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Original Article

Risk-informed approach to the safety improvement of the reactor protection system of the AGN-201K research reactor

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ABSTRACT

Periodic safety reviews (PSRs) are conducted on operating nuclear power plants (NPPs) and have been mandated also for research reactors in Korea, in response to the Fukushima accident. One safety review tool, the probabilistic safety assessment (PSA), aims to identify weaknesses in the design and operation of the research reactor, and to evaluate and compare possible safety improvements. However, the PSA for research reactors is difficult due to scarce data availability. An important element in the analysis of research reactors is the reactor protection system (RPS), with its functionality and importance. In this view, we consider that of the AGN-201K, a zero-power reactor without forced decay heat removal systems, to demonstrate a risk-informed safety improvement study. By incorporating risk- and safety-significance importance measures, and sensitivity and uncertainty analyses, the proposed method identifies critical components in the RPS reliability model, systematically proposes potential safety improvements and ranks them to assist in the decision-making process.

1. Introduction

The routine review of nuclear power plants (NPPs) is a primary means for ensuring safety. As such, some countries have initiated systematic safety reassessments, known as periodic safety reviews (PSRs), to assess the cumulative effects of plant aging and modifications, operating experience, technical developments and siting aspects [1]. A PSR is a comprehensive safety review of all of the important aspects of safety, including the implementation and timescale of safety improvements and an assessment of plant design and operation against applicable safety standards and operating practices to ensure safety throughout the NPPs operating lifetime [1].

Among the fourteen safety factors considered in the PSR, the probabilistic safety assessment (PSA) is safety factor number 6 under *safety factors relating to safety analysis* [1]. PSA is a tool for analyzing the safety of complex systems and has been used extensively to investigate the safety of NPPs [2–6]. PSAs are also useful for allocating limited resources during design, while maintaining safety in risk-informed applications [7–10]. During review, a PSA can be done to identify weaknesses in the design and operation of the NPP, and to evaluate and compare proposed safety improvements. A PSA provides important insights to the risk-informed decision-making process [11] when evaluating the potential outcomes of alternative safety measures.

PSRs have been enacted in Korea for both research reactors and commercial NPPs to improve safety standards since the Fukushima accident. The practical performance of a PSA for an entire research reactor is difficult, due to the scarce data availability. One system that is important for the safety of research is the reactor protection system (RPS). The RPS is more safety-critical for zero-power research reactors not having forced decay heat removal systems than for NPPs.

Several risk-informed design optimization approaches have been developed and applied to the NPP system design process, mostly for the system conceptual design phases [7–10,12]. However, in Ref. [13] the authors have highlighted that risk information has not been used in the design of instrumentation and control (I&C) architecture; rather, a conservative approach has been adopted because of safety concerns. As research reactors are primarily used for research and experimental work, they produce little revenue compared to NPPs, which lead to the need for cost optimization while preserving the required level of safety. In addition,

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Nomenclature

Acronyms and Abbreviations

AGN	Aerojet general nucleonics
AVR	Automatic voltage regulator
BI	Birnbaum importance
CCF	Common cause failure
CDF	Core damage frequency
DC-PS	Direct current power supply
EF	Error factor
ETA	Event tree analysis
FPGA	Field programmable gate array
FTA	Fault tree analysis
FV	Fussell–Vesely
GDC	General design criteria
HE	Human error
HEP	Human error probability
I&C	Instrumentation and control
IAEA	International Atomic Energy Agency
IEC	International electrotechnical commission
ISA	International Society of Automation
KHU	Kyung Hee University
NIC	Neutron instrumentation channel
NPP	Nuclear power plant
PSA	Probabilistic safety assessment
PSR	Periodic safety review
RAW	Risk achievement worth
RPS	Reactor protection system
RRW	Risk reduction worth

Notations

M	Number of models
m	Model index
j	Basic event index
r_j	Reliability of the basic event j
U_p^m	Point estimate unavailability of model m
FV_j^m	FV importance of basic event j in model m
BI_j^m	BI importance of basic event j in model m
U_p^b	Point estimate unavailability of baseline model b
$U_{95^{th}}^m$	95% upper bound unavailability of model m
$U_{5^{th}}^m$	5% lower bound unavailability of model m
U_μ^m	Mean value unavailability of model m
α_m	Reduction factor of model m
β_m	Uncertainty factor of model m
T_m	Max allowable time until post-diagnosis action
T_a	Time for the post-diagnosis action
T_d	Time difference between T_m and T_a
HEP_{dp}	Diagnosis HEP
HEP_{pp}	Post-diagnosis HEP
HEP_{tp}	Total failure HEP in median value
HEP_{tp}	Total failure HEP in mean value
V_{tt}	Trip signal voltage
V_{tm}	Trip signal monitoring voltage
V_{mk}	Cathode monitoring voltage

the regulatory requirements for research reactors [14] are more flexible for I&C systems than for NPPs [15]. This indicates that a risk-informed design method for research reactor I&C systems may be advantageous. For this reason, a probabilistic approach to the design and optimization of the I&C architecture of research reactors has been proposed conceptually using RPS as a case study [13]. As an extension of this approach, a hybrid RPS architecture was, then, considered as a case study in Ref. [16], taking into consideration the benefits of both analog and digital configurations. However, the approaches proposed in Refs. [13,16] focused on the design phase, where the decision makers can formulate and evaluate various configurations to select an optimal configuration. Yet, the implementation of a completely new configuration on an operating research reactor is mostly infeasible and expensive. Rather, a systematic methodology to identify feasible modifications that can improve safety and availability of the baseline RPS configuration, without excessive changes to the existing configuration, is necessary.

In this view, this work presents a risk-informed procedure for safety improvement of the RPS of the AGN-201K zero-power research reactor. The AGN-201K is a research and educational reactor located at the global campus of Kyung Hee University (KHU), Republic of Korea and has been in operation since 1982 [17]. The design of the AGN-201K has inherent safety features by which decay heat is passively cooled to ensure that an accident cannot affect the integrity of the fuel assembly: thus, there is no need to consider the full scope PSA and, rather, only the availability analysis of the RPS is examined. Sensitivity analyses are performed on potential safety improvements. To increase the safety and reliability, the development of a monitoring system is, then, proposed for the most critical components identified.

The remainder of this paper is organized as follows. The risk-informed methodology is presented and discussed in Section 2. In Section 3, this is, then, applied to the case study of the RPS of the AGN-201K reactor. The results and discussions, including the sensitivity analyses performed on the safety improvements, are presented in Section 4. Concluding remarks are given in Section 5.

2. Methodology

2.1. Risk-informed safety improvement process

Generally, risk-informed applications involve utilizing the information provided by PSA and to make requirements more effective by using the risk information. Hence, they are applications which incorporate risk information into assessments and decision-making. They can also incorporate deterministic and non-risk information into the decision-making. However, while several insights other than insight from PSA can be considered in risk-informed decision-making, risk-informed design, considering the insight from PSA, has been used in literatures [8,10,12]. In Ref. [8], both the deterministic and probabilistic criteria are used for the design of the emergency core cooling system for future reactor systems, in which the general design criteria (GDC) regulatory requirement of design basis accidents and core damage frequency (CDF) were used for deterministic and probabilistic criteria, respectively, for the selection of the alternative design option. In Ref. [10], risk-informed design of International Reactor Innovative and Secure (IRIS) using level-1 PSA was presented for conceptual design phase. The analysis of the design alternatives for the design optimization of APR+ during low power shutdown operation was performed based on PSA criteria in Ref. [12]. The cost of the alternative design options, in addition to the unavailability, has been considered also as an analysis criteria in Refs. [13,16] during the conceptual design phase of the research reactor RPS. However, as stated in Section 1, it is worth repeating here that the full scope PSA is not considered in this paper, rather, only the availability analysis of the RPS is evaluated. This is because the design of AGN-201K has inherent safety features by which decay heat is passively cooled to ensure that an accident cannot affect the integrity of the fuel assembly. Furthermore, the current operating RPS of AGN-201K has been designed with three redundant safety channels as detailed in Section 3, hence, satisfying the deterministic criteria of single failure GDC requirement. Therefore, the present work is not to design and formulate difference RPS design options; rather, it is to analyze the present operating RPS of AGN-201K and systematically proposes the potential safety improvement options based on risk-informed approach by utilizing the risk information from unavailability analysis.

