Assessing the risk related to the spent fuel pool and interactions with the reactor core for WWER type reactors

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Abstract:

The Fukushima accident has shown that a severe accident may occur in the spent fuel pool (SFP) and also highlighted the potential for a simultaneous severe accident in two places: reactor core and SFP. In-depth assessment of SFP accident sequences is necessary for improvement of severe accident management strategies to further enhance the safety of nuclear power plants.

This paper deals with the specifics of risk assessment for the SFP and the SFP interaction with the reactor core. Methodological aspects and preliminary results of SFP PSA Level 1 and Level 2 are presented, as well as general issues on SFP interaction with the reactor core for WWER type reactors are discussed.

1 INTRODUCTION

The spent fuel pool in nuclear power plants can represent a significant risk of radiation releases in case of melting/damage of stored spent fuel. NPP safety analyses performed up to date are mainly focused on the reactor core. The accident at Fukushima Daiichi unit 4 has demonstrated the vulnerability of SFP with regard to extreme hazards and loss of residual heat removal.

Conservative deterministic SFP targeted safety re-assessments have been performed as a part of the EU stress-tests [1]. ETSON member states recognized the necessity to perform an in-depth analysis of fuel damage in SFP and the associated accident progression and consequences, which resulted in a relevant position paper [2].

At present existing methodologies on PSA development [3] - [5] are not well matured relating to peculiarities of PSA for SFP and SFP interactions with the reactor core.

This paper summarizes joint efforts of the group of researchers from SSTC NRS (Ukraine), UJV Rez (Czech Republic) and SEC NRS (Russia) on SFP PSA and SFP interactions with the reactor core for WWER type reactors trying to attract attention to this issue and presenting preliminary work results.

Two operating WWER-440/213 and WWER-1000/320 designs are subject to studies because of the specific design features and location of SFPs. The WWER-1000 houses the SFP inside the containment, while the SFP in WWER-440 is located outside the hermetic compartments.

2 WWER SFP GENERAL DESCRIPTION

This chapter briefly describes design of spent fuel pools of WWER-440/213 (Dukovany NPP) and WWER-1000/320 (Kalinin NPP, Rostov NPP, Khmelnitskiy NPP).

2.1 Design of WWER-440 SFP

WWER-440/213 is a six-loop reactor design that includes a passive pressure suppression tower (referred to as a bubble tower) and a set of sealed boxes that together fulfill the containment function. There are three independent trains of both high- and low-pressure emergency core cooling systems (ECCS), as well as four hydroaccumulators.

Each of the two twin units shares a reactor building, an auxiliary building, a central pumping station with four cooling towers, a diesel generator station (with 3 DGs per unit), an emergency feedwater pumping station, and an operational building with laboratories. Each unit has its own control room from which its condition and operation can be monitored and controlled.

The containment is usually referred to as the “hermetic zone”, while other rooms in the reactor building are referred to as the “airtight zone”. The hermetic zone is more leaktight and stronger than the airtight zone, and it is tested regularly for leaktightness during reactor outages. The airtight zone is equipped with its own ventilation systems, some of which contain aerosol and iodine filters. The airtight zone is kept at slight underpressure during normal operation. Spent fuel pool is located in the reactor building, but outside the containment.

The spent fuel pool is a “room” with trapezoidal floor projection. It is approximately 15 meters high with total volume 335 m^3. Used fuel assemblies are placed vertically in a stable storage rack (max. 699 fuel assemblies) in the SFP for at least 5 years. During extended outage, when all fuel rods are removed from the reactor, a reserve storage rack is used for temporary storage of the fuel rods (max. 350 fuel assemblies). The reserve storage rack is placed on the stable storage rack, so the fuel assemblies are placed in two layers during the extended outage.

Cooling of the fuel assemblies in the SFP is secured by SFP normal cooling system (TG11,12). The cooling system consists of 2 independent trains: 1 in operation and 1 in standby. Each train includes a centrifugal pump with a normal flow rate of 300 m^3/h and a heat exchanger cooled by service water.

2.2 Design of WWER-1000 SFP

WWER-1000 type reactors have spent fuel pools under the containment. SFP of WWER-1000/320 consists of four sections.

To supply water to SFP, three pumps and three heat exchangers are used. Each pump can supply water to any SFP compartment. Heat from the heat exchangers is transferred to the essenetial cooling water system. Flowsheet for SFP systems of WWER-1000/320 is presented in Figure 1.

The SFP cooling system interacts with the following systems:

- boron concentrate system, through which water is supplied to SFP in case of failure of pumps, heat exchangers or relevant fittings of the three-train system
- spray system, through which emergency water is suplied to SFP in case of failure of pumps, heat exchangers or relevant fittings in three trains of the cooling system or boron concentrate system
- essential cooling system, system of cooling heat exchangers
- power supply system
- ventilation system TL10, cooling room which houses the system equipment
- industrial control system, through which the pumps and valves are controlled.

