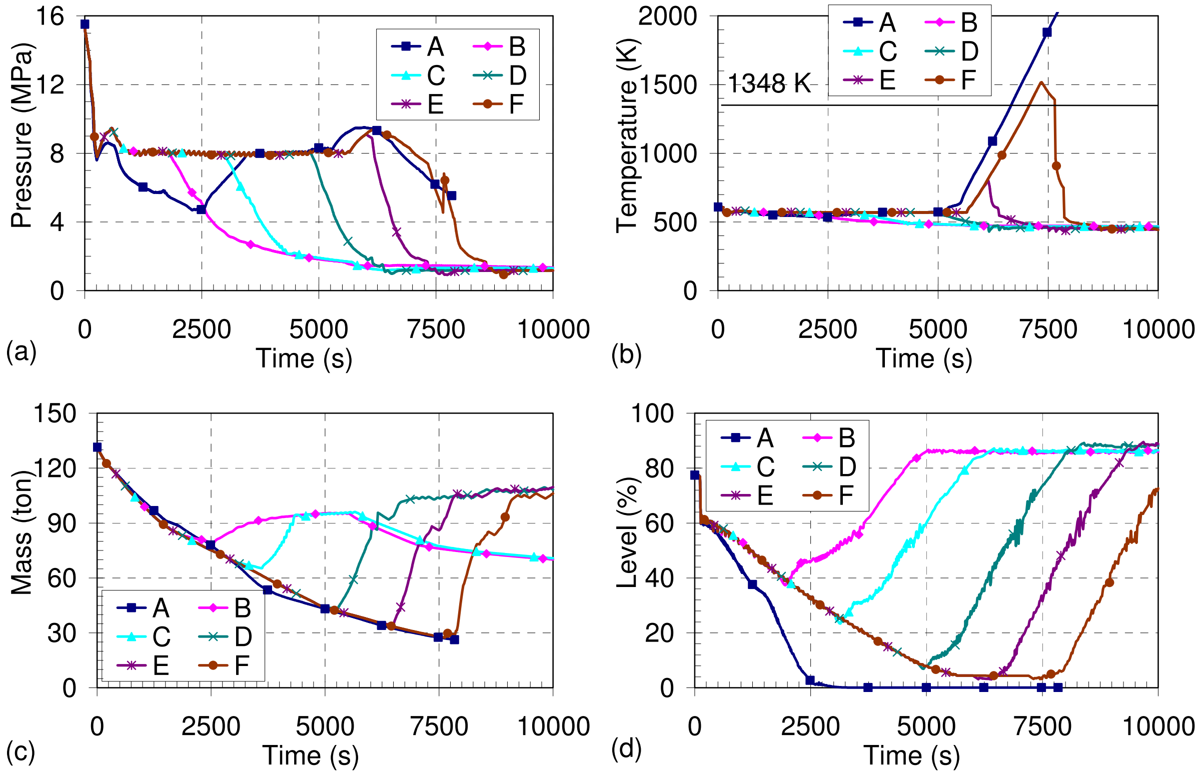

Figure 3**:** 2.54 cm break scenarios with two operator actions: (a) RCS pressure, (b) Core cladding temperature, (c) RCS mass inventory, (d) SG no.1 wide range level

3.2 LOFW Calculations with Manual Actuation of AFW

The delays of AFW pump start from 40 min. to 70 min. were simulated to determine the time window for AFW start. The pressurizer pressure shown in Fig. 4(a) does not exceed 18.95 MPa what means that primary system was not ovepressurized. From Fig. 4(b) the core heatup could be determined, while Fig. 4(c) shows the RCS mass inventory. Finally, in Fig. 4(d) it is shown the SG no. 1 level, which starts to efficiently fill after AFW pump start thus enabling RCS depressurization.

At the time when one AFW pump was started to inject into the secondary side, cooling of the secondary side caused the pressurizer pressure to drop below the pressurizer (PRZ) PORV closure setpoint and then below the maximum pressure capacity of HPSI pump (see Fig. 4(a)). The closure of the pressurizer PORV and coolant injection into primary system resulted in recovering the RCS inventory as shown in Fig. 4(c) and quenching the core as shown in Fig. 4(b). From Fig. 4(c) it can be seen that the RCS mass depletion depends mainly on the delay of one AFW pump start. The parametric analysis showed that the core significantly heats up with start of AFW pump delayed for 60 minutes or greater. The case with start of one AFW pump delayed for 50 minutes cause small core heatup and with delay of 60 minutes the core temperature is still below criterion 1348 K for core damage, while in the case with delay of 70 minutes this value is exceeded. Finally, in Fig. 4(d) is shown the steam generator wide range level. As already mentioned the start of AFW caused filling of steam generator and RCS system depressurization. Also it can be seen that the steam generator fills in approximately one hour.