An overview of the proposed risk-informed safety improvement process for the RPS of a representative operating research reactor is presented in Fig. 1.

System reliability importance measures can be used to develop effective risk-informed safety improvements. These measures can quantify the risk- and safety-significance of components (basic events), given their role within the system and their important characteristics [18–20]. Some of the most commonly used importance measures are risk achievement worth (RAW), risk reduction worth (RRW), Fussell–Vesely (FV) and Birnbaum Importance (BI) [18–20]. In this work, risk is defined in terms of RPS unavailability, i.e. the probability that the RPS fails to trip the reactor when demanded. Risk- and safety-significance were regarded as complementary ways for characterizing the importance of basic events for the system risk [20,21]. FV has often been used as measure of risk-significance. BI is often preferred over RAW to represent safety-significance [22], as it is independent of the present value of the system unavailability. In this work, FV and BI measures were, thus, used to characterize the importance of basic events for the risk of RPS failure to trip the reactor when demanded.

The failure rates associated to the basic events and used to quantify the system's unavailability were mostly sourced from generic failure databases, and have a certain degree of uncertainty. Uncertainty analysis [23] was informed to incorporate the uncertainties of the failure rates associated to the basic events and propagate them onto uncertainty in system unavailability. The unavailability analysis was performed by Fault Tree Analysis (FTA) using AIMS-PSA developed by the Korea Atomic Energy Research Institute (KAERI) [24]. The results were, then, analyzed and a systematic modification to the baseline RPS was proposed through the information obtained using the basic events identified as critical components contributing most to the unavailability of the RPS. The potential solution proposed to improve the safety and availability of the RPS has been examined critically to ensure that there is no significant change to the structure and configuration of the RPS. Potential safety improvement options considered include:

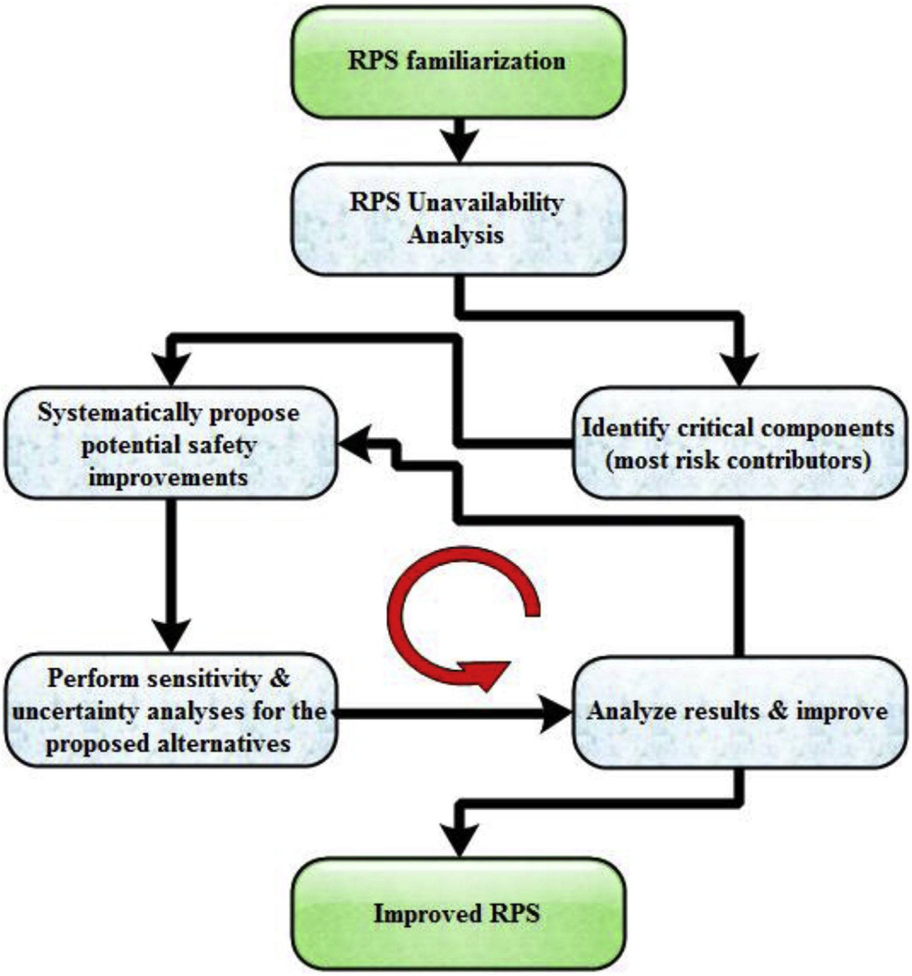

RPS familiarization
RPS Unavailability Analysis
Systematically propose potential safety improvements
Identify critical components (most risk contributors)
Perform sensitivity & uncertainty analyses for the proposed alternatives
Analyze results & improve
Improved RPS

Fig. 1. Overview of the proposed risk-informed safety improvement process.

(a) reduction of test interval
(b) increase of condition monitoring of components
(c) replacement of critical components with components of higher reliability
(d) increase in channel redundancy
(e) change of architectural configuration.

However, options (d) and (e) were not considered because their implementation in an operating research reactor was deemed too costly and, therefore, infeasible. Therefore, only options (a) through (c) were eventually considered.

A sensitivity analysis was, then, performed on the proposed improved system. The results were analyzed and compared with the baseline configuration to verify whether or not the availability has been improved. The details of the procedure are discussed in the next subsection.

2.2. Analysis procedure for the determination of improvement options

Let M represent the number of option models including the baseline model, where each model represents a specific RPS configuration. FV and BI were used as a measure of risk- and safety-significance, respectively, defined as

$$FV_j^m = \frac{U_p^m(base) - U_p^m(r_j = 1)}{U_p^m(base)}, \tag{1}$$

$$BI_j^m = U_p^m(r_j = 0) - U_p^m(r_j = 1), \tag{2}$$

where FV_j^m is the FV importance of basic event j in option model m, $m = 1, 2, \cdots, M$, BI_j^m is the BI importance of basic event j in option model m, $U_p^m(base)$ is the present point estimate unavailability of option model m, quantified based on all basic events in option model m, $U_p^m(r_j = 1)$ is the decreased point estimate unavailability of option model m quantified with basic event j optimized or assumed to be perfectly reliable (i.e., the reliability of the basic event j, r_j is 1) and $U_p^m(r_j = 0)$ is the increased point estimate unavailability of option model m quantified with basic event j assumed to be failed (i.e., the reliability of the basic event j, r_j is 0).

Here we introduce two more metrics: α_m is the unavailability reduction factor of option model m as a way of showing improvement effectiveness, and β_m is the uncertainty factor of option model m which is the improvement variation when a design is taken.

$$\alpha_m = \frac{U_p^m(base)}{U_p^b(base)}, \tag{3}$$

$$\beta_m = \frac{U_{95^{th}}^m(base) - U_{5^{th}}^m(base)}{U_\mu^m(base)}, \tag{4}$$

where $U_p^b(base)$ is the point estimate unavailability of baseline model b, $U_{95^{th}}^m(base)$ is the 95th percentile of the present unavailability of option model m, $U_{5^{th}}^m(base)$ is the 5th percentile of the present unavailability of option model m, $U_\mu^m(base)$ is the mean value of the unavailability of option model m, α_m is the unavailability reduction factor of option model m, defined as the ratio of the point estimate unavailability of option model m to the point estimate unavailability of the baseline model and β_m is the uncertainty factor of option model m, defined as the ratio of the difference between the 95th and 5th percentiles of the present unavailability to the mean unavailability value of option model m. Since those uncertainty bounds are based on the confidence interval that are meant to estimate the degree of uncertainty in a sample statistic resulted from the Monte Carlo simulation of the system, the wider confidence interval means greater uncertainty level and narrower one indicates the opposite. Hence, a lower value of β_m for model m, compare to other models, implies a lower uncertainty level in model m.

