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Human Reliability Analysis for Steam Generator Feed-and-Bleed Accident in Bushehr NPP-1

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ABSTRACT

According to the incident/accident reports, unsuccessful implementation of steam generator feed-and-bleed procedure is one of the most important events in nuclear power plants operation which greatly contributes to the level of risk of the plants. Generally, the loss of all feed water pumps flow (as one of the precursors) results in failure to maintain adequate cooling of the reactor core unless the operating crew initiate and follow the feed-and-bleed procedure correctly and timely.

In this paper, firstly, a Human Reliability Analysis (HRA) event tree is presented delineating the major human activities and errors in the implementation of the steam generator (SG) feed-and-bleed procedure following the loss of (both normal and emergency) water feed to four SGs of Bushehr Nuclear Power Plant unit1 (BNPP-1).

Secondly, the graphical method of task analysis as a part of HRA is used as a means of delineating correct and incorrect human actions. To be used in the probabilistic risk assessment (PRA), the outputs of the HRA event trees are fed into the system event trees, functional event trees or system fault trees. As a part of a probabilistic risk assessment of BNPP-1 and to assess the reliability of control room operators, a human reliability analysis model is applied based on the THERP (Technique for Human Error Rate Prediction) technique. The THERP method is used in the form of event trees named as the probability tree diagrams. In this research the Human Reliability Analysis event tree is constructed based on the background information and assumptions made and on a similar NPP task analysis. It is done so because the BNPP-1 is not an operational nuclear power plant.

Thirdly, based on NUREG/CR-1278 Handbook, a computer program has been developed in Visual Basic language and used to illustrate the major human activities and determination of error rates of operators in the course of the implementation of the steam generator feed-and-bleed procedure.

Finally, total failure rate of BNPP-1 control room operators in the relevant steps of "immediate Actions" and "Follow-up Actions" is determined in the framework of a team work in which a modified concept of dependence among the control room operators is used.

1 INTRODUCTION

On the basis of BNPP-1 safety analysis report, complete loss of steam generator feedwater supply is considered as beyond-design basis accident [1]. The typical conditions of complete loss of main and emergency feedwater are:

- All SG boiler water level decreases;
- Primary coolant heating-up to boiling with pressure increase up to inadvertent operation of PRZ[1] safety valves;
- Loss of primary coolant resulting from PRZ safety valve operation and impossible of ECCS[2] cooling water supply due to high primary coolant pressure.

Table 1 shows sequence of events (system and device operation), as well as interlocking and setpoints initiating their operation.

Table 1: Sequence of system and device operation [1]

Time moment, s	Event	Interlocking, setpoint for operation or other reason
0.0	Complete loss of main and emergency feedwater	Initial event
0.1	Turbogenerator stop valve closing	By the fact of trip of all feedwater pumps
2.1	BRU-K[3] opening and subsequent operation under load-follow conditions	MSH[4] pressure reaches the setpoint for BRU-K opening 6.67 MPa
4.0 to 41.0	The period of BRU-A[5] operation steam lines of all SG	Pressure in SG 1, 2, 3, 4 reaches the setpoint for BRU-A opening 7.154 MPa
4.2 to 28.3	The period of injection into PRZ from RCPS[6] delivery side	Primary and secondary pressure reaches setpoints for the first and second injection valves – 16.07 MPa and 16.27 MPa respectively
11.6	Normalization of signal for EP[7] operation	By core outlet pressure increase up to 17.5 MPa
35.6	Disconnection of RCPS in calculated loop 1	SG1 level decrease by 500 mm of nominal value
50.3	RCPS trip in calculated loops 2, 3, 4	Decrease of SG 2, 3, 4 level by 500 mm of nominal
1312 to 1337 1670 to 1696	Periods of the first PRZ PSD[8] operation	By primary pressure increase up to 18.11 MPa
1800	Onset of operating personnel actions: • Opening of valves of the emergency gas removal system; • Opening at full section of all three PRZ PSD; • Disconnection and prohibition to connect PRZ heaters	Emergency Instruction
2395	Change of boron emergency injection pump from recirculation to primary boron solution supply line	Reaching of pressure for the onset of boron solution supply from emergency boron injection pumps (7.8 MPa)
2640	ECCS hydroaccumulator operation	By primary pressure decrease to 5.9 MPa

