HUMAN RELIABILITY ANALYSIS OF THE TEHRAN RESEARCH REACTOR USING THE SPAR-H METHOD

by

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The purpose of this paper is to cover human reliability analysis of the Tehran research reactor using an appropriate method for the representation of human failure probabilities. In the present work, the technique for human error rate prediction and standardized plant analysis risk-human reliability methods have been utilized to quantify different categories of human errors, applied extensively to nuclear power plants. Human reliability analysis is, indeed, an integral and significant part of probabilistic safety analysis studies, without it probabilistic safety analysis would not be a systematic and complete representation of actual plant risks. In addition, possible human errors in research reactors constitute a significant part of the associated risk of such installations and including them in a probabilistic safety analysis for such facilities is a complicated issue. Standardized plant analysis risk-human can be used to address these concerns; it is a well-documented and systematic human reliability analysis system with tables for human performance choices prepared in consultation with experts in the domain. In this method, performance shaping factors are selected via tables, human action dependencies are accounted for, and the method is well designed for the intended use. In this study, in consultations with reactor operators, human errors are identified and adequate performance shaping factors are assigned to produce proper human failure probabilities. Our importance analysis has revealed that human action contained in the possibility of an external object falling on the reactor core are the most significant human errors concerning the Tehran research reactor to be considered in reactor emergency operating procedures and operator training programs aimed at improving reactor safety.

Key words: human reliability analysis, standardized plant analysis risk – human method, Tehran research reactor, probabilistic safety analysis

INTRODUCTION

In spite of all preventive and mitigative measures considered in the design of nuclear reactors, they still represent a residual risk to the outside world. To reduce it, probabilistic safety analysis (PSA) has been used as a powerful method for the survey of nuclear reactor safety. In addition, any meaningful PSA needs to account for human action (HA) and their effects, both in the probability of risk significant events, as well as their consequences. This is because HA is an unavoidable part of the operation and maintenance in a nuclear power plant (NPP), both in normal and abnormal situations [1]. A Reactor safety study [2] revealed that more than 60% of the potential accidents in nuclear industry are related to human errors. Also, in some references, the contribution of human errors to PSA results were reported to be as high as 88% [3] (accidents at the three miles island (TMI), in 1979, and Chernobyl, in 1986, have yielded additional information about the importance of human reliability [4, 5]).

As for research reactors, the situation is even worse because of the role humans play in ensuring the safety of such installations. Many safety functions, performed automatically in power plants, must be performed manually in research reactors.

Human reliability analysis (HRA) as a part of PSA is defined as follows [6]: human reliability represents the probability of a person (1) correctly performing an action required by the system in the required time and (2) not performing any extraneous activity that could degrade the system. Any method by which human reliability is assessed may be designated as HRA [7]. The analysis typically includes the following phases: (1) identification of HA, (2) modelling of important actions and (3) assessment of probabilities of HA. The identification and modelling of important HA, from the PSA point of view, most often take place as a part of system and accident sequence modelling, as demonstrated, for example, in [8].

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In general, there are three main approaches in HRA: task-related (discrete nodal) models, task-related (group action) models, and time reliability models. Based on these three approaches, a wide range of various HRA models and techniques are available, each with their own characteristics: technique for human error rate prediction (THERP), cause-based decision tree (CBDT) [9], human error assessment and reduction technique (HEART) [10], nuclear action reliability assessment (NARA) [11], standardized plant analysis risk-human (SPAR-H) [12], human cognitive reliability (HCR) [13], time reliability curve (TRC) [6], operator reliability experiments/human cognitive reliability time reliability curve (ORE/HCR TRC) [14], cognitive reliability and error analysis method (CREAM) [15], holistic decision tree (HDT) [16], technique for human event analysis (ATHEANA) [17], cognitive reliability and error analysis method II (CREAM II) [15], method for assessing the completion of operator's action for safety (Mermos) [18], and success likelihood index method (SLIM) [19].

In this research, SPAR-H was chosen from all other available methods for the HRA of the TRR for the following reasons:

– well-documented and systematic HRA system,
– human performance choices tabulated based on expert opinion,
– PSF selected via tables,
– HA dependencies accounted for, and
– method appropriate to the intended use

In addition, the method is one of the newest developed for HRA and its basic error rates are calibrated against other HRA methods, such as the TEHRP, accident sequence evaluation program (ASEP), HEART, *etc*.

The SPAR-H method is applied to the TRR, which is a 5 MW pool-type research reactor with light water as a moderator for a heterogeneous, solid fuel reactor, in which the water is also used for cooling and shielding. The reactor core is immersed in either of the sections of the two-sectioned concrete pool filled with water. One of the sections of the pool contains an experimental stall in which beam tubes and other experimental facilities converge. The other one is an open pool area for bulk irradiation studies. The pool is spanned by a manually operated bridge from which an aluminum tower supporting the reactor core is suspended. The control of the reactor is accomplished by the insertion or removal of neutron absorbing control rods suspended from control-drives mounted on the reactor bridge. Additional control is provided by the inherent negative temperature coefficient of the reactivity of the system. A general symbolic scheme of the TRR is presented in fig. 1.