3. Case study: AGN-201K

The AGN-201K research and educational reactor was considered for the case study. The AGN-201K is a zero-power reactor that has been in operation at the global campus of KHU in the Republic of Korea since 1982 with a rated power of 0.1 W [17]. The obsolete control system was refurbished from 2004 to 2007 with a governmental research fund [25]. During this time, the reactor power was up-rated, the old analog operational console and I&C parts were replaced, shielding walls were installed, and a new digital console, for monitoring purposes only, was installed [25]. All safety shutdown functions are kept by analog console with original safety logics. The upgraded reactor has a maximum thermal power of 10 W and has been in operation since October 2007 [25].

In addition to the shutdown system (RPS), the AGN-201K has inherent safety features, including high negative feedback effects and low excess reactivity, that can make the reactor subcritical if the shutdown system fails. An additional thermal fuse located at the central part of the core protects against abnormal power excursion; this fuse was designed to melt at 120 °C, before the fuel would melt at 200 °C, thereby making the bottom half of the core to drop down, resulting in subcriticality due to the separation of the core [25]. Nevertheless, as the AGN-201K is a homogeneous reactor with a core comprised of polyethylene homogeneously mixed with uranium dioxide, there is no forced decay heat cooling system. The RPS, thus, plays an important role in ensuring that the reactor is safely shut down in the event of an accident.

3.1. Reactor Protection System (RPS) description

The RPS of the AGN-201K research reactor has three single-wired neutron instrument channels connected to an analog console by two BF3 ionization chambers and one proportional counter. There are three shutdown signals from these chambers and three additional interlock shutdown signals: low temperature of shielding water, low level of shielding water and earthquake vibration signals. For safety consideration and simplicity, this work only considered the three redundant reactor safety neutron channels, whereas it excluded the interlock signals, which are related to system's operation. All components not part of the safety/protection function or that do not influence the safety function were excluded.

A block diagram of the three redundant safety channels, each comprised a neutron instrument channel, meter, sensitrol relay and reset button, is shown in Fig. 2. The neutron instrument channel of safety channel 1 comprises a proportional counter, pre-amplifier, main amplifier, and associated power-supply electronics, which are not shown in the diagram. Safety channels 2 and 3 comprise an ionization chamber, pre-amplifier, main amplifier, and associated power-supply electronics. The signals from the independent safety channels are fed to their respective independent meters whose outputs are input to their respective sensitrol relay. The trip signals from the sensitrol relay trigger the 6L6 power tube, which de-energizes the electromagnets holding the control rods (two safety rods and one coarse rod), allowing the fueled control rods to exit the core due to gravity and springs, thereby shutting the reactor down. Additionally, a period trip signal is generated from the period thyratron schematic if the reactor power increase rate exceeds the allowable limit. There are also two manual reactor trip

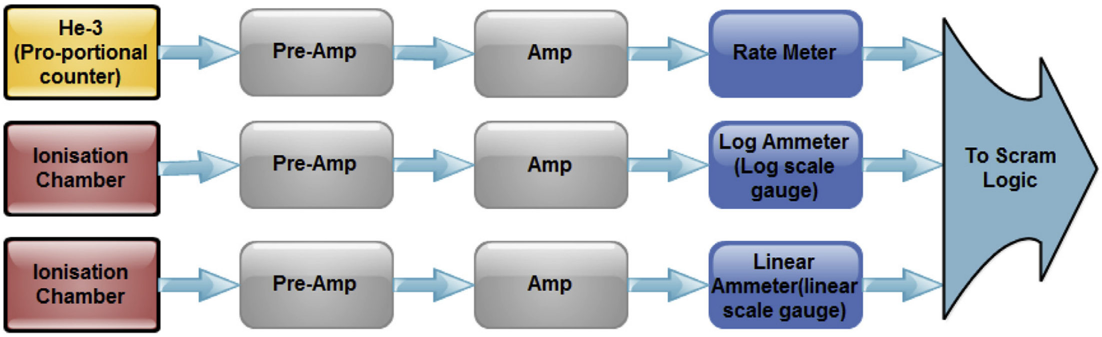

He-3 (Pro-portional counter)
Pre-Amp
Amp
Rate Meter
Ionisation Chamber
Pre-Amp
Amp
Log Ammeter (Log scale gauge)
Ionisation Chamber
Pre-Amp
Amp
Linear Ammeter(linear scale gauge)
To Scram Logic

Fig. 2. Block diagram of the AGN-201K's neutron instrument channels.

buttons that shut down the reactor when pressed by the operator. The I&C system is supplied from two power sources: automatic voltage regulator (AVR) 1 and 2. AVR #2 supplies power to the neutron instrument channels and their respective meters and AVR #1 supplies power to the other I&C components.

To obtain failure rate data, all components of the neutron instrument channel (i.e., the neutron detector, amplifier, and associated power supplies) were considered as a single component. This was done in accordance with the component boundary definition of the nuclear instrument channel given in the related IAEA document [26], where the boundary of a nuclear instrument channel includes the sensor (detector), power-supply electronics and associated signal amplifiers. As failure data specific for the AGN-201K are not available, generic failure rate data was used and the following assumptions were made when analyzing the RPS [27]:

1. The RPS failure was defined as the inability of the RPS to trip the reactor on demand by interrupting power to the electromagnets holding the control rods.
2. The system boundaries of analysis were defined to include the components within the trip signal paths from the neutron detectors to control rod drops that must operate successfully to trip the reactor when required. The paths of the other trip signals from interlocks were outside of the system boundary.
3. The failure of an offsite AC power source was not directly considered. Rather, the failure of internal power supplies were considered and modeled explicitly in the FTA. Based on assumption #1, modeling the offsite power loss in the fault tree was unnecessary.
4. Generic failure rate data from different sources were used for this analysis, as no specific AGN-201K failure rate data were available when this study was performed.
5. Manually tripping the reactor involves manual actuation of any of the push-button switches (two manual trip switches and four reset buttons). This action was assumed to be strongly coupled among the push-button switches. Failure to manually initiate a reactor trip was therefore modeled as a single operator error.