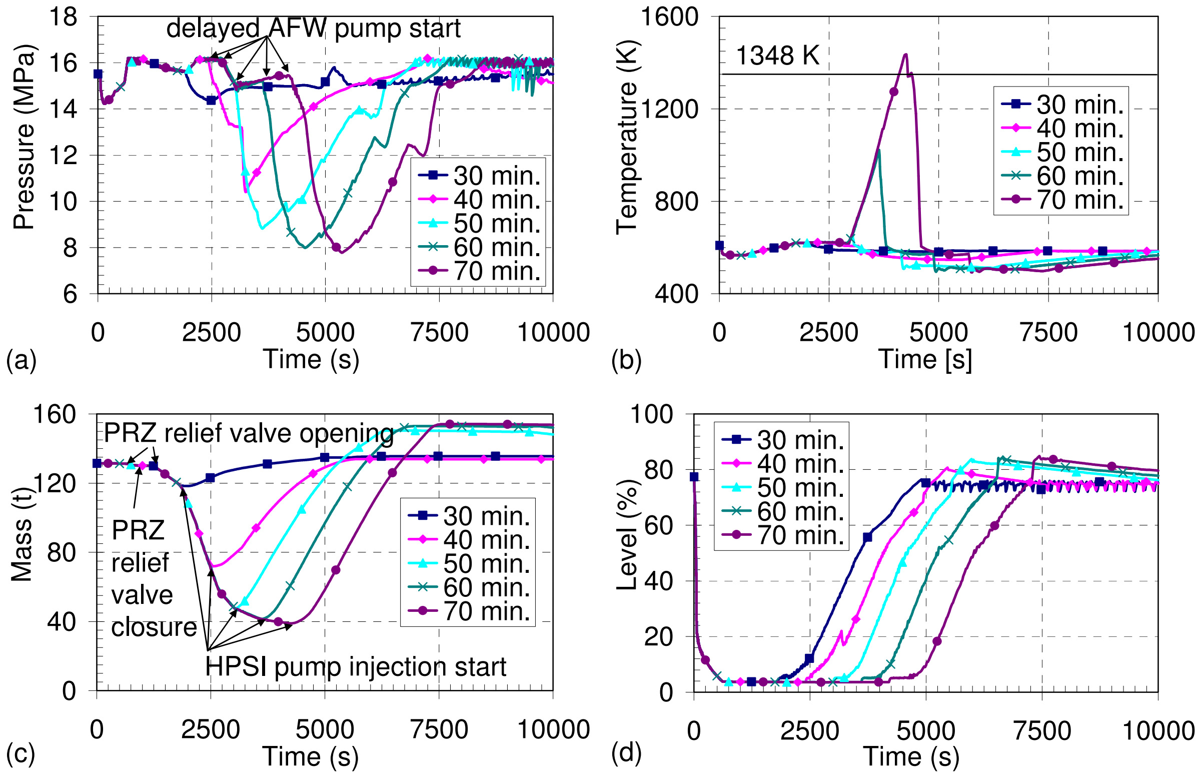

Figure 4: LOFW transient: (a) RCS pressure, (b) Core cladding temperature, (c) RCS mass inventory, (d) SG no.1 wide range level

3.3 LOCA Calculations with Manual Actuation of SI

The results are shown in Fig. 5. At breaks smaller than 5.08 cm the RCS was not sufficiently depressurized as shown in Fig. 5(a) to enable accumulator injection, while larger breaks depressurize the RCS. Figure 5(b) shows that the temperature criterion 1348 K is first exceeded for 15.24 cm (case 6”), then for 10.16 cm break (case 4”), 7.62 cm (case 3”), 1.91 cm (case 0.75”) and the last for 5.08 cm (case 2”). The reason is that for 5.08 cm break the accumulators were sufficient to cool the core until they emptied. At breaks larger than 5.08 cm the core starts to significantly heatup after the accumulators emptied. In general it can be concluded, the larger is the break the faster is the core uncovery. For the 15.24 cm break the core starts to heatup at 20 minute. For the 5.08 cm break core cladding temperature could exceed criterion at first peak in the case of considering the uncertainty. When SI signal was actuated at that time further core heatup was prevented (case 6” SI). Similarly this was the case for 5.08 cm break (case 2” SI). Therefore, at least 20 minutes are available for the operator action. In the case of this scenario the treatment of uncertainty is not needed as the time window is the shortest for the largest break in the spectrum.

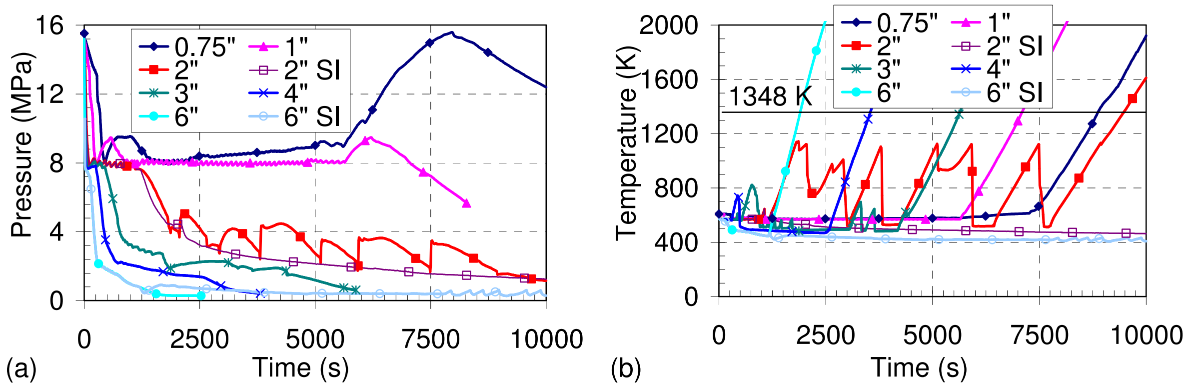

Figure 5: LOCA with manual actuation of SI: (a) RCS pressure, (b) Core cladding temperature

3.4 Results discussion

The times needed for performing operator actions were determined based on the simulator experience [7]. For starting the AFW the operator needs from 1 to 10 minutes, while for SI signal actuation 2 minutes are needed. When the time window is large, much of the additional time is available and there is no need to very accurately determine the time window even if the human factor event is an important contributor to the risk. For example, the time needed to start SI signal is 2 minutes and there is additional 18 minutes to perform this action. Considering typical uncertainties in peak cladding temperatures of 200 K based on previous uncertainty evaluations [8] and adiabatic heatup rate for 15.24 cm break, the criterion would be reached 3 minutes earlier. Equally important is also time uncertainty of reaching maximum temperature which is approximately 2 minutes according to [9]. The additional time considering uncertainties is still sufficient.

