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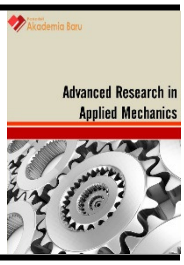

A review of safety features of Small Modular Reactor (SMR): Malaysian nuclear program perspective

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Muhammad Adil Khattak[1,*], Rafiziana Md Kasmani[1], Muhammad Zulhelmi Mahadi[1], Nurfarahin Miftah[1], Tiu Hor Yen[1], Khairulnadzmi Jamaluddin[1], Muhammad Faiz Farhan Noorizab[1], Abdoulhdi Amhmad Borhana Omran[2], Ali Najah Ahmed[3]

[1] Department of Nuclear Engineering, Faculty of Chemical and Energy Engineering Universiti Teknologi Malaysia, 81310 Skudai Johor, Malaysia
[2] Department of Mechanical Engineering, College of Engineering Science & Technology, Sebha University, Sebha, Libya
[3] Department of Civil Engineering, Universiti Tenaga Nasional, Selangor, Malaysia

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ABSTRACT

Mobile nuclear reactor is a new technology which is still under development in most of the countries. Based on the proposed design idea, definitely there are some possible constraints that need to be considered during its design process. Basically, constraints are the rules or limitations through which design is conceived and created. Safety assessment method used in the project is based on the method proposed by Department of Safety and Health, DOSH. The key to safety is to create multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer is exclusively relied upon. Defence–in–depth is one of the strategies that will be implemented in designing a reactor. It includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures. Defence–in–depth can be grouped into three levels. Level one is the prevention of occurrence of abnormal state, level two is prevention of development of an abnormal state to an accident and level three is protection of abnormal release of radioactive materials into environment. According to IAEA-TECDOC-626, the concepts of passive and active safety systems are defined and discussed. A passive safety system is defined as: either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation.

Keywords:
Mobile nuclear reactor, safety, defense–in–depth, passive system

1. Introduction

The unique feature of nuclear power plants, as distinct from other power-generating facilities, is the presence of large amounts of radioactive materials, primarily the fission products, same goes to the buoyant mobile nuclear power reactor. Hence, the central safety problem in the design of the nuclear reactor is to assure that, in so far as is possible or practical, these fission products remain

* *Corresponding author*
E-mail address: Muhammad Adil Khattak (muhdadil@utm.my)

safely confined at all times-during the operation of the plant, refuelling of the reactor, and preparation and shipping of spent fuel.

2. Overall Status of Malaysia Nuclear Program

As Malaysia has established a nuclear agency with the country has periodically reviewed the nuclear option to meet the increasing demands of energy, there is a need to build nuclear power generation plant, with plans for a nuclear plant are at the feasibility stage. However, due to concern prior from the Fukushima Daiichi nuclear disaster in 2011, plans to have a nuclear reactor have been postponed with neighbouring Vietnam have made a declaration to ditch their future nuclear energy plans. Minister in the Prime Minister's Department Nancy Shukri stated in 2016 that Malaysia will only built their plant after 2030, although the country has meet the requirements based from observations by the International Atomic Energy Agency [1-6].

The current activities focus on detail studies to identify issues and considerations as well as to objectively determine and assess the current level of national capabilities and state-of-preparedness pertaining to the development of a national nuclear power program, as one of the 131 Entry Point Projects (EPP) under Malaysia Economic Transformation Program (ETP). Pre-Project Activities are spearheaded by Malaysia Nuclear Power Corporation (MNPC)[7-12], as NEPIO and Nuclear Malaysia as TSO with tentative Nuclear timeline. The actual decision to implement nuclear power projects, however, will be guided by the Government's decision, after considering the recommendations in the details studies.

2.1 National Power Consumption

Despite the existence of Five-Fuel Diversification Policy, Five-Fuel Diversification Policy, i.e. oil, gas, coal, hydropower & renewable energy, there are only three major energy sources used for power generation, with coal mostly imported, indigenous gas supply uncertain beyond 2019, and hydropower resources located mostly in Sarawak & adequate to only around 2030.