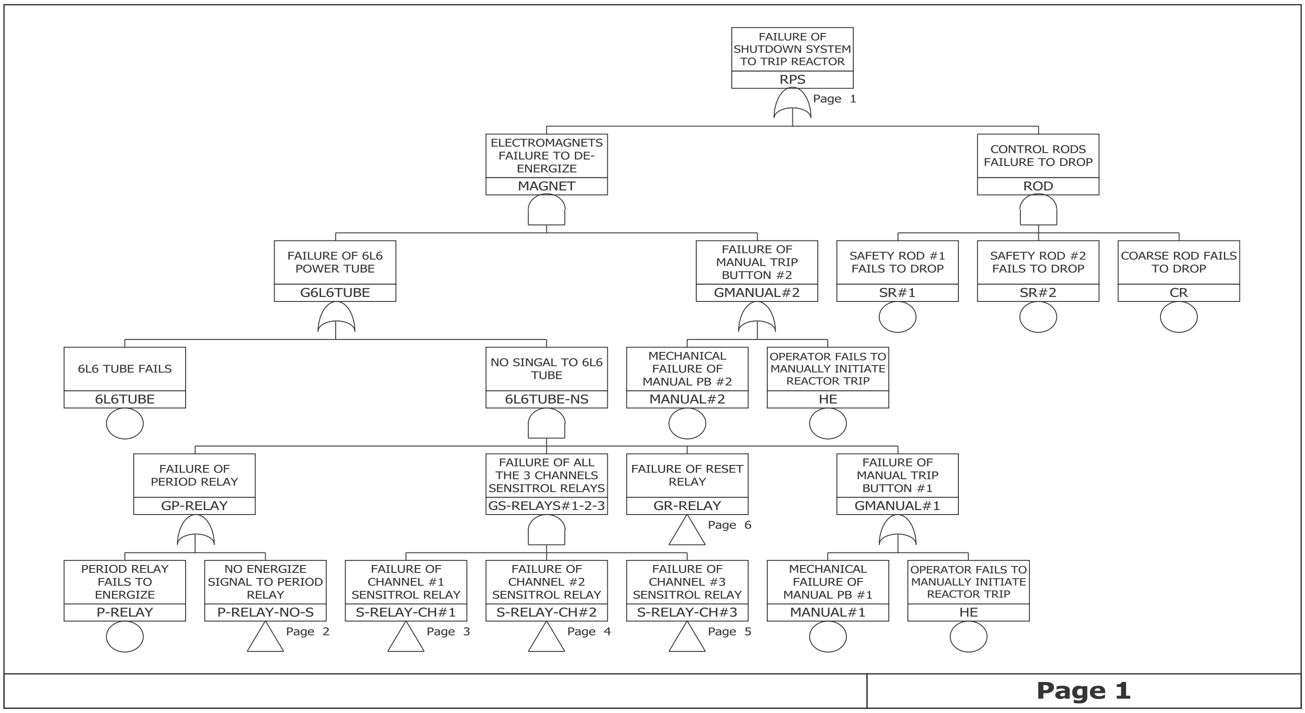

FAILURE OF SHUTDOWN SYSTEM TO TRIP REACTOR
RPS
Page 1
ELECTROMAGNETS FAILURE TO DE-ENERGIZE
MAGNET
CONTROL RODS FAILURE TO DROP
ROD
FAILURE OF 6L6 POWER TUBE
G6L6TUBE
FAILURE OF MANUAL TRIP BUTTON #2
GMANUAL#2
SAFETY ROD #1 FAILS TO DROP
SR#1
SAFETY ROD #2 FAILS TO DROP
SR#2
COARSE ROD FAILS TO DROP
CR
6L6 TUBE FAILS
6L6TUBE
NO SINGAL TO 6L6 TUBE
6L6TUBE-NS
MECHANICAL FAILURE OF MANUAL PB #2
MANUAL#2
OPERATOR FAILS TO MANUALLY INITIATE REACTOR TRIP
HE
FAILURE OF PERIOD RELAY
GP-RELAY
FAILURE OF ALL THE 3 CHANNELS SENSITROL RELAYS
GS-RELAYS#1-2-3
FAILURE OF RESET RELAY
GR-RELAY
Page 6
FAILURE OF MANUAL TRIP BUTTON #1
GMANUAL#1
PERIOD RELAY FAILS TO ENERGIZE
P-RELAY
NO ENERGIZE SIGNAL TO PERIOD RELAY
P-RELAY-NO-S
Page 2
FAILURE OF CHANNEL #1 SENSITROL RELAY
S-RELAY-CH#1
Page 3
FAILURE OF CHANNEL #2 SENSITROL RELAY
S-RELAY-CH#2
Page 4
FAILURE OF CHANNEL #3 SENSITROL RELAY
S-RELAY-CH#3
Page 5
MECHANICAL FAILURE OF MANUAL PB #1
MANUAL#1
OPERATOR FAILS TO MANUALLY INITIATE REACTOR TRIP
HE
Page 1

Fig. 3. System modeling fault tree for AGN-201K RPS (Top-level model).

6. Only post-accident tasks were considered in the human reliability analysis (HRA), where a screening HRA for post-accident tasks was employed.
7. Only a *failed-short* failure mode of the 6L6 power tube was considered, i.e., the failure to disconnect the current conduction between the anode and cathode when a trip signal is applied to the control grid terminal of the tube. The *failed-open* failure mode was not considered; this failure mode results in reactor shutdown and is, thus, fail-safe.

3.2. Unavailability analysis of RPS

An unavailability analysis was, then, performed to determine the probability that the shutdown system failed to trip the reactor when demanded.

3.2.1. System modeling and failure data collection

A system fault tree model was developed from the identification of the system failure criteria via analysis of scram logic. Fig. 3 shows the top-level fault tree model of the RPS of AGN-201K.

After the development of the system fault tree, failure data was collected for all of the basic events. Component failure data based on plant operating experience was not used; rather, generic data sources were used (see Assumption #4 in subsection 3.1). A list of components used and their respective failure mode, mean failure rate/probability, error factor (EF) and failure data reference sources is given in Table 1, where the EFs are calculated as the ratio of the 95th percentile to the mean failure rate/probability by adopting the definition of EF in the IAEA document [26]. The EF represents a quantitative measure of uncertainty associated with component failure rate/probability and was thus used in FTA for uncertainty analysis. In cases where only the mean failure rate was provided by the reference, the EF value was typically assumed as 2.4. This value is justifiable since the maximum EF value of all the components failure data collected from the IAEA research reactor failure database [28] was found to be 2.4, as can be seen in Table 1.

3.2.2. Common cause failures

Common cause failures (CCFs) were considered to represent multiple failures originating from a common cause that impacts the associated system's unavailability. A CCF event tends to nullify any redundancy incorporated in the design and can make the system incapable of tripping the reactor when demanded. A screening approach was used to identify the basic events and failure modes of CCFs that were most likely to contribute to AGN-201K RPS unavailability. The components selected for common cause treatment contained redundant partners and included the sensitrol relay, reset buttons and neutron instrument channel. Only the neutron instrument channels of safety channels 2 and 3 were considered during CCF modeling, as the neutron instrument channel of safety channel 1 had a different component composition, operating mechanism and neutron detector. The CCF of the AVR was not considered because the power-supply system basic event was explicitly modeled in the fault tree. The CCF of the manual trip button was also not considered because, even though the two buttons performed the same function, they were situated in different locations with a different operating mechanism.

Having identified the redundant components to be considered in the fault tree model, the alpha factor method was used to calculate the CCF failure data in the fault tree model. The generic CCF alpha factors used were obtained from Ref. [32] and are shown in Table 2.

3.2.3. HRA

A HRA was also performed to quantify the impact of human (operator) errors on RPS. The procedure outlined in NUREG/CR-4772 entitled “the Accident Sequence Evaluation Program Human Reliability Analysis Procedure” (ASEP HRA) was followed [33]. The ASEP HRA methodology is a simplified version of the HRA approach model from NUREG/CR-1278 [34] and is separated into different guidelines for pre- and post-accident tasks. Only post-accident tasks were considered here and the screening HRA for post-accident task procedures outlined in Tables 7–1 of NUREG/CR-4772 [33] was employed (see Assumption #6). Any post-accident operator action required for the system to successfully function when demanded was specified and added directly to the fault tree. In this case of AGN-201K RPS, the post-accident operator action identified as potentially either reducing or eliminating an abnormal event is manual reactor scram. The standard operating procedure of the AGN-201K directs the reactor operator to immediately scram the reactor upon any annunciation of abnormal event occurring

Table 2
Generic CCF alpha factors [32].

Alpha Factor	**CCCG = 2**		**CCCG = 3**	
	Rate CCF	Demand CCF	Rate CCF	Demand CCF
α_1	9.70E-1	9.80E-1	9.71E-1	9.79E-1
α_2	3.05E-2	1.95E-2	1.74E-2	1.45E-2
α_3			1.19E-2	6.28E-3

Table 1
List of components and their failure data.

S/No	Component	Event Name	Failure Mode	Failure Rate/prob. (mean)	EF	Ref.
1	Neutron Instr. Ch.	CH[b]X-NEUTRON-INT	Fail to function	8.89E-5/h	1.5	[28]
2	Rate Meter	R-METER	Fail to function	3.00E-6/h	2.4[a]	[29,30]
3	Log Meter	LG-METER	Fail to function	3.00E-6/h	2.4[a]	[29,30]
4	Linear Meter	LN-METER	Fail to function	3.00E-6/h	2.4[a]	[29,30]
5	Thyratron	P-THYRATRON	Fail to function	5.00E-5/h	2.4[a]	[30]
6	Sensitrol Relay	S-RELAY[b]X	Fail to function	8.30E-6/h	2.4	[28]
7	Reset Relay	R-RELAY	Fail to de-energize	1.25E-4/d	2.4	[31]
8	Period Relay	P-RELAY-F	Fail to energize	1.25E-4/d	2.4	[31]
9	6L6 Power Tube	6L6TUBE	Fail to function	2.00E-5/h	2.4[a]	[30]
10	Manual Scram BT	MANUAL[b]X	Fail to contact	1.25E-5/d	2.4	[31]
11	Reset Button	RESET[b]X	Fail to contact	1.25E-5/d	2.4	[31]
12	Single Rod Ass.	SR[b]1, SR[b]2, CR	Fail to drop	3.00E-5/d	2.6	[31]
13	Power Supply to I&C	AVR[b]1, AVR[b]2	Fail to function	5.00E-6/h	2.4	[28]
14	Rectifier (DC Supply)	DC-PS	Fail to function	1.14E-5/h	2.4	[28]

[a] These are assumed EF values.
[b] X represents #1, #2, or #3.

during normal operations. Consequently, manually scramming the reactor was considered a post-diagnosis task.

