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ANALYSIS OF CONTROL ROD STUCK INITIATING EVENT AT PAKISTAN RESEARCH REACTOR-2 BY USING PSA

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Probabilistic Safety Analysis (PSA) is an analytic method used for nuclear reactors safety. Large developments have been made in this field for better understanding about risks of those events about which there is often very little information is known. This analysis is being increasingly used to complement the deterministic approach in nuclear safety. This paper presents the work and results of level-1 PSA performed for Pakistan Research Reactor-2 (PARR-2) located in PINSTECH Complex Nilore, Islamabad, Pakistan. Event tree was drawn to study the response of the system against initiating event while fault tree were used for modeling of failure of the safety of system. Generic and maintenance data of PARR-2 were used to calculate the frequencies of the accident that may cause core damage. Thus the total frequency of core damage for control rod as initiating event was found to be 1.25E-06per reactor year. These results were used to identify the system and components of PARR-2 that are important for the safety the reactor.

Keywords: PSA, Reliability, Research reactor, MNSR, PARR-2, Risk analysis

1. Introduction

Probabilistic Safety Analysis (PSA) is now fundamental tool that provide guidance to safety related decision making. It is being increasingly used to complement the deterministic approaches to nuclear safety. This paper presents the results of Level-1 PSA performed according to the procedures suggested in relevant IAEA publication [1, 2]. PSA identifies the events and their sequence that can lead the core damage and quantifies their likehoods of occurrence.

PSA has been widely used in nuclear industry. The first comprehensive application of this method and techniques dates back to 1975 to NRC reactor safety study known as WASH-1400 [3]. Since then substantial methods and technique has become a standard tool in a safety of nuclear reactors. Being consist and integrated model source, PSA also provides insights into plant design performance and environmental impacts. PSA identify the dominant risk contributors and compare the different options that are available for risk reduction in nuclear power plants PSA being conceptual and mathematical tool drives numerical results.

2. Outline Methodology

The main steps of PSA that are used to identify the potential plant failure which leads to core damage and quantify their occurrence are following: hazard identification, accident sequence modeling, data assessment and parameter estimation and accident sequence quantification.

2.1. *Short description of reactor*

PARR-2 is a tank-in-pool type reactor with anominal power rating of 30 kW. It is moderated and cooled by demineralized light water through natural convection mode. This demineralized light water also acts as reflector and shielding agent. The reactor has peak power of 87 kW with 4 mk reactivity release in cold clean core with an inherent characteristic of self-limiting power. PARR-2 is fueled with highly enriched uranium (HEU) fuel pins which are about 90% enriched in ^{235}U. The fuel meat is uranium-aluminum alloy with aluminum alloy as cladding. Each fuel pin contains 2.88 g of ^{235}U [4]. The reactor core is surrounded by radial as well as axial beryllium reflectors to improve neutron economy in this under-moderated system.

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There is a single control rod in the reactor core that is used to control the neutron flux and hence the reactors power. The reactivity insertion through control rod can only be a ramp function of time. The coolant flow path is restricted by placing two orifice plates on the top and the bottom of the core. The restricted coolant flow adds the self power-limiting feature to this under moderated reactor. The coolant enters the reactor core through the gap between the lower orifice plate and the annular reflector, rises up due to buoyancy and leaves through the upper orifice plate while cooling the fuel pins.

The core of PARR-2 is an under-moderated array with hydrogen to fuel ^{235}U atomic ratio of about 201.2 at 20°C and provides a strong negative temperature and void coefficient of reactivity. The excess reactivity of cold and clean core (at 23°C) is limited to 4.0 mK .This reactivity is much less than effective delayed neutron fraction of 0.00795 which therefore, eliminate the possibility of prompt critical accident. Also the core has a fairly large prompt neutron life-time of 4.633×10^{-5}s [4] which results in large reactor periods due to perturbations or transients.

For cooling the heat rejected from the core is first transferred to the water in the tank and then to the outer pool by natural convection.

2.2. Initiating event selection

The initiating event (IE) identification and selection is an important task after the plant familiarization. Initiating event is an event that creates disturbance in the plant and has potential to lead the core damage depending upon successful operation of the various mitigating system in the plant. These event triggers the accident scenarios. In this study control rod stuck was taken as an initiating event.

