OUTCOMES OF AN INTERNATIONAL INITIATIVE FOR HARMONIZATION OF LOW POWER AND SHUTDOWN PROBABILISTIC SAFETY ASSESSMENT

by

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Many probabilistic safety assessment studies completed to the date have demonstrated that the risk dealing with low power and shutdown operation of nuclear power plants is often comparable with the risk of at-power operation, and the main contributors to the low power and shutdown risk often deal with human factors. Since the beginning of the nuclear power generation, human performance has been a very important factor in all phases of the plant lifecycle: design, commissioning, operation, maintenance, surveillance, modification, decommissioning and dismantling. The importance of this aspect has been confirmed by recent operating experience.
This paper provides the insights and conclusions of a workshop organized in 2007 by the IAEA and the Joint Research Centre of the European Commission, on Harmonization of low power and shutdown probabilistic safety assessment for WWER nuclear power plants. The major objective of the workshop was to provide a comparison of the approaches and the results of human reliability analyses and gain insights in the enhanced handling of human factors.

Key words: probabilistic safety assessment, low power and shutdown, human reliability analyses, WWER, human factors

BACKGROUND

Probabilistic safety assessment is a powerful technique that allows assessing the risks implied by the operation of complex industrial facilities, and it is fully applicable to nuclear power plants (NPP). Many utility organizations have carried out detailed probabilistic safety assessment (PSA) studies to assess core damage frequency (CDF) and radioactive release frequency from their plants considering various hazard sources[**] and plant operational modes. The latter typically includes the operation with nominal power and low power and the shutdown operational mode.

Many PSA studies have demonstrated that for WWER the risk dealing with low power and shutdown (LPSD) operation is often comparable with the risk of at-power operation or may exceed it. Such factors as taking multiple components out of service for maintenance, and changes in the configuration of plant compartments dealing with refueling and maintenance activities may cause potential challenges to safety features. It has also been demonstrated that for the risk dealing with LPSD operation the main contributors are often related to human factors, what is a logical consequence of the fact that the human element intervenes more in the related organizational processes during the low power and shutdown operation in comparison with the full power operation.

By means of PSA, weaknesses in human performance and human factors can be identified and, sometimes, the appropriate corrective actions can be taken with the aim of a further enhancement of nuclear safety.

Possible applications of the results of LPSD PSA include:

– determination of the actual safety level during shutdown states,

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** The hazards considered typically are: internal initiating events caused by random component failures and human errors, internal events such as fires, floods, turbine missiles initiated inside the plant, and external events, both natural and man-made.

– risk-informed improvements of operating procedures relating to low power and shutdown states,
– better outage planning and balancing between effectiveness and safety,
– justification for moving some tests and preventive maintenance activities from the outage to the full power operation,
– risk monitoring for outage, and
– radiation protection improvement.

In 2007, the IAEA launched the Regional technical co-operation (TC) Project RER9087 'Harmonization of PSA & PSA Applications'. The overall objectives of the project were:
– to enhance and maintain high level of nuclear safety in member states operating WWER reactors by strengthening their capabilities for conducting and applying PSA, and
– to harmonize and support the development of PSA and its uses by delivering state-of-the-art technology, comparing and improving the analyses carried out and creating infrastructures for co-operative efforts between relevant organizations in member states.

The ultimate goal of PSA harmonization effort has been to identify which differences in relation to risk insights are driven by differences in design features and operational practices, and which ones are due to differences in data and modeling approaches. For the latter, the role of the best practices should be highlighted.

In the framework of the above project, the IAEA organized, in co-operation with the Institute for Energy of the Joint Research Center of the European Commission (JRC-IE), a workshop on "Harmonization of Low power and shutdown probabilistic safety assessment for WWER NPP". The workshop was held on October 29 – November 2, 2007, at the headquarters of the Nuclear Regulatory Authority of the Slovak Republic in Bratislava, Slovak Republic. This workshop was the second one in the series of workshops conducted on harmonization of LPSD PSA for WWER NPP in 2007; one of the key topics was the analysis of the impact of human factors on NPP safety [1]. Specifically, the workshop was aimed at continuing the work on harmonization of PSA for WWER-type NPP for LPSD states that was started at the first workshop held at the same place in March 2007 [2]).

The focus of the workshop was placed on the detailed comparison of the approaches and data used for modeling several important initiators that, on the basis of the results of the first workshop, appeared to be ranked differently in different PSA in terms of risk significance. The aim was also to provide a more detailed comparison of the results of human reliability analysis and gain insights in dealing with the human factors.