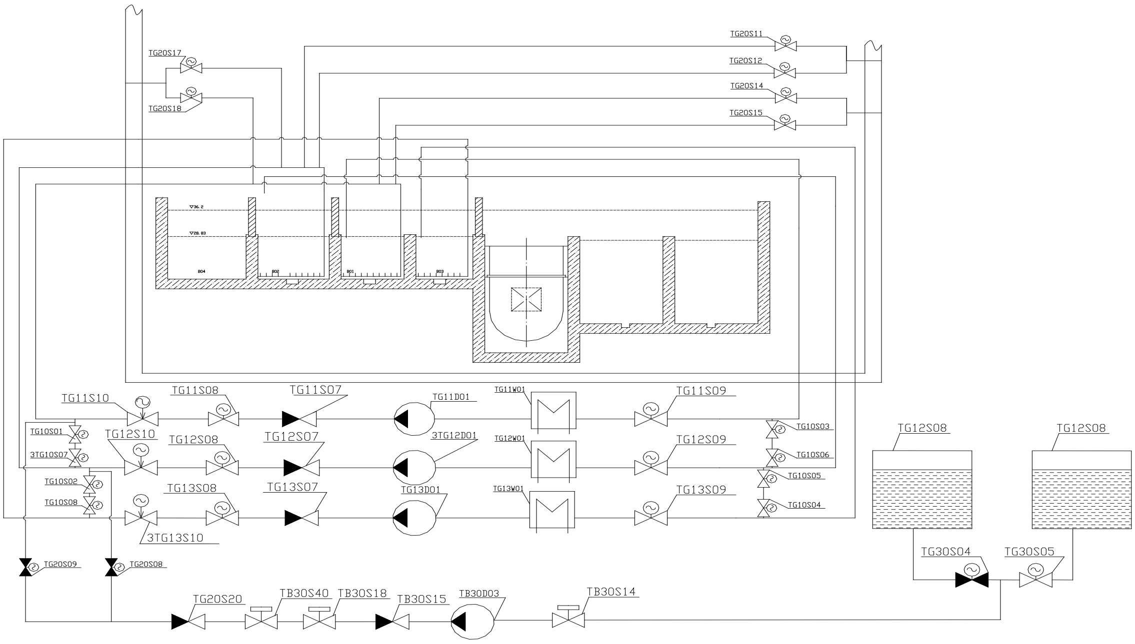


Figure 1 – Flowsheet for SFP systems of WWER-1000/320

During long-term storage of fuel, one pump cooling pond is sufficient to provide the required water temperature in SFP (50 ^{0}C). Three pumps are used in the design: one pumps is operating, the second is in standby and the third can be repaired. Similar considerations apply to heat exchangers. During an outage, all fuel can be unloaded to the SFP from the core. In this case, to provide the required water temperature in the cooling pool (70 ^{0}C) requires the operation of two pumps and two heat exchangers.

The system operates in emergency modes, except for break of primary or secondary piping inside the containment, leading to an increase in pressure under the containment and actuation of valves in the containment system. In the event of accidents involving primary or secondary piping inside the containment, the cooling water circulation ceases. Temperature of water in the SFP increases up to the boiling temperature.

To compensate for evaporation of water in the SFP, pool makeup from tank GA-201 is envisaged, for which the design provides for feeding from the spray pumps.

After pressure in the containment is decreased, valves of the containment system can be opened and regular SFP cooling is put into operation. When an initiating event with station blackout occurs, the operating pumps are disconnected and then, after diesel generators are actuated, all pumps that were in operation before the event are connected to the emergency power supply system.

3SPECIFICS OF PSA FOR SFP

3.1 PSA Level 1

In general, PSA Level 1 for SFP is based on similar methodology as PSA for the reactor core at nominal power and in shutdown states [4], [5]. Research results on development of a SFP PSA Level 1 model for the French EPR reactor is presented in [6].

3.1.1 Definition of the operational states and initiating events

3.1.1.1 SFP operational states

The current practice for dealing with the variety of plant states in PSA is to define a limited number of plant operational states (POS) during which the plant status and configuration are sufficiently stable and representative.

Three groups of physical and technical aspects are identified as important for the selection of the SFP operation states:

- parameters in SFP (boric acid concentration temperature and water level)
- availability and configuration of safety systems and SFP cooling systems
- initiating events that may occur in the operational state.

Operational states for SFP should be specified on the basis of actual operational experience and according to present practices and procedures. SFP operational modes are usually defined in the plant operating documentation and WWER design typically includes five modes: storage of spent fuel, refueling, complete reactor core unloading, emergency operation of SFP systems, switching in the SFP systems. Table 1 presents aspects for consideration in establishing the SFP operational states.

Table 1– Aspects important for definition of SFP operational states

Aspect	Description
Boric acid concentration	16-20 g/kg at all operational modes
Water temperature	< 50 °C in storage mode, < 70 °C in refueling
Water level	7.3 m in storage mode, 14.4 in refueling
Availability and configuration of safety systems and SFP cooling systems	Depends on the reactor and SFP operational modes (e.g. during refueling 1 train of the safety system may be in maintenance)

As a minimum, two SFP operational states should be selected for further consideration:

1) “Long term fuel storage”, situations arising during power or shutdown states with the core in the reactor vessel. One train of the SFP cooling system is required to be in operation.

2) “Refueling”, situations arising during shutdown for refueling, including activities on the SFP and system preparation for refueling and refueling itself. Two trains of the SFP cooling system is required to be in operation.

Operational experience and schedules of annual repairs are taken into account to evaluate the duration of SFP operational states. It is possible to perform evaluation based on average four-year cycle, or with a focus on the years with full unloading from the core and therefore with increased risk.

3.1.1.2 Initiating events

A systematic process should be used to identify the set of initiating events to be addressed in SFP PSA Level 1, as well as in PSA for the reactor core. As a starting point, a generic list can be compiled following analysis of full power and shutdown modes for the reactor core, taking into account its applicability to the SFP. SFP operational experience is a valuable source for specification of the list of initiating events. The list of potential initiating events is compiled for each of the identified SFP operational states.

Initiating events for SFP do not significantly depend on the operational state of the power unit (reactor operation at full/low power or in shutdown state). Nevertheless, there are peculiar initiating events for the refueling shutdown state, which differ from operation at full power. For example, the success criteria for SFP cooldown in case of emergency or complete regular core unloading differ from those for long-term fuel storage in operation at full power. On the other hand, shutdown modes may lead to new initiating events, such as drop of heavy loads or events caused by SFP overdraining. Table 2 shows a list of initiating events for consideration in SFP PSA.