In the case of small and medium break LOCAs with the assumption that HPSI is not available, the depressurization is needed for breaks smaller than 5.08 cm. The break 5.08 cm is limiting as for this and larger breaks the RCS depressurize by itself. However, when the pressure drops below the accumulator injection point, the core is already heated up for 5.08 cm break. Considering the typical cladding temperature uncertainty of the best estimate calculation to be 200 K [8] the criterion 1348 K could be exceeded. The recovery action would be questionable because of short time window. The uncertainty analysis was not needed, as the risk contribution of this event to the plant risk is insignificant.

On the other hand, establishing AFW at LOFW event is significant contributor to the risk, but the calculated time window gives sufficient additional time, even if conservative time window is considered in the human reliability analysis.

For the case of LOCA with delayed SI signal actuation it was shown that the additional time available is sufficient, therefore uncertainty analysis is not needed in spite of the fact that event is significant contributor to the risk.

All these examples showed that uncertainty analysis was not needed, as additional time was available and/or the event was not significant contributor to the risk. If the event is significant contributor to the risk or not, it is answered by PSA. Based on this it can be concluded that uncertainty analysis may be valuable only for significant risk contributors, when additional available time is small. For the selected examples this was not the case. In reference [10] it is proposed to estimate the uncertainty for an operator's action in the PSA work scope by considering conservative time windows.

4 CONCLUSIONS

The operator action success criteria time windows were estimated using RELAP5/MOD3.3 for updated human reliability analysis. For the three selected cases the results of deterministic safety analysis were examined in sense how late after the required human intervention the operator performs its action that the safety criteria are not exceeded. This gives available time for operator to act. The results of deterministic analyses showed that in some cases the treatment of uncertainty for variables compared with safety criterion could significantly change the time window. However, based on the information from PSA regarding the contribution to the risk, uncertainty analysis was not needed, what greatly support the use of best estimate codes for probabilistic safety assessment. It can be concluded that uncertainty evaluation of realistic safety analysis may be needed only when there is little time for recovery action and the affected human factor event is an important contributor to risk.

ACKNOWLEDGMENTS

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AN EVALUATION ON THE EMERGENCY OPERATION STRATEGY FOR THE RECIRCULATION SUMP BLOCKAGE AT KORI UNITS 3&4

Chang Gyun Lee, Cheol Shin Lee, Sung Hyun Shin, Eun Kee Kim
Korea Power Engineering Company, Inc.
150 Deokjin-dong, Yuseong-gu, Daejeon, 305-353, Republic of Korea
Email: cglee@kopec.co.kr, cslee@kopec.co.kr

Jong Woon Park, Chang Hyun Kim
Korea Hydro & Nuclear Power Co., Ltd.
25-1 Jang-dong, Yuseong-gu, Daejeon, 305-343, Republic of Korea
Email: parkjw@ khnp.co.kr

ABSTRACT

The purpose of this paper is to evaluate the emergency operation strategy on the recirculation sump blockage to address the recommendation in USNRC Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors" for Kori Units 3&4 in Korea. This Bulletin requires that the licensees evaluate the ECCS and CSS performance in compliance with 10CFR50.46(b)(5) (Option 1) or implement interim compensatory measures to reduce the potential risk due to LOCA-generated debris (Option 2) before the design improvement and installation of the strainer to resolve the GSI 191, "Assessment of Debris Accumulation on PWR Sump Performance". WOG has developed a guidance to prepare the plant-specific responses to the Bulletin and performed qualitative evaluations and quantitative analyses for selected COAs to be incorporated into a plant's EOP. The results of these evaluations and analyses were summarized in WCAP-16204-NP, "Evaluation of Potential ERG and EPG Changes to Address NRC Bulletin 2003-1 Recommendations" for both Westinghouse and Combustion Engineering type plants. In this paper, the analyses for operator actions securing containment spray pump and safety injection pump prior to recirculation alignment were performed to evaluate the applicability of the recommended operator actions to Kori Units 3&4 using computer codes, RELAP5/MOD3 and CONTEMP4/MOD5. Based on the analysis result, it is concluded that the operator action for early termination of one containment spray pump is applicable to the current EOP for Kori Units 3&4 if three or four Containment Fan Coolers are operating.

1. INTRODUCTION

A major safety issue in relation to long-term recirculation cooling after a LOCA is that LOCA-generated debris may be transported to the recirculation sump screen, eventually resulting in the blockage in the sump screen and loss of ECCS and CSS pump's NPSH margin. The USNRC issued Bulletin 2003-01[1] to inform licensees of the potential adverse effects due to sump blockage by LOCA-generated debris in the recirculation operation mode during long-term cooling. In addition, the Bulletin requested the verification of conformance to the existing regulatory requirements or implementation of any interim compensatory measures to reduce the risk due to LOCA-generated debris before the sump design improvement.