Twenty-two participants from seven countries (Armenia, Bulgaria, the Czech Republic, Hungary, the Russian Federation, Slovakia, and Ukraine) attended the workshop.

APPROACH FOR PSA COMPARISON

A questionnaire was developed in advance and sent to the participants a few weeks before the workshop. The questionnaire was aimed at collecting detailed information on initiating event frequencies, human errors, and modeling details for six selected initiators for WWER-440 and WWER-1000 plants. The selected initiators represented the major contributors to the CDF in PSA studies for respective WWER plant units. These had been identified at the first workshop [2]. The original responses can be found in Annex III of ref. [1]. The information in questionnaires was extensively used during the PSA comparison activities in working groups.

Two working groups (WG) were established:
– WG#1: "Comparison and Harmonization of LPSD PSA for WWER-440 NPP", and
– WG#2: "Comparison and Harmonization of LPSD PSA for WWER-1000 NPP".

For both groups of WWER NPP the following information was collected for the six initiators recognized at the previous workshop to be the major source of differences in the risk profiles:
– contribution to the total CDF for the initiator from different plant operational states (POS) [1, 2],
– twenty top minimal cut sets (MCS)*, and
– human errors modeled.

The following main analysis areas were covered in the discussions carried out in the WG participating in harmonization:
(1) comparison of the results of analysis of selected initiators from LPSD PSA for WWER-440 NPP,
(2) comparison of the results of analysis of selected initiators from LPSD PSA for WWER-1000 NPP,
(3) exploring the feasibility of consolidation of data for selected initiators and providing generic data for WWER plants for LPSD modes **, and
(4) drawing insights on the results and specific features of human reliability analysis (HRA) for PSA for LPSD states.

While doing the comparison exercise, the actual design differences were analyzed and taken into account, as well as the fact that the original designs were not identical.

SPECIFIC FEATURES OF HRA FOR LPSD PSA

The HRA done within a PSA for LPSD states has some specific features that may be different from the

* A minimal cut set is a combination of an initiating event and component failures and /or human errors that could lead to undesirable consequences (*e. g.* core damage). It means that: (1) the given combination of events would cause core damage, and (2) if any event is selected and eliminated from the minimum cut set, the remaining subset of events does not cause core damage anymore. Each MCS has a frequency assessed by PSA technique.

** This topic is presented in the ref. [1].

features of HRA performed within an at-power PSA. These new features may have both positive and negative impact on the quality of operators' work. These include:

– different spectrum of time windows available for operators to mitigate consequences (generally, low power and shutdown time windows are longer),
– the level of detail of the procedures used during shutdown, which is less in comparison with the procedures used for full power operation,
– more requirements for manual activation of plant equipment in response to the initiating event due to unavailability of some emergency interlocks,
– local detection of initiating events is available in many cases, and
– potentially very strong interaction between a human-induced initiator and the subsequent operator response (positive or negative dependencies).

OVERVIEW OF HRA QUANTIFICATION APPROACHES

During the workshop, the emphasis was put on the comparison of the methodologies used for HRA in the LPSD PSA for WWER-440 NPP, in particular regarding the quantification of the probability of occurrence of human failure events. An overview is presented in tab. 1.

Additional information concerning the table

(1) Specific methodology developed on the bases of THERP taking into consideration performance shaping factors that are most relevant for the wrong valve positioning and the measurement of miscalibration failure.
(2) Significantly initiating event specific, very detailed (and original) analysis made for some initiating events.
(3) Similar methodology to the full power case, some specific LPSD features needed to be addressed (long time windows, absence of signals, absence of procedures).
(4) The same contractor as for Bohunice V-1 plant.
(5) Version modified by J.Vaurio (for further reading [10, 11]).
(6) Specific version of decision tree set developed by VEIKI, Budapest.
(7) Specific version of decision tree set developed by VEIKI, Budapest, taking into consideration data records from simulator exercises.
(8) No LPSD was finished at the time of the workshop; the method was used in the full power PSA, but the plant foresaw to use the same method for LPSD PSA.