Table 2 General list of initiating events for consideration in SFP PSA

#1: Loss of heat removal	**# 2: Leakages**
1.Equipment failures of SFP cooling system 2.Break of SFP cooling system piping 3.Loss of the essential service water system 4.Loss of power supply on in-house 6 kV buses 5.Total loss of power supply	1.Direct leakage of SFP (leakiness of welds bottom or walls, damage to drainage pipes) 2.Loss of water caused by human errors in maintenance and repair 3.Inadvertent SFP drainage due to failure of system of control water level in the SFP
# 3: Reduction of the boron concentration leading to a criticality accident	**#4: Heavy load drops**
1. Water with low boron concentration in the core and SFP when the reactor is opened after unloading 2. Inadvertent boron dilution due to ingress service water in the SFP cooling system for leak at heat exchanger 3. Uncontrolled flow of excess distillate or decrease in the concentration of boric acid in the ECCS tanks 4. Human error in conducting chemical analysis before loading of SFP cells 5. Decrease in concentration of homogeneous absorbers in SFP water	1.Drop of heavy loads to SFP 2.Drop of a hydraulic lock into SFP 3. Drop of a cover with fresh assemblies to SFP and falling of assemblies out of the cover 4. Drop of spent fuel assembly onto the core and end pieces of assemblies in SFP 5. Drop of a cask with spent fuel

To ensure a structured analysis and reduce the number of event trees to be developed, IEs may be grouped. IEs are grouped for each identified operational state based on the following criteria:

- identical set of safety functions required to mitigate accident consequences
- identical set of operable systems that may perform the required safety functions
- identical success criteria for all required systems.

3.1.2 Accident sequence modeling

3.1.2.1 Safety functions, success criteria and end states

Two safety functions identified for SFP are as follows:

- Reactivity control

The physico-chemical characteristics of water in the spent fuel pool should allow not exceeding the effective neutron multiplication factor 0.95.

- Residual heat removal

The residual heat removal from the fuel in SFP should be ensured.

The quantity and damage rate of fuel elements should not exceed the maximum design limit for damage of fuel claddings including the maximum temperature on the surface of fuel claddings should not exceed 1200 °C.

The water temperature in SFP and compartments has the following operational limits:

- no more 50 °C in storage of spent fuel
- no more than 70 °C in full core unloading to SFP
- 100 °C on the top emergency border.

The basic acceptability criterion with respect to damage of fuel in PSA used for the Ukrainian NPPs is fuel uncovering.

End states of emergency sequences depending on compliance with acceptability criteria are classified as OK (if the stable safe condition is reached) or FD (if there is fuel damage in SFP).

The Main Plant Logical Diagram provides linking between SFP safety functions and required front and support safety systems (Figure 2).

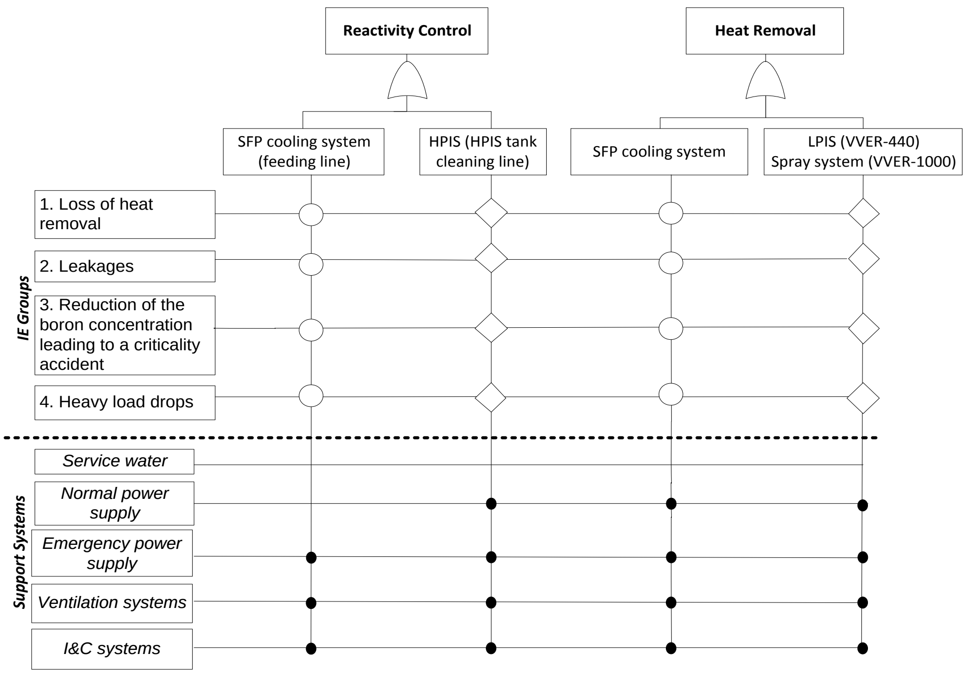


Figure 2 – Main Plant Logical Diagram

3.1.2.2 Human Reliability Analysis

Specifics of the human reliability analysis for different PSA studies are presented in Table 3.

Table 3 – Specific aspects of the HRA for PSA studies

PSA for reactor core at nominal power	PSA for reactor core at low power and shutdown states	PSA for SFP
Timing performances from 0.5 to several hours	Considerable time intervals. Timing performances reach tens of hours	Considerable time intervals. Timing performances reach from tens of hours to about several days
Rather the considerable quantity or simple alarm system, clear display	The alarm system and indication are absent or is insufficient	The alarm system and indication are attend
It is easier to carry out correct diagnostics of process	More possibilities for wrong diagnostics	It is easier to carry out correct diagnostics of process
Considerable quantity of actions of automatics on management of systems and equipment	Considerable quantity of actions for the management systems and equipment, actions carried out in place, and also regenerative actions	Considerable quantity of actions on remote control of systems and the equipment
Rather a small amount of personnel actions on performance approbation, maintenance service, repair and equipment adjustment	Significant amount of personnel actions on performance approbation, maintenance service, repair and equipment adjustment	Rather a small amount of personnel actions on performance approbation, maintenance service, repair and equipment adjustment

Considerable quantity of demanded safety functions (reactivity control, heat removal from reactor, pressure control of the primary coolant circuit etc.)	Considerable quantity of demanded safety functions (reactivity control, heat removal from reactor, pressure control of the primary coolant circuit etc.)	Required to perform only one safety function - residual heat removal

There are several positive aspects for HRA in SFP PSA:

1) A substantial (up to several days) margin of available time (less stress, more time to think about the situation, more opportunities to perform actions to restore the failed equipment or the use of alternative systems).

