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Technical Report

Human Error Probability Evaluation as Part of Reliability Analysis of Digital Protection System of Advanced Pressurized Water Reactor – APR 1400

KAERI

Korea Atomic Energy Research Institute

Statement of Submission

To: President of Korea Atomic Energy Research Institute

This report is submitted as a result of work performed on

"Human Error Probability Evaluation as Part of Reliability Analysis of Digital Protection System of Advanced Pressurized Water Reactor – APR 1400".

This work has been carried out by Dr. P.V. Varde, a Scientist from Bhabha Atomic Research Centre, Mumbai (INDIA), as part of his postdoctoral research work with the Instrumentation & Control - Human Factors Division of KAERI.

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요 약 문

이 보고서는 한국형 가압 경수로인APR 1400의 디지털 보호계통에 대한 신뢰도 분석의 일환으로 수행한 인간신뢰도 분석 관련 내용을 기술하고 있다. 디지털 원자로 보호계통은 필요한 상황이 발생하면 원자로를 자동 정지하게 된다. 그러나, ATWS와 같은 경우 원자로 정지의 다변화 수단으로 운전 원의 수동정지가 필요하게 된다. 이 보고서에서는 원자로 정지 관련 인적오류와 원자로 보호계통 계장설비의 Calibration 오류에 대한 분석 내용을 기술하였으며 필요한 경우 파라미터 분석을 포함하였다. 이 보고서는 한국형 가압 경수로인 APR 1400의 디지털 보호계통에 대한 신뢰도 분석의 일환으로 수행한 인간신뢰도 분석 관련 내용을 기술하고 있다. 디지털 원자로 보호계통은 필요한 상황이 발생하면 원자로를 자동 정지하게 된다. 그러나, ATWS와 같은 경우 원자로 정지의 다변화 수단으로 운전 원의 수동정지가 필요하게 된다. 이 보고서에서는 원자로 정지 관련 인적오류와 원자로 보호계통 계장설비의 Calibration 오류에 대한 분석 내용을 기술하였으며 필요한 경우 파라미터 분석을 포함하였다.

SUMMARY

A case study on human reliability analysis has been performed as part of reliability analysis of digital protection of the Korean Advanced Pressurized Water Reactor – APR 1400. The digital protection system of the reactor automatically actuates the shutdown system of the reactor when demanded. However, the safety analysis takes credit for operator action as a diverse mean for tripping the reactor for, though low probability, ATWS scenarios. Based on the available information two cases, viz., human error in tripping the reactor and calibration error for instrumentations in protection system, have been analyzed. Wherever applicable a parametric study has also been performed.

1. Introduction

The human factor plays an important role in determining the safety of the operating nuclear plants. The literature and data on operating performance of the nuclear plant world over observes that contribution of human factor to accidents is ~ 70%. This is the reason why the nuclear community is paying increasing attention to the evaluation of human performance in the nuclear plant operations. Though the deterministic approach is the primary and basic approach for the safety analysis, it does not offer a suitable framework for the integration of human factor in the evaluation of plant safety. In this regard, the Probabilistic Safety Assessment (PSA) methods have definite edge over the deterministic methods. PSA approach enables the integration of human factor with the model of the plant towards giving the statement of safety. At system level human factor is integrated into the system fault tree for giving the quantified estimate of system ‘unavailability’. At plant level the human factor forms the part of accident sequences in the evaluation of Core Damage frequency (CDF).

Human reliability analysis was performed for the critical man-machine interactions as part of reliability analysis of digital protection system of the Korean Advanced Pressurized Water Reactor (APR-1400).

Section 2 presents the approach for the analysis. Section 3 describes the models / methods relevant for this analysis. The reliability evaluation for the selected human actions has been described in section 4. Section 5 gives salient observation related to the analysis and concluding remarks.

2. The Approach

The design of digital protection system of APR-1400 ensures implementation of defense-in-depth principles through diversity and redundancy. Accordingly, the monitoring system comprised of two-out-of-four voting logic. Nevertheless, the design of the protection system takes credit for human interactions for:

1. Manually tripping of the reactor in case of the very low probability occurrence

of failure control system to trip the reactor on demand.

2. Routine calibration checking of the analog instrumentation in the plant.

These human actions are very critical and could have safety implications. Hence, a systematic human reliability analysis was performed for the scenarios which were considered to have direct bearing on plant safety. The development of the fault trees and an overview of the projected accident sequences provided vital inputs on the human actions to be considered for the evaluation of quantified estimates of human reliability.

The input data for this analysis have been taken from generic sources. Some assumptions have been made in the analysis. Most of these assumptions have to be made as the finer details of the control room design and other factors related to human-machine interface are yet to be determined. However, the experts' opinion formed an important source of information in this regards.