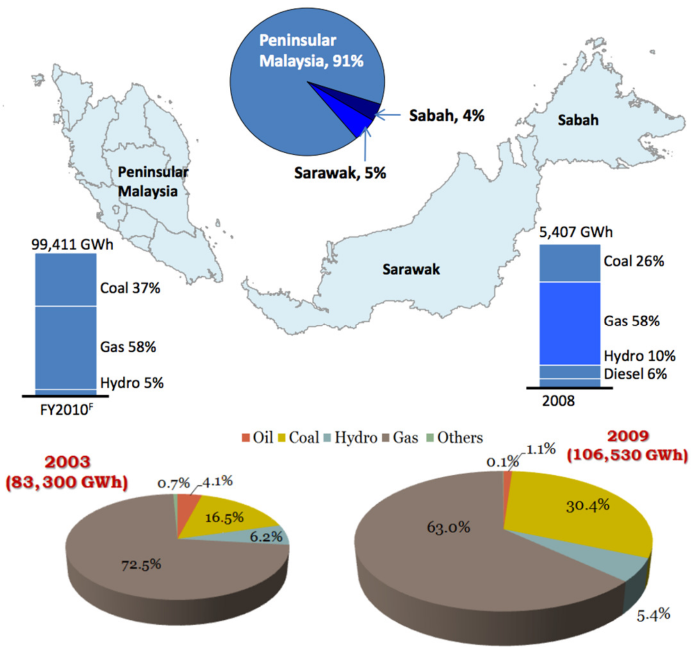


Fig. 1. National Power Consumption by Geographical Region & Power Generation Mix

2.2 Brief Economic Review

There has always been a strong correlation between the country's GDP and its electricity consumption where the predominance of certain types of economic activities affected the electricity demand at the sectoral basis. However, during these recent years, the encouraging economy growth did not reflect the Peninsular Malaysia electricity growth trend and this could be contributed by the energy efficiency and renewable energy initiatives aggressively implemented in the country. Nevertheless, the electricity intensity stood at 0.16 GWh/GDP for the year 2013 and recently achieved 0.123 GWh/GDP during the first half of 2015.

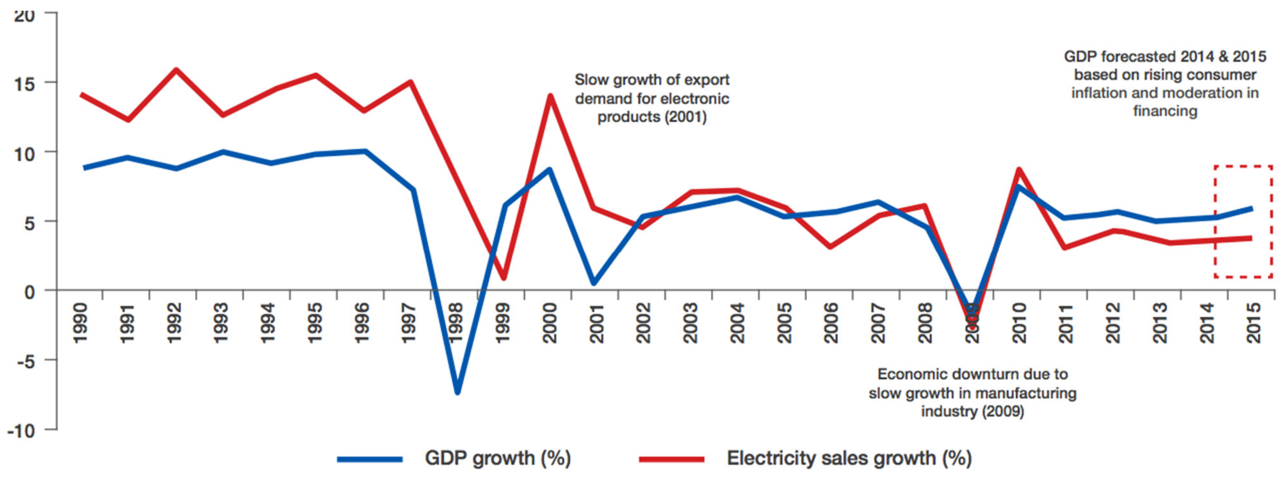
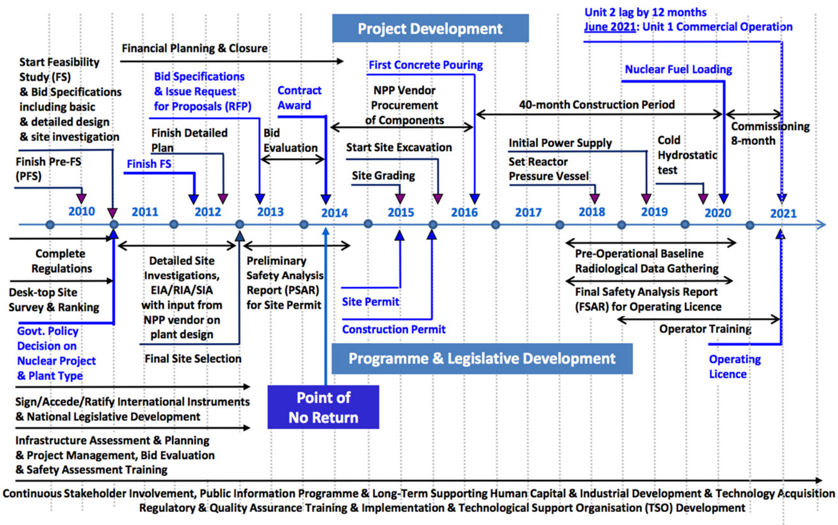