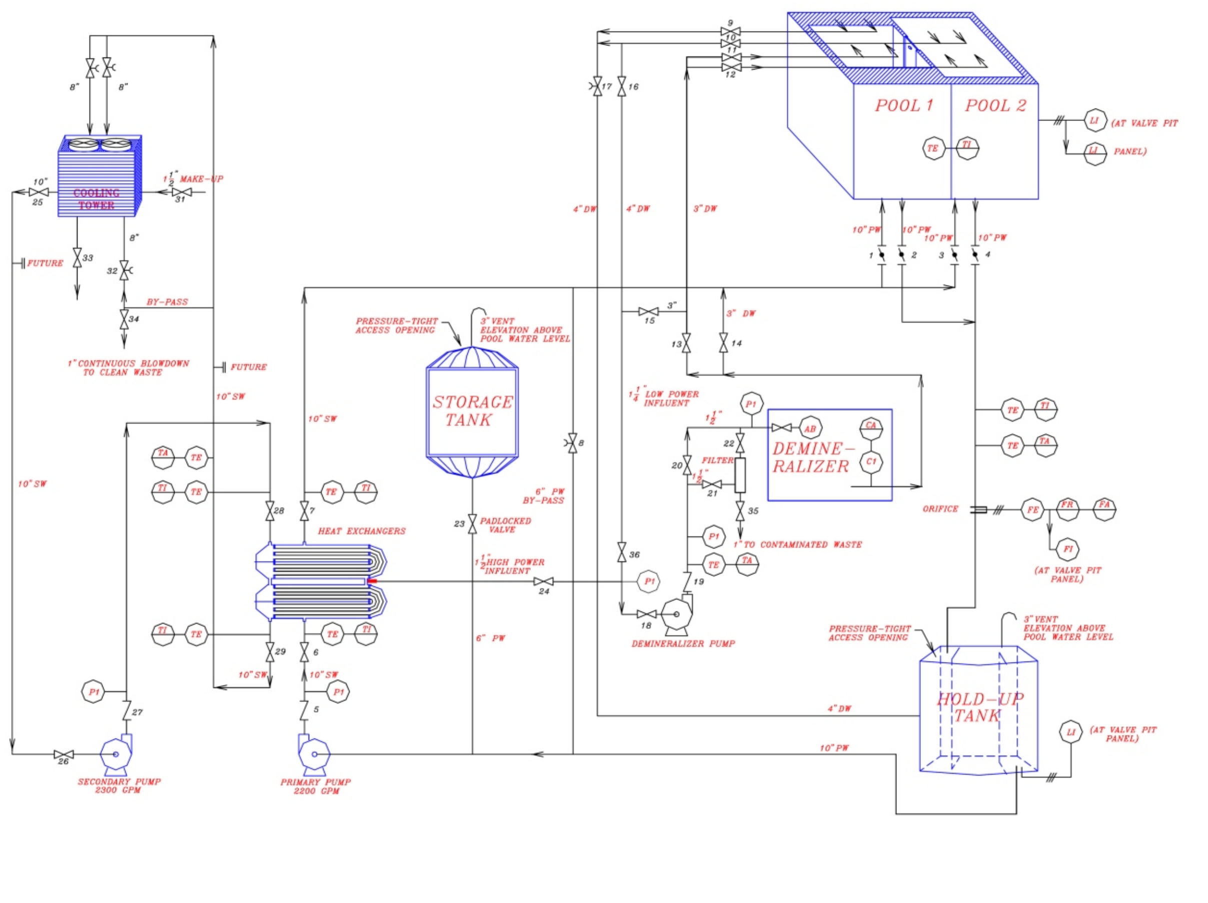

Figure 1. Symbolic scheme of the Tehran research reactor [20]

Its main components are the reactor core, control and safety systems, pool, holdup tank, pumps, heat exchanger, connecting pipes, check valves, gate valves and butterfly valves. Some of the main reactor data are outlined in tab. 1, while detailed specifications data are given in [20].

Our research has been organized as follows. *Human reliability analysis* describes the human reliability analysis and a brief description of the TRR. In Spar-H method, the SPAR-H method is addressed. The methodology for the quantification of HA is addressed in Methodology. The HRA for the TRR is considered in Human reliability analysis results for the Tehran research reactor. Results and discussion encompasses results and discussions.

HUMAN RELIABILITY ANALYSIS

The most important performance measure of interest in any PSA is human reliability. HRA is an important part of any risk analysis. It has long been recognized that human error has a substantial impact on the reliability of complex systems. To obtain a precise and accurate measure of system reliability, human error has to be taken into account.

Regardless of the methods applied, HRA must be performed within a defined general framework, nearly the same for all HRA. In other words, to perform a HRA, one has to perform a number of tasks. To put these tasks in the right order, a procedural framework has been developed. There are four phases, each of which contains a number of steps, summarized in tab. 2. Interested readers are to refer to [12, 21] for detailed information about the said steps.

For a more accurate modelling of HA in a PSA system, HA are classified as follows:

– category A: Pre-initiating event interactions (also called routine actions), *e. g*., maintenance of errors, testing errors, calibration errors,
– category B: IE-related interactions, *e. g*., human errors causing system trips, human errors causing loss of power), and
– category C: Post-initiating event interactions (also called emergency actions), *e. g*., all actions actuating a manual safety system backup of an automatic system.

Figure 2 depicts the locations of the various HA in a simplified logic tree. Categories A and B of HA are accounted for in the fault tree analysis as being the basic human error probabilities (BHEP), existent throughout generic databases [6, 22]. Upon that, the BHEP was modified with specific plant data by means of the Bayesian updating technique [23]. Almost all software for reliability and risk analysis contains a toolbox for updating data via the Bayesian approach. In this research, SAPHIRE [24] was applied for the task.