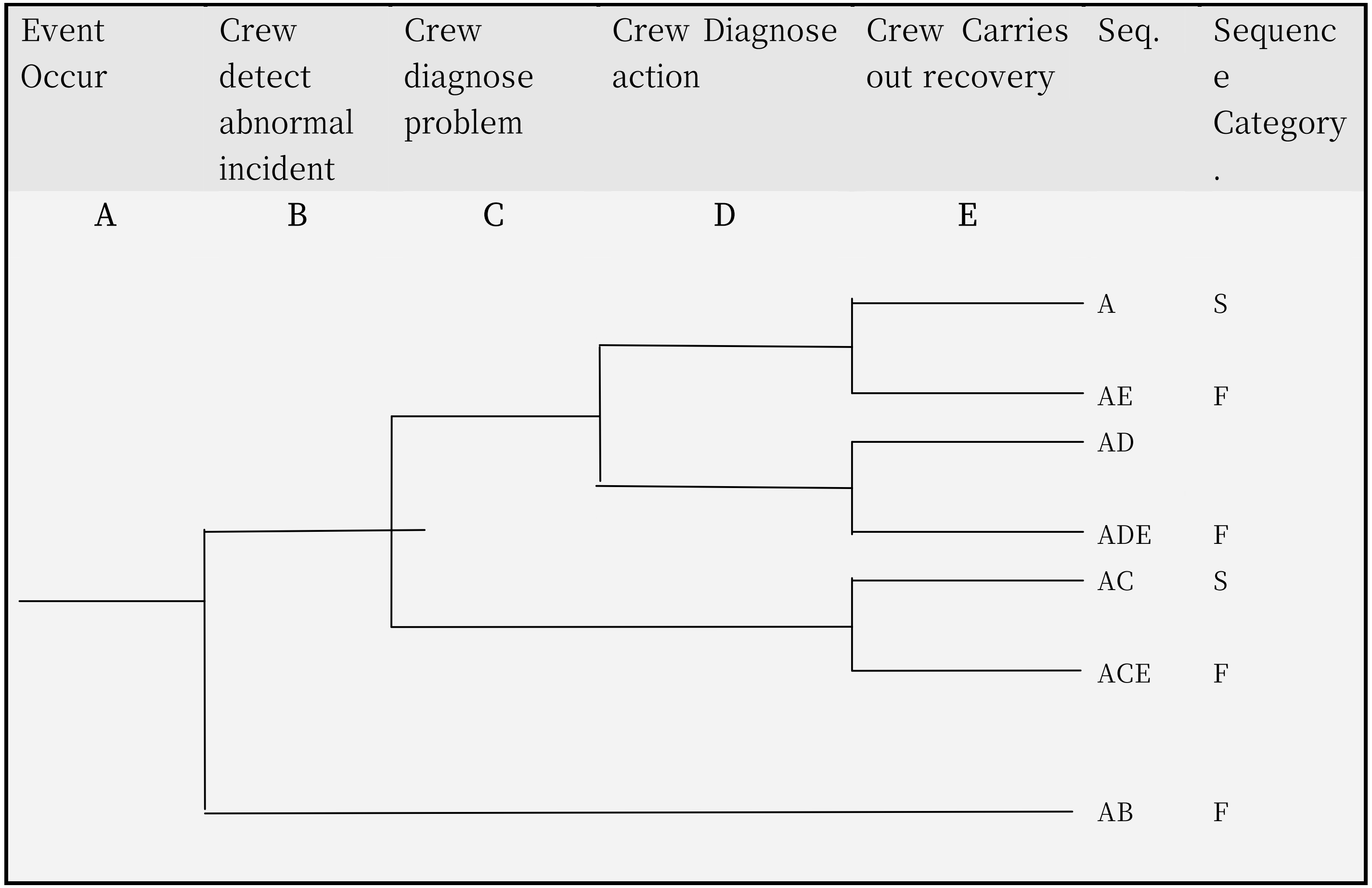
The Operator Action Tree (OAT) framework, as shown in Figure 1, was used for modeling the sequences of human actions. The Human Cognitive Reliability (HCR) Model was used to determine the operator's / crew mental ability in processing plant information and condition. The classification of the human actions into the a) Skill based b) Rule based and c) Knowledge Based actions was done using the guidelines given in IAEA-TECDOC-592. Performance Shaping Factors (PSFs) were used for taking into account recovery actions. The data on available 'time window' were obtained either using the results of related deterministic analysis or using expert opinions. The final estimate of human error probability was given by summing up the contribution from cognitive and action parts.

Keeping in view the uncertainties which are inherently part of any human reliability model, the result of this study were used with due care. This uncertainty was addressed by using relatively large error factors in the uncertainty analysis. Wherever applicable parametric studies were performed to see the sensitivity of results to various PSFs and other parameters.