Table 1. Methodologies used for analysis and quantification of human error probability (HEP) for the individual categories of human actions

Plant	Type of human failure event		
	Pre-accident	Initiator	Post-accident
Armenian 2	THERP* (8) [3, 4]	... (9)	HCR* [5], ASEP* (8)
Bohunice V-1	THERP	THERP, ASEP	TRC*, THERP
Bohunice V-2	THERP (4)	THERP, ASEP (4)	TRC, THERP (4)
Dukovany	THERP (1)	THERP, CREAM* [6], HEART* [7], Decision trees (2)	Decision trees + ASEP (3)
Mochovce	THERP, ASEP [8]	THERP, ASEP	SLIM [9]
Paks	ASEP (5)	Decision trees (6)	Decision trees (7)
Kozloduy NPP Units 3,4	THERP	Information not available	HCR

* THERP – Technique for human error rate prediction; HCR – Human cognitive reliability; HEART – Human error assessment and error reduction technique; CREAM – cognitive reliability and error analysis method; TRC – time reliability curves; ASEP – accident sequence evaluation program; SLIM – success likelihood index methodology

(9) Since the information has been taken from the full power PSA, no human induced initiators were identified and, therefore, no specific HRA method was deemed necessary for this part of HRA.

The following conclusions can be made based on tab. 1:

– a broad spectrum of HRA quantification methods was used by the individual teams (seven methods in total, considering TRC and HCR to be the same),
– the "old" THERP method is still very popular, it is used far most frequently; in addition, the second most popular method ASEP can be seen as a version of THERP, too,
– the THERP/ASEP pair is the most used HRA methods for analysis of pre-accident human errors,
– for the most important category of post-accident human errors, there is a big variability in the methods used, and
– TRC (HCR) is the most popular method for post-accident human errors analysis.

IMPACT OF SYMPTOM-BASED PROCEDURES

Nowadays, symptom-based procedures commonly provide support for most of operators' activities carried out in response to initiating event occurrence, not only during the full power operation, but in some limited manner in low power and shutdown mode. The general aim of planning plant strategy of response to initiating event occurrence (which is still being an interesting subject for objections and long discussions) is to remove cognition-based activities from the profile of control room crew actions performed under the accident circumstances, the typical attributes of which are high stress and dynamics. The effects of implementation of symptom-based procedures for LPSD operation in terms of CDF value have been estimated for different plants. On the basis of PSA analysis, a remarkable decrease of CDF for all NPP for which LPSD-PSA and symptom-based procedures for LPSD were available has been noted.

For two NPP among the participants of the harmonization project, the CDF value "after" includes not only installation of new symptom based procedures, but some minor modifications, as well. Still, symptom based procedures do represent a dominant contributor to CDF decreasing. For two NPP a big impact on CDF value has been partly caused by the effect of extending the scope of procedures (feed and bleed added).

The values of the CDF decrease differ significantly. The difference is such that it may encourage further work explaining it, *i. e*., an analysis defining more deeply the individual, elementary risk decreasing effects of symptom based procedures existence. For now, it can be pointed out that even the risk drop of 30% value is a very good achievement – it means that almost as much as one third of the plant operation risk was addressed just with changes in procedures, without any costly requirements to plant design and hardware changes.

EXAMPLE OF COMPARISON OF THE MODELING OF HUMAN-INDUCED LOSS OF COOLANT ACCIDENT IN LPSD PSA FOR WWER-440 NPP

This section illustrates the application of the approach for comparison of selected initiators across the WWER NPP. Information on the analysis process and the results of the comparison of modelling approaches and the results for the initiating event "Human Induced LOCA" in WWER-440 NPP is provided.

The event "Human Induced LOCA" has been chosen as an example for the illustration of the comparison process because it has been shown to be one of the most significant contributors to the risk for LPSD states for WWER-440 plants. The analysis results for other initiating events can be found in ref. [1].

General description of the event

The loss of reactor coolant inventory can cause the loss of residual heat removal and at the same time reduce the time available for operator recovery. During the plant evaluations when the operators are lowering the level, human mistakes could lead to a reduced inventory condition. Both plant procedures and operational experience should be reviewed to identify these initiating events.

Overview of the risk impact of the event

The CDF, conditional core damage probability (CCDP) having human-induced loss of coolant accident (LOCA) as an initiating event (IE), and IE frequency presented respectively the following margins, for the considered NPP:

CDF: (2.25E-06, 1.51E-05)

CCDP (3.59E-05, 2.02E-03)

IE frequency (4.57E-03, 6.72E-02)

It can be seen that the estimations of IE frequency varies by approximately one order of magnitude, *i. e*., in a similar way as the final CDF, whereas CCDP varies even by two orders of magnitude. This great deal of variation is typical for the results obtained by means of different HRA methods for the analysis of the same problem. The high variance of CCDP values may be caused by application of different approaches to evaluation of dependency between human errors causing an initiating event (inappropriate actions causing loss of primary circuit integrity) and human errors in response to the initiating event occurrence (the identification and localization of coolant loss, isolation or interruption of the loss, *etc*.)