Fig. 2. Country GDP vs. Electricity sales (1990–2015)

2.3 Detailed Timeline on Nuclear Power Deployment

A Detailed Timeline on Nuclear Power Deployment is shown in Fig. 3.



Source: Nuclear Malaysia; Malaysia NKEA OGE Laboratory 2010.

Fig. 3. A Detailed Timeline on Nuclear Power Deployment

3. Implementing Defence-In-Depth

3.1 Level One: Prevention of Occurrence of Abnormal State

In this level, the prevention of accidents is primarily on virtue of the design, construction and surveillance of the plant. The design is utilized inherently on safe properties. There are physical properties such as Doppler effect, and moderator temperature effect in the core region which tend to keep automatically a steady state operation [14-16]. The properties can be effectively incorporated into the design. Below show the process of the effect.

3.1.1 Doppler effect

The process occurred when unexpected reactivity insertion into the core happened [17-18]. This caused when the fuel temperature increases then lead to increase neutron absorption of Uranium-238. After that, the number of neutrons drop in the core which causes fission rate of Uranium-235 decreases. The power will eventually fall and thus, create a drop in fuel temperature. The operation is back to normal. Figure 4 shows the flow of the Doppler effect.

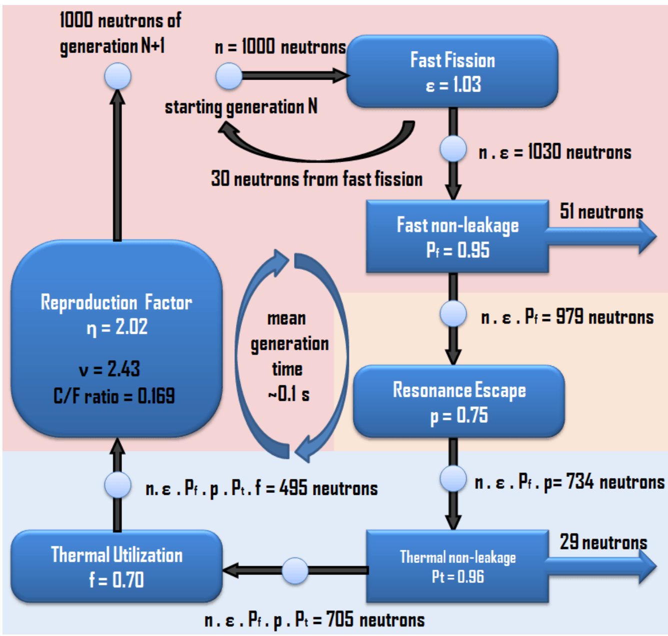


Fig. 4. Doppler Effect

3.1.2 Moderator temperature effect

In moderator temperature effect, the process starts with an unexpected reactivity insertion into the core which can lead to increase of moderator temperature [19]. The increasing of moderator temperature causes a drop on moderator density which then decreasing the number of moderator atoms in the core region. This makes the neutron less moderated. This causes a drop of Uranium-235 fission rate and then decrease the power. Decreasing of power lead to reduction of moderator temperature and the plant is operated in normal condition. Figure 8.2 shows the flow of moderator temperature effect.

Interlock system can be prevented from human errors of operators such as withdrawal of control rods in wrong way and it is required to be designed so that it does not accept wrong human action either mechanically, electrically or electronically. The failsafe system on the other hand is designed to prevent propagation or development of the local incident of equipment or subsystems to avoid expansion to an accident, which can lead to the situation where it is always in the safety side. Furthermore, all equipment and control systems must be inspected not only during fabrication and installation but also in service regularly at the legally determined intervals in the presence of the governmental regulation officers to keep their perfect performances.

3.2 Level 2: Prevention of Development of an Abnormal State to an Accident

