Proceedings of the Korean Nuclear Society Spring Meeting
Kwangju, Korea, May 1997

Human Reliability Analysis in Wolsong 2/3/4 Nuclear Power Plants Probabilistic Safety Assessment

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Abstract

The Level 1 probabilistic safety assessment(PSA) for Wolsong(WS) 2/3/4 nuclear power plant(NPPs) in design stage is performed using the methodologies being equivalent to PWR PSA. Accident sequence evaluation program(ASEP) human reliability analysis(HRA) procedure and technique for human error rate prediction(THERP) are used in HRA of WS 2/3/4 NPPs PSA. The purpose of this paper is to introduce the procedure and methodology of HRA in WS 2/3/4 NPPs PSA. Also, this paper describes the interim results of importance analysis for human actions modeled in WS 2/3/4 PSA and the findings and recommendations of administrative control of secondary control area from the view of human factors.

1. Introduction

Operating experience of nuclear power plants (NPPs), after TMI and Chernobyl accidents, shows that human factors play a significant role on the safety of NPP. Also, since the publication of WASH-1400, human factors are identified as dominant contributors to the results of Level 1 Probabilistic Safety Assessment(PSA)[1]. As too much conservative or optimistic approach in human reliability analysis(HRA) can lead to wrong insights being drawn from the PSA, the proper handling of human actions is very important aspects of any PSA. The objectives of HRA in the of Wolsong(WS) 2/3/4 NPPs PSA are to identify human actions to be modeled in the PSA logic structure and to quantify them. Also, the objective of HRA is to provide information for abnormal operating procedure(AOM) writers in accordance with CANDU's design practice.

Human error can be reduced through decreasing the chance of vulnerability to it and the human error probability(HEP)[2,3]. The former can be obtained by the introduction of system automation and the redundancy of system design or the use of safer substitute material. The latter can be obtained by the improvements of procedures, and man-machine or man-man interfaces or the enforcement of operator training. Another kind of an approach to reduce human error is the hardware and software improvements of the systems and environments which can adversely affect human performance.

The purpose of this paper is to describe the interim HRA results of internal Level 1 PSA of WS 2/3/4 NPPs, mainly focusing on the procedure and methodology of HRA, the importance analysis results, and the insights for future administration procedure preparation of WS 2/3/4 NPPs .

2. Overview of HRA in WS 2/3/4 NPPs PSA

The HRA of WS 2/3/4 NPPs PSA has some limitations since PSA is performed during the design stage. These limitations mainly come from the differences between WS 1 NPP EOPs and WS 2 NPP AOMs[4]. The EOPs of WS 1/2/3/4 NPPs would follow the present AOMs of WS 2 NPP.

The overall of an HRA procedure for WS 2/3/4 NPPs PSA is based on the Systematic Human Action Reliability Procedure (SHARP) approach[5]. The first task is to give a guidance to event/fault tree analyst and to compile a list of the human actions that are modeled in fault and event trees. After then, human reliability analysts identify the functions of human actions in event and fault tree. The second task is to perform an preliminary quantification to determine which human actions are significant to system unavailability or core damage frequency. The dependencies between multiple human actions and the recovery actions are also identified through the preliminary sequence quantification. The third task is to perform an in-depth analysis of those human actions that are known to be important. In this study, accident sequence evaluation program(ASEP) HRA Procedure[6] and technique for human error rate prediction(THERP)[7] is used for the detailed analysis of significant human actions. The comprehensive review of WS 2 NPP AOMs and the administration, test, and maintenance procedures of WS 1 NPP, and the interview with WS 1 NPP personnel are performed during the detailed analysis. The fourth task is to produce a traceable and understandable description of the process used to quantify human actions. Work sheets are prepared to document information such as the event information, the performance shaping factors, and the quantification process.

According to general classification scheme of human interaction in PSA, human actions are classified into following three types;

Type A(pre-accident human actions): actions prior to an initiating event

Type B: actions inducing the initiating events

Type C(post-accident human action): actions following an initiating event, broken down further into two categories CP: actions directed by procedures, and

CR: actions to recover failed equipment or systems to terminate accident

Type A human actions consist of maintenance, test and calibration actions that improve or degrade system availability. Type B human action is related to the initiating events. These human actions are usually included in the outage-frequency data base. Type CP human actions are that operator fails to perform proceduralized actions during transients or LOCA. Type CR human actions include one to restore or return to service that equipment that has failed during a transient or LOCA.