[1] Pressurizer
[2] Emergency Core Cooling System
[3] Steam Dump Valve to Turbine Condenser
[4] Main Steam Header
[5] Steam Dump Valve to Atmosphere
[6] Reactor Coolant Pumps
[7] Emergency Protection of the Reactor
[8] Pulse-Safety Device

As a result of all feedwater pumps trip all SG feedwater supply is lost. Turbogenerator stop valves are closed by trip of all feedwater pumps that results in secondary pressure increase up to the setpoints for BRU-K opening (at 2.1 s of the transient) and BRU-A (at 4.0 s of the transient). Deterioration of secondary heat removal results in primary pressure increase. At 4.2 s the setpoint for opening of the first PRZ injection valve from RCPS delivery side is reached and at 11.6 s EP operation signal is normalized by core outlet pressure increase up to 17.5 MPa. Caused by the discharge through the secondary dump devices SG level is decreased by 500 mm of the nominal value. As a result loop 1 RCPS (the loop with PRZ) is tripped by SG level decreased by 500 mm of nominal value at 35.6 s, and at 50.3 s RCPS in loops 2, 3 and 4 are tripped by the same symptom. On RCPS trip and termination of their coastdown primary coolant natural circulation stabilized. At 1010 s SG level is decreased by 900 mm of nominal value, resulting normalization of the signal for emergency feedwater electric pump start-up (since SG emergency feedwater supply is not available as a result of the initial event). SG tubing uncovering continued results in primary coolant pressure and temperature increase. At 1312 and 1670 s actuation of the first PRZ PSD occurs.

It is assumed that the operating personnel activity initiated at 1800 s purposed to decrease primary coolant pressure.

For this purpose an operator takes the following measures at 1800 s:

- Forced opening of all three PRZ PSD at full flow section;
- Opening valves of the emergency gas removal system;
- Disconnection of PRZ heaters with prohibition to connect.

At 1822 & 2000 s of the transient coolant discharge through PRZ PSD and emergency gas removal results in RCC[1] and RPC[2] levels normalization respectively. At 2395 s when coolant pressure is decreased to 7.8 MPa, primary coolant system boron solution from emergency boron injection pumps is initiated. Proceeding pressure decreases results in ECCS hydroaccumulators operation at 2640 s. Boron solution supply from ECCS results in RPC filling at 2840 s. RCC is filled at 3520 s of the transient. Reactor coolant system pressure is stabilized at the level 20 MPa at 4000 s. reactore core inlet and outlet temperature gradual decrease is preceded due to reactor primary circuit boron solution supply from emergency injection pumps.

To identify the possible human errors a task analysis has been done in which the operator actions in the case of occurrence of the accident are analyzed. The critical errors are identified as the following:

- Failure to diagnose the event correctly within 30 minutes [1], which includes failure to respond appropriately to an annunciator that warns the saturation of the pressurizer.
- Failure to perform the procedures "immediate action" and "follow up action" according to the list of procedures.

The graphical method of task analysis "HRA" is used as a means of delineating correct and incorrect human actions (figure 1). For PRA use, the outputs of the HRA event trees are fed into system event trees, functional event trees or system fault trees.

2 THE BASIS MODEL AND METHOD

As a part of a probabilistic risk assessment of BNPP-1 and to assess the reliability of control room operators, human reliability analysis model is applied using the THERP[3]

[1] Reactor Collection Chamber
[2] Reactor Pressure Chamber
[3] Technique for Human Error Rate Prediction

technique [2], [3]. The basis method of THERP is used as a form of event tree named as the probability tree diagram (figure 1).

In the HRA event tree, the limbs represent a binary decision process, i.e., correct or incorrect performance are the only choices. Thus, at every binary branching, the probabilities of the events must sum to 1.0.