3. Relevant Tools for HRA

This section reproduces the information from the available literature, in brief, which was considered relevant for study. The objective is to provide the information readily

Fig. 1: Operator Action Tree



available in this report for reference and better understanding of the work described in this report.

3.1 Classification of Human Actions

This is the basic step in human reliability analysis. In human reliability parlance the human actions are classified into the three man categories;
a) Skill based
b) Rule based
c) Knowledge based

Figure 2 depicts the criteria for classifying the human actions into any of the above categories [1].

The Electric Power Research Institute approach on human reliability analysis deals with the five types of human actions for modeling human behavior in a nuclear plant. They are:

1. Testing and maintenance actions prior to an initiating event
2. Human actions which result into accident initiation
3. Amelioration of an accident by correctly responding to the event
4. Exacerbation of an accident by taking incorrect action
5. Amelioration of an accident in progress by improvisations which were not included in the procedure

The scope of this analysis takes into account the Type 1 and 3 human actions.

3.2 Human Cognitive Reliability Correlation

This is an analytical method that enables the quantification of reliability of control room personnel in respect of response to abnormal plant conditions. The HCR correlation is basically a normalized time-reliability correlation which takes into account the factors like i) time window available to complete a task / action, ii) expected operator stress level, iii) the type of human-machine interface, etc. This model, as shown below, facilitates estimation of cognitive reliability for a) skill based, b) rule based and c) knowledge based behavior. Table 1 lists the value of the coefficients corresponding to