Results of this analysis are presented in Table 3, where T_m is the maximum allowable time to have correctly diagnosed the abnormal event and to have completed the required post-diagnosis actions to achieve the system success criteria established by systems analysts, T_a is the estimated time needed to get to proper locations and to perform any required post-diagnosis actions, and T_d is the estimated allowable time for a correct diagnosis that will still permit sufficient time to perform the required post-diagnosis actions prior to T_m, $T_d = T_m - T_a$. A value of 60 min was determined for T_m from the temperature profile along with time calculated by thermo-hydraulic simulation. In order to obtain conservative results, the maximum reactivity insertion was modeled. The initial condition was 0.9 W operating at the atmospheric conditions, considering fuel temperature feedback. Heat transfer modeling was also conservatively implemented under the assumption that only the radial direction of thermal conduction is allowed. In this case, the maximum temperature at the core centre was much less than the melting point and the temperature profile kept was less than its maximum temperature at least during 5 h; hence, the selected value of 60 min is justifiable. As the post-accident action was to be performed in the control room, T_a was assumed as 1 min, taken as the combined required travel and manipulation time for each control room action taken on the primary operating panels, which are normally in visual access of the control room operator, as stated in Tables 7–1 of NUREG/CR-4772 [33]. The appropriate diagnosis human error probability (HEP) (i.e., HEP_{dp}) and appropriate post-diagnosis HEP (HEP_{pp}) were estimated, and the total failure probability (HEP_{tp}) was calculated based on the procedure outlined in NUREG/CR-4772 [33]. As the obtained HEP_{tp} was a median value, it was converted to a mean HEP (HEP_{mp}) using Eq. (5) with a conservative EF value of 10 [27] for consistency with the failure data presented in Table 1. The result was, then, used in the FTA.

$$HEP_{mp} = HEP_{tp} exp\left[0.5\left(\frac{ln(EF)}{1.645}\right)^2\right] \tag{5}$$

3.2.4. Quantification results

In this section, the quantification results are presented and the basic events contributing most to the risk of failure of the RPS are identified. Based on the discussion with the AGN-201K operator, the testing interval used in this study was 1 month (720 h). Uncertainty analysis was performed to calculate the uncertainty that exists for RPS unavailability due to the uncertainty in the values used for the basic event failure rates/probabilities. The results are presented by the point estimate unavailability, mean value and EF in Table 4. The fault tree basic event uncertainties were propagated to obtain the RPS unavailability distribution. A Monte Carlo sample size of 100,000 was used for uncertainty propagation. It is importance to note that, the point estimate unavailability was calculated as the sum of the cusets' probabilities, while the mean unavailability value was estimated from the Monte Carlo simulation. The distribution of the unavailability of the RPS failure to trip the reactor on demand is shown in Fig. 4.

Table 4
Unavailability distribution result of baseline RPS.

Point estimate unavailability	Distribution			
	5%	mean	95%	EF
5.316E-4	1.704E-5	5.376E-4	2.038E-3	3.79

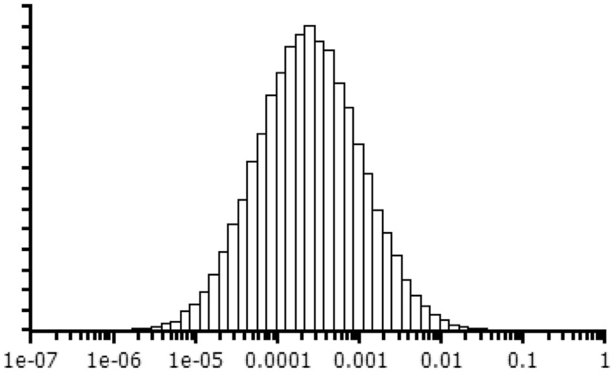


Fig. 4. Unavailability distribution of baseline RPS.

Five dominant cutsets contributing to the baseline RPS unavailability are presented in Table 5. The cutsets represent various combinations of basic events that prevent the AGN-201K reactor from tripping when demanded. The cutset including the failure of the 6L6 tube and of the operator to manually initiate the reactor trip contributes most to the RPS unavailability, nearly 72%. To identify individual basic event's contribution to the failure of the RPS, the FV and BI are used. In PSA, the FV could be high by either high basic event unavailability or weak defense in depth. When both FV and BI are high simultaneously for a particular basic event, safety can be improved by decreasing the basic event unavailability or by improving the defense in depth. Therefore, a plot of BI against FV can easily identify those potential components for safety improvement as demonstrated in Fig. 5. Fig. 5 shows a plot of BI against FV in a log scale for all the basic events in the baseline model, which indicates the fraction contribution of a basic event to the unavailability. Human error, 6L6 power tube failure and AVR #2 failure contribute the most and were, thus, identified as the potential basic events that can lead to safety improvements, if taken care of.

3.3. Safety improvements of RPS

Based on the results shown in Table 4, the quantified point estimate unavailability of the RPS baseline model was 5.316E-4. This value falls within the Safety Integrity Level 3 (SIL 3) of the reliability requirement specifications of the International Electrotechnical Commission (IEC) 61226: *Nuclear power plants—Instrumentation and Control important to safety—Classification of Instrumentation and Control functions, IEC 61508: Functional Safety of electrical/electronic/programmable electronic safety-related systems*, and the International Society of Automation (ISA), ISA 84.00.01: *Functional*

Table 3
Post-accident HRA result.

Action	T_m	T_a	T_d	HEP_{dp}	HEP_{pp}	HEP_{tp}	HEP_{mp}	EF
Manual Scram	60 min	1 min	59 min	0.001	0.01	0.01099	0.0266	10

Table 5
Dominant cutsets contributing to RPS unavailability (baseline).

No.	Cutset Probability	Basic Events	Event Description	Relative Contribution
1	3.83E-4	6L6TUBE **AND** Human Error (HE)	6L6 tube fails to disconnect and operator fails to manually initiate reactor trip (HE)	71.99%
2	9.58E-5	AVR#2 **AND** HE	AVR#2 fails to supply power to the designated safety channel's components and HE.	18.00%
3	1.40E-5	DC-PS **AND** CH#2-NEUTRON-INT **AND** HE	DC-PS originated from AVR#1 to other components fails, neutron instrument channel (NIC) #2 fails to function and HE	2.63%
4	7.86E-6	P-THYRATRON **AND** DC-PS **AND** HE	Period thyratron schematic fails to function, DC-PS originated from AVR#1 to other components fails and HE	1.48%
5	6.98E-6	CH#1-NEUTRON-INT **AND** CH#2-NEUTRON-INT **AND** CH#3-NEUTRON-INT **AND** HE	NIC #1 fails to function, NIC #2 fails to function, NIC #3 fails to function and HE	1.31%