3. HRA for Type A Human Actions

The HRA for Type A human actiona is briefly described as following;

- Screening
- Identify situation and critical human actions
- Determine dependency effects
- Determine recovery factors

- Quantification

Based on general PSA approach, interview with WS 1 NPP personnel, and the review of administration procedure of WS 1 NPP, the judgmental screening criteria is made to give a guidance for fault tree analyst. Almost human actions are quantified as the value of 4.84E-3. A screening value of 4.84E-2 is used with a verification error of 1.0E-1.

4. HRA for Type C Human Actions

The HRA for Type C human actions are briefly described as following;

- Screening
- Identify human actions and dependency
- Identify accident sequences
- Perform task analysis
- Identify level of training/experience and procedure
- Determine type of cognitive behavior
- Determine type of task and stress level
- Quantification

Detailed analysis of approximately 40 human actions is performed with the careful consideration of human action dependencies in the accident sequences. The communication error is included in the quantification of human actions for systems shared by both NPPs. Type C human actions are quantified as the human error probability(HEP) of diagnosis error and execution error. The HEP for execution error and diagnosis error is evaluated using the Table 8-2 and the Table 8-5 of the ASEP[5], respectively. In this study, all HEPs in ASEP procedure are assumed to be the medians of a log-normal distribution. For all HEPs of execution error except special cases, basic HEPs of 3.22E-2 for step-by-step action under high stress conditions and 8.05E-2 for dynamic action under high stress conditions are used.

The Performance Shaping Factors (PSFs) considered in the process of quantification of human error is allowed time, available time, execution time, stress level, level of operator training/experiences, availability of and type of operating procedure related to the task, needs of communication between operators of both NPPs, type of cognitive behavior, hesitancy of operator, and the location and the degree of environmental hazard where the task is performed.

Dependencies can occur between the sub-tasks that make up an task, between various parallel tasks(e.g. fails to depressurize primary heat system using shutdown cooling system or emergency water system) in accident sequence and between operators. The level of dependencies between the sub-tasks is evaluated as complete or zero. The dependency between operators is implicitly considered in the process of the quantification of recovery probability of execution error. The level of dependencies between human actions in accident sequences are evaluated in accordance with THERP approach[7,8]; zero, low, medium, high, and complete. Level of dependencies is determined on mainly time difference between two human actions, cues of operator actions, structure of AOMs , and interview with operators.

5. Importance Analysis

Significant human actions to total core damage frequency can be represented as Fusel-Vesley importance[9]. The Fusel-Vesely measure of importance of human action H_j is defined as the probability that jth human contributes to the frequency of core damage given that the jth human action H_j failed. The Fussel-Vesely importance can be expressed as

$$I_{FV}(H_j) = \frac{\Sigma\,(CH)_j}{\Sigma\,C_j} = \frac{\text{Sum of cut set contributions to CDF containing the human action } H_j}{\text{CDF}}$$

where, $I_{FV}(H_j)$ the Fussel-Vesely importance of jth human actions, $(CH)_j$ is the frequency of an accident sequences which include jth human actions, and C_j is the frequency of jth accident sequences.

The importance analysis is performed with KIRAP code[10]. Top 10 human actions to total core damage frequency are presented in Table 1 with the descriptions, the HEP, the importance, and the ranking of them. Type A human actions are identified as an insignificant contributors to total core damage frequency, which are similar to those of other PSA. The importance analysis results in this study can be used for future EOP and accident management procedure preparation, risk-informed regulation, and operator training.

Table 2 shows the accident sequences related to human actions mentioned above. For example, OMV2-EW1D represents both failures of operator action for main steam safety valves(MSSVs) open and emergency water system(EWS) operation. As the allowed time is short and there is no distinct cues to classify the initiation time of both human actions, the dependency level of diagnosis error and execution error for both human actions are assumed to be complete and high, respectively. Description of accident sequence for OMV2-EW1D is following;

IE-DCC: Initiating event of digital control computer failure

/RS: success of reactor trip

/HTPC: success of liquid relief valves(LRVs) coarse pressure control

/SGPRC1: success of SG pressure removal

/OSIHIS*/SMPCIP: success of solid mode pressure control

FW: failure of feed water system

6. Administrative Control of Secondary Control Area

WS 1 NPP share D2O supply system, EWS from reservoir and emergency power system(EPS), which are modeled in WS 2/3/4 PSA, with WS 2 NPP. According to WS 1 NPP operating procedures, D2O supply system is operated at WS 1 NPP local place, EWS and EPS can be operated at WS 1 secondary control area(SCA) or local places. The SCA is originally designed to operate major safety systems when the main control room(MCR) is not available in the case of seismic event. As administrative control and test procedures of those systems are not decided from the operation aspects of WS 2 NPP during WS 2/3/4 PSA, it is assumed that those systems can be operated by only WS 1 NPP operators. Therefore, communication error is considered in the process of HRA. The core damage frequency of WS 2/3/4 NPPs can be decreased approximately 3% without the communication error.