2) Reduced number of required safety functions (more possibilities to identify the emergency situation and make the right decision).

Taking into account the comparatively long duration of the accident at the SFP, an important issue is to the choice of an appropriate methodology to quantify the probability of human error in terms of diagnosis and decision-making.

The most accurate tool that takes into account the dependence of the available time to the probability of human errors is the TRC (g. Swain curves presented in NUREG \ CR-1278), however, the curves for the nominal rated probability of human errors have a time limit as 1000 min. This issue requires more detailed further study and are not discussed in this paper.

3.1.2.3 Analysis of specific IE groups

Loss of heat removal

IEs caused by failures in the SFP cooling system and support systems are considered in this group.

These IEs lead to failure in residual heat removal from the spent fuel stored in SFP and from the fuel placed in the pool during total inspection of the reactor coolant system (including emergency fuel unloading). In case of such IEs, residual heat is removed from the stored fuel through evaporation of SFP water. To prevent inadmissible decrease of the water level in fuel compartments and fuel uncovering, operating personnel should take actions to ensure the required water inventory in SFP using SFP makeup pumps or low-pressure injection system (spray system for WWER-1000).

To develop event trees (ETs), the following design and operating features of WWER SFP are taken into account:

1) Operating personnel have sufficient time to take actions to prevent fuel uncovering in SFP. The operator has several options for SFP emergency makeup.

2) When failure in the SFP operating pump is revealed and there is no possibility to recover water circulation in SFP, repair actions should be started immediately to recover at least one train of the SFP cooling system.

3) Failure of trains of the essential service water system affect the operability of LPIS pumps and primary makeup and boron control system, as well as the reliable power supply system or WWER-1000.

4) The capacity of storage batteries is sufficient to ensure emergency power supply to category 1 loads at least within one hour in loss-of-power conditions. If normal power supply is not restored within the discharge time of storage batteries but at least one standby diesel generator is actuated, normal power supply can be restored within any period of time.

For example, event tree for IE “loss of normal power supply” during refueling is shown in Figure 3.

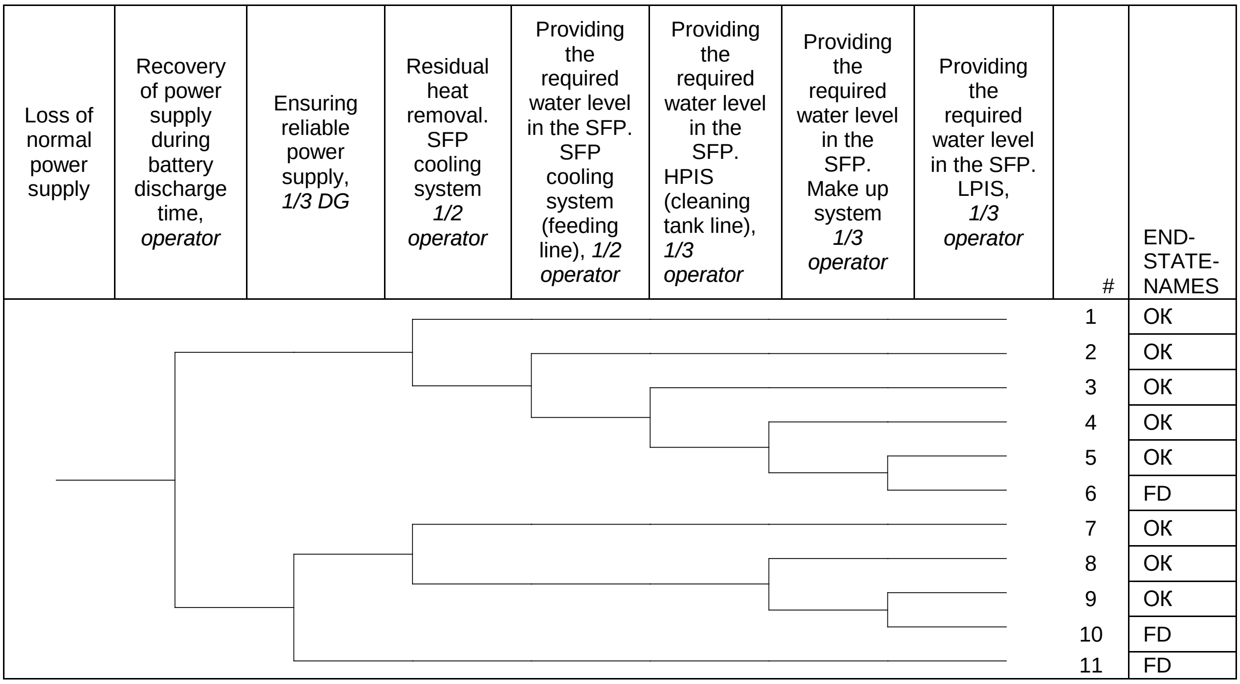


Figure 3 – Event tree for IE «Loss of normal power supply»

SFP leakages

These IEs lead to decrease in water level in SFP compartments and affect heat removal from spent fuel loaded into SFP.