In this level, many systems such as early detection system, reactor protection system, reactor and decay heat removal system had been created to prevent the progress of any abnormal state which can causes accidents.

Early detection system is created to cope with the case where an incident or an abnormal state occurs in spite of the efforts in the Level 1. The installation of early detection systems for the abnormal states of the plant are required, such as detection systems for temperature, pressure, flow rate, leakage of fluid or radiological values.

Besides, the installation of reactor protection system is to detects the occurrence of an abnormal state or malfunction of an equipment of the plant and sends signals to the reactor scram system which then actuating the alarm. There are two circuits in this system which are circuit actuating the reactor emergency shutdown system and emergency core cooling system (ECCS) which is so designed with multiple and independent structure that even if an equipment consisting the system loses its function, the function as the overall system does not lose its original function [20-21].

Furthermore, the reactor shutdown system has an extremely important function to secure the safety of a plant. In Pressurized Water Reactor (PWR), the reactor shutdown system has two independent systems based on different actuation principles such as control or safety rods and injection of boric acid into the core as a backup [22-27]. Lastly, decay heat removal system is needed to remove the decay heat due to the fission product in the fuel generate the decay heat in long period of time.

3.3 Level 3: Protection of Abnormal Release of Radioactive Materials into Environment

Engineered safety system, reactor containment vessel and reactor containment vessel spray system are needed to protect the releasing of radioactive materials into environment. Engineered safety system must be designed to have functions to suppress or protect the release of large quantity of radioactive materials from the reactor vessel into the environment due to break or failure of reactor system. The system contains emergency core cooling system (ECCS) and reactor containment vessel which includes separation valves and purification system.

Emergency core cooling system (ECCS) is provided to avoid the temperature of the fuel increases steeply due to the decay heat and resulting possibly in the core melting. The ECCS consists of independent multiple subsystems actuated by different principles such as high pressure water injection system in the Pressurized Water Reactor.

Reactor containment vessel on the other hand is one of the engineered safety systems of Pressurized Water Reactor and contains reactor vessel, main steam piping, recirculation piping and related equipment. The functions of the reactor containment vessel are barrier of high pressure steam due to Loss of Cooling Accident (LOCA) and barrier of diffusing radioactive materials caused by the fuel fracture due to the LOCA.