It is time that the strategy of administration control for EWS and EPS be decided from the safety and operation aspects of both NPPs. Recommendation of reducing the human error is to keep the EWS and EPS operations under the control of both NPP operators. The examples of recommendation for the familiarization of EWS and EPS operations and the minimization of human errors for them during the emergency situation may be followings;

1) Participation of both NPP operators in the periodic test of both systems
2) Use of common administration and test procedures for both systems

7. Conclusion

The HRA in WS 2/3/4 NPPs PSA is systematically performed through the comprehensive review of various procedures and the interview with WS 1 NPP personnel. Information for future EOP and accident management procedure preparation, and operator training is obtained from the importance analysis. Also, some findings which can affect human performance are obtained from the review of AOMs being prepared and administration procedure of SCA. Recommendations of administrative control of SCA are given to minimize human errors for the operation of systems shared by both NPPs.

This paper shows that the comprehensive review of various procedures and the communications between HRA analyst, procedure writers, and NPP personnel are essential to get the insights from HRA. This paper also shows that PSA in design stage can provide the guidance for the preparation of administration procedures.

Acknowledgments

The authors wish to thank WS 1 NPP operators and maintenance technicians for their sincere and valuable assistance. In addition, the authors acknowledge the help and the discussion with AOM writers, Mr. Kim Seong Rae, Dr. Lee Ki Won and Dr. Lee Jeong Pyeo, at KOPEC. This work is supported by KEPRI(Korea Electric Power Research Institute) project 93N-J10.

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Table 1 Top 10 human Actions in WS 2/3/4 PSA

Event Name	HEP	Description of Operator Action	Importance	Ranking
OMV2-EW1D	1.28E-2	failure of MSSV open and EWS operation	8.05E-2	7
OEWS-ECCHX	7.34E-2	failure of EWS operation for ECCS HX	7.56E-2	8
OCC1	1.20E-2	failure of crash cool-down operation	6.25E-2	13
OR-DA-MKP-D	5.0E-1	failure of de-aerator makeup using demi- water system - high dependency with previous human actions	5.91E-2	16
OBP2-SD2C-EW1C	4.64E-5	failure of boiler pressure cool down, shut down cooling operation, and EWS operation	5.75E-2	18
OR-N2-FW-D	5.00E-1	failure of connection of IA tank to IA FW valve	4.96E-2	22
OPTHT	1.31E-2	failure of HTS pump trip	4.91E-2	23
OMV1-SD2C-EW1C	8.03E-5	failure of MSSV open, SDC operation, and EWS operation	4.84E-2	26
OR-RFT-AFW-D	5.0E-1	failure of transfer from de-aerator to reserve feed water tank - high dependency with previous human actions	3.51E-2	29
OBP-SD2C-EW1C	8.34E-5	failure of boiler pressure control, shutdown cooling system, EWS operation	3.39E-2	31

Table 2 Accident Sequences of Top 10 Human Actions

Event Name	Accient Sequences
OMV2-EW1D	IE-DCC*/RS*/HTPC*/SGPRC1*/OSIHIS*/SMPCIP*FW*OMSSV*EW1D
OEWS-ECCHX	various IEs*/RS*/LI*/CC*/ECC-D*ECC-LT
OCC1	IE-ESCB*/RRS-SETB1*CC1
OR-DA-MKP-D	IE-MSL3*/RS*/SGPR*/AFW
OBP2-SD2C-EW1C	IE-MSL3* /SGPR*/FW*OBPCC2*CND*SDCS*EWS
OR-N2-FW-D	IE-IA* /RS/*/FW
OPTHT	IE-SW*/RS*/CLPS*PTHT
OMV1-SD2C-EW1C	IE-IA*/RS*/FW*/SGPRC*/OMSSV1*SDCS*EWS
OR-RFT-AFW-D	IE-LOCD*/RS*/SGPR*/AFW
OBP-SD2C-EW1C	various IEs*/RS*/SGPRC*/FW*OBPCC*SDCS*EWS