WOG launched a research program to respond to the Bulletin 2003-01 and recommended COAs to be incorporated into a plant's EOP based on the qualitative evaluation and quantitative analysis for both Westinghouse and Combustion Engineering type plants. The results of this program were summarized in Reference [2]. The reference provides generic support and guidance for those licensees that choose to include operational changes as part of their response to the Bulletin 2003-1. WOG selected eleven COAs from the Bulletin 2003-1 and operator input from Procedures Working Group of WOG. Selection of COAs was based on 1) Were they identified in USNRC Bulletin 2003-01, 2) Could they increase the time to automatic switchover to recirculation and 3)Could they reduce the velocity of recirculation through the sump. Eleven COAs are as follows:

(1) Secure one containment spray pump before recirculation alignment,
(2) Manually initiate one train of containment sump recirculation earlier,
(3) Terminate one train of HPSI/high-head injection after recirculation alignment,
(4) Terminate LPSI/RHR pump prior to recirculation alignment,
(5) Refill refuelling water storage tank,
(6) Inject more than one RWST volume from refilled/diluted RWST or by bypassing RWST,
(7) Provide more aggressive cooldown and depressurization following a small break LOCA,
(8) Provide guidance on symptoms and identification of containment sump blockage,
(9) Develop contingency actions in response to containment sump blockage, loss of suction, and cavitation,
(10) Terminate HPSI/high-head injection prior recirculation alignment,
(11) Delay containment spray actuation for small break LOCA in ice condenser plants.

Some COAs require quantitative analysis to verify the benefit of their implementation to plant-specific EOP. However, some COAs need only qualitative evaluation like the review of relevant steps in the current EOP. In particular, operator actions securing containment spray pump and safety injection pump before the alignment of containment sump recirculation mode need to be analyzed quantitatively. These quantitative and qualitative evaluations should be performed for the justification and implementation of these COAs into plant-specific EOP.

The purpose of this paper is to evaluate the emergency operation strategy on the recirculation sump blockage to address the recommendation in USNRC Bulletin 2003-01 for Kori Units 3&4, Westinghouse type PWRs, which consist of 3 RCS loops. Kori Units 3&4 have two trains of safety injection system. Each train consists of one HPSI pump and one LPSI pump. The plants have also two trains of containment heat removal system to maintain the safety function of containment pressure and temperature. Each train includes one containment spray pump and two containment fan coolers. The system is designed such that two containment spray pumps, four containment fan coolers or one train of the system provide enough heat removal to maintain the safety function of containment pressure and temperature. At the time of recirculation actuation, ECCS and CSS pumps transfer their suction from the RWST to the containment sump.

In this paper, the analyses for operator actions securing containment spray pump and safety injection pump prior to recirculation alignment were performed to evaluate the applicability of the operator actions to Kori Units 3&4 using computer codes, RELAP5/MOD3 and CONTEMP4/MOD5.

2. EVALUATION METHODOLOGY

Two major operator actions of COAs which require quantitative analyses were selected to evaluate the emergency operation strategy on the recirculation sump blockage. The analyses of two major operator actions were performed to justify quantitatively the implementation of the operator actions into Kori Units 3&4 EOP[3] in terms of reducing the flowrate through the sump screen and delaying the time to the start of containment recirculation mode during LOCA.

One category is to secure containment spray pump(s) before the alignment of containment recirculation mode (COA 1). This category is to verify the RWST depletion time and containment pressure and temperature within the EQ curve limit when any containment spray pump is secured.

The other category is to terminate safety injection pump(s) prior to the alignment of containment recirculation mode (COAs 4 & 10). This category is to verify PCT of LOCA licensing requirements when any safety injection pump is terminated. Although this operation action is conflicted with the current operator action steps, the analysis for this category was performed to provide the bases in case it is required to modify the emergency operation strategy related to safety injection pump termination.

The methodology applied in this paper referred the same methodology provided in Reference [2] including the selection of accident scenarios, aspect of evaluation, etc.

2. 1 ANALYTICAL TOOL

Best-estimate simulation tools were used in the analyses. Thermal hydraulic response of the NSSS to LOCA was simulated using RELAP5/MOD3[4] and the containment pressure and temperature response to mass and energy release was simulated by coupling RELAP5/MOD3 and CONTEMP4/MOD5[5].

Figure 1 shows the RELAP5/MOD3 nodalization for Kori Units 3&4. The nodalization consists of 296 hydraulic volumes and 342 junctions for modelling primary and secondary systems including steam generator, pressurizer, reactor coolant pumps and various safety systems.

2.2. INITIAL CONDITIONS AND ANALYSIS SCENARIO

Analyses for two categories above were performed to evaluate the emergency operator actions for securing containment spray pump and safety injection pump. Table 1 enumerates the initial conditions for key parameters used in the analyses.