Fig. 2: Depiction of criteria for classification of human behavior

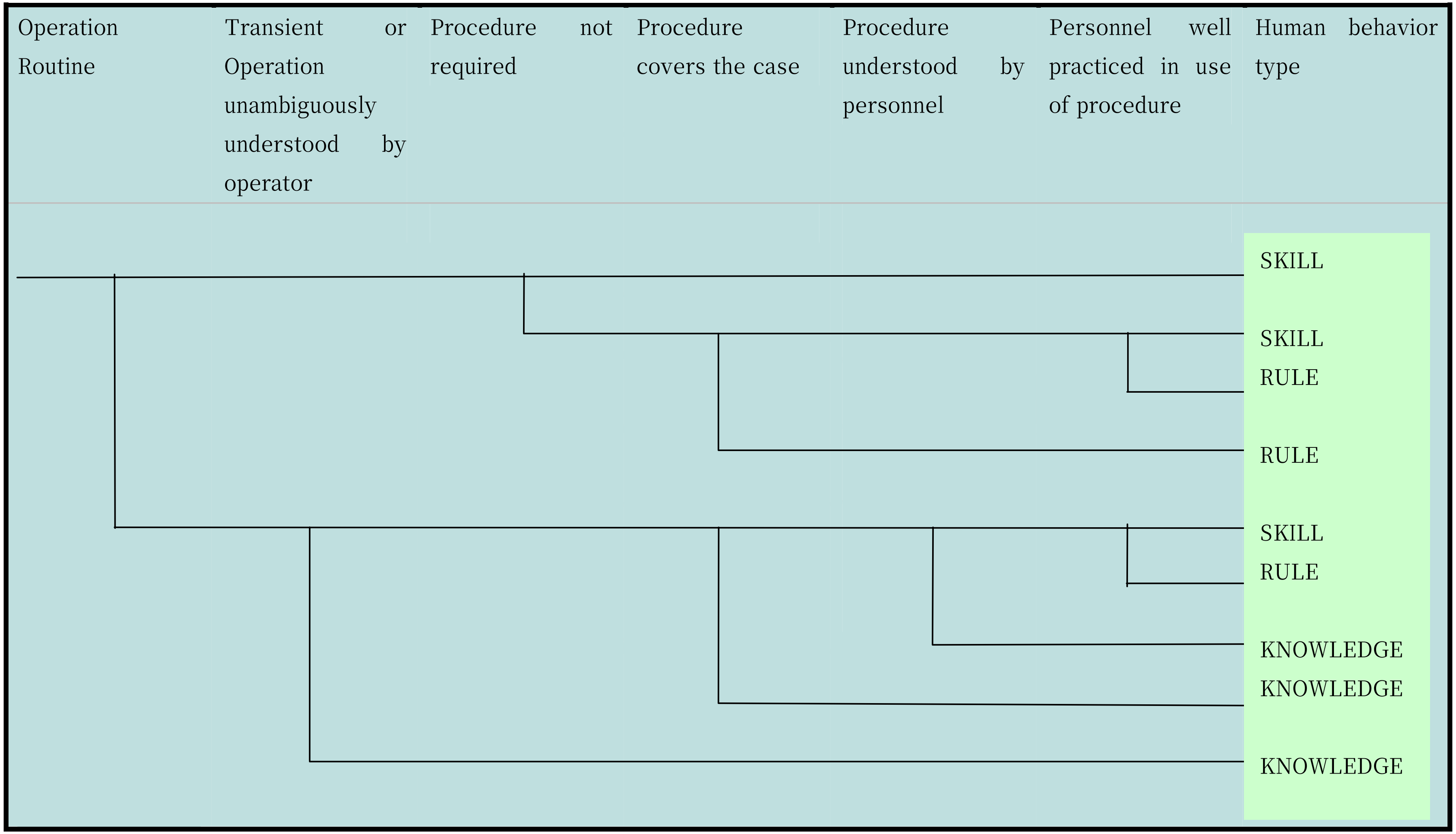


Table 1: HCR interim parameters

Cognitive behavior	Empirical HCR coefficients		
	Ai	**Bi**	**Ci**
Skill based	0.407	0.7	1.2
Rule based	0.601	0.6	0.9
Knowledge based	0.791	0.5	0.8

the above mentioned factors.

$$P(t) = \exp-\left[\left\{\frac{(t/T_{1/2}) - B_i}{A_i}\right\}^{C_i}\right]`$$

where, t : time available for the crew / operator(s) to complete a given action or set of activities following a stimulus

$T_{1/2}$: estimated median time taken by the crews / operator(s) to complete an action or the task

Ai, Bi, Ci : correlation coefficients associated with the i^{th} type of cognitive behavior

P(t) : crew / operator(s) non-response probability for a given time window

Keeping in view the advantages and limitations of this approach this model has been used for actions like manual tripping of reactor from control room consequent to the failure of protection system to actuate the CEDMs (Control Element Drive Mechanisms) on demand.

3.3 Performance Shaping Factors

The factors which are considered responsible for directly affecting the operator performance, like operator experience, stress level, the quality of man-machine interface, etc are called performance shaping factors (PSFs). Table 2 lists these factors and corresponding values which have been determined by simulator experiments and the associated criteria used for the classification of these factors.

Table 2: Interim HCR model performance shaping factors and related coefficients

S.N.	Performance shaping factors	Coefficients	Criteria (in brief)
1	*Operator Experience*		
	1a. Expert, Well trained	-0.22	License > 5.0 year
	1b Average knowledge, training	0.00	License > 6.0 month
	1c Novice, minimum training	0.44	License < 6.0 months
2	*Stress level*		
	2a. Situation of grave emergency	0.44	Operator feels threatened
	2b. Situation of potential emergency	0.44	Mild stress
	2c. Active, no emergency	0.00	Optimum situation
	2d. Low activity, low vigilance	0.28	Unexpected transients with no precursor
3.	*Quality of operator / plant interface*		
	3a. Excellent	-0.22	Operator advisor systems
	3b. Good	0.00	Integrated plant displays
	3c. Fair	0.44	Display human engineered
	3d. Poor	0.78	Display available but not human engineered
	3e. Extremely poor	0.92	Display needed to alert operator

3.4 Human Dependency Analysis

For the cases when more than one human error probabilities exists in a minimal cut set in the fault tree dependence must be considered. The level of dependence could be given by equation of conditional probability. HEP of failure given failure of the previous task and with Po as its dependent HEP value

Zero dependence $HEP = Po$

Low dependence $HEP = \frac{1+19Po}{20}$

Moderate dependence $HEP = \frac{1+6Po}{7}$

High dependence $HEP = \frac{1+Po}{2}$

Total dependence $HEP = 1$

4. ESTIMATION OF HUMAN ERROR PROBABILITIES FOR APR-1400

As mentioned earlier two human interactions were analyzed in this study as part of reliability analysis of digital protection system in APR-1400. These interactions are; a) Manual tripping of reactor consequent to the failure of digital protection system to trip the reactor, and b) Test and calibration related errors in the digital and analog instrumentation. The subsequent section presents the detailed analysis related to above human interactions.

4.1 Manual Reactor Tripping

4.1.1 The Problem Definition

The digital protection system of the reference Advanced Pressurized Water Reactor – APR 1400 trips the reactor automatically on demand. The probability of failure of the automatic tripping of the reactor, achieved through the actuation of digital plant protection system and there upon gravity fall of the associated shutdown system called

Control Element Drive Mechanisms (CEDMs), has been demonstrated to be very low. However, the credit for operator action has been taken in tripping of reactor as part of incorporation of diverse means for tripping the reactor. Accordingly, the task definition for the purpose of the assessment of human reliability is as follows:

"Assessment of probability of failure to manually trip the reactor by the operator from the operating state of 100 % FP in the event of failure of automatic protection system to actuate the shutdown devices (CEDMs) on demand".