The following design and operating features of WWER SFP are taken into account in ET development:

1) Integrity of the fuel pool is ensured by two-layer coating. There are drainage lines with valves passing from the coating to a special tank with a level annunciator providing display of information on MCR. This permits monitoring and measurement of SFP leaks. The water level in SFP is monitored with level meters that provide data to MCR.

2) The design provides for measures to prevent SFP dewatering during fuel storage including: use of double coating and piping for SFP filling, overflow and cooldown above the minimum allowable water level. Hydraulic locks are placed on the lines for coolant supply from ECCS pumps in SFP to prevent its emptying in break or leak of pressure piping. There are air vents in the upper part of the hydraulic locks to prevent reverse coolant flow because of discharge (siphon). These air vents are installed under the normal level in SFP so break of the pressure piping would not result in significant level decrease in SFP.

3) Operating personnel have sufficient time to take actions to prevent fuel uncovering in SFP. The operator has several options for SFP emergency makeup.

Reduction of the boron concentration

When boric acid concentration in SFP or reactor decreases below 16 g/kg, all refueling operations are terminated and the cause of this failure is eliminated or the required boric acid concentration is restored.

To develop ETs, the following design and operating features of WWER SFP are taken into account:

1) SFP subcriticality is not lower than 0.05 in pure water in all operating modes, which is due to the design of SFP racks. The boric acid credit is not taken into account in nuclear safety analysis.

2) During fuel long-term storage and reloading, boric acid concentration is continuously monitored in the reactor and SFP (at least twice per shift).

Therefore, boron concentration in water is not a determining safety factor for WWER-440 and WWER-1000 SFP and the group of reactivity accidents may be not modeled further.

Heavy load drops

Events caused by drop of spent or fresh fuel were quite frequent at NPPs. Possible causes are failures or damage in handling equipment or human error in reloading of spent fuel from the reactor to spent fuel pool or loading of fresh fuel into the reactor [7] - [9].

To develop ETs, the following design and operating features of WWER SFP are taken into account:

1) The hydraulic lock is removed and installed using a crane located in the SFP area adjacent to the transport corridor. The movement of loads above SFP is blocked by the crane design. This blocking is designed so that it cannot be terminated with a single, inadvertent personnel action (i.e. by human error). Nevertheless, drop of the hydraulic lock in SFP can lead to fuel damage, deformation and damage of the support plate of SFP racks, but required Keff for potential damage configurations is ensured. Drop of the hydraulic lock may lead to damage of SFP compartment (event tree should be developed for this case).

2) Design, engineering and administrative measures are provided to prevent criticality conditions and activity release during fuel reloading, storage and transportation.

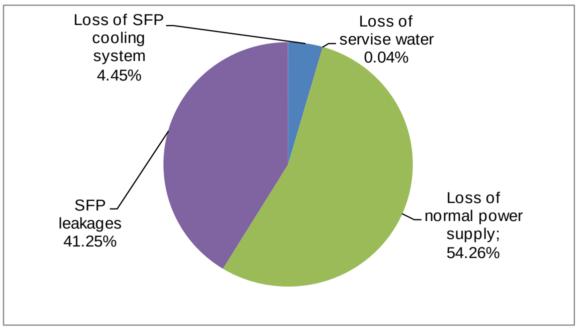
3) Drop of assemblies from the cover and their contact may lead to nuclear hazardous conditions. Nevertheless, the cover is transported in dry conditions and is unfastened on a platform. In addition, to prevent drop of assemblies from the cover, it is provided with a lid that is reliably fixed, so there is no actual possibility to reach criticality.

3.1.3 Summary of the results

According to the available calculations for operating WWER-400 and WWER-1000, fuel damage frequency with conservative criterion on fuel uncovering is estimated at the level of 10^{-6} 1/year.

The dominant event is “loss of heat removal” due to failure of normal power supply system. Practically equal risk distribution was assessed for operational states with long-term fuel storage and refueling.

The results of SFP PSA Level 1 for RNPP unit 1 [10] and KhNPP unit 2 [11] are shown in Figure 4.



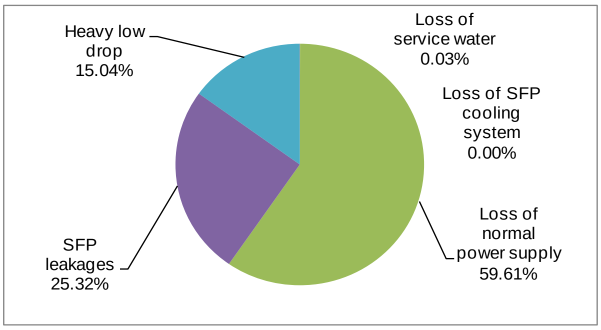


Figure 4 - IE contribution to FD frequency for RNPP-1 (left) and KhNPP-2 (right)

3.2 PSA Level 2

Level-2 PSA model, described in this chapter, was performed for Czech nuclear power plant Dukovany, which is pressurized water reactor WWER 440, Model V-213.

3.2.1 Interface Between Level 1 and Level 2 PSA

The main purpose of the interface between non-integrated Level 1 and 2 PSA models is to characterize the status of a plant at the onset of core damage by grouping fuel damage sequences into a limited number of categories for the further analysis. The concept of Plant Damage States (PDS) is used for grouping of those fuel damage sequences of the Level 1 PSA that cause similar progression of events following damage of the reactor core. The first step of a PDS analysis is identification of a representative set of PDS attributes, i.e. events and systems or features that have a potential to influence the progression of a severe accident. 9 PDS attributes were identified for Dukovany NPP Level 2 PSA, ref. [14]:

- Type of scenario (called “initiating event” in the interface)
- Leakage isolation
- Status of SFP normal cooling system (TG11,12)
- Status of power supply
- Location of emergency core cooling system (ECCS) water inventory
- Location of bubble tower (BT) water inventory
- Status of containment ventilation systems
- Status of reactor hall (RH) ventilation systems
- Containment and reactor hall isolation.