Table 1. Specifications and main operating conditions of the Tehran research reactor [20]

Core material	
Coolant	Light water
Fuel element	Plate-type clad in aluminum
Moderator	Light water
Nuclear fuel	Material test reactor (MTR) (low enriched uranium)
Reflector	Graphite/light water
Thermohydraulics	
Cladding thermal conductivity [$W^{-1}mK^{-1}$]	167.0
Cooling method	Forced flow
Fuel thermal conductivity [$Wm^{-1}K^{-1}$]	10.0
Holdup tank water volume [m^3]	37.417
Inlet coolant temperature [°C]	37.8
Pool water volume [m^3]	477.8
Primary cooling loop mass flow rate [m^3h^{-1}]	500
Pump head [m]	30.48
Secondary cooling loop mass flow rate [m^3h^{-1}]	522
Total heat transfer surfaces [cm^2] for standard fuel elements (SFE)	14,022.0
Total heat transfer surfaces [cm^2] for control fuel elements (CFE)	10,332.0
Total power peaking factor	3.0
Fuel element dimensions	
Fuel height [cm]	70.5
Fuel length [cm]	8.1
Fuel width [cm]	7.07
Number of plates in standard fuel elements	19
Passing cooling water cross-section [cm^2] at CFE	25.81
Passing cooling water cross-section [cm^2)] at SFE	33.92
Plate clad thickness [mm]	0.4
Plate clad height [cm]	61.5
Plate clad width [cm]	6.0
Plate meat [mm]	0.7
Water channel between plates [mm]	2.7
Fuel meat	
^{235}U [%]	12.44
^{238}U [%]	49.78
O [%]	11.17
Al [%]	26.50

* gpm – means gallon per minute

Category C human errors need to be accounted for comprehensively and in a plant-specific manner.

Table 2. Procedural framework for performing HRA

No.	Phase	Steps
1	Familiarization	Collection of information
		Plant visit
		Review of written procedures
2	Qualitative analysis	Identification of potential human errors
		Modelling of human errors in PSA
3	Quantification of human failures	Different methods in HRA (SPAR-H) in this research
4	Evaluation	Sensitivity and uncertainty analysis
		Recommendations
		Documentation

As stated above, in this research, the SPAR-H method was used to account for HA.

SPAR-H METHOD

The SPAR-H was a revision and a replacement of the U. S. Nuclear Regulatory Commission's accident sequence precursor (ASP) HRA screening method. The revisions made were intended to contribute to a more realistic characterization of human performance involving SPAR methods and techniques and to reflect the newest trends in HRA methods and data. Some of the goals of the SPAR-H include easy use and better representation of the uncertainty and dependency of the information gained for use in SPAR PRA models originating from the US NPP. The SPAR-H has been applied to over 70 US NPP. It was originally developed as a screening methodology, but the method was later on extended to full human error probability (HEP) quantification.

Task types

The SPAR-H model is built on years of experience of the authors in the nuclear energy field, especially in human factors and HRA. The underlying psychological basis for the SPAR-H construct is the informational model of humans. This being the case, they have in mind a diagnosis and action model for crew and personnel responses to accident conditions. They further realize that the responses of persons are affected by the context of the condition under which these persons operate. The model consists of probabilities associated with diagnosis and action *i. e*., HEP values as 0.01 and 0.001 for diagnosis and actions, respectively. An effective HEP consists of these elements, along with modifiers stemming from the context (PSF).

SPAR-H PSF

The terms PSF or performance influence factor (PIF) cover the same item and refer to anything that could increase or decrease performance (*i. e*. HA) and ,thus, error probability for a particular type of task. In this paper, only the term performance shaping factor is being used. PSF are hypothetical, since one does not know for certain if they will have a particular effect in a specific situation and consulting with operators and experts results in better PSF assignment. SPAR-H

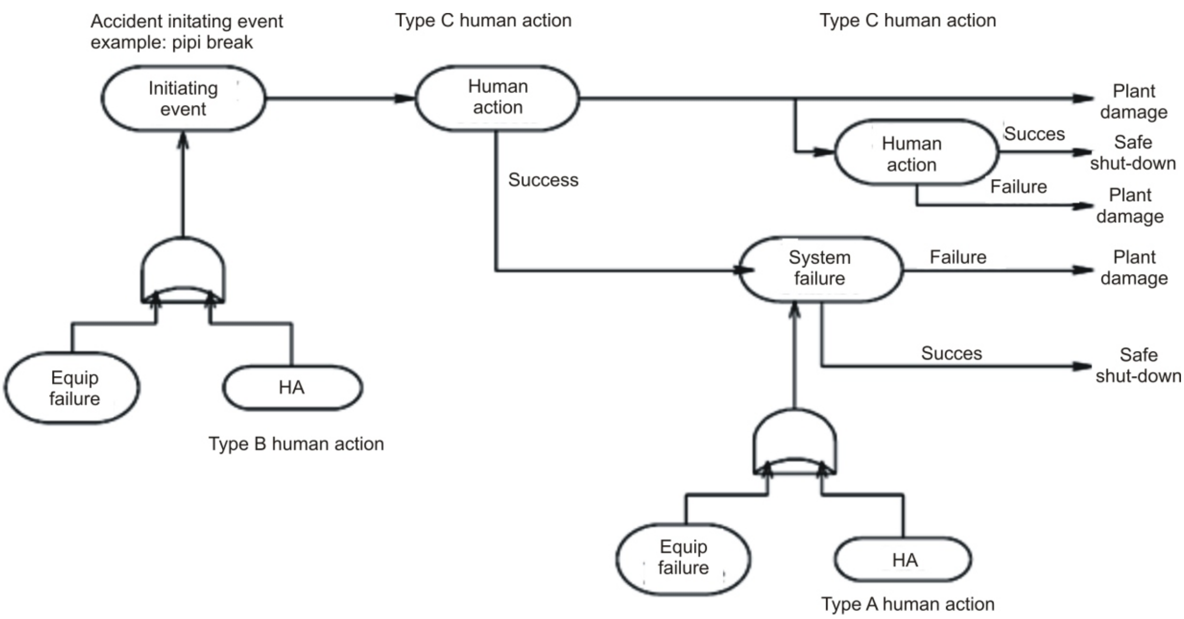

Figure 2. The location of human error types within the plant logic tree [25]

deals with this shortcoming in a systematic way, which is explained here.

SPAR-H is based on an information-processing model of human cognition, yielding a causal model of human error. SPAR-H also provides a discussion of the interdependencies of PSF, often ignored in other HRA methods. This being said, the interdependencies are not available to the reader in terms of correlation coefficients. The eight PSF applied to both the action and diagnosis phase used by the method are:

– available time,
– stress/stressors,
– complexity,
– experience/training,
– procedures,
– ergonomics/human-machine interface,
– fitness for duty, and
– work processes

Those interested can find detailed explanations for each PSF in [12].