4.1.2 Failure Criteria

The scenario considered for the above operator action is failure of reactor trip system at the onset of failure of feedwater system and consequent requirements of manual tripping of the reactor from control room by the operator. The deterministic analysis requires that the reactor should be tripped in 79 sec to avoid the over pressurization of the primary coolant system. Hence, *the failure to manually trip the reactor in the available time window of 79 sec will be considered as failure of operator to initiate reactor trip.*

4.1.3 Assumptions

Following assumptions have been made as follows:

1. The control room crew comprised of well qualified senior operator and junior operator to assist the senior operator.
2. The reactor trip buttons are well marked such that there is negligible chance of committing the error in identifying and pressing the trip button.
3. The operator is well trained in handling this situation, i.e. requirements of manual tripping of the reactor, as part of loss of feed water event coupled with failure of reactor trip on auto.
4. Chances for misdiagnosis with regards to tripping of reactor are very small.
5. No credit has been taken for the registration of Pre-trip before a reactor trip occurs.

4.1.4 Scenario Representation

The Operator Action Tree for the given scenario has been given in Fig. 3. The sequence of process involved in tripping the reactor has been divided into two parts; a)detection and diagnosis part and b) the action part.

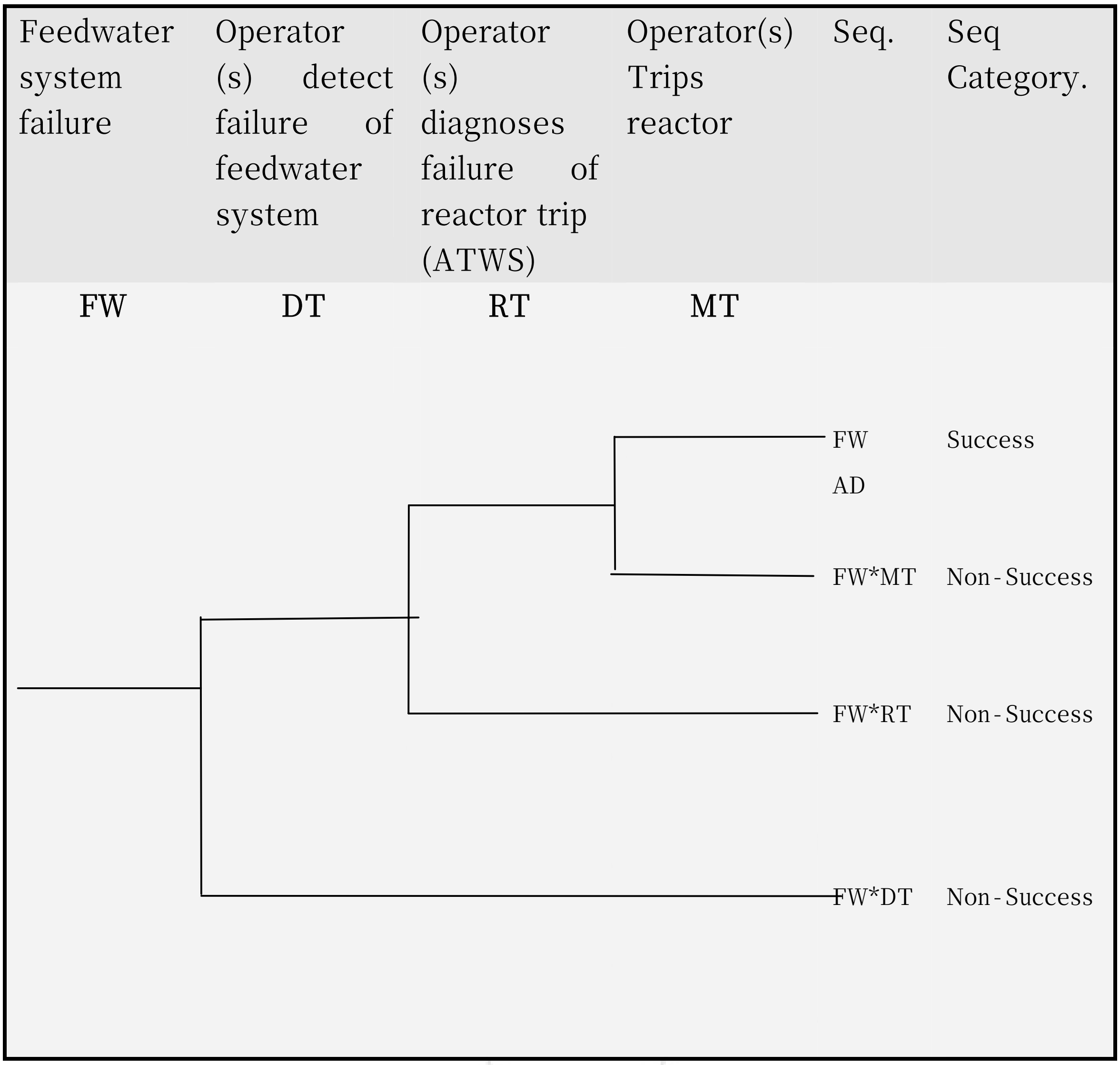


Fig. 3: Representation of operator interaction with the plant

As mentioned earlier the header event on the left side of in Fig. 3, FW and DT, have not been evaluated. The estimation of failure frequency of event 'feedwater system failure' is not within the scope of this work. The failure of operator to detect the feedwater system failure has been considered to be low probability event.

The event 'Operator diagnoses the ATWS' has been evaluated using the Human Cognitive reliability (HCR) model.

4.1.5 HCR Model

The mental model of the reference operator has been formulated considering applicable PSFs for the cognitive processing. This analysis does not take the credit for the registration for the registration of the 'Pre-trip' which enables operator to understand the scenario in the offing. Accordingly, the available time window for the performance of the above task has been taken as 79 sec.

The operator(s) have enough indications suggesting that an ATWS scenario has occurred. The indications that the CEDMs should have tripped are as follows: i) Indication of turbine trip, ii) reactor trip annunciation trip alarm in control room, iii) Reactor trip monitor light. The indication that reactor ha not tripped are : a) CEDM position indicator showing rods have not been inserted, ii) CEDM bottom indication off, iii) Reactor power meter showing no drop in power and the iv) the trip circuit breakers (TCBs) closed indication showing that breakers have not tripped. The experts opinion and the data available in the literature suggests that operator will take ~ 30 seconds to perform the diagnosis of ATWS and initiate the action to trip the reactor.