The second step of the PDS analysis is identification of a preliminary set of PDS vectors (= the unique combination of PDS attributes) based on the accident sequences from the Level 1 Dukovany PSA and assignment of probabilities to the defined PDS vectors. In case of Dukovany NPP, 17 different preliminary PDS vectors were defined.

The last step of PDS analysis is application of screening criteria described in [12] or [13] and definition of the final set of the PDS vectors. An example of the PDS vectors for Dukovany NPP Level 2 PSA is shown in the next table.

Table 4 – PDS vectors for Dukovany NPP Level 2 PSA

PDS	IE	Leakage Isolation	TG 11,12	Power supply	ECCS Inv.	BT Inv.	Cnt m Vnt	RH Vnt	RH Isol.	FDF
1O10	1O_FALL	VE_Isol_BS	SP_CS	EP	Tank	nDrain	Vnt	nVent_RH	Is/nIs	2,62 E-06
1O11	1O_FALL	VE_Isol_BS	SP_CS	EP	Tank	nDrain	Vnt	Vent_RH	Is/nIs	3,40 E-08
6O10	6O_FALL	VE_Isol_BS	SP_CS	EP	Tank	nDrain	Vnt	nVent_RH	Is/nIs	1,11 E-06
6O11	6O_FALL	VE_Isol_BS	SP_CS	EP	Tank	nDrain	Vnt	Vent_RH	Is/nIs	1,40 E-08
7C10	7C_TRANS	VE_Isol_BS	fSP_CS	nEP	Tank	nDrain	Vnt	nVent_RH	Is/nIs	6,10 E-08
7C20	7C_UNC_NIZ	fVE_Isol_BS	fSP_CS	EP	Unrecov	Drain	Vnt	nVent_RH	Is/nIs	1,80 E-08
7O10	7O_FALL	VE_Isol_BS	SP_CS	EP	Tank	nDrain	Vnt	nVent_RH	Is/nIs	2,40 E-07

As seen from the previous table (column “IE”), 3 different scenarios went through the screening process and subsequently entered the PSA Level 2 analysis. These scenarios are:

- xO_FALL = **Heavy load drops**,
- xC_TRANS = **Loss of SFP cooling system** (due to loss of power supply, loss of essential service water, etc.),
- xC_UNC_NIZ = **Spent fuel pool leakage**,

where x stands for the number of the plant operating mode.

3.2.2 Deterministic Accident Progression Analysis

Termohydraulic analysis [15], analysis of dynamic effects of heavy load drops [16], and severe accident analysis, [17], were performed for the above mentioned scenarios, which enter Level 2 PSA model. This chapter describes main outcomes of the analyses.

Heavy load drops

Heavy load drops (HLDs) scenarios are considered in three different plant operational modes – modes 1, 6 and 7. HLDs are modeled together for modes 1 and 6, since there are no major differences between these two cases (except for residual heat, where the more conservative case - 2,1 kW per 1 fuel rod - was considered). Fuel rods are stored in a stable storage rack only (one layer) in those modes.

The main differences between modes 1, 6 and 7 are the following: 1) The fuel rods are stored in two layers (the stable storage rack and the reserve storage rack) in mode 7 and 2) residual heat is slightly higher – 3,9 kW per 1 fuel rod - in mode 7, since mode 7 starts immediately after removing all fuel from the reactor into the SFP.

Deterministic analyses show that after a heavy load drop into the SFP, no more than seven fuel rods can be damaged in such a way, which prevent circulation of water around damaged fuel assemblies. Damage of fuel is very limited in modes 1 and 6. In total, 14.5 kg of radioactive products (=0.02% of total inventory) is released from the fuel, but most of them are confined in the SFP. In case of mode 7, release of radioactive products is similar to the previous case, but the progression of the severe accident is faster due to the higher residual heat.

Spent fuel pool leakage

Based on Level 1 PSA model assumptions, a leakage is considered at the SFP normal cooling system, specifically from the outlet line from the SFP, which would result in SFP drainage as low as 10 cm above the fuel rods in mode 7 (two layers of fuel in the SFP). The water from SFP would flow through the break into the room with SFP cooling pumps, which would cause the complete unavailability of the system. The leakage from the certain piping segments is non-isolable. There are two main options for the emergency SFP make up: from 1) ECCS tanks or from 2) bubble trays (in bubble tower). Anyway, after some time (approximately 7 days) all emergency water inventory would be depleted and water level in the SFP would start to decrease. From this point, the progression of the severe accident is very similar to the scenario “Loss of SFP cooling system”, so results of those deterministic analyses can be applied also to this case, see the next paragraph.

Loss of SFP cooling system

According to deterministic analyses, this is a type of a scenario with the most dynamic progression and the worst consequences. The initiating event (the most likely loss of power supply) is considered in the operational mode R (two layers of fuel in the SFP). Water saturation starts in the SFP approximately after 7 hours after the IE, fuel uncovery after 32 hours, larger releases of radioactive products after 40 hours and a melt-through of the bottom of SFP after 12 days following the start of the IE.

During the 12 days, 860 kg of radioactive products are released from the fuel into the reactor hall. Core concrete interaction starts approximately after 4 days, which causes rapid increasing of production of combustion gases (H_2 and CO). The analysis shows that 4700 kg of hydrogen and 6100 kg of carbon dioxide would be produced until the moment of the floor melt-through. Risk of hydrogen combustion is decreased by a huge volume of the reactor hall and by a high concentration of steam in reactor hall atmosphere, as illustrated in Shapiro diagram in the picture bellow.