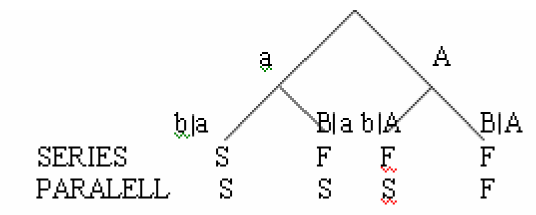

Figure 1**:** HRA event tree for series or parallel system [3]

Limbs in the HRA event tree show different human activities as well as different conditions or influences upon these activities.

2.1 BNPP-1 Control Room Organization

In BNPP-1, Main control room (MCR) is provided at the unit, the operating personnel of which implement the control of the process equipment of normal operation systems and the safety systems in normal operation and under emergency conditions [4]. The MCR layout for BNPP-1 is based on state-of-art technology with respect to human factors engineering, ergonomics and annunciation.

At the MCR with the unit under operation there should be not less than 3 persons licensed to operate the reactor plant. A minimum strength of shift personnel at the MCR is presented in table 2.

Table 2: Minimum staff at the BNPP MCR [4]

Position	Quantity
Plant Shift Supervisor	1
Unit Shift Supervisor	1
Reactor Compartment Shift Supervisor	1
Reactor Operator	1
Turbine Compartment Shift Supervisor	1
Turbine Operator	1

According to BNPP-1 configuration the Plant Shift Supervisor is the senior leader of the shift operating personnel at the BNPP-1. The plant shift supervisor manages work of operative shift personnel at start-up or shut down of the unit for repair, normal operation. He is the person who is in charge of all matters related to the BNPP-1 operation.

The Unit Shift Supervisor is the operative leader of shift personnel operating the main BNPP-1 equipment (reactor, turbine plants); he follows instruction of the plant shift supervisor and informs him about any changes in equipment status. In the transient cases at the unit, the unit shift supervisor (SS) manages actions of reactor shift supervisor, reactor operator and turbine shift supervisor.

The Reactor Compartment Shift Supervisor manages the reactor operator and the reactor compartment personnel. He is senior operative head of reactor unit during shift time and informs the plant shift supervisor of all activities at equipment assigned to him.

Reactor Operator is directly responsible for operative control of reactor and ensures safe operation of equipment and system at the reactor. Reactor Operator is directly subordinate to shift supervisor.

The Turbine Compartment Shift Supervisor manages the Turbine operator and personnel of turbine compartment. He is subordinate to unit shift supervisor.

The Turbine Operator is responsible for the BNPP-1 turbine plant operation during his duty hours. He is subordinate to the turbine compartment shift supervisor.

2.2 Effect of Recovery Factors

Recovery factor is the probability of timely detecting and corrections of incorrect task performance to avoid undesirable consequences. In any man-machine system, there are usually several recovery factors, routine inspections that can increase the probability of detecting errors before they affect the system [5]. For example the routine inspections can be a possible way of detection and correction of operators (maintenance) errors.

2.3 Dependence between BNPP-1 Control Room Operators

Dependence is a continuum, and it is necessary to judge the appropriate level existing between any pair of task performances. Figure 2 shows the relative positions of the five discrete points in this model. There are five points showed in this figure:

The two end points of zero dependence (ZD) and complete dependence (CD) plus three points in between. We call these intermediate points low dependence (LD), moderate dependence (MD), and high dependence (HD).

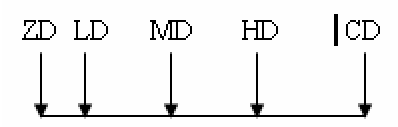

Figure 2**:** Continuum of dependence levels

The levels of dependence among the control room personnel are:

- High dependence (HD) between Reactor Compartment Shift Supervisor and Reactor Operator as well as Turbine Compartment Shift Supervisor and Turbine Operator.
- Low to moderate dependence (LD to MD) between Unit Shift Supervisor and the other operators, and
- Low to complete dependence (LD to CD) between the Plant Shift Supervisor and the licensed operators depending on the nature of the task.