2.3 Reactor safety system

Owing to the under moderated core design, large negative temperature and void coefficient of reactivity, natural convection cooling and small excess reactivity and large neutron life time, PARR-2 belongs to inherently safe reactors. Thus such design of this reactor incorporates a number of safety functions that prevent core damage. To implement these safety function one or more safety function are incorporated in the reactor design. The system directly that perform safety function are called frontline system (see Table I) and those required for proper function of front-line system are termed as support system.

2.3.1. Reactor protection system

A rapid shutdown of the reactor by releasing and dropping the control rod into core occurring at maximum hazard conditions is defined as scram. In PARR-2 a scram switch in the control console can be pressed to de-energies the electromagnetic clutch for manual shutdown of the reactor. Reactor scram system be also initiated automatically if the measured neutron flux reaches 120% of the rated flux of 10^{12} n cm^{-2} Sec $^{-1}$ or when temperature drop across the core exceeds 25^{0}C.For these abnormal condition an alarm system is also provided. But in our case due to control rod stuck this system is of less impotence.

2.3.2 Primary heat removal system

The heat generated in the core is removed by natural convection in reactor vessel. The lower orifice controls the flow of water through the core. The heat absorbed by the water in the core is transferred to the reactor pool.

2.3.3. Reactor pool heat sink

Reactor pool is underground reinforced cement concrete (RCC) structure of size 6*3.5and 7m deep. It is stainless steel lined on its inner surface containing 126 m^{3} of demenralized water. To facilitate pool cleaning 1% slope towards sump is also provided. The heat from the reactor vessel is transferred to this pool which acts as heat sink.

2.3.4. Makeup supply system

The water level in the vessel is maintained between upper and lower limit of 400-500mm below the reactor vessel lid. In case water level falls below the lower limit makeup supply water from the makeup tank is supplied to the vessel under gravity through the manually operated valve.

2.4. Plant system requirements

For successful operation of the each front-line system a short description is given by assuming that all components are functioning properly with correctly performed operator action. The success criterion for negative reactivity insertion is achieved by interruption of the chain reaction through the insertion of cadmium capsule into the core irradiation sites. Reactor sub criticality is also successful phenomenon while the success criterion for the removal of primary heat is that transient does not damage the core.

3. Accident Sequence Modeling

Accident Sequence development consist of logic tools to graphically represents the sequence of events starting with initiating event(IE) and concluding with the release of radioactivity to the environment or outside the reactor confinement. Event tree was drawn to study the response of the system against initiating event while the fault trees were used for the modeling of the failure of the safety of the system. Conservative approach has been used in considering failure of components or system.

3.1. Event sequence modeling

The event tree (ET) against control rod stuck as initiating event (as shown in Fig.2) comprises on following events :

3.1.1. Control rod stuck

The event tree shown in Fig.2. models the possible response of the reactor to the control rod stuck.

3.1.2. Operator action 1

In case of control rod stuck the availability of the operator make the insertion of cadmium capsule for introducing the negative reactivity in the system. Success of this event will prevent the reactor to go into abnormal state.

3.1.3. Negative reactivity insertion system

After the occurrence of initiating event it is assumed that this system should shutdown the reactor or to make the reactor sub-critical detailed is given in section of reactor protection system. Success of this system makes the reactor scram or sub-criticality and avoids the accident to occur. As control rod is stuck therefore success of this system will be the sub-criticality or shutdown of the reactor.

3.1.4. Heat sink

Reactor pool being heat sink removes the heat transferred from the reactor vessel. It provides the deficiency of coolant in the core even in case of the rupture of reactor water level which is primary source of the coolant. Success of this system avoids the core to meltdown.

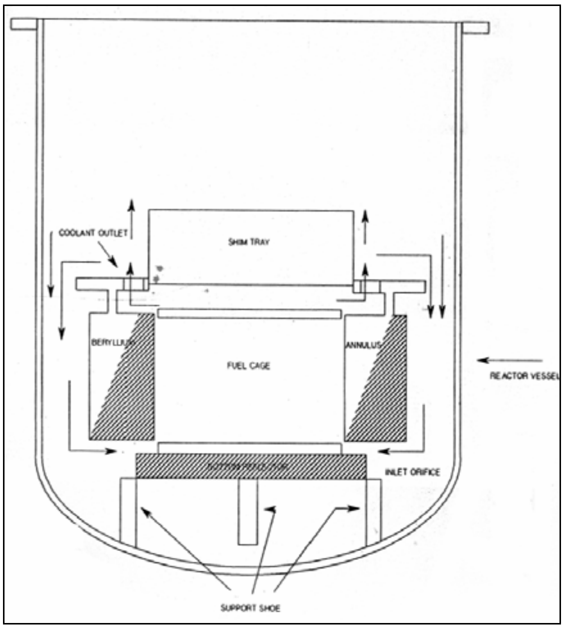

Figure 1. Core layout of PARR-2.