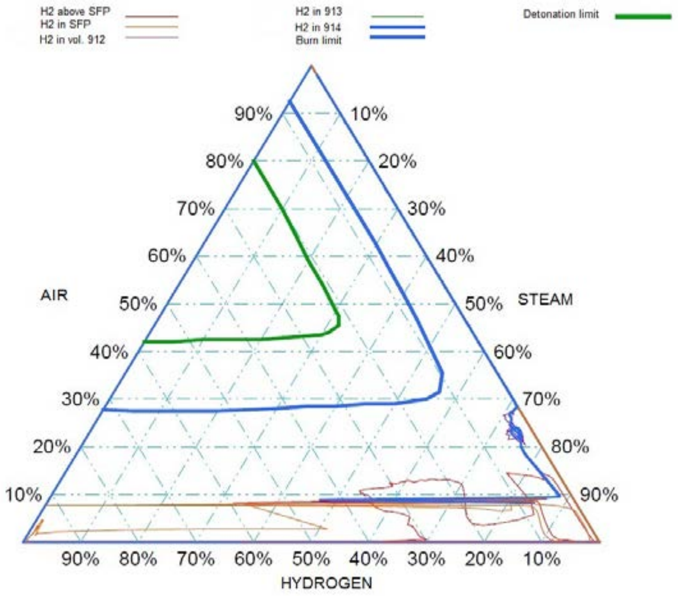


Figure 5 – Hydrogen flammability in the SFP and in the reactor hall

3.2.3 Accident Progression Event Trees

The accident progression event tree (APET) for Dukovany NPP [18] was performed in EVNTRE software and is described by one large event tree. This tree integrates also PDS vectors from PSA between level 1 and 2 interface in nodes (the first 12 questions) and radioactivity release calculation (the last 13 questions). The APET tree includes 39 questions in total. Note that the APET tree for SFP is quite simple compared with an APET for full power operation, which contains more than 100 questions.

The APET events (questions) may represent:

- PDS vectors and / or may be closely related to PDS
- system status (available…) or behavior (failure, repairing…)
- human actions
- physical phenomena (an example of an complex phenomenon is the hydrogen risk).

Four different containment/reactor hall failure modes were defined and used for final source term evaluation:

- early isolation failure - ISE
- late reactor building rupture – CFL_Rp (due to hydrogen risk)
- late melt-through – CFL_MT
- intact reactor building nCF (natural release)

where the term “early” is used for the time period ≤ 24 hours after the occurrence of the initiation event. Preliminary results of fuel damage frequencies for different failure modes and scenarios defined by Level 1 PSA are shown in the Table 5:

Table 5 – Fuel damage frequencies of reactor building failure modes for different scenarios

IU	CNTM Status	FDF [1/year]		FDF [%]
1O_FALL	ISE	2,65E-06	2,69E-08	1,0%
	CFL_Rp		3,82E-08	1,4%
	nCF		2,59E-06	97,5%
6O_FALL	ISE	1,12E-06	1,23E-07	10,9%
	CFL_Rp		1,38E-08	1,2%
	nCF		9,88E-07	87,9%

7C_TRANS	ISE	6,10E-08	6,65E-09	10,9%
	CFL_Rp		4,46E-08	73,1%
	CFL_MT		4,84E-09	7,9%
	nCF		4,94E-09	8,1%
7C_UNC_NIZ	ISE	1,80E-08	1,96E-09	10,9%
	CFL_Rp		1,44E-08	80,2%
	CFL_MT		1,59E-09	8,8%
	nCF		1,60E-11	0,1%
7O_FALL	ISE	2,40E-07	4,56E-08	19,0%
	CFL_Rp		1,94E-08	8,1%
	nCF		1,75E-07	72,9%
Total			**4,10E-06**	

The preliminary results in the previous table show relatively high fraction of intact reactor building (nCF). This mode does not mean that there is no release of radioactive products into environment, but it means that there is a natural release from the airtight zone. Calculations of a decontamination factor of the reactor hall and final magnitudes of the releases into environment are still in progress.

4 SFP AND REACTOR INTERACTIONS FOR WWER DESIGN

Accidents at the reactor and SFP, as well as simultaneous accidents at both places, should be considered to provide a complete picture of risk. The pilot results for Spent Fuel Pool Risk Assessment Integration Framework (Mark I and II BWRs) are given in [19].

Several general aspects of SFP and reactor interactions may be directly derived from the Fukushima lessons:

1) Risk of the common cause failure of the reactor core and SFP cooling systems (due to loss of the power supply, seismic or flooding impact).

2) Importance of SFP and reactor core shared systems and existing connections. This is confirmed by hydrogen explosion at unit 4. First it was believed that the explosion was caused by hydrogen generated by the spent fuel, in particular, by the recently unloaded core. Further analysis have proved that hydrogen causing the explosion was apparently generated in Unit 3 and transported through exhaust ducts that shared the same chimney with Unit 4.

3) Difference in the rate of the accident processes in SFP and reactor core (due to low decay heat of fuel assemblies, the processes in SFP are slow in comparison to the processes in the reactor core).

4) Complication of the operator response in case of the simultaneous accident in the reactor core and SFP.

Relating to WWER the specified interaction between the core and SFP is mainly typical for WWER-1000 designs. Because WWER-1000 design houses the spent fuel pool in the reactor building and SFP accidents can impact the reactor and vice versa. Since SFP of WWER-440 reactor is placed outside the containment interactions related to changes of the containment atmosphere does not occur as all.

First of all significant difference in accident progression rate in the core and SFP should be noted. This can be shown graphically to emphasize the significant difference in accident timing. Time of SBO progression for WWER-1000 [20] is presented at Figure 6.