2.4 Application of Dependence Equations

The initiating task normally is assigned a basic HEP[1] (BHEP), that is, an HEP without considering the effects of dependence. The HEPs for successing tasks represent conditional probabilities and may reflect one of the five dependence levels. Based on the dependence model, to obtain the probability of human error on all tasks, conditional human error probability (CHEP) can be calculated using BHEP estimations.

[1] Human Error Probability

The equations for conditional HEPs are listed in table 3, where task "N" follows task "N-1".

Table 3: Equations for (CHEPs) on task "N" given on previous task "N-1"

Level of Dependence	Failure Equations	
ZD	$pr\left[f_N \big/ f_{N-1} \left| ZD \right| \right] = N$	(1)
LD	$pr\left[f_N \big/ f_{N-1} \left| LD \right| \right] = \dfrac{1+19N}{20}$	(2)
MD	$pr\left[f_N \big/ f_{N-1} \left| MD \right| \right] = \dfrac{1+6N}{7}$	(3)
HD	$pr\left[f_N \big/ f_{N-1} \left| HD \right| \right] = \dfrac{1+N}{2}$	(4)
CD	$pr\left[f_N \big/ f_{N-1} \left| CD \right| \right] = 1$	(5)

3 FLOW DIRECTION OF HUMAN RELIABILITY ANALYSIS

The human reliability analysis is calculated using software, developed in Visual Basic 6 language. The program begins with the "START" ellipse as shown in figure 3, and is followed by the path from the abnormal event to rule-based action.

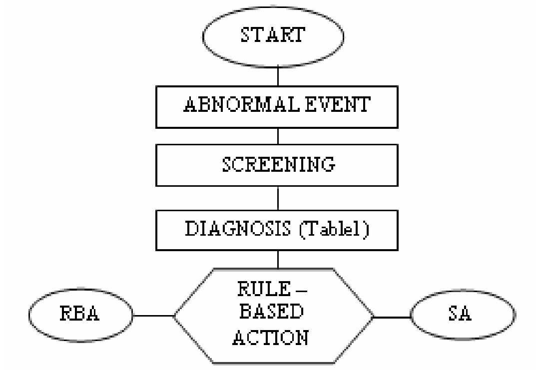

Figure 3**:** First step of HRA schematic

This path leads to table 4 for screening diagnosis model.

Table 4: Initiate-screening model of estimated HEPs and EFs for diagnosis [3]

Item	t minutes after t_0	Median joint HEP for diagnosis of a single or the first event	EF
1	1	1.0	---
2	20	0.01	10
3	**30**	**0.001**	**10**

According to BNPP-1 Safety Analysis Report [1], loss of all feedwater should be diagnosed within 30 minutes to allow time to carry out the feed-and-bleed procedure. In the next step we obtain HEPs and EFs[1] for rule-based actions (RBA) by control room personnel after diagnosis of the event.

These potential errors are divided in two steps:

1. Errors per critical step without recovery factors.

[1] Error Factors

2. Error per critical step with recovery factors.

The amounts of these errors for two above steps are 0.05 and 0.025 respectively [3].

The next decision node in rule-based actions is determining type of error (figure 4). We assume that the BNPP-1 operators are well trained in carrying out the feed-and-bleed procedures, so that errors of commission are negligible. Thus the only errors to be considered are errors of omission.

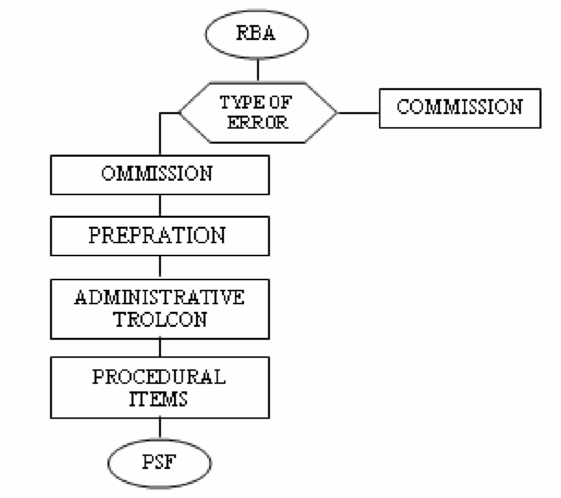

Figure 4: Second step of HRA schematic

The path leading to the performance shaping factors "PSFs", ellipse is presented in figure 5.