Estimation of IE frequency

In one NPP the quantification of human-induced LOCA events was based on the HRA. It has been found that all identified scenarios are LOCA that can be isolated by re-closing the erroneously opened valve. The valves, motor operated valves (MOV) or manual valves, located on the boundary of the reactor coolant system (RCS), which are closed and locked to prevent the boron dilution in the operating mode 6 (ref 1), are not considered to be erroneously open due to the involved measures and their independent checking. The man induced LOCA events are considered only for the selected plant operating regimes [1].

In another plant HRA quantification methods technique for human error rate prediction, decision trees (THERP, DT) were used for small LOCA, and statistics and direct numerical estimation based on the plant specific experience (precursors) for a very small LOCA.

In a third NPP, during the estimation of the initiating event frequencies, an analysis has been made in each plant operational state separately. The identified effects include (1) internal causes and similar, (2) erroneous change of the operational loop, (3) erroneous draining of the operational loop, (4) maintenance activities, (5) erroneous system alignment, (6) heavy load drop, and (7) other erroneous interactions. The effects (2-5) and (7) represent the potential direct human origin of the initiating events. The human contribution to the heavy load drop is also possible but, in general, this initiator is not considered as a human induced one.

Dominant human errors

In more details, the dominant human errors identified in different PSA proposed in the workshop included the following sample of specific human errors:
– operator fails to initiate boron injection,
– operator fails to initiate component cooling,
– operator fails to initiate make-up of the refueling cavity,
– operator fails to compensate losses,
– no RCS make-up initiated by the operator,
– operator fails to isolate the man induced LOCA,
– operator fails to vent the primary side of steam generator and reactor vessel,
– operator fails to identify the loss of natural circulation,
– plant staff fails to isolate the leakage following the loss of cooling/natural circulation,
– plant staff fails to isolate the leakage before the loss of cooling/natural circulation (short time window),
– operator fails to identify the leakage before the loss of cooling/natural circulation (medium time window),
– operator fails to provide long-term make-up of the refuelling pool,
– execution error for water-water cooling of the reactor,
– execution error for start-up of RCS filling pumps (PZR),
– cognitive error for start-up of low pressure injection (LPI) system,
– operator fails to identify a man-induced LOCA,
– operator fails to align water supply into the reactor via hydroaccumulator (HA) trains by the gravity spilling of suppression pool trays or by the pumps with suction from low pressure emergency core cooling system tanks,
– system train unavailability due to a latent human error introduced during the performed activities, and
– small LOCA due to maintenance activities or erroneous system alignment in specific plant operation states.

Insights derived from the considered case

It was noted that one of the considered NPP has reliable safety systems to mitigate the accident. The reason is in the limiting conditions for operation (LCO). The NPP was, at the time of the workshop, the only plant where the LCO require the availability of safety trains to the maximum possible extent, which means that in the specific operating modes the availability of two safety trains is required. The third train may be in preventive maintenance. In the other plants, only a single safety train is required to be available in these operating modes. Two trains can be in preventive maintenance and the estimated probability that two trains be simultaneously in maintenance is high (the 3rd train is postulated to be in maintenance for the whole duration of POS).

If a plant reveals a significant contribution of unavailability of safety systems due to planned maintenance, it is recommended to control the plant configuration during maintenance activities, *e. g.* using risk monitor. It is also recommended to consider as good practice the reported approach which requires the availability of safety trains to the maximum possible extent.

A detailed consideration has been given in many PSA for modeling various operator actions. Attention should be paid mainly to the description of accident scenarios involving operator actions and to dependencies between human errors.

CONCLUSIONS AND RECOMMENDATIONS

The ultimate goal of the PSA harmonization effort is to identify which differences in relation to risk insights are driven by differences in design features and operational practices, and which ones are due to differences in data and modeling approaches; for the latter the best practices should be highlighted.

The effort on LPSD PSA harmonization will be especially useful for the plants that are planning or being currently in the process of developing a LPSD PSA. Other countries can effectively use the outcome of the IAEA/EC workshops to adjust their analyses within a refinement programme or a next cycle of PSA update.

The following insights and recommendations from the workshop are given:

(1) PSA scope

Not all WWER plants cover a full scope PSA. Considering the increasingly wide use of PSA in different applications, a full-scope PSA (*i. e.* comprising the risk assessment for various hazards and plant operational modes) should be encouraged and emphasized.

(2) Status of PSA review

LPSD PSA for WWER reactors are lacking independent peer and regulatory review, which is seen to be an important aspect of robustness of PSA. The reviews performed by the IAEA were mostly done for at-power PSA. This issue deserves further attention and efforts on pursuing PSA review.