Table 1. Safety function and corresponding front-line systems.

Safety function	Front-line system
1-Reactivity control	Reactor protection system a- Automatic b- Manual
2- Removal of decay and stored heat	a-Natural convection b- Pool as heat sinks

3.1.5. Operator action 2

In case of decreased level of water in the reactor vessel the availability of the operator opens the valve of makeup water supply system. Success of this action compensates the deficiency of water level in the core and keeps the core cool.

3.1.6. Natural convection system

Heat generated in the core is removed by natural convection in water .The lower orifice control the flow through the core while the upper orifice provides the outlet path for the coolant. When temperature difference across the core increases, buoyant and circulating pressure head also increases which as a result increase the flow rate through the core thus cools the core. Success of this event cools the core while failure may cause core damage.

3.2. System analysis

The fault trees for each top event in the event tree are shown below.

3.2.1. Fault tree with top event "control rod stuck"

Control rod may stuck due to failure of clutch, gear failure, or may be due to reason that signal which indicate the control rod position may fail to sent back control rod the into the core. The fault tree is shown in Fig. 3.

3.2.2. Fault tree with top event "Pneumatic capsule transfer system failure"

Negative reactivity insertion system may fail to shutdown or sub-critical the reactor if scram signal fails or an error made in sending the capsule into irradiation sites in the core. The fault tree is shown in Fig. 4.

3.2.3.. Fault tree with top event "loss of Heat sink"

As external event were ignored in this study therefore pool was assumed to fail through rupture or any accidental drainage (see Fig 5).

3.2.4. Fault tree with top event "natural convection failure"

Natural convection may fail due to blockage of coolant channel, orifice or loss of coolant level in vessel. Water level in the vessel is assumed to fall down due to presence of rupture in vessel tank to such level that it should not stop the fission process due to unavailability of the moderator but it effect the natural convection cooling. The rupture is assumed to occur just below the upper orifice. Makeup supply system and pool water were also assumed to fail during that condition. The fault tree is shown in Fig. 7.

3.3. Core damage state

Core damage state Core damage is assumed whenever the available means for fission and decay heat removal fail to cool down of the core. In this study five types of core damage states have been considered on the basis of core damage definition and engineering judgment. Core damage state 1 (D1) is assumed to occur when core cooling fails along with no scram. Core damage state 2 (D2) is assumed to occur when transient without scram and means for heat removal are not available. Core damage state 3 (D3) is assumed to occur when transient without scram and natural convection is the only means of heat removal. Core damage state 4 (D4) is assumed to occur when transient with scram without any heat removal means. Core damage state 5 (D5) is assumed to occur when only one fuel element particularly with a blocked or by-passed flow channel.

4. Data Assessment and Parameter Estimation

This was major procedural step aims at aims at acquiring and generating all necessary informa-tion for the quantification of model that has been developed during the previous step. Inadequate availability of such data effect the model. Efforts were made to use PARR-2 specific data from control room log books. The average failure rate was then calculated by using accumulated PARR-2 services hours(~2800 h).Plant operating factor defined as average operated hours per year divided by total calendar hours in a year was also used to find frequencies. Generic data were used where specific data was not available [5-7] .In some cases expert opinion has to used to predict the frequency. The corresponding components unavailability models used are shown in Table II, while the reliability data used is shown in Table III.

5. Accident Sequences Quantification

Each accident sequence starts from initiating event followed by number of successes or failures of different systems. If these events do not depend upon each other then accident sequence may be quantified by simply multiplying their frequency values and unavailability In case of their dependences laws of Boolean algebra are use in this study RISKSPECTRUM computer code was used for these accident sequences analysis. The total core frequency is 1.25E-06 per year for this initiating event. The core damage states D1and D4 has maximum occurrence frequency, 5.32E-07and 5.30E-07 per reactor year among all others states. The core damage state D2 has negligibly low frequency of 1.23E-13 per year is least in this case. The core damage state and their corresponding frequency are shown in Table 4.