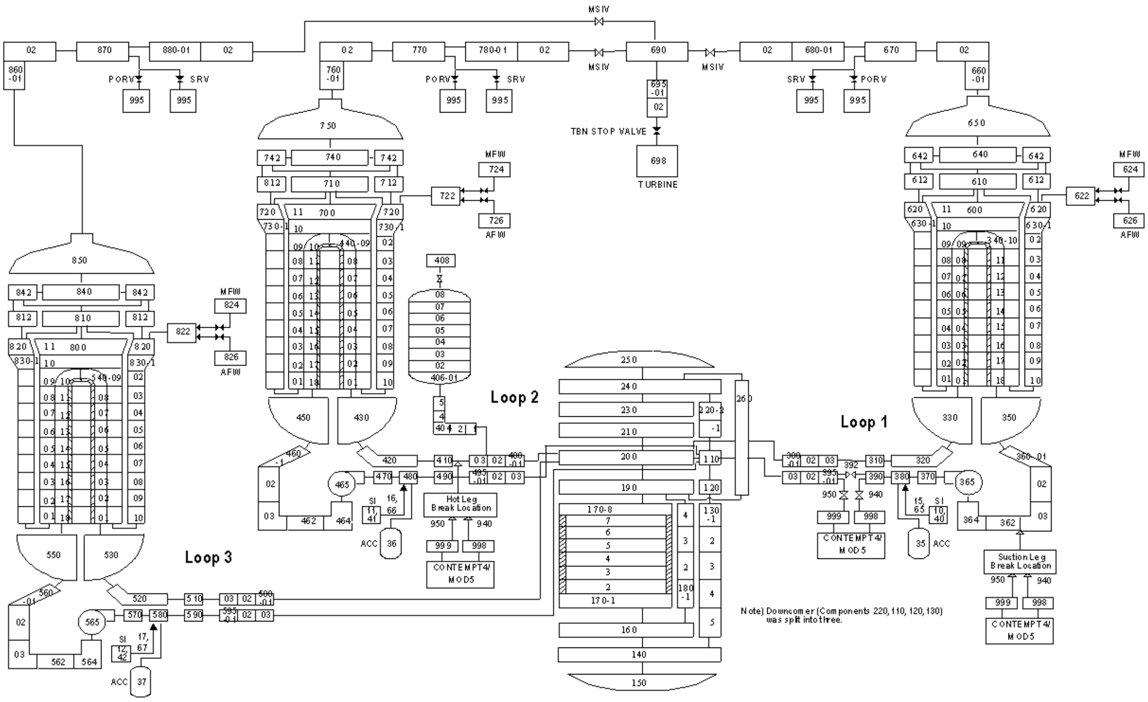

Figure 1: RELAP5/MOD3 Nodalization for Kori Units 3&4

Table 1: Initial Conditions for Key Parameters

Key Parameter	Value *
Core Thermal Power, MWt	2,958
Pressurizer Pressure, MPa(psia)	15.51 (2,250)
Pressurizer Level, %	58.93
RCS LOOP Flowrate, kg/sec	4443.52
Cold-Leg Temperature, K(^{o}F)	562.59 (553)
Hot-Leg Temperature, K(^{o}F)	600.37 (621.0)
Core Average Temperature, K(^{o}F)	584.54 (592.5)
S/G Pressure, MPa (psia)	6.38 (926)
S/G Level, m	12.75
RWST Inventory, m^3 (gal)	1,639.8 (433,200)
Containment Spray flowrate (for 1 CSP) , kg/sec (gpm)	173.34 (2,750)

* Best-estimate value

- <u>Category I : Securing Containment Spray Pump</u>

Operator action securing containment spray pump is intended to delay the time up to the start of recirculation and to reduce the flow rate through the sump screen when recirculation begins. It is also expected to reduce the differential pressure across the containment sump screen if there is any debris buildup.

To evaluate the duration of RWST depletion time, it is assumed that one containment spray pump is secured for a small break LOCA when two containment spray pumps are operating. It is also assumed that one containment spray pump is secured on CSAS and at 10 minutes after 2 inch- and 6 inch-small break LOCA, respectively. Operator action time considered is 10 minutes for analysis of emergency operation strategy generally, and the representative smaller break sizes for LOCA are selected because it will have a negligible effect on large breaks. It is also assumed that two trains of SIS are operating. It is expected that this operator action delays the initiation of recirculation operation.

In addition, one remaining operating containment spray pump may also stop if its electric bus is lost. This case needs a quantitative analysis to justify that containment temperature and pressure will be bound by the current licensing basis. To demonstrate the conformance with environmental qualification requirements in case of securing all containment spray pumps, it is assumed that one of the two operating containment spray pumps is turned off by operator at 10 minutes after LOCA, and at the same time the remaining spray pump is stopped due to the loss of electric bus to this pump. This case was performed for large break LOCA at the discharge leg of RCP by varying operable CFCs to quantitatively support the implementation of this operator action. The cases for operator action securing containment spray pump are summarized in Table 2.

Table 2: Cases of Securing Containment Spray Pump

Input / Case	Break Size	Containment Spray Pump	CFC
1	2 in SB	1 Pump Stop At CSAS	N/A
2	6 in SB	1 Pump Stop after 10 min	N/A
3	LB	All Pumps Stop after 10 min	1 CFC Available
			2 CFCs Available
			3 CFCs Available
			4 CFCs Available

- <u>Category II : Securing Safety Injection Pump</u>

Operator action securing safety injection pump is aimed at reducing the total flow through the sump screen and the rate of debris transport. It is expected to reduce the risk of sump blockage following a LOCA. This analysis includes shutting down redundant safety injection pumps that are not necessary to provide the required flows for the heat removal of the reactor core.