The reference may be made to Table 1 of this report for various PSFs. Applicable PSFs and their justifications are as follows:

The operator has been considered to have average training and the stress level for reference operator for this transient could be represented by the phrase 'situation of potential emergency'. The control room of APR-1400 has features like critical function monitoring, on-line estimation of thermal margin using core protection calculator and other on-line operator aids. Hence, the quality of operator / plant interface has been assumed to be 'excellent'. Keeping in view above discussion the value of coefficients, K1 (Operator experience), K2 (Stress level) and K3 (Quality of operator / plant interface was 0.00, 0.44 and -0.22, respectively.

Hence, considering above coefficients the median time, $T_{1/2,}$ can be given as

$$T_{1/2} = 30*(1-0)*(1+1.44)*(1-0.22) = 33.7 \sim 34 \text{ seconds}$$

Given the background that the above scenario, i.e., ATWS involving feedwater system failure, is 'not' a routine phenomenon and operator gets rare opportunity to practice this type of scenario, the action could be conservatively categorized as 'Rule based'(refer Fig. 1 for the same). Accordingly, the applicable interim HCR parameter (Table 1) Ai, Bi and Ci for rule based cognitive processing could be given as 0.601, 0.6 and 0.9 respectively.

Considering the above factors the contribution from the cognitive part to failure probability of reactor trip P(t) could be estimate as follows:

$$P(t) = \exp - \left[\left\{ \frac{(79/34) - 0.6}{0.601} \right\}^{0.9} \right]$$

$$P(t) = 7.5 * 10^{-2} \text{ per demand}$$

Hence, the contribution from cognitive part to net probability of failure of manual reactor trip has been estimated as 7*.5 * 10*$^{-2}$ *per demand.*

4.1.6 Considerations for action part

Once the diagnosis has been completed successfully the parameters which determine the failure probability of action in initiating the reactor trip could be summarized as follows;

The operator is well trained in actuating the trip button in case of manual demand. The stress level could be considered to be moderate and is not expected to affect the action manual actions of the operator. There are two reactor trip button provided in the control room with distinct marking. These marking and the prominence given to these push buttons in the control room ergonomics rather eliminates the chances of error of commission from the operator. The presence of second well trained operator to assist the senior operator excludes to a great extent erroneous action of the operators part. The prompt response / non response of the plant on pressing the trip button informs the operator about the success / failures of his actions. This provides a recovery mechanism to the operator. After accounting all above factors the failure probability from the action part works out to be very low.