6. ConclusionS

The results of Level-1 PSA performed for PARR-2 indicated total core damage frequency for initiating event of control rod as 1.25E-06 per year of reactor operation. This frequency is not expected to change significantly when external

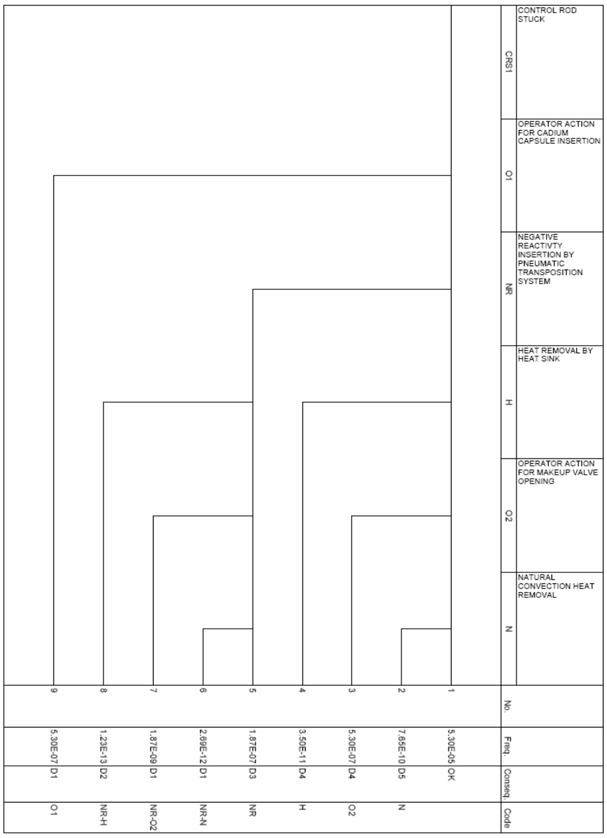

Figure 2. System event tree.

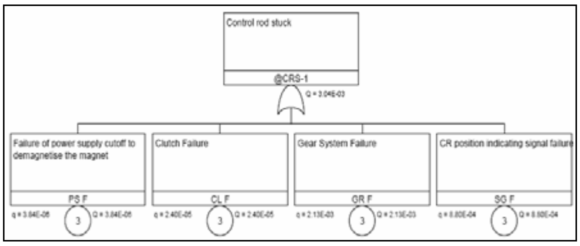

Figure 3. System fault tree with top event 'control rod stuck'.

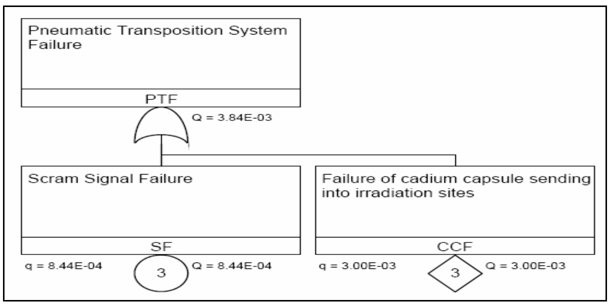

Figure 4. System fault tree with top event ‘Negative reactivity insertion failure.

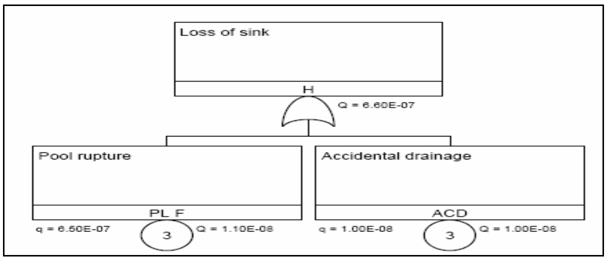

Figure 5. System fault tree with top event 'Loss of sink .

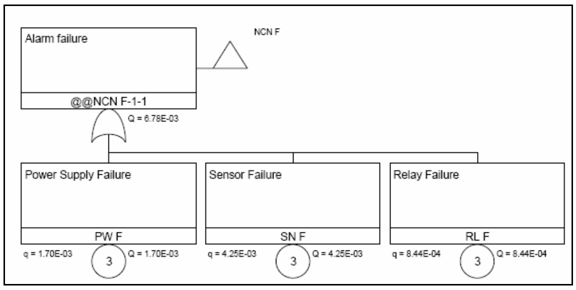

Figure 6. System fault tree with top event 'Alarm failure'.