To evaluate the impact on the LOCA consequences, it is assumed that 1) all safety injection pumps, 2) one HPSI pump and two LPSI pumps, and 3) one LPSI pump and two HPSI pumps are secured at 10 minutes after large break LOCA for each case. The case of securing all safety injection pumps simulates the scenario for which the operator turns off flow from one safety injection train (1 HPSI and 1 LPSI) with a single failure causing the loss of safety injection flow from the other safety injection train.

This analysis was performed to evaluate the conformance with LOCA licensing requirements for peak cladding temperature when any safety injection pump prior to recirculation alignment is early terminated to delay ECCS suction switchover from the refuelling water tank to containment sump. These cases are summarized in Table 3.

Table 3: Cases of Securing Safety Injection Pump

Input / Case	Break Size	HPSI Pump	LPSI Pump
4	LB	All Pumps Stop after 10 min	All Pumps Stop after 10 min
5	LB	Only 1 Pump Available after 10 min	All Pumps Stop after 10 min
6	LB	All Pumps Stop after 10 min	Only 1 Pump Available after 10 min

3. EVALUATION OF ANALYSIS RESULTS

Figures 2 and 3 show the depletion of the RWST during 2 inch- and 6 inch-small break LOCA, respectively. As shown in the figures, RWST depletion time for securing one of two operating containment spray pumps after small break LOCA is longer by 11.5 to 16.4 minutes than the case of two operating containment spray pumps. This operator action is effective for the prolongation of the RWST depletion time and delaying the recirculation operation as one of interim compensatory measures in response to Bulletin 2003-01.

Figures 4 and 5 show the containment pressure and temperature, respectively, when all containment spray pumps are secured during LBLOCA. As shown in the analysis results, the CFCs maintain the containment pressure and temperature within environment qualification curve. However, in case of one or two running CFCs, the containment pressure and temperature increase slowly after securing two containment spray pumps and start to exceed environment qualification limit at about $3x10^5$ seconds and $3x10^4$ seconds, respectively. Therefore, at least three CFCs or more should be operated to maintain the safety function of the containment pressure and temperature control.

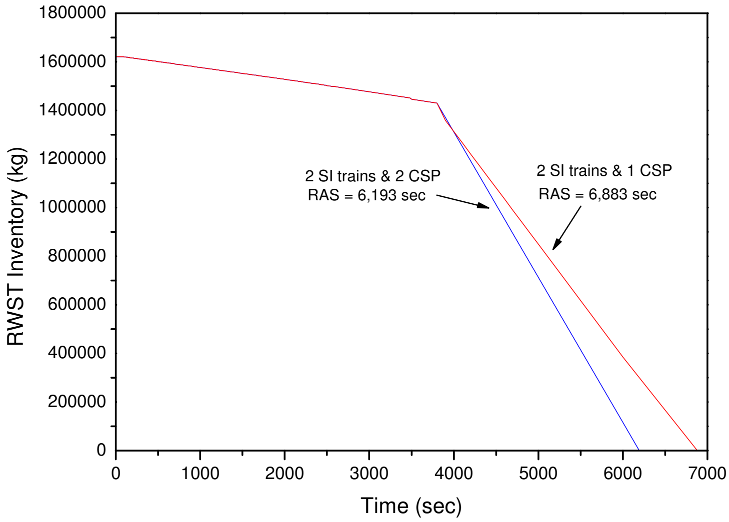

Figure 2: RWST Remaining Inventory during 2 inch SBLOCA (Case 1)

(1 CSP Stop @ CSAS)

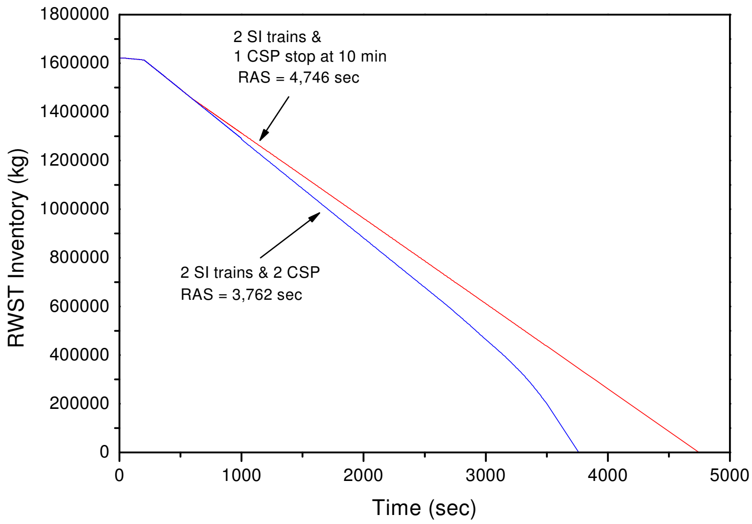

Figure 3: RWST Remaining Inventory during 6 inch SBLOCA (Case 2)

(1 CSP Stop @ 600 sec)

The results of this analysis show that the operator action securing one of two operating containment spray pumps is applicable to the current EOP for Kori Units 3&4 if three CFCs or more are operating. In addition, these results for Kori Units 3&4 are similar to the results of analysis for securing containment spray pump in Reference [2].