4. Passive Safety System

As part of the IAEA's overall effort to foster international collaborations that strive to improve the economics and safety of future water-cooled nuclear power plants, an IAEA Coordinated Research Project (CRP) was started in early 2004. This CRP, entitled Natural Circulation Phenomena, Modelling and Reliability of Passive Safety Systems that Utilize Natural Circulation, focuses on the use of passive safety systems to help meet the safety and economic goals of a new generation of nuclear power plants. According to IAEA-TECDOC-626, 619, 1560, 1616, 1056 and 1421, the concepts of passive and active safety systems are defined and discussed [28-33]. A passive safety system is defined as: either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation. Four categories were established to distinguish the different degrees of passivity which shown in Table 1. In the following subsection, passive safety system of category B to D implemented in the emergency mobile nuclear power reactor are discussed. Most of the passive safety system design is similar to AP600/AP1000 design, product of Westinghouse, USA. This conventional passive safety system gives more confidence and reliability to our design.

Table 1
Characteristic of four categories of passive safety system in the emergency mobile nuclear power reactor

Categories	Characteristic	Passive Safety System
Category A	• no signal inputs of 'intelligence' • no external power sources or forces • no moving mechanical parts • no moving working fluid	• physical barriers against the release of fission products (nuclear fuel cladding, pressure boundary systems) • hardened building structures for the protection of a plant against seismic and or other external events • core cooling systems relying only on heat radiation and/or conduction from nuclear fuel to outer structural parts, with the reactor in hot shutdown; and static components of safety related passive systems (tubes, pressurizers, accumulators, surge tanks) as well as structural parts (supports, shields).
Category B	• no signal inputs of 'intelligence' • no external power sources or forces • no moving mechanical parts • moving working fluids	• Automatic Depressurization System (ADS) 1-3 Steam Vent into IRWST • Passive Containment Cooling System
Category C	• no signal inputs of 'intelligence' • no external power sources or forces • moving mechanical parts, whether or not moving working fluids are also present	• Automatic Depressurization System (ADS) 1-3 Steam Vent into IRWST • Accumulator Tanks
Category D	• signal inputs of 'intelligence' to initiate the passive process • energy to initiate the process must be from stored sources such as batteries or elevated fluids • active components are limited to controls, instrumentation and valves to initiate the passive system • Manual initiation is excluded	• Passive Residual Heat Removal System • Core Make-up Tanks • IRWST injection • Lower Containment Sump Recirculation • Passive Containment Cooling System • Sea Water Cooling Jacket

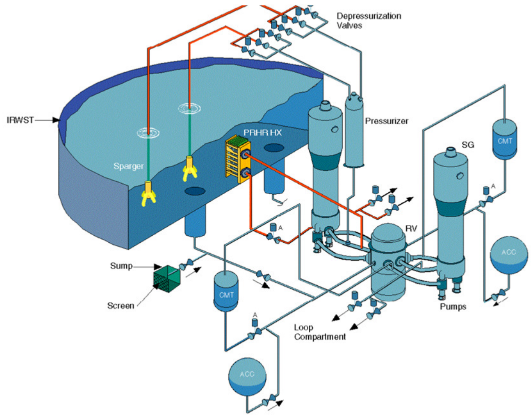


Fig. 5. Passive safety system in AP600/AP1000 design (which similar to proposed design)

4.1 Passive residual heat removal system

Passive residual heat removal (PRHR) heat exchanger is incorporated into the primary coolant system and resides in the water-filled in-containment refuelling water storage tank (IRWST) [34-39]. Their primary function is to provide primary coolant heat removal via single-phase liquid natural circulation loop.

The PRHR heat exchanger loop is pressurized and ready for service. Single-phase liquid flow is actuated by opening the isolation valve at the bottom of the PRHR heat exchanger. Hot water rises through the PRHR inlet line attached to one of the hot legs. The hot water enters the tubesheet in the top header of the PRHR heat exchanger at full system pressure and temperature. The IRWST is filled with cold borated water and is open to the containment. Heat removal from the PRHR heat exchanger occurs by boiling on the outside surface of the tubes. The cold primary coolant returns to the primary loop via the PRHR outline line that is connected to the steam generator lower head.

PRHR system is particularly useful in mitigating the station blackout scenario. In general, it eliminates the need for ‘bleed and feed’ operations for plant cool-down.