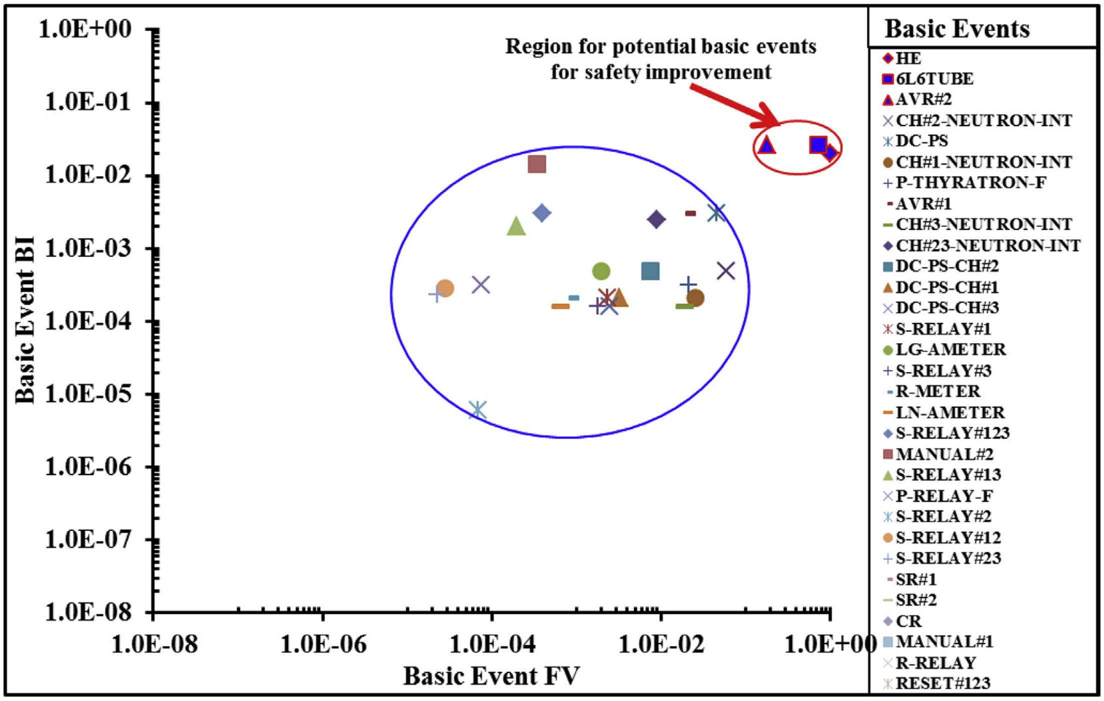


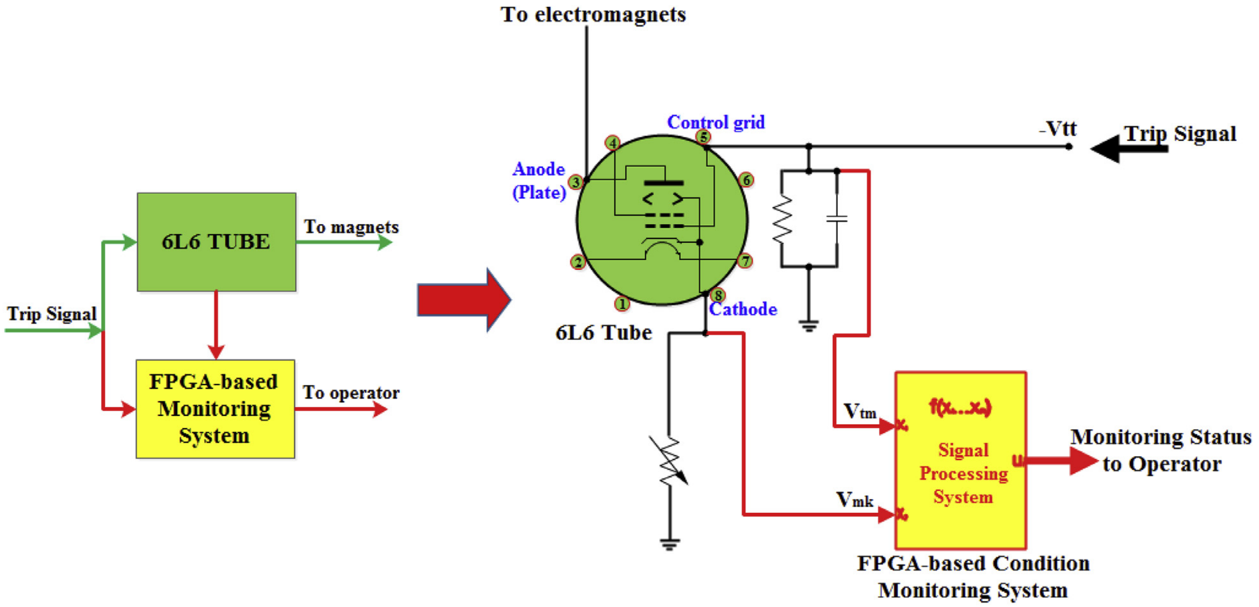
Fig. 5. Risk- and safety-significance metrics: plot of BI against FV in log scale.

Safety—Safety Instrumented Systems for the Process Industry Sector for safety systems. These standards require that the probability of failure on demand, i.e., the unavailability of the safety system, be between ($\geq 10^{-4}$ to $< 10^{-3}$) [35,36] for SIL 3 (high risk). Thus, the studied RPS satisfies these reliability specifications. However, if the uncertainties of the basic event failure rates are considered, the requirement is not strictly satisfied; although the mean unavailability value falls within the range of the requirement, the upper bound is out of the range, implying that a further improvement of the safety and availability of the RPS can be a good thing.

Potential safety improvements to the baseline model, based on the dominant contributors to the baseline RPS unavailability identified in subsection 3.2.4 were, therefore, systematically proposed and analyzed via a sensitivity analysis:

1. Increase of condition monitoring of components via the addition of a Field Programmable Gate Array (FPGA)-based monitoring system for the 6L6 power tube.
2. Add an AVR (AVR #3); i.e., connect the components and systems (NICs and meters) supplied by AVR #2 to AVR #3 via an automatic changeover device (or automatic power transfer switch). The power-supply systems are redundant to those components and systems via a power transfer switch in which only one of those AVRs at a time will be connected to those components. Thus, the 110 VAC power-supply system to the instrument channels and meters fails only if both AVR #2 and AVR #3 fail simultaneously, under the assumption that the failure of the automatic power transfer switch is negligible and fail-safe to AVR #2 or AVR #3.
3. Reduce the testing interval from one month to two weeks, one week, or one day.

A conceptual model of the FPGA-based monitoring system proposed for the 6L6 tube in improvement #1 is presented in Fig. 6. This monitoring system is proposed to monitor the functionality and operations of the 6L6 tube and return the status to the operator. The 6L6 tube is a beam-power-tetrode type, a vacuum tube with auxiliary beam-focusing plates designed to augment power-handling capability and help reduce unwanted emissions. Vacuum tubes are electronic devices that come in many forms, including diode, triode, tetrode, and pentode. Although low-power vacuum tubes have been largely replaced by solid-state



(a) Block diagram (b) Concept development diagram

Fig. 6. FPGA-based monitoring system for 6L6 power tube.

(semiconductor) devices, many vacuum tubes are still used in a variety of applications and continue to perform a valuable service at high powers and particularly at high frequencies in high-power radio transmitters. For the foreseeable future, if high power is required, electron/vacuum devices will remain the best solution [37].

The control grid terminal of the tube allows the flow of current from anode to cathode to be controlled. A block diagram of the proposed monitoring system is shown in Fig. 6(a) and the concept development showing pin configuration of 6L6 tube is shown in Fig. 6(b). The pins are: pin 2—and pin 7—Heaters (filaments), pin 3—Anode (also called Plate), pin 4—Grid 2 (Screen grid), pin 5—Grid 1 (Control grid), and pin 8—Cathode & beam-forming. Here, a current passed through the heater heats the cathode, causing it to emit electrons by thermionic emission. A positive voltage applied to the magnets between the anode and cathode causes a flow of electrons from the cathode to the anode through the two grids, thereby allowing a current flow. The control grid operates at a negative potential with respect to the cathode. By varying the applied voltage to the control grid, the current flow in the anode can be controlled. Hence, during normal operation, there is no trip signal and the control grid is connected to the ground potential, allowing current flow from the magnets at the anode through the tube to the cathode. In an abnormal condition or when a reactor scram is required, the trip signal is converted to a negative voltage ($-V_{tt}$) and triggers the control grid, limiting the flow of electrons from the cathode to the anode, to thereby stop the current flow and de-energize the magnet to shutdown the reactor.