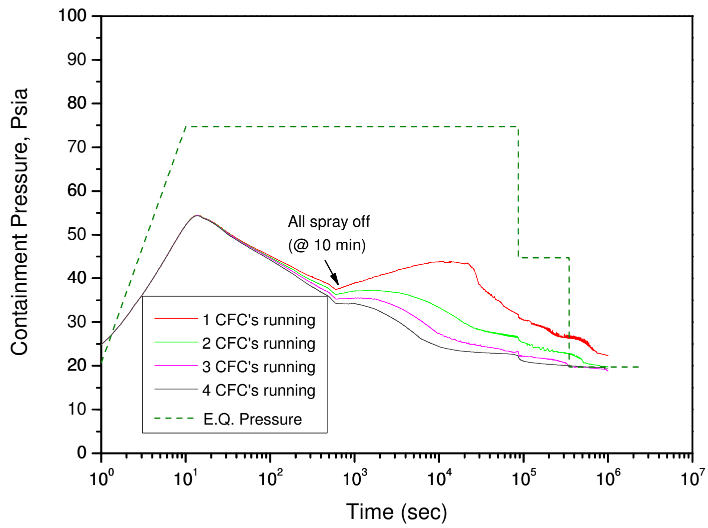

Figure 4: Containment Pressure during LBLOCA (Case 3)
(2 CSPs Stop @ 600 sec)

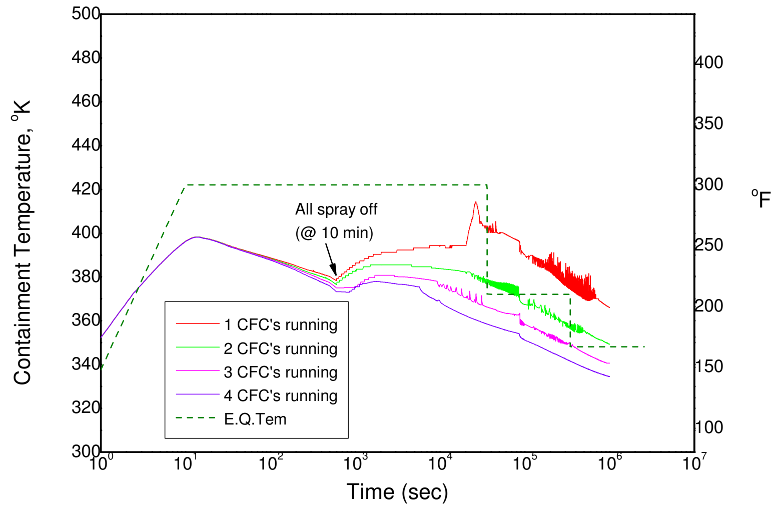

Figure 5: Containment Temperature during LBLOCA (Case 3)
(2 CSPs Stop @ 600 sec)

Figures 6 through 8 illustrate the peak cladding temperature for cases 4, 5 and 6, respectively. Figure 6 shows the variation of the cladding temperatures when all safety injection pumps are stopped at 10 minutes after LBLOCA. The temperature of higher fuel regions starts to increase due to the lack of SI flow from about 1,000 seconds. Since these temperatures are rising very rapidly, it is expected that the cladding temperature could exceed the acceptance criterion for licensing analysis within a few minutes.

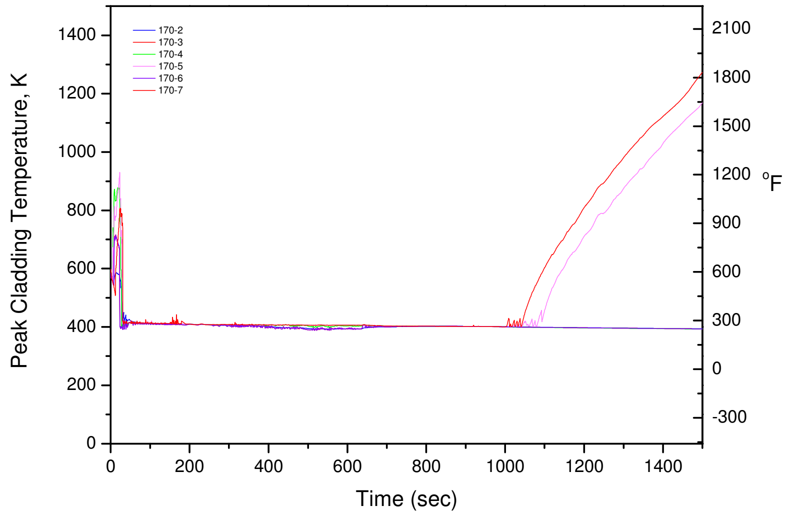

Figure 6: Peak Cladding Temperature (Case 4)
(2 SI Trains Stop @ 600sec)

Figures 7 shows the variation of the cladding temperature when all safety injection pumps except only one HPSI pump are stopped at 10 minutes after LBLOCA. As shown in the figure, the cladding temperature for all fuel regions is less than 300℉ as the core region is fully covered by the safety injection flow. Figure 8 depicts the variation of the cladding temperature when all safety injection pumps except only one LPSI pump are stopped at 10 minutes after LBLOCA. As shown in the results, the cladding temperature for all fuel regions is also maintained at less than 300℉ as in the previous case. These indicate that the safety injection flow of only one HPSI or LPSI pump is sufficient to keep the core covered and to remove decay heat.