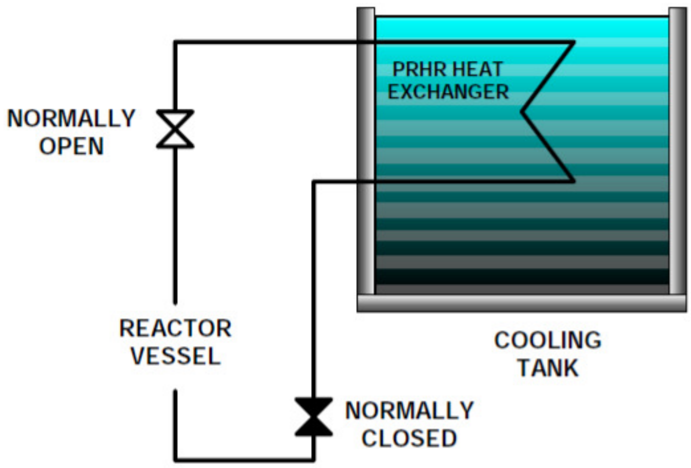


Fig. 6. Simple layout of PRHR concept

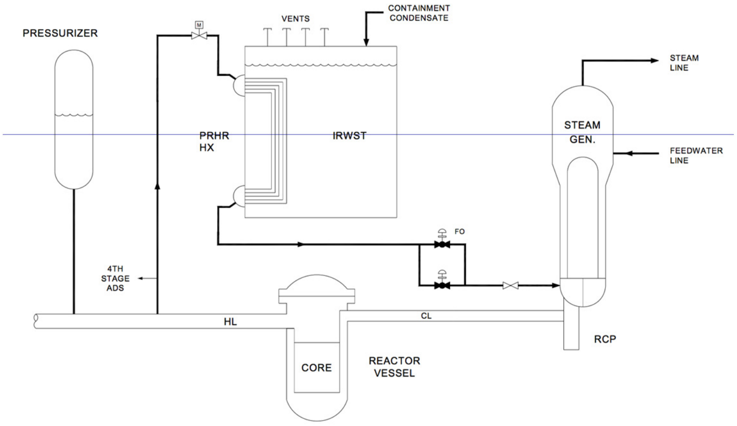


Fig. 7. Passive decay heat removal (PRHR) system

4.2 Core Make-Up Tanks (CMT)

Natural circulation loops represent an effective means of providing core cooling. Our designs implement elevated tanks connected to the reactor vessel at the top and bottom of the tank as shown in Figure. This tank is known as core make-up tank (CMT) and it effectively replace the high pressure safety injection systems in conventional PWR. Each CMT consists of a large volume stainless steel tank with an inlet line that connects one of the cold legs to the top of the CMT and an outlet line that connects the bottom of the CMT to the direct vessel injection (DVI) line. The DVI line is connected to the reactor vessel downcomer. Each CMT is filled with cold borated water. The CMT inlet valve is normally open and hence the CMT is normally at primary system pressure. The CMT outlet valve is normally closed, preventing natural circulation during normal operation. When the outlet valve is open, a natural circulation path is established. Cold borated water flows to the reactor vessel and hot primary fluid flows upward into the top of the CMT.

In order to reduce the number of pipelines connected with the reactor pressure vessel, the delivery line of the core make-up tank (CMT) is in common with the emergency core coolant delivery line. In case of a number of accident scenarios, the CMT delivery can start before the accumulator delivery and end-up after the accumulator emptying. In those situation the CMT delivered flow-rate can be affected by the accumulator delivered flow-rate to a noticeable extent.