As stated in assumption #7 in Section 3.1, two failure modes of 6L6 power tube exist: *failed-short* and *failed-open*. The *failed-open* mode is a fail-safe condition, in which the current flow is disconnected from the anode to the cathode terminal, thereby de-energizing the magnet and shutting down the reactor. In the *failed-short* mode, the current continues to flow through the magnets from anode to cathode when a reactor scram is requested and the trip signal is applied to the control grid terminal of the tube. This failure type is caused by an interelectrode short circuit [37], which allows current to flow regardless of the applied voltage at the control grid. The FPGA-based condition monitoring system in Fig. 6(b) is to be designed to monitor not only the *failed-short* mode, but the entire normal and abnormal operating conditions of the tube. Therefore, from Fig. 6(b), the following conditions are to be considered during the development and implementation, and processed by the monitoring system:

(a) *failed*-short mode: when $[(V_{tm}=V_{tt})\boldsymbol{AND}(V_{mk}>0)]$,
(b) *failed-open* mode: when $[(V_{tm}>V_{tt})\boldsymbol{AND}(V_{mk}=0)]$, and
(c) *normal* operating mode: when $[(V_{tm}>V_{tt})\boldsymbol{AND}(V_{mk}>0)]$.

A block diagram showing the addition of AVR #3 (i.e., proposed improvement #2) is shown in Fig. 7. The failure of the automatic power transfer switch was assumed fail-safe, as the switch can be either *failed-open* or *failed-short,* resulting in contact with AVR #2 or AVR #3, respectively. In either case, power is still supplied to the safety channels.

3.4. Sensitivity Analysis

The following sensitivity analyses of the proposed improvements were, then, investigated:

✓ sensitivity analysis 1: Impact of an FPGA-based monitoring system for the 6L6 tube on the baseline model,

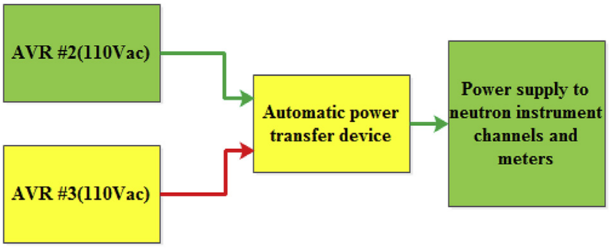

AVR #2(110Vac)
AVR #3(110Vac)
Automatic power transfer device
Power supply to neutron instrument channels and meters

Fig. 7. Block diagram of additional AVR connection (AVR #3, i.e., proposed improvement #2).

Table 6
Unavailability distribution from sensitivity analyses #1 to #3.

Model	Modification	Point Estimate Unavailability	Distribution			
			5%	Mean	95%	EF
Baseline	–	5.316E-4	1.704E-5	5.376E-4	2.038E-3	3.79
Sen. Analysis #1	Monitoring	1.490E-4	4.877E-6	1.484E-4	5.663E-4	3.82
Sen. Analysis #2	AVR#3	4.362E-4	1.307E-5	4.384E-4	1.674E-3	3.82
Sen. Analysis #3	Monitoring + AVR#3	5.357E-5	1.785E-6	5.364E-5	2.048E-4	3.82

- ✓ sensitivity analysis 2: Impact of AVR #3 in redundancy with AVR #2 on the baseline model,
- ✓ sensitivity analysis 3: simultaneous consideration of FPGA-based monitoring system and AVR #3 in redundancy with AVR #2 on the baseline model,
- ✓ sensitivity analysis 4: reduction of the testing interval from one month (720 h) to two weeks (336 h) on the baseline and other proposed models,
- ✓ sensitivity analysis 5: reduction of the testing interval from one month (720 h) to one week (168 h) for the baseline and other proposed models, and
- ✓ sensitivity analysis 6: reduction of the testing interval from one month (720 h) to one day (24 h) for the baseline and other proposed models.

As AVR #3 was proposed to be redundant with AVR #2, the failure data of AVR #2 are used for AVR #3. For FPGA-based monitoring system, the failure rate was conservatively assumed as in Ref. [38] as 2.08E-6 h^{-1}. This value was used for these analyses with an EF of 2.4. The results of these sensitivity analyses are presented in Section 4.

4. Results and discussion

The results of sensitivity analyses #1, #2, and #3 are presented in Table 6 for a Monte Carlo sample size of 100,000, as used in the baseline model. The unavailability of the baseline model was reduced by a factor of 0.2803, 0.8205 and 0.1008 due to the changes implemented in sensitivity analyses #1, #2 and #3, respectively. The implementation of the proposed improvements significantly reduced the unavailability of the AGN-201K RPS system. The proposed safety improvements implemented in sensitivity analysis #3, i.e., adding a monitoring system for the 6L6 tube and an AVR #3, led to a high reliability and availability of the RPS of AGN-201K. The scatterplot of the basic events BI against the corresponding FV for the model, shown in Fig. 8 in a log scale, shows that the BI values significantly reduced, indicating that the safety was improved significantly.

The unavailability results from sensitivity analyses #4 to #6 of the current model, the model including the monitoring system, the model including AVR #3, and the model including both the monitoring system and AVR #3 are shown in Table 7, at various test intervals. The total number of modeling options is 16. There are four options for the base model ($m = 1,\ 2,\ \cdots,\ 4$), and the additional options ($m = 5,\ 6,\ \cdots,\ 16$) which are arranged as shown in Table 7. The models were, then, ranked based on the reduction factor α_m, as shown in Fig. 9(a), the uncertainty factor β_m, as shown in Fig. 9(b), and the point estimate unavailability of model m, as shown in Fig. 9(c). Ranking via the reduction factor and point estimate unavailability (i.e., Fig. 9(a) and (c), respectively) produced the same ranking, as the reduction factors were derived from the point estimate unavailability of the models (see Eq. (3)). The rankings