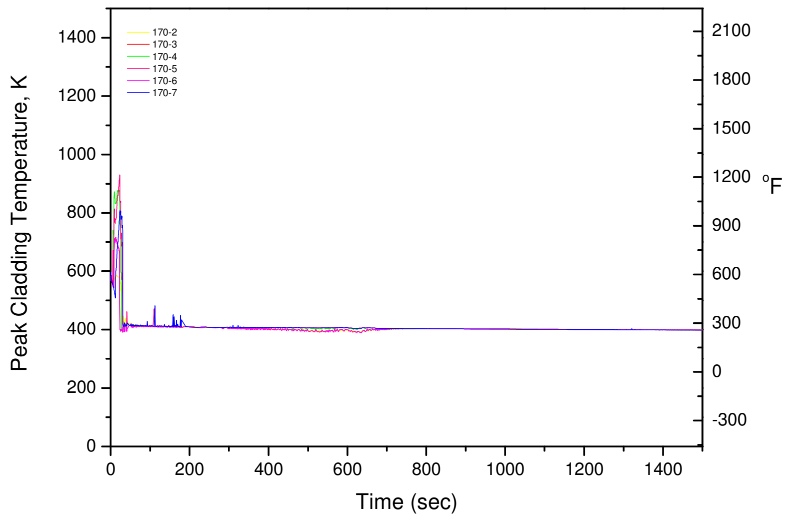

Figure 7: Peak Cladding Temperature (Case 5)
(1 HPSI & 2 LPSI Pumps Stop @ 600sec)

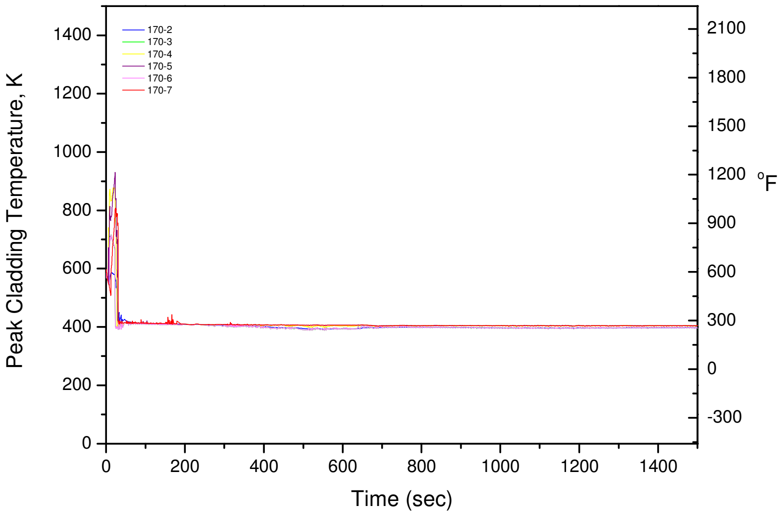

Figure 8: Peak Cladding Temperature (Case 6)
(2 HPSI & 1 LPSI Pumps Stop @ 600sec)

The analysis results for securing one HPSI or one LPSI pump show that this operator action is effective for the prolongation of the RWST depletion time delay with LOCA licensing requirements satisfied. However, as described above, this operator action conflicts with the operation step for safety injection termination of the current EOP for Kori Units 3&4. The results of this analysis will provide the backgrounds in case it is required to modify the emergency operation strategy related to safety injection pump termination. Therefore, this operator action securing safety infection pump is not applicable to the current EOP for Kori Units 3&4.

4. CONCLUSIONS

The evaluation of the emergency operation strategy on the recirculation sump blockage was performed to address the recommendation in the Bulletin 2003-01 for Kori Units 3&4. The analyses for operator actions securing containment spray pump and safety injection pump prior to the alignment of containment sump recirculation mode were carried out to evaluate the applicability of the recommended operator actions to Kori Units 3&4. It is concluded that the operator action securing one of two operating containment spray pumps before recirculation alignment is applicable to the EOP for Kori Units 3&4 if three CFCs or more are operating.

In addition, the COA for RWST refill has already incorporated into the current EOP for Kori Units 3&4. The COA related to symptoms and contingency actions of containment sump blockage is considered for its incorporation into the EOP for Kori Units 3&4.

ACKNOWLEDGMENTS

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NOMENCLATURE

BL	Bulletin
CFC	Containment Fan Cooler
COA	Candidate Operator Action
CSAS	Containment Spray Actuation Signal
CSS	Containment Spray System
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guideline
ERG	Emergency Response Guideline
GL	Generic Letter
GSI	Generic Safety Issue
HPSI	High Pressure Safety Injection
KHNP	Korea Hydro & Nuclear Power Company, Ltd.
LPSI	Low Pressure Safety Injection
LOCA	Loss of Coolant Accident
NPSH	Net Positive Suction Head
NPP	Nuclear Power Plant
PWR	Pressurized Water Reactor
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWST	Refueling Water Storage Tank
SIS	Safety Injection System
USNRC	United States Nuclear Regulatory Committee
WOG	Westinghouse Owners Group.

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[2] WCAP-16204-NP, Rev.1, "Evaluation of Potential ERG and EPG Changes to Address NRC Bulletin 2003-01 Recommendations (PA-SEE-0085)," March 2004.
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[5] Korea Hydro & Nuclear Power Company, “The Specific Technical Topic Report of the Best Estimate for the ECCS,” TR-KHNP-0002, 2002.12.

European Nuclear Society

Rue Belliard 65
1040 Brussels
Belgium
Telephone +32 2 505 30 54
Fax + 32 2 502 39 02

topsafe2008@euronuclear.org

www.euronuclear.org