Dependency

In 1994, a dependency method was developed that yielded a dependency rating from zero to complete dependency. These levels were then matched to the nomenclature in THERP. In 2003, the SPAR-H method was again updated, this time to allow for analysts to acknowledge additional aspects of context when considering dependency. The approach is meant to highlight those actions or diagnoses that should be further reviewed and for which higher failure rates can be assumed.

Table 3 presents the dependency table that analysts use to assign the dependency level.

Note: If the error is the 3rd error in a sequence, then the dependency is at least moderate. If the error is the 4th error in a sequence, then the dependency is at least high.

The final task failure dependency is calculated via well known dependency equations addressed in [6].

Caution: dependencies are analyzed based on the analysis after minimal cut sets used for calculating the frequency of core damage consequences for all initiating events. So, it is obvious that the dependency analysis must be postponed and performed after minimal cut sets are generated. Then, if the success of a task requires the success of ORed operator actions, (in which case, the operator error of the task is the failure of ANDed actions), dependency modelling is applied. This assumption is based on the belief that if the operator fails in the first step in a series or group of ORed actions, it is more likely that he will fail in subsequent steps of the group. In that regard, the nominal HEP (multiplied by the performance shaping factor) is applied to the first step. Upon this, different levels of dependency are derived applying aforementioned equations to the HEP of the first step.

Uncertainty analysis

To take uncertainty into account, the SPAR-H method employs a beta distribution which requires two parameters, α and β. A table of applicable α parameters (as functions of the mean HEP) is supplied in [26]. Figure 3 shows the numerical value of α as a function of the HEP. Once α is obtained, β, is found via the equation, $\beta = \alpha$ (1-HEP)/HEP.

Table 3. Dependence condition table [12]

Condition number	Crew (same (s) or different (d))	Time c – close in time nc – not close in time	Location (same or different)	Cues a – additional na – no additional	Dependency
1	s	c	s	na	complete
2				a	complete
3			d	na	high
4				a	high
5		nc	s	na	high
6				a	moderate
7			d	na	moderate
8				a	low
9	d	c	s	na	moderate
10				a	moderate
11			d	na	moderate
12				a	moderate
13		nc	s	na	low
14				a	low
15			d	na	low
16				a	low
17					zero

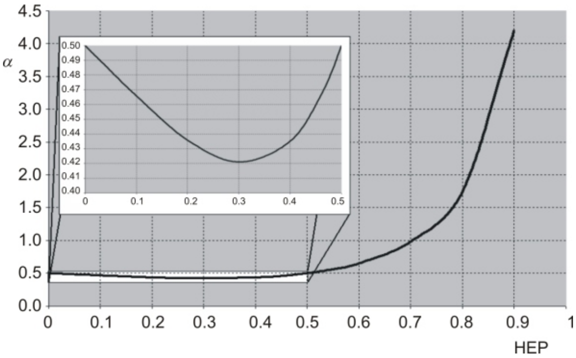

Figure 3. Alpha (α) as a function of the mean HEP

SPAR-H method in quantification

Final HEP values are arrived at by multiplying the nominal HEP (NHEP) (*i. e.* 1.0E-02 for the diagnosis phase and 1.0E-03 for the action phase) by the weighting factors derived from tab. 4 and tab. 5, for the diagnosis and action phase, respectively. This process is carried out for diagnosis and action items, and the overall value is given by the addition of both the diagnosis and action contributions. Note that, in the case when the number of PSF (for which the weighting factor is greater than 1.0) is greater than or equal to 3, the base HEP value is given by the following formula: HEP = NHEP × $PSF_{composite}$/[NHEP ($PSF_{composite}$ – 1) + 1].

METHODOLOGY

After reviewing the PSA of the TRR to find HA, the quantification of each task is performed in following steps:

Category A HA

During operation, all stand-by safety systems which may be in a state of unavailability because of pre-accident human errors are searched for possible testing, maintenance and calibration activities. Once the errors are identified, a basic event is created in the relevant system fault tree near the unavailable component and quantified using the THERP approach as follows:

In accordance with the THERP approach (tab. 20.6), the possibility of a human error to be committed during the testing and maintenance of a safety system is taken to be *P* = 1.0E-2. Also, in consultations with TRR operators, it was established that the results of maintenance and checks of the safety system's protection and interlocks are entered into maintenance logs. They are checked by a supervisor, so a recovery from human error is possible and this is modelled as the recovery factor (RF). The value of the recovery factor is *RF* = 1.0E-01 (based on table 20.22 in THERP). Under the assumption of middle level dependency between operators, the value of RF1 could be modified by dependency equations mentioned earlier. Hence, the probability of a pre-accident human error regarding this component is *P* = BHEP RF1.

Note: Type-A human failure events are characterized as follows:

– misalignment of PSA components in their normal operation or standby status after testing and maintenance, and
– miscalibration after calibration activities

In contrast with NPP, in research reactors, misalignments of components are in many cases easily detected by the plant's personnel in the control room during operation and by the walk – around supervisor outside the pool. In addition, they are immediately corrected after detection. Therefore, there is a very low probability of these misalignment situations and an initiating event occurring at the same time. Thus, such type-A human failure events can be screened out (the same goes for calibration activities that are screened out due to criteria addressed above). But it is conservatively assumed that every safety system operating in standby mode during operation may be in an unavailability state because of a pre- accident human error, so a basic event is introduced into the fault tree of each safety system which is out of the screening criteria previously addressed in this paper.