4.1.7 Parametric Study

Parametric study or to put it straight a ‘sensitivity analysis’ was carried out to see the effect of variation of values of coefficients and PSFs on the result of the analysis. For instance if the cognitive process is assumed to be skill based, then the corresponding value of the coefficients Ai, Bi and Ci are 0.407, 0.7 and 1.2 respectively. Accordingly, the failure probability for the cognitive part works out to be ~ $3.1*10^{-3}$ /d. It shows that the result of the analysis is very sensitivity to the parameters considered for the analysis. Similarly, if the cognitive processing is considered to be skill based (as in previous case) and the median time taken is changed from 34 sec to 50 sec then the failure probability comes to $8*10^{-2}$ / d. In this manner more cases could be analyzed using the model given in this report. Based on the observations made out in this parametric study it the result of this study could be said to be conservative and suitable for incorporation in the main fault tree of the digital protection system.

4.1.8 Result

Keeping in view that the i) cognitive processing is rule based and the operator has median time of 34 sec to trip the reactor manually when he has the available time window of 79 sec and ii) the contribution from the action part is relatively low **the failure probability of manual actuation of reactor trip has been estimated to be $7.5*10^{-2}$ / demand. Keeping in view the uncertainty associated with the result an error factor of 5 has been assumed for this estimate.**

4.2 Manual Error in Calibration

This type of error, as can be seen in the list on classification of type of errors, which occurs prior to the accident, is classified as type 1 error. Since, such conditions do not deal with any urgency and provide enough time to the operator (here it means a person who is performing the calibration / checking of instrument). Hence, the ‘Technique for Human Error Rate Prediction’ (THERP) approach has been used for estimating the failure probability of committing error during test / calibration in the digital processors. Since, the ‘drift free operation’ is one of the important features of the digital system, the possibility of time related drift phenomenon in the processor modules could be ruled out. However, the data entry job related to trip parameter settings prior to the reactor startup (or during the routine fuel cycle) leave some room for error in setting the trip parameter bounding values in the bi-stable processors module of DPPS. The error may be committed due to any of the causes like, the error in using the procedure, and / or wrong entry of set points from computer to the processor chip. As could be gathered from the discussion with the designers, the operator follows the written procedures for data entry. Also, an independent checker verifies the entry and related aspects. So there exists an independent recovery mechanism. Apart from this, provision exists in the digital system module for automatically checking the upper / lower bounds of the processor parameters based on the design safety limits.

4.2.1 Fault tree basic event

Human error in setting the value of the process trip parameter such that the trip parameter bounding values are set outside the allowable outage range in the bi-stable processor module.

4.2.2 Task description

The calibration of the bi-stable processor is carried out prior to the reactor start up and the data are considered to be fuel cycle specific. The operator follows the written and approved procedure to perform this task and the whole job is independently supervised. The job involves are entry of data into the computer and transfer of these set point data to the processor module

4.2.3 Failure criteria

Setting of trip parameter values outside the allowable limit by the operator which goes undetected by the checker/supervisor.

4.2.4 Initial Conditions

The opportunity to commit this type of error exists prior to reactor start up when the fuelling specific settings are configured for the process parameters. One fuel cycle has been assumed to last for 18 months. These settings remain unaffected during one fuel cycle.

4.2.5 Calibration procedure

The designers were consulted for the procedure adopted for entry of trip setting from computer to the digital system processors. Accordingly, two broad steps were considered for this evaluation;

a) Select the applicable procedure for entry of trip setting,
b) Mark the check list as part of the procedure
c) Enter the data in the Trip Entry parameter window in the computer
d) Transfer the data to the processor module
e) Mark the check list
f) Freeze the setting.

It may be noted that the task is of routine nature and of low criticality hence nominal PSFs will be considered.

4.2.6 Quantification

As could be seen in the step a), the operator may committee error in selecting the right procedure. This involves verification of the applicable procedure for specific fuel cycle. According to the THERP [2] approach the probability of using wrong written procedure for routine task / surveillance has been given as $5 * 10^{-2}$ /d. with an error factor of 5. Again in step c) the operator enters the trip setting of the parameter. The table no. 20-6 of THERP handbook indicates a failure probability of $3 * 10^{-3}$ for entering the wrong (in unsafe direction) set point in to the bi-stable processor. Hence the probability of setting wrong setting by the operator could be given as

` $$= 5 * 10^{-2} + 3 * 10^{-3} = 5.3 * 10^{-2} \text{ /d}$$

There is an independent checker / supervisor to ensure that settings have been correctly entered. Using the dependency equation as given in the earlier section of this report, a low dependence interaction could be assumed between the operator and checker and a conditional human error probability could be given as follows:

$$HEP_c = \frac{1 + 19 * 0.1}{20} = 0.15$$

Hence, the probability of setting wrong value during the calibration of trip parameters in the bi-stable processor could be given as:

$$= 5.3 * 10^{-2} * 0.15 = 8.0 * 10^{-3} \text{ /d}$$

Here, the credit was been taken for the detection of the error of calibration during routine trip / alarm checking. It was assumed that weekly trip and alarm checks will be performed on the routine basis. Since, the bi-stable processors are accessible during reactor operations the correction could be incorporated after detecting the miscalibration. As given in IAEA-TECDOC-592, the credit for these weekly check could be taken by modifying the basic value with 0.01 for each successive week passed. Accordingly, the final result works out as follows:

$$= 8 * 10^{-3} * (1/4)(1+10^{-2}+10^{-4}+10^{-6} \ldots .) = 2 * 10^{-4} \text{ /d}$$