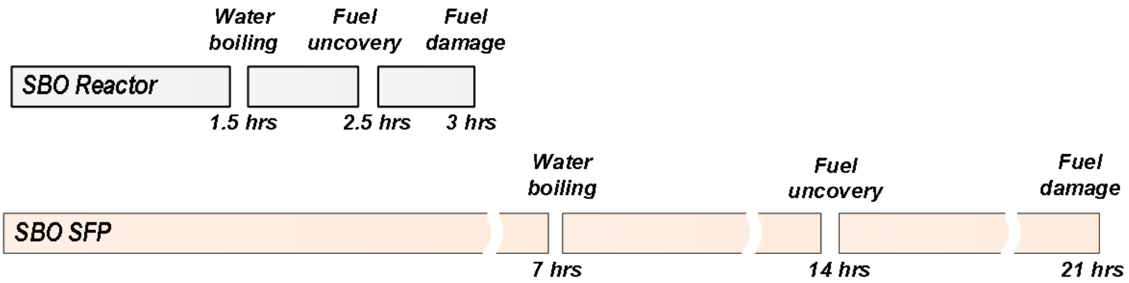


Figure 6 – Time of SBO progression for WWER-1000

Faster accident progression in the reactor determines predominant impact of the core processes during short-term accident phase. SFP effects appear at a later phase (in 7-10 hours).

Examples of mutual impact of the reactor and SFP for WWER-1000 are presented below:

1) Sharing the reactor and SFP systems:

- essential service water system (cooling water supply to heat exchangers of ECCS and SFP, ECCS pumps cooling)
- use of spay system for SFP emergency makeup (it is also necessary to consider SFP spraying in containment, according to rough estimate based on SFP area and flow-rate ratio spraying is sufficient to compensate water evaporation from SFP and level maintenance over the top part of the fuel assembly)
- SFP emergency makeup using water from the sump tank (GA-201) of ECCS systems
- common support systems (power supply, ventilation).

2) Impact of accidents in the reactor on SFP:

- in case of LOCA accidents in the reactor pressure in the containment will reach the safety operational limits and isolating valves will be closed. Isolating valves at TG piping will be also closed that will result in failure of SFP cooldown system. It is necessary to consider post-accident operator's actions on re-opening of pneumatic valves.

3) Impact of accidents in SFP on the reactor:

- it is necessary to consider a possibility of reactor system equipment failure caused by floods resulting from SFP system piping rupture (sources of floods from SFP are usually considered under Flooding PSA for reactor);
- accidents in SFP may lead to conditions which require emergency unit shutdown by the personnel (should be taken into account during frequency calculation of events which may lead to scram).

4) Impact of simultaneous accidents:

- the most important factor is to consider generation of additional quantity of hydrogen in SFP. According to the results of performed assessments within stress-tests [21], mass of hydrogen generated in SFP may be comparable to mass of hydrogen in the core (intensive generation of hydrogen starts in several days after accident beginning).

5) Consideration of accident management for the reactor and SFP:

- during implementation of actions on accident management, it is necessary to consider mutual impact of these actions on the reactor and SFP. For example, to arrange mode of the containment venting system it is necessary to take into account that pressure decrease may lead to more intensive water boiling in SFP and to significant loss of it which reduces time margin to the beginning of uncovering and heating of spent nuclear fuel.

5 CONCLUSIONS

PSA for SFP has certain specific features related to the identification of initiating events, determination of SFP operational states, human reliability analysis, interface between Level 1 and Level 2 PSA, deterministic assessments and probabilistic modeling.

Four main groups of internal IEs are identified for SFP PSA: loss of SFP heat removal, SFP leakages, criticality accident and heavy load drops. SFP parameters such as boric acid concentration, temperature and water level, availability and configuration of safety systems and SFP cooling systems, initiating events that may occur are the key factors for defining the SFP operational states. Human reliability analysis should take into account the positive factor associated with substantial time to take appropriate actions. Assessment of potential types of containment/reactor hall failures should consider early isolation failure, late reactor building failure (due to hydrogen risk), late melt-through and intact reactor building.

According to availiable evaluations, the fuel damage frequency for operating WWER-400 and WWER-1000 is estimated at the level of 10^{-6} 1/year. This evaluation do not take into account the safety upgrades on SFP makeup in case of station black-out from portable diesel-driven pumps or from additional protected stationary equipment under implementation as one of the lessons learnt from the Fukushima accident.

Regarding reactor building for SFP of WWER-440, upon the preliminary results of SFP PSA Level 2 the most dominant mode is intact reactor building This mode does not mean that there is no release of radioactive products into environment, but it means that there are releases from the airtight zone through untightnesses. Magnitude of release depends on a scenario. Risk of hydrogen combustion is decreased by a huge volume of the reactor hall and by a high concentration of steam in reactor hall atmosphere.

This paper also presents several examples of mutual impact of the reactor and SFP for WWER-1000 as the first step to understand the problem of SFP and reactor interactions to be further analyzed and carefully considered in probabilistic and deterministic models of PSA.

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Acronyms:

APET… Accident progression event tree

BT… Bubble tower

CFL… Late reactor building rupture

CFL_MT… Late melt-through

DG… Diesel generator

ECCS… Emergency core cooling system

EHRS…Emergency heat removal system

ET… Event tree

ETSON… European Technical Safety Organisations Network

FD…Fuel damage

HLD… Heavy load drops

HRA…Human reliability analysis

IAEA… International Atomic Energy Agency

IE… Initiating event

ISE… Early isolation failure

KhNPP… Khmelnitskiy nuclear power plant

LOCA… Loss of coolant accident

MCR… Main control room

nCF… Intact reactor building

NPP… Nuclear power plant

NUREG…US Nuclear Regulatory Guide

PDS… Plant damage state

POS… Plant operational state

PSA…Probabilistic safety assessment

RH… Reactor hall

RNPP…Rivne nuclear power plant

SBO… Station black out

SEC NRS… Scientific and Engineering Centre for Nuclear and Radiation Safety

SFP… Spent fuel pool

SSTC NRS… State Scientific and Technical Centre for Nuclear and Radiation Safety

TG… Turbine generator
TRC… Time-reliability correlation method
UJV…Ustav jaderneho vyzkumu (=Nuclear research institute)
WWER… Water-Water Power Reactor