Category B HA

Standard PSA do not distinguish the root cause of an initiator and, often, these types of events are not modelled separately in risk analyses. Usually, generic databases are used to assign a BHEP to these actions [6, 22]. Then, the BHEP will be modified with specific plant data by means of the Bayesian updating technique [23]. The objective of the Bayesian update method is to combine generic data and plant-specific data in such a way that the influence of the plant-specific data on the updated data increases with the lengthening of the period in which the data is collected or the number of failures increases. The method is especially useful if little plant-specific data is available or little confidence in the plant specific data exists. Almost all software for reliability and risk analysis contains a toolbox for updating data by the Bayesian approach. As mentioned, in this research, SAPHIRE was used for the task.

Category C HA

The recipe for quantifying category-C HA is as follows:

Table 4. PSF evaluation criteria for the diagnosis phase [12]

PSF	PSF levels	Multiplier for diagnosis	Criteria for evaluation
Available time	Inadequate time	*P*(failure) = 1.0	– Available time < required time – Deterministic safety analysis (DSA) results must be used
	Barely adequate time (˜2/3 nominal)	10	DSA results must be used
	Nominal time	1	– DSA results must be used – Nominal required time
	Extra time (between 1 and 2 nominal and > than 30 minutes)	0.1	DSA results must be used
	Expansive time (>2 nominal and >30 minutes)	0.01	DSA results must be used
	Insufficient information	1	Lack of sufficient information
Stress/stressors	Extreme	5	– The onset of the stressor is sudden – The stressing situation persists for long – Feeling of threat to one's physical well-being or to one's self-esteem or professional status – Accident sequences that go well beyond expected conditions (*e. g.*, a small loss of coolant accident (SLOCA) with failure of safety injection) – Catastrophic failures (due to potential or radioactive release) – LOCA, (loss of offsite power) LOOP, anticipated transient without scram (ATWS), steam generator tube rupture (SGTR)
	High	2	– Multiple instruments and alarm go off unexpectedly and at the same time – The consequences of the task represent a threat to plant safety – Transients
	Nominal	1	Normal plant operating conditions
	Insufficient information	1	Lack of sufficient information
Complexity	Highly complex	5	– There is much ambiguity as to what needs to be diagnosed (*e. g.*, a SLOCA which is not depressurized) – Many variables are involved, with concurrent diagnoses
	Moderately complex	2	– There is some ambiguity as to what needs to be diagnosed – Several variables are involved, perhaps with some concurrent diagnoses
	Nominal	1	– There is little ambiguity – Single or few variables are involved
	Obvious diagnosis	0.1	– The problem is so obvious that it would be difficult for an operator to misdiagnose it – Compelling cues such as SGTR
	Insufficient information	1	Lack of sufficient information
Experience/ training	Low	10	– Less than 6 months of experience and/or training – Level of experience/training does not provide adequate practice in those tasks – Level of experience/training does not expose individuals to various abnormal conditions
	Nominal	1	More than 6 months of experience and/or training – Level of experience/training provides an adequate amount of formal schooling
	High	0.5	Level of experience/training provides operators with extensive knowledge and practice in a wide range of potential scenarios
	Insufficient information	1	Lack of sufficient information

Table 4. (continuation)

Procedures	Not available	50	
	Incomplete	20	Information is needed that is not contained in the procedure or procedure sections
	Available, but poor	5	A procedure is available but it is difficult to use because of factors such as formatting problems, ambiguity, or such a lack in consistency that it impedes performance
	Nominal	1	Procedures are available and enhance performance
	Diagnostic/symptom oriented	0.5	Diagnostic procedures assist the operator/ crew in correctly diagnosing the event
	Insufficient information	1	Lack of sufficient information
Ergonomics/ HMI	Missing/misleading	50	The instrumentation is inaccurate
	Poor	10	The design of the plant negatively impacts task performance
	Nominal	1	The design of the plant supports correct performance, but does not enhance performance
	Good	0.5	The design of the plant positively impacts task performance
	Insufficient information	1	Lack of sufficient information
Fitness for duty	Unfit	P (failure) = 1.0	The individual is unable to carry out the required tasks, due to illness or other physical or mental incapacitation (*e. g*., having an incapacitating stroke).
	Degraded fitness	5	The individual is able to carry out the tasks, although performance is negatively affected
	Nominal	1	The individual is able to carry out tasks; no known performance degradation is observed
	Insufficient information	1	Lack of sufficient information
Work processes	Poor	2	Performance is negatively affected by the work processes at the plant (*e. g*., shift turnover does not include adequate communication about ongoing maintenance activities; poor command and control by supervisor(s); performance expectations are not made clear)
	Nominal	1	Performance is not significantly affected by work processes at the plant, or work processes do not appear to play an important role (*e. g*., crew performance is adequate; information is available, but not necessarily proactively communicated)
	Good	0.8	Work processes employed at the plant enhance performance and lead to a more successful outcome than would be the case if work processes were not well implemented and supportive (*e. g*., good communication; well understood and supportive policies; cohesive crew).
	Insufficient information	1	Lack of sufficient information

– evaluate the PSF for the diagnosis portion of the task, if any,
– calculate the diagnosis failure probability,
– calculate the adjustment factor, if negative multiple (3) PSF are present,
– record the final diagnosis for HEP,
– evaluate PSF for the action portion of the task, if any,
– calculate action failure probability,
– calculate the adjustment factor, if negative multiple (3) PSF are present,
– record the final HEP action,
– calculate task failure probability without formal dependence (Pw/od), and
– dependency analysis.

HUMAN RELIABILITY ANALYSIS RESULTS FOR THE TEHRAN RESEARCH REACTOR

First of all, all initiating events, along with their accident sequence modelling and fault trees for event trees' headings, as well as for core damage states, are considered in the PSA of the TRR in search of possible human actions.