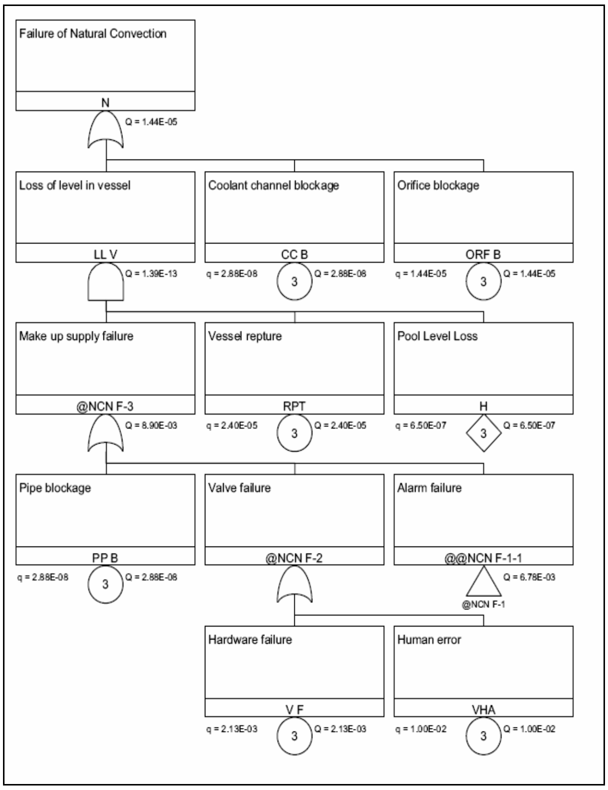

Figure 7. System fault tree with top event 'Loss of natural convection cooling.

Table 2. Models used for data analysis.

Component model	*Calculation formula*	*Parameters definition*
Stand-by Systems		
Tested components hardware failure	$1-(1-e^{-\lambda T})/\lambda T$	λ is stand-by failure rate, T is component test period
Untested components	$1-(1-e^{-\lambda T})/\lambda T$	λ is stand-by failure rate, T is fault exposure time
Monitored components	$\lambda t/(1+\lambda t)$	λ is stand-by failure rate,T is mean waiting and repair time
On-line Systems		
Non-repairable components	$1-e^{-\lambda T}$	λ is operating failure rate, T is mission time
Repairable components	$\lambda t/(1+\lambda t)$	λ is operating failure rate, T is mean repair time

Table 3. Unavailability data used in fault tree analysis.

Bottom event	*Failure rate (/hr)*	*Failure probability*	*Type of data*	*Source of data*
Valve	1.78E-04	0.00213	Specific	Log Book
Relay	3.55E-04	8.44E-04	Specific	Log Book
Power supply	7.11E-05	1.70E-03	Specific	Log Book
Sensor	1.78E-04	4.25E-03	Specific	Log Book
Pipe blockage	1.20E-09	2.88E-08	Generic	TECHDOC-478
CR position indicator	3.55E-04	8.48E-03	Specific	Log Book
Gear failure	1.78E-04	2.13E-03	Specific	Log Book
Clutch failure	1.00E-06	2.40E-05	Generic	TECHDOC-478
Electromagnetic failure	1.78E-04	2.13E-03	Generic	TECHDOC-478
Cadmium capsule entry failure		3.00E-03	Estimate	Expert Opinion
Pool failure	2.7E-8	6.5E-07	Generic	TECHDOC-478
Vessel rupture	1.00E-06	2.40E-05	Generic	TECHDOC-478
Orifice blockage	6.00E-07	1.44E-05	Generic	TECHDOC-478
Channel blockage	1.20E-09	2.88E-08	Generic	TECHDOC-478

Table 4. Core damage states.

Core damage state	*Frequency (/yr)*	*% of total core damage*
D1	5.32E-07	42.60
D2	1.23E-13	~0
D3	1.87E-07	15.00
D4	5.30E-07	42.40
D5	8.0E-10	0.064
Total	1.25E-06	

events are added. The results were based on conservative assumptions and have been used for ranking the components and systems of PARR-2 important to safety. The Probabilistic safety criteria of IAEA for generation-II reactors are assigned a frequency of one core damage in 10000 years of reactor operation. This frequency for PARR-2 one out of 80000 reactors year for studied initiating event. This shows quite safe status of this reactor. However it is recommended that to increase its further safety feature an automatic signal actuated system should be added in the reactor to eliminate human errors e.g. sending the cadmium capsule into irradiation sites when the control rod is being stuck, and an alternate power source in the form of battery cell etc. should be provided for alarm system so that failure due to power may be avoided.

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