(3) Status of harmonization in treatment of specific initiators

The comparison of specific initiators for the same type of NPP has showed that for some of the analysed initiating events (*e. g*. human-induced LOCA, loss of non-essential service water for WWER-440), the differences in the assessed risk impact of the events can be explained by certain differences in operating practices or design, while for other events (*e. g*. reactivity accidents, heavy load drops), the differences are driven by the differences in the used approaches and data. The latter require a more detailed further analysis. Specific conclusions on the status of harmonization in the treatment of specific initiating events can be found in refs. [1] and [2].

(4) Human reliability analysis

One of the issues causing a high contribution of LPSD states in the overall risk profile is the human factor. Based on the comparison of HRA from the PSA included in the scope of the workshop, a general observation can be made that the human error probability (HEP) value ranges in the LPSD studies are driven mainly by the differences among HRA approaches (different teams carrying out HRA for different plants) much more than by the differences amongst the individual accident scenarios and plant features. It is suggested to conduct a dedicated co-ordinated research project, a benchmark exercise or a series of focused workshops to promote harmonization of HRA in the PSA for WWER plants.

One of the insights is dealing with the improvement of normal and emergency procedures for LPSD states that may significantly contribute to risk decrease. A specific issue that deserves further harmonization effort is the modeling of the impact of symptom-based procedures that have been implemented at several NPP. The comparison exercise showed that a significant variation existed in this area.

(5) Other recommendations for further research

The contribution of LPSD states in the overall risk profile (that includes various hazards and operational modes) appeared to be significant in all PSA studies for WWER plants. The comparison of PSA performed within the workshop has revealed important elements of LPSD PSA that being treated differently in the methodological sense sometimes causes significant differences in the distribution of the risk contributors. In the two workshops, the comparison of LPSD initiators involved from 20 to 50 percent of the calculated specific core damage frequencies. To achieve a better coverage of the risk profile, more meetings of similar nature could be recommended. The same harmonization effort would be also useful for internal floods and internal fires PSA for WWER plants.

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ACRONYMS

ASEP	– accident sequence evaluation program
CCDP	– conditional core damage probability
CDF	– core damage frequency
CREAM	– cognitive reliability and error analysis method
DT	– decision trees
HA	– hydroaccumulator
HCR	– human cognitive reliability
HEART	– human error assessment and error reduction technique
HEP	– human error probability
HRA	– human reliability analysis
IE	– initiating event
JPC-IE	– Joint Researcs Centre of the European Commission
LCO	– limiting condition for operation
LOCA	– loss of coolant accident
LPI	– low pressure injection
LPSD	– low power and shutdown
MCS	– minimal cut sets
MOV	– motor operated valves
NPP	– nuclear power plant
POS	– plant operational states
PSA	– probabilistic safety assesment
PZR	– RCS filling pump
RCS	– reactor coolant system
SLIM	– success likelihood index methodology
TC	– technical co-operation
THERP	– technique for human error rate prediction
TRC	– time reliability curves
WG	– working group

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РЕЗУЛТАТИ МЕЂУНАРОДНЕ ИНИЦИЈАТИВЕ ЗА УСКЛАЂИВАЊЕ ПРОБАБИЛИСТИЧКЕ ПРОЦЕНЕ СИГУРНОСТИ РАДА НУКЛЕАРНЕ ЕЛЕКТРАНЕ ПРИ НИСКОЈ СНАЗИ И ИСКЉУЧЕЊУ

Многе студије засноване на пробабилистичкој процени сигурности, до сада окончане, показале су да је ризик управљања на ниској снази и искључењу често упоредив са ризиком управљања на пуној снази и да људски фактори често највише доприносе овом ризику. Од почетка рада нуклеарних електрана, хумани допринос био је веома значајан чинилац у свим фазама животног циклуса електране: пројектовању, укључавању у рад, управљању, одржавању, надзору, прилагођавању, декомисији и уклањању. Новије искуство у управљању електраном такође потврђује значај људског фактора.

Овај рад указује на садржај и закључке семинара о Усклађивању пробабилистичке процене сигурности рада ВВЕР електрана при ниској снази и искључењу, организованог 2007. године од стране Међународне агенције за атомску енергију и Центра здруженог истраживања Европске комисије. Главни циљ семинара био је да упореди приступе и резултате анализа људске поузданости и да пружи увиде о побољшаном управљању хуманим чиниоцима.

Кључне речи: пробабилистичка процена сигурности, ниска снага и искључење, анализа људске поузданости, ВВЕР, људски фактор