Initiating events

Only internal initiating events, *i. e*. hardware failures in the plant or faulty operations of plant hardware through human error or computer software deficiencies have been considered. Two major categories of initiating events can be distinguished. Loss of coolant accident (LOCA) initiator is an event that directly causes loss of integrity of the primary coolant pressure boundary. Transient initiators are those that could create the need for a reactor power reduction or shutdown and a subsequent removal of the decay heat [27]. Based on the responses of the safety systems, we have considered 11 groups of initiating events of which four are LOCA initiators and the others are transient initiators [28]. Transient initiators are also subdivided into following categories:

Table 5. PSF evaluation criteria for the action phase [12]

PSF	PSF levels	Multiplier for diagnosis	Criteria for evaluation
Available time	Inadequate time	P(failure) = 1.0	Such as diagnosis phase
	Time available is the time required	10	
	Nominal time	1	
	Time available 5 the time required	0.1	
	Time available is 50 the time required	0.01	
	Insufficient information	1	
Stress/stressors	Extreme	5	
	High	2	
	Nominal	1	
	Insufficient information	1	
Complexity	Highly complex	5	
	Moderately complex	2	
	Nominal	1	
	Insufficient information	1	
Experience/training	Low	3	
	Nominal	1	
	High	0.5	
	Insufficient information	1	
Procedures	Not available	50	
	Incomplete	20	
	Available, but poor	5	
	Nominal	1	
	Insufficient information	1	
Ergonomics/HMI	Missing/misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
	Insufficient information	1	
Fitness for duty	Unfit	P(failure) = 1.0	
	Degraded fitness	5	
	Nominal	1	
	Insufficient information	1	
Work processes	Poor	5	
	Nominal	1	
	Good	0.5	
	Insufficient Information	1	

- loss of offsite power supply (LOPS),
- loss of flow, forced circulation unavailable (LFFCU),
- loss of flow, forced circulation available (LFFCA), and
- excess reactivity insertion (ERI).

Core damage states (CDS)

Core damage has been conservatively assumed to occur when the available thermohydraulic models cannot support a successful cooling-down of the reactor core, given the particular states of the various safety systems. More detailed calculations might indicate that in some cases core damage is not actually occurring. All accident sequences identified do not lead to the same degree of core damage. Depending on the initiating event, operating safety systems and indications stemming from the thermohydraulic analysis, eight states have been defined, of which two correspond to abnormal states (which do not lead to core damage), while the others belong to core damage states. The said eight states are described below as:

- CDS1: when the reactor shutdown takes place successfully, but the natural circulation system fails (with no primary heat removal),
- CDS2: when the reactor fails to shut down and there is no primary heat removal,
- CDS3: when the reactor fails to shut down in the case of a fuel channel blockage accident,
- CDS4: when the reactor does not shut down in the case of a reactivity accident, although the primary heat removal system works normally,
- CDS5: when the reactor does not shut down in the case of a reactivity accident and the primary heat removal system also fails,

– CDS6: when reactor shutdown takes place, but the core is bared because of a failure in the pool-isolation system in case of LOCA,
– CDS7: when the reactor shutdown does not take place, but the core is bared because of the failure of the pool isolation system in case of LOCA, and
– CDS8: when the reactor shutdown does not take place and both the Natural Circulation and Forced Circulation systems work normally but, because of the opening of the safety flapper, the core is bypassed.

Specific accident sequences consisting of an initiating event group, specific system failures and successes and possible human responses are defined here. These system failures are, in turn, modelled in terms of basic event component unavailability and human error, so as to identify the basic causes underlying them and to allow for the quantification of system failure probabilities (unavailability) and accident sequence frequencies. The list of event tree headings representative of different safety functions/systems is summarized in tab. 6.

Considering initiating events, core damage states, accident sequence modelling and fault trees for their headings, HA of the TRR are clarified and assigned numbers in tab. 7. Then, dependency analysis is taken into account to generate the final HEP. It is worth mentioning that after all minimal cut sets are generated, the dependencies are analyzed based on an analysis of minimal cut sets used for calculating consequences ending CDS for all initiating events. Probabilities of all Category C HA are set to 1. After this, minimal cut sets containing two or more HAs are identified. If the frequency of a minimal cut set is more than 1.00E-08, it will be analyzed for dependency. Table 8 shows identified human actions for 11 initiating events, along with dependency analysis.

With all HA clarified, an importance analysis is performed to rank the most significant HA to have a backfitting, both in emergency operating procedures and operator training programs for TRR.

Importance analysis

Importance analysis was used to determine the most important HA at the plant. There are different measures in importance analysis such as Fussell-Vesely (FV), risk reduction worth, risk achievement worth (RAW), and differential importance measure. Among them, FV and RAW are more common in analyses.

Fussell – Vesely importance

This measure was introduced by Vesely [29, 30] and later applied by Fussell [31]. The FV of compo-

Table 6. Event trees headings

No.	Heading description
1	Containment sealing
2	Emergency electrical power supply
3	Electrical power supply
4	Emergency ventilation
5	Forced cooling system
6	High power scram fail
7	High radiation scram fail
8	High power scram system common cause failures
9	Low flow scram fail
10	Manual shutdown
11	Natural circulation
12	Pool isolation system for loss of coolant accident 1 (LOCA1)
13	Pool isolation system for LOCA2
14	Pool isolation system for LOCA3
15	Pool isolation system for LOCA4
16	Pool level scram fail
17	Primary pump scram fail
18	Period scram fail
19	Reactor protection system for excess reactivity insertion
20	Reactor protection system for loss of flow, forced circulation available 2
21	Reactor protection system for loss of flow, forced circulation available 3
22	Reactor protection system for loss of flow, forced circulation unavailable 1
23	Reactor protection system for loss of flow, forced circulation unavailable 2
24	Reactor protection system for LOCA1
25	Reactor protection system for LOCA2
26	Reactor protection system for LOCA3
27	Reactor protection system for LOCA4
28	Reactor protection system for loss of power supply
29	Water recovery fail