4.2.7 Calibration Error for Analog Instrumentations

As the data and information required for the modeling of calibration error for analog instrumentation was not available it was assumed that the this value will be 10 times higher than the digital instrumentation calibration. Accordingly, the probability of calibration error in analog instrumentation has been worked out as $2.0 * 10^{-3}$ /d.

4.2.8 Result

The human error probability of carrying out wrong calibration taking into account dependence between the operator and checker has been worked out as $2.0 * 10^{-4}$ /d with an error factor of 5 and the value for calibration error for analog instrumentation based on the assumptions worked out as $2.0 * 10^{-3}$ /d with an error factor of 5

5. Conclusions

Human Reliability Analysis has been performed for the two operator actions namely a) operator failed to initiate manual reactor trip consequent to the failure of digital system to initiate automatic reactor trip and b) Human error of calibration for digital as well as analog instrumentations, i.e. setting wrong value of trip parameters for specific fuel cycle. Keeping in view the design considerations that operator action in tripping the reactor is a diverse mechanism; all care has been taken in selecting the associated PSFs and coefficients in the calculation. The result could be considered to be on conservative side. The uncertainty in the result of error of calibration is expected to be more as many times the calculation parameters were based on the expert opinion for the want of exact information, hence, rather high value of error factor ~ 5 was assumed. Both these events form part of the protection system fault tree analysis. The assumption that the calibration error in analog instrumentation is 10 fold more is based on expert opinion and the figure quoted in the literature.

Reference

1. *International Atomic Energy Agency: Case Study on the use of PSA methods : Human Reliability Analysis, IAEA-TECDOC-592, International Atomic Energy Agency, Vienna, 1991.*
2. *Swain, A.D. and Guttmann, H.E., : Handbook on Human Reliability Analysis with Emphasis on Nuclear Power Plants Applications,NUREG-CR-1278, Sandia National Laboratories, Aug. 1983.*

서 지 정 보 양 식

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초록 (15-20줄내외)

이 보고서는 한국형 가압 경수로인APR 1400의 디지털 보호계통에 대한 신뢰도 분석의 일환으로 수행한 인간신뢰도 분석 관련 내용을 기술하고 있다. 디지털 원자로 보호계통은 필요한 상황이 발생하면 원자로를 자동 정지하게 된다. 그러나, ATWS와 같은 경우 원자로 정지의 다변화 수단으로 운전 원의 수동정지가 필요하게 된다. 그러나, ATWS와 같은 경우 원자로 정지의 다변화 수단으로 운전 원의 수동정지가 필요하게 된다. 이 보고서에서는 원자로 정지 관련 인적오류와 원자로 보호계통 계장설비의 Calibration 오류에 대한 분석 내용을 기술하였으며 필요한 경우 파라미터 분석을 포함하였다. 이 보고서는 한국형 가압 경수로인APR 1400의 디지털 보호계통에 대한 신뢰도 분석의 일환으로 수행한 인간신뢰도 분석 관련 내용을 기술하고 있다. 디지털 원자로 보호계통은 필요한 상황이 발생하면 원자로를 자동 정지하게 된다. 그러나, ATWS와 같은 경우 원자로 정지의 다변화 수단으로 운전 원의 수동정지가 필요하게 된다. 이 보고서에서는 원자로 정지 관련 인적오류와 원자로 보호계통 계장설비의 Calibration 오류에 대한 분석 내용을 기술하였으며 필요한 경우 파라미터 분석을 포함하였다.

주제명키워드 (10단어내외)	Human Reliability Analysis, Pressurized Water Reactor, Protection System, Manual Action

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Abstract (15-20 Lines)

A case study on human reliability analysis has been performed as part of reliability analysis of digital protection of the Korean Advanced Pressurized Water Reactor – APR 1400. The digital protection system of the reactor automatically actuates the shutdown system of the reactor when demanded. However, the safety analysis takes credit for operator action as a diverse mean for tripping the reactor for, though a low probability, ATWS scenario. Based on the available information two cases, viz., human error in tripping the reactor and calibration error for instrumentations in protection system, have been analyzed. Wherever applicable a parametric study has also been performed.

Subject Keywords (About 10 words)	Human Reliability Analysis, Pressurized Water Reactor, Protection System, Manual Action