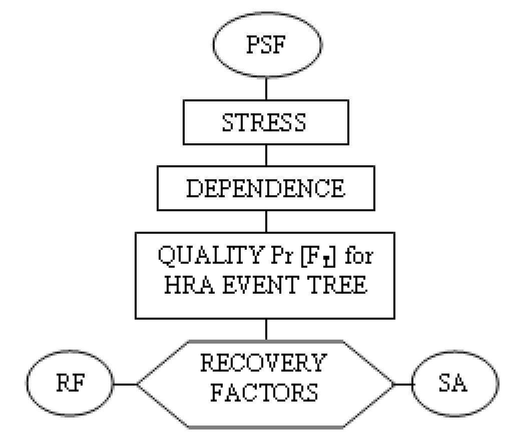

Figure 5: Third step of HRA schematic

According to NUREG/CR-1278 the stress level in this accident is moderately high. Bushehr NPP-1 operators have not any operation experience and are considered novice. According to table 5 a modification (multiple by 4) is applied to the BHEP for step-by-step activities under moderately high stress.

Table 5: Modification coefficient for the effects of stress and experience level [3]

Stress level	Modifiers for nominal HEPs	
	Skilled	**Novice**
Moderately high (heavy task load):	(a)	**(b)**
Step-by-step	*2	***4**

In the next step (figure 6) we determine recovery factors. In NPP operations, few human errors cause damage or reduce the availability of individual systems because the potentially adverse effect of the error is prevented or compensated by other components or systems or by other human actions. We call these preventive or compensatory factors recovery factors. The error that did not result in some undesirable consequences to the system is a recovered error, or a no-cost error. If an error is not recovered and it results in some undesirable consequences, it is called an un-recovered error [6].

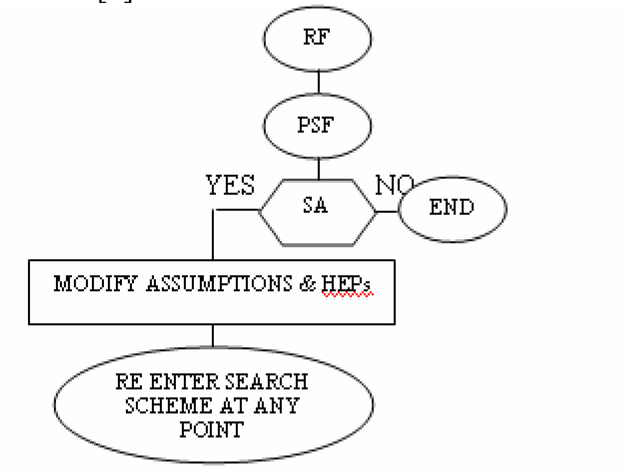

Figure 6**:** Last step of HRA schematic

In the last step, if we are satisfied with the HRA, the No path leads to the "END" ellipse, and the HRA is finished. If there is need to vary certain assumptions the Yes path in figure 6 enables us to re-enter the search scheme at any point and modify estimates and recalculate the Pr [F_T].

4 CALCULATION OF TOTAL FAILURE TERM OF BNPP-1 OPERATOR'S FUNCTION

In accordance with diagrams described in previous sections, the total amount of failure during steam generator feed-and-bleed accident can be obtained by the summation of failure rates in each stage of the procedure as the following equation:

$$F_T = F_1 + F_2 + F_3 + F_4 + F_5 \qquad (6)$$

In equation 6 the factors F_1 through F_5 indicate the failure rate of 5 different stages described below:

1. Diagnosis of abnormal event by control room personnel.
2. Verification of availability of steam-driven and electric emergency feedwater pumps.
3. Verification of correct status of feedwater or bypass valves.
4. Implementation of the procedure in case of unavailability of both EFW[1] and main feedwater.
5. Initiation of HPI[2].