Table 7. HA in TRR

No.	Human actions
1	Bypass high radiation scram
2	Detection of fuel channel blockage
3	Detection of high pool level
4	Determination of LOCA 1 procedure
5	Determination of LOCA 2 procedure
6	Determination of LOCA 3 procedure
7	Determination of LOCA 4 procedure
8	Detection of containment sealing necessity
9	Detection of excess reactivity insertion
10	Detection of LOCA1
11	Detection of LOCA2
12	Detection of LOCA3
13	Detection of LOCA4
14	Detection of low pool level
15	Forced circulation necessity
16	Hold up tank high level
17	Turning on generator

Table 8. Results of HRA for TRR

No.	Initiating event	Human action(s)	$\text{Alpha}_{w/od}$	$\text{Beta}_{w/od}$	Pw/od*	Pwd**
1	LOPS	17	4.99E-01	1.24E+02	4.00E-03	N/A***
		15	4.98E+01	1.06E+02	4.70E-03	5.45E-02
		8	4.99E-01	1.24E+02	4.00E-03	1.46E-01
2	ERI	9	4.19E-01	1.26E+00	2.50E-01	N/A
		1	4.68E-01	4.21E+00	1.00E-01	1.45E-01
		17	4.99E-01	1.24E+02	4.00E-03	1.46E-01
		15	4.99E-01	1.24E+02	4.00E-03	5.00E-01
		8	4.98E+01	9.92E+01	5.00E-03	5.00E-01
3	LFFCA1	2	4.77E-01	6.34E+00	7.00E-02	N/A
		17	4.99E-01	1.24E+02	4.00E-03	5.38E-02
		8	4.99E-01	1.33E+02	3.75E-03	1.46E-01
4	LFFCA2	17	4.99E-01	1.24E+02	4.00E-03	N/A
		8	4.98E+01	9.92E+01	5.00E-03	5.47E-02
5	LFFCA3	3	4.99E-01	1.66E+02	3.00E-03	N/A
		17	4.99E-01	1.24E+02	4.00E-03	5.38E-02
		8	4.99E-01	1.27E+02	3.90E-03	1.46E-01
6	LFFCU1	14	4.99E-01	1.66E+02	3.00E-03	N/A
		16	4.95E-01	3.25E+01	1.50E-02	6.42E-02
		17	4.99E-01	1.24E+02	4.00E-03	1.46E-01
		8	4.99E-01	1.18E+02	4.20E-03	5.02E-01
7	LFFCU2	14	4.99E-01	1.42E+02	3.50E-03	N/A
		17	4.99E-01	1.24E+02	4.00E-03	5.38E-02
		8	4.98E-01	1.04E+02	4.75E-03	1.46E-01
8	LOCA1	10	4.99E-01	1.66E+02	3.00E-03	N/A
		16	4.97E-01	4.92E+01	1.00E-02	5.9E-02
		17	4.99E-01	1.24E+02	4.00E-03	1.46E-01
		4	4.84E-01	9.19E+00	5.00E-02	5.25E-01
		8	4.99E-01	1.24E+02	4.00E-03	5.02E-01
9	LOCA2	11	4.99E-01	1.66E+02	3.00E-03	N/A
		16	4.97E-01	4.92E+01	1.00E-02	5.9E-02
		17	4.99E-01	1.24E+02	4.00E-03	1.46E-01
		5	4.84E-01	9.19E+00	5.00E-02	5.25E-01
		8	4.99E-01	1.33E+02	3.75E-03	5.01E-01
10	LOCA3	12	4.99E-01	1.66E+02	3.00E-03	N/A
		16	4.97E-01	4.92E+01	1.00E-02	5.9E-02
		17	4.99E-01	1.24E+02	4.00E-03	1.46E-01
		6	4.19E-01	1.26E+00	2.50E-01	6.25E-01
		8	4.99E-01	1.24E+02	4.00E-03	5.02E-01
11	LOCA4	13	4.99E-01	1.66E+02	3.00E-03	N/A
		16	4.97E-01	4.92E+01	1.00E-02	5.9E-02
		17	4.99E-01	1.24E+02	4.00E-03	1.46E-01
		7	4.19E-01	1.26E+00	2.50E-01	6.25E-01
		8	4.99E-01	1.24E+02	4.00E-03	5.02E-01

* Without format dependence, ** with dependence, *** not applicable,first in sequence and zero dependency

nent X is the fraction the baseline core damage frequency (CDF) would be reduced if component X was always available (never failed and never out of service)

$$FV(X) = \frac{CDF - CDF(X = 0)}{CDF} \qquad (1)$$

CDF is the baseline CDF, with basic events assigned probabilities, and CDF($X = 0$) is the CDF, setting probability equal to 0 for the basic events representing the component for which FV is calculated.

Risk achievement worth

Chadwell and Leverenz [32] discuss RAW (also referred to as risk increase factor – RIF) as a measure in which input variable probability or frequency is set to unity, and the effect of this change on the system risk is measured. Therefore, RAW is the ratio of the new (increased) risk to the baseline risk of the system when the probability of the specified risk element is set to unity.

$$RIF(X) = \frac{CDF(X = 1)}{CDF} \qquad (2)$$

Based on the importance analysis, the most significant HA for each initiating event are shown in tab. 9.

Table 9. Importance of HA

Core damage state	Human action (s)
CDS1	Detection of forced cooling necessity
	Turning on generator
CDS2	Detection of containment sealing necessity
CDS3	Determination of fuel channel blockage
	Detection of containment sealing necessity
CDS4	Detection of containment sealing necessity
CDS5	Forced cooling necessity
	Turning on generator
CDS6	Determination of LOCA 3 procedure
	Determination of LOCA 4 procedure
	Determination of LOCA 2 procedure
	Determination of LOCA 1 procedure
	Detection of containment sealing necessity
CDS7	Determination of LOCA 3 procedure
	Determination of LOCA 4 procedure
	Determination of LOCA 2 procedure
	Determination of LOCA 1 procedure
	Detection of containment sealing necessity
CDS8	Detection of containment sealing necessity

RESULTS AND DISCUSSION

As stated, tab. 8 shows identified human actions for 11 initiating events along with the dependency analysis, while tab. 9 shows the results for the importance analysis of the TRR. Also, to show the significance of operator actions at the TRR, calculations of total CDS were performed with all HA failures and successes (*i. e.* in failure, all HA set to 1). Results shows that the total core damage frequency of TRR is 1.8E-01 and 2.12E-05 for all HA set to failure and success, respectively.