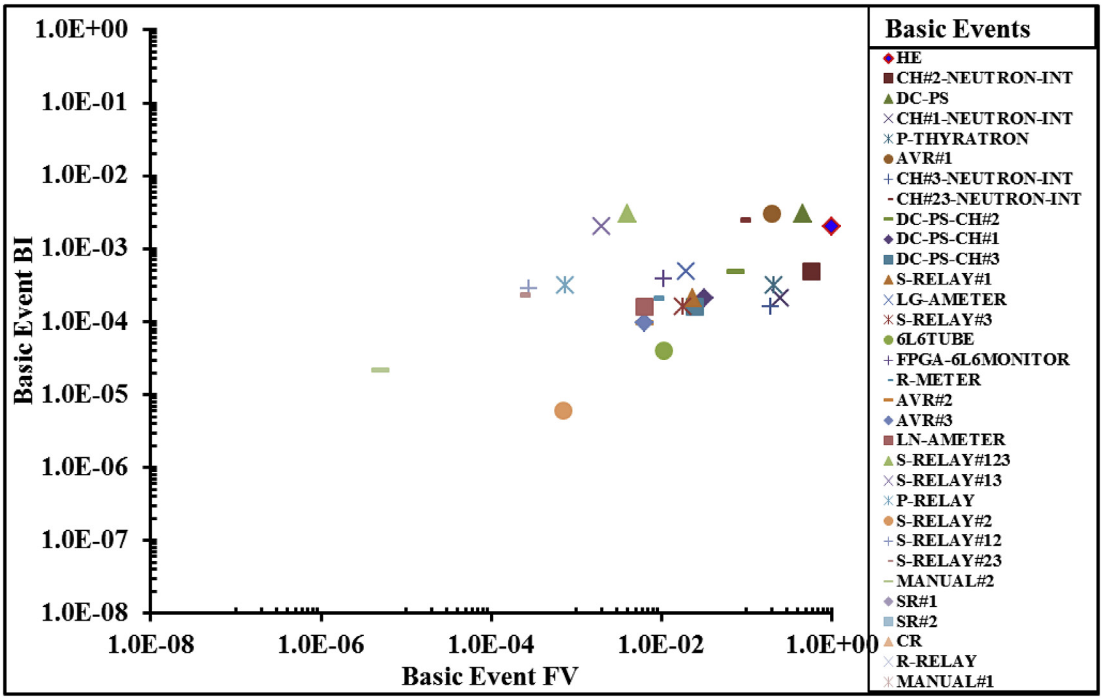
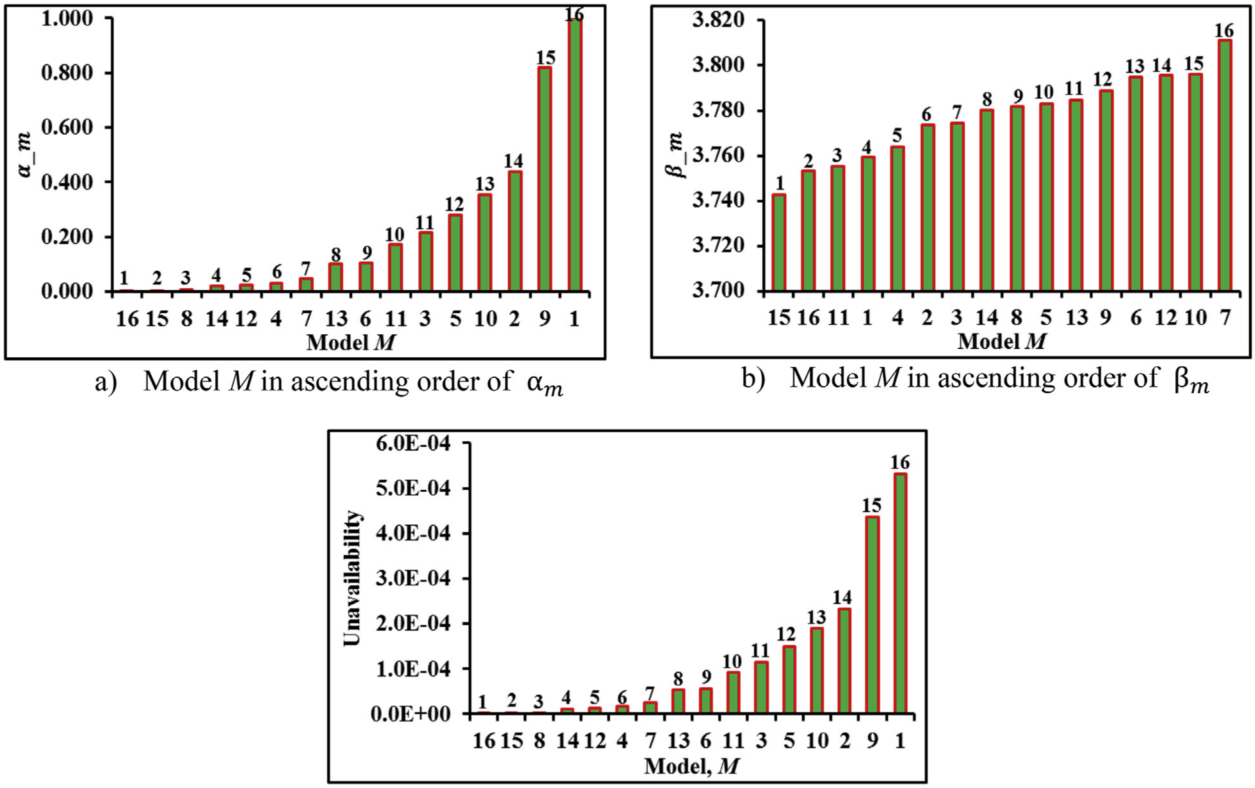


Fig. 8. Plot of BI against FV in log scale for the addition of both monitoring and AVR #3 (Sensitivity Analysis #3).

Table 7
Unavailability distribution of models from sensitivity analyses with various test intervals.

Model	*m*	Test Period	Test Interval (hours)	Point est. unavail.	Distribution			
					5%	mean	95%	EF
Base (Current PPS)	1	Monthly (Baseline)	720	5.32E-4	1.70E-5	5.38E-4	2.04E-3	3.79
	2	Bi-weekly	336	2.34E-4	7.34E-6	2.36E-4	8.99E-4	3.80
	3	Weekly	168	1.14E-4	3.55E-6	1.16E-4	4.40E-4	3.81
	4	Daily	24	1.60E-5	4.98E-7	1.58E-5	6.00E-5	3.80
Addition of Monitoring System	5	Monthly	720	1.49E-4	4.88E-6	1.48E-4	5.66E-4	3.82
	6	Bi-weekly	336	5.48E-5	1.69E-6	5.46E-5	2.09E-4	3.83
	7	Weekly	168	2.47E-5	7.32E-7	2.47E-5	9.47E-5	3.84
	8	Daily	24	3.24E-6	9.24E-8	3.24E-6	1.23E-5	3.81
Addition of AVR#3	9	Monthly	720	4.36E-4	1.31E-5	4.38E-4	1.67E-3	3.82
	10	Bi-weekly	336	1.89E-4	5.48E-6	1.88E-4	7.21E-4	3.83
	11	Weekly	168	9.18E-5	2.61E-6	9.28E-5	3.51E-4	3.78
	12	Daily	24	1.28E-5	3.59E-7	1.30E-5	4.96E-5	3.82
Addition of both Monitoring & AVR#3	13	Monthly	720	5.36E-5	1.79E-6	5.36E-5	2.05E-4	3.82
	14	Bi-weekly	336	1.02E-5	3.33E-7	1.02E-5	3.89E-5	3.81
	15	Weekly	168	2.39E-6	7.72E-8	2.40E-6	9.07E-6	3.77
	16	Daily	24	4.70E-8	1.50E-9	4.73E-8	1.79E-7	3.78



a) Model M in ascending order of α_m

b) Model M in ascending order of β_m

c) Model M in ascending order of unavailability

Fig. 9. Ranking of model *M* based on various measures.

produced by the uncertainty factor (Fig. 9(b)) were different because the uncertainty factor was calculated using uncertainty intervals and normalized by the mean uncertainty value of the respective model, and it is independent of the point estimate unavailability.

The results presented in Fig. 9 indicate that models 15 and 16 showed the lowest values of both unavailability and uncertainty. However, model 13 included the addition of a monitoring system and AVR #3 on a monthly test period and satisfied the reliability specifications of the IEC and ANSI/ISA standards, and was thus considered a better alternative than the models 14, 15 and 16, as a model that is reliable over a long test period would be preferred to those with shorter test periods. Taking the test interval into consideration, the recommended safety improvements options are, in descending order, models 13, 14, 6, 5, 10, and 2.

5. Conclusions

The RPS is most safety-critical in the I&C system of a research reactor, as it provides vital functions of protection and shutdown of the reactor. This work demonstrates a risk-informed methodology for the safety improvement of a RPS of an operating research reactor. Using the RPS of an operating AGN-201K research reactor as case study, unavailability, uncertainty, minimal cutsets, and risk- and safety-significance metrics were first analyzed. The results were, then, used to identify basic events that contributed most to the risk and safety of the RPS, and to propose several potential improvements to increase the reliability and availability of the RPS, while avoiding/minimizing tampering with safety channels. Based on the results of the sensitivity analyses performed on the potential safety improvement options, the safety and availability of the RPS have been significantly improved. Taking the longer test interval into consideration, the best top three potential safety improvement options are found to be model 13, 14, and 6, which are *addition of both monitoring system and AVR #3 on monthly test period, addition of both monitoring system and AVR #3 on bi-weekly test period, and addition of only monitoring system on bi-weekly test period*, respectively. This is because the model that is reliable over a long test period, having satisfied reliability requirement specification, would be preferred to those with shorter test periods.

The analysis of the procedure and the results of the case study indicated how safety can be improved in the RPS of an operating research reactor, while minimizing tampering with safety channels. The results of these analyses help to understand the safety-critical characteristics of the research reactor and to base any backfitting on a cost-benefit analysis for ensuring that only necessary changes are made.

Acknowledgments

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