[1] Emergency Feed Water
[2] High Pressure Injection

Failure rate for each stage is the multiplication of human error probability (HEP) of whole control room personnel during procedure. For example failure rate of first stage is calculated by equation 7 in which A_1 to A_7 are estimated HEP of involving personnel.

The calculated HEP for the first stage is shown here. The total failure rate of operators error for the other stages can be obtained with following the same direction.

$$F_1 = A_1 * A_2 * A_3 * A_4 * A_5 * A_6 * A_7 \tag{7}$$

- A_1
All six control room personnel fail to correctly diagnose the abnormal event within 30 minutes after annunciation, based presence of Reactor Operator, Turbine Operator, Reactor Compartment Shift Supervisor, Turbine Compartment Shift Supervisor, Unit Shift Supervisor, and Plant Shift Supervisor. The estimated HEP in this stage is 0.001 [3].
- A_2
Reactor Operator fails to respond to annunciator cues of the misdiagnosis. The estimated HEP in this stage is 0.006 [3].
- A_3
Reactor Compartment Shift Supervisor fails to respond to the annunciator cues. The conditional HEP in this stage is related to existence of high dependence (HD) levels between Reactor Compartment Shift Supervisor and Reactor Operator. Thus, the estimated conditional HEP is about 0.5 when using equation 4.
- A_4
The procedure to estimate Turbine Operator error probability in this stage is the same as what presented in section 2 for Reactor Operator probability estimation.
- A_5
The procedure to estimate Turbine Compartment Shift Supervisor in this stage is similar to Reactor Compartment Shift Supervisor error probability estimation which is presented in section 3.
- A_6
Unit Shift Supervisor fails to respond annunciator cues. The conditional HEP in this stage is related to existence of medium dependence (MD) levels between Unit Shift Supervisor and the other operators. Thus, the estimated conditional HEP is around 0.14 when using equation 3.
- A_7
Plant Shift Supervisor fails to respond the annunciator cues. The Plant Shift Supervisor will not pay attention to the saturation annunciators since this is a "detailed operation" and he belives that a correct diagnosis has been made. So the level of dependence to estimate conditional HEP is complete dependence. Thus the estimated conditional HEP in this stage is approximately 1.0 when using equation 5.

The calculated approximately amount of F_T for BNPP-1 for two relevant steps of "immediate action" and "follow up action" is presented in table 6.

Table 6: BNPP-1 control room operators' total failure rate Pr [F_T]

IA[1]				FA[2]	
F_1	F_2	F_3	F_4	F_5	F_T
$<<10^{-8}$	10^{-6}	$<<10^{-8}$	$<<10^{-8}$	$<<10^{-8}$	10^{-6}

[1] Immediate Action
[2] Follow up Action

CONCLUSION

The HRA calculation results for beyond-design basis accident with complete loss of main and emergency feedwater has proved that the accident process control by operating personnel by opening of three PRZ PSD and use of emergency gas removal system at 1800 s since the accident initiation results in primary coolant pressure decrease to the level at which cooling water supply from four channels of boron emergency injection and subsequently from ECCS hydroaccumulators is initiated. Hereby the hottest fuel rod cladding temperatures do not exceed nominal value (350°C) within the whole accident period. Thus, with operating personnel interfere (action); within the period considered in the analysis, reactor plant is brought into controlled condition, at which reactor core reliable cooling, RPC, RCC, PRZ filling and primary coolant cooldown is provided. Thus, reactor plant is brought into safe condition by operating personnel actions.

Furthermore as table 6 shows, the amount of F_2 is too larger than the other factors and is the dominate factor in F_T calculation. Also in the second stage of procedure (F_2) Reactor & Turbine Compartment Shift Supervisors with the highest amount of error probability (HEP = 0.57) has the most influence on the total failure rate.

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