Also, to justify the results of the importance analysis, some tests have been conducted as follows. In each CDS, analysis was performed with failure and success of human actions and the results obtained have been considered for possible backfitting in reactor emergency operating procedures and also operator trainings programs.

CDS1
- Detection of forced cooling necessity set to failure (*i. e.* set to 1) and the CDS1 frequency increased to 1.2E-6, compared with 1.4E-7 (in success mode).
- Turning on generator set to failure CDS1 frequency increased to 1.9E-7, compared with 1.4E-7.

CDS2
- Results with failure and success of detection of containment sealing necessity are 7.1E-8 and 8.3E-9, respectively.

CDS3
- Results with failure and success of determination of fuel channel blockage are 2.0E-5 and 2.1E-6, respectively.
- Results with failure and success of detection of containment sealing necessity are 1.9E-5 and 2.1E-6, respectively.

CDS4
- Results with failure and success of detection of containment sealing necessity are 2.2E-7 and 2.7E-8, respectively.

CDS5
- Results with failure and success of forced cooling necessity are 2.0E-9 and 2.6E-10, respectively.
- Results with failure and success of turning on generator are 1.9E-9 and 2.6E-10, respectively.

CDS6
- Results with failure and success of determination of LOCA 3 procedure are 2.3E-5 and 1.7E-5, respectively.
- Results with failure and success of determination of LOCA 4 procedure are 2.1E-5 and 1.7E-5, respectively.
- Results with failure and success of determination of LOCA 2 procedure are 2.0E-5 and 1.7E-5, respectively.
- Results with failure and success of determination of LOCA 1 procedure are 1.85E-5 and 1.7E-5, respectively.
- Results with failure and success of detection of containment sealing necessity are 1.77E-5 and 1.7E-5, respectively.

CDS7
- Results with failure and success of determination of LOCA 3 procedure are 7.7E-10 and 7.3E-11, respectively.
- Results with failure and success of determination of LOCA 4 procedure are 7.5E-10 and 7.3E-11, respectively.
- Results with failure and success of determination of LOCA 2 procedure are 7.2E-10 and 7.3E-11, respectively.
- Results with failure and success of determination of LOCA 1 procedure are 7.0E-10 and 7.3E-11, respectively.
- Results with failure and success of detection of containment sealing necessity are 6.5E-10 and 7.3E-11, respectively.

CDS8
- Results with failure and success of detection of containment sealing necessity are 4.8E-6 and 4.4E-7, respectively.

CONCLUSIONS

This work is an important step in our quest to enhance the operation safety of the TRR. In consultation with the operators, a comprehensive study of HA was performed using SPAR-H as a systematic method in human reliability analysis for a true representation of human errors at the TRR. Also, importance analysis showed the most significant HA that should be taken into account in order to improve the safety of the TRR. In our future works, we intend to use the concepts of this paper in live probabilistic safety assessment of the TRR.

According to the conducted HRA, results obtained and the discussion section, it is clear that humans play a crucial role in TRR safety. This is obvious when we see that the total CDF of the plant has increased to1.8E-01, setting all HA to failure compared with 2.12E-05.

Also, considering the CDS results with HA set to failure and success, it is clear that among HA, the determination of fuel channel blockage and detection of containment sealing necessity in CDS3 which are representative of an external object falling on the reactor core and determination of LOCA 3 procedure, determination of LOCA 4 procedure, determination of LOCA 2 procedure, determination of LOCA 1 procedure and the detection of containment sealing necessity in CDS6, are the most significant HA demanding great attention in emergency operating procedures and operator training programs for the TRR.

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АНАЛИЗА ЉУДСКЕ ПОУЗДАНОСТИ ИСТРАЖИВАЧКОГ РЕАКТОРА У ТЕХЕРАНУ КОРИШЋЕЊЕМ SPAR-H МЕТОДЕ

Циљ овог рада је да размотри анализу људске поузданости на Истраживачком реактора у Техерану, користећи прикладну методу за приказивање вероватноће људске грешке. У досадашњем раду примењена је техника за предвиђање учесталости људске грешке као и методе стандардизоване анализе ризика (људске поузданости) код електрана, које се широко користе у нуклеарним електранама ради квантификовања различитих категорија људских грешака. Анализа људске поузданости је заиста интегралан и значајан део студија пробабилистичке анализе сигурности и без ње ова анализа сигурности не би била систематична и комплетна представа стварних ризика у електрани. Додатно, могуће људске грешке код истраживачких реактора чине значајан део постојећег ризика оваквих постројења и њихово увођење у анализу сигурности је сложен задатак. Стандардизована анализа ризика може се користити за суочавање са оваквим проблемима јер је добро документована и систематична са табелама могућих перформанси људи, које су припремљене уз сарадњу са експертима из ове области.У овој методи фактори обликовања перформанси бирани су из табела, урачунате су зависности људских акција и метода је прилагођена за намењену употребу. У сарадњи са оператерима на реактору, идентификоване су људске грешке и одговарајући фактори обликовања перформанси придружени су како би се добиле вероватноће људских грешака. Наша анализа је показала да људске акције садржане у вероватноћи да страни објекат падне на језгро реактора су најзначајније људске грешке које се тичу овог реактора, те треба да буду унете у реакторске процедуре за случај акцидента, као и у програме обуке техничара који за циљ имају побољшање безбедности реактора.

Кључне речи: анализа људске поузданости, стандардна анализа ризика код електрана, истраживачки реактор у Техерану, анализа сигурности