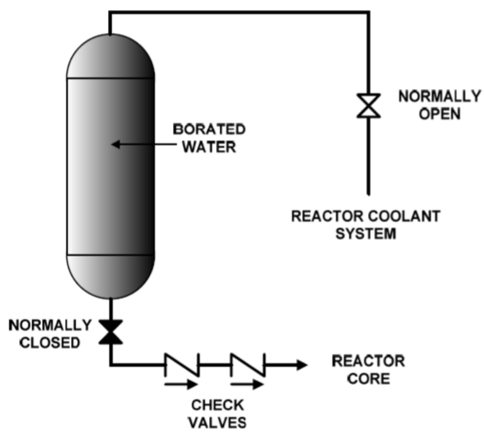


Fig. 8. Simple layout of core make-up tanks

4.3 Automatic Depressurization System (ADS)

The automatic depressurization system (ADS) consists of four stages of valves that provide for the controlled reduction of primary system pressure [40-43]. The first three stages consist of two trains of valves connected to the top of the pressurizer. The first stage opens on CMT liquid level. ADS stages two and three open shortly thereafter on timers. The ADS 1-3 valves discharge primary system steam into a sparger line that vents into the in-containment refuelling water storage tank (IRWST). The steam is condensed by direct contact with the highly subcooled water in the IRWST. The fourth stage of the ADS consists of two large valves attached to ADS lines on each hot leg. The ADS-4 valves open on low CMT liquid level and effectively bring primary side pressure down to containment conditions. The ADS-4 valves vent directly into the containment building.

4.4 In-Containment Refueling Water Storage Tank (IRWST) injection

The In-containment refueling water storage tank (IRWST) is a very large concrete pool filled with cold borated water. It serves as the heat sink for the PRHR heat exchanger and a source of water for IRWST injection. The IRWST has two injection lines connected to the reactor vessel DVI lines. These flow paths are normally isolated by two check valves in series. When the primary pressure drops below the head pressure of the water in the IRWST, the flow path is established through the DVI into the reactor vessel downcomer. The IRWST water is sufficient to flood the lower containment compartments to a level above the reactor vessel head and below the outlet of the ADS-4 lines.

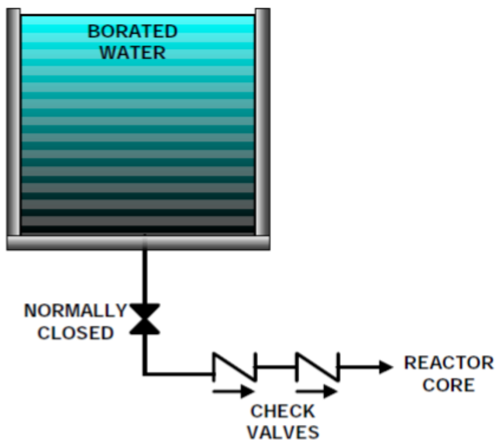


Fig. 9. Elevated gravity drain tank a.k.a IRWST

4.5 Accumulator Tanks

The accumulators are similar to those found in conventional PWRs. They are large spherical tanks approximately three-quarters filled with cold borated water (75%) and pre-pressurized with nitrogen. The accumulator outlet line is connected to the DVI line. A pair of check valves prevents injection flow during normal operating conditions. When system pressure drops below the accumulator pressure (plus the check valve cracking pressure); for example, during the event of a loss of coolant accident (LOCA), the check valves open allowing coolant injection to the reactor downcomer via the DVI line.

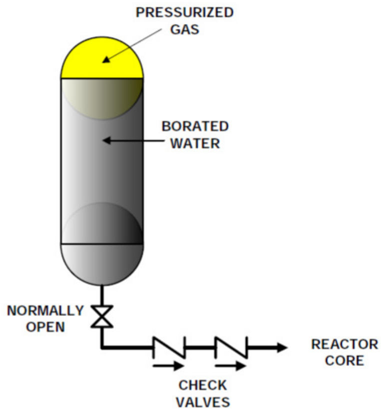


Fig. 10. Pre-pressurized core flooding tank (accumulator)

4.6 Lower Containment Sump Recirculation

After the lower containment sump and the IRWST liquid levels are equalized, the sump valves are opened to establish a natural circulation path. Primary coolant is boiled in the reactor core by decay heat. This low-density mixture flows upward through the core and steam and liquid is vented out of the ADS-4 lines into containment. Cooler water from the containment sump is drawn in through the sump screens into the sump lines that connect to the DVI lines.

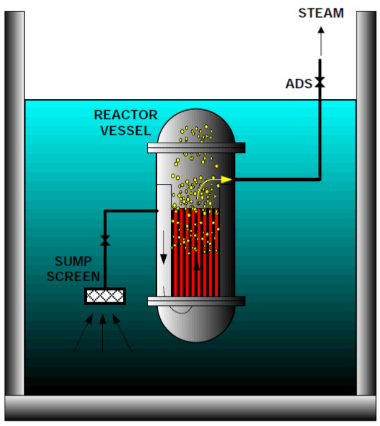


Fig. 11. Core cooling by sump natural circulation

4.7 Passive Containment Cooling System (PCCS)

Figure 12 shows a design that implements a natural draft air cooled containment. Subsequent to a LOCA, steam in contact with the inside surface of the steel containment is condensed. Heat is transferred through the containment wall to the external air. An elevated pool situated on top of the containment provides a gravity driven spray of cold water to provide cooling in a LOCA scenario. The air flow for the cooling annulus, that is generated by a chimney-like type effect, is a Category B passive safety system [44-47]. The containment vessel sprays are a Category D passive safety system.

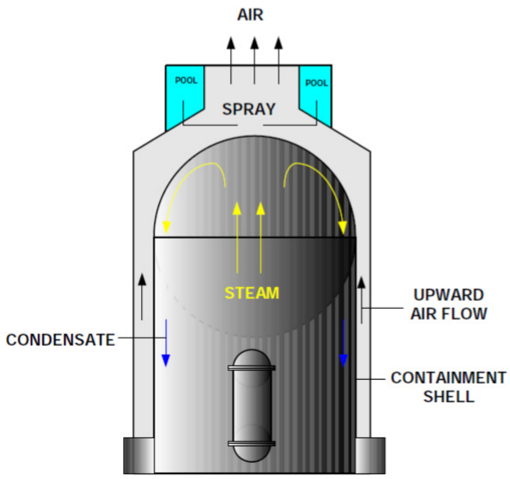


Fig. 12. Containment pressure reduction and heat removal following a LOCA using a passive containment spray and natural draft air

4.8 Sea Water Cooling Jacket

Sea water cooling jacket is a unique passive safety system implemented in this mobile nuclear power reactor which mounted on platform. It utilizes the lower containment compartments as a reservoir of coolant for core cooling in the event of a break in the primary system, also in the event of overheating of core and other components when the reactor exceeds the limit of super criticality at emergency situation. All components in primary system are under the sea water level. Sea water can easily flow from the environment into the reactor cavity through piping and opening of check valves. The water inlet will be at the top of the containment cavity while the outlet will be at the bottom. Eventually the reactor is completely immersed in water and heat is removed due to natural circulation (different in density). Sea is an ultimate heat sink. Hence, it can reduce the temperature of core in very short time. However, this system is started up at the last level of defence as sea water is extremely corrosive.

5. Safety Analysis

Safety analysis of this designed nuclear power plant is done through risk assessment. Risk assessment is an essential and systematic process for accessing the impact, occurrence and the consequences of human activities on systems with hazardous characteristics and constitutes a needful tool for the safety policy of a company. “Risk” has been defined as the chance that someone or something that is valuated will be adversely affected by the hazard, and also a measure under uncertainty for the severity of a hazard while “hazard” is any unsafe condition or potential source of an undesirable event with potential for harm or damage.

5.1 Identification of Hazard Sources

Risk analysis (or alternatively safety analysis) is an approach to identify the factors that may lead to accidents or according to an international standard (ISO/IEC, 1999) is the systematic use of available information to identify hazards. They focus on parameters and guidewords (like ‘more’ or ‘other than’), to discover meaningful deviations of normal interactions involving complex systems that may be risky and cause damage or injury. Accident scenarios which are based on technical knowledge and human factors knowledge may cover many of the real accidents, because they provide the way of identifying the contributing factors that are essential for the accident to occur.

The hazard sources considered in the analysis is based on the initiating events of accidents at Surry Nuclear Power Station, as shown in Table 8.1 with its mean annual occurring frequency.

Table 2
Initiating events considered in analysis for accidents resulting from internal initiators at Surry

Initiating event	Mean annual frequency (year^{-1})
Loss of offsite power	7.70E-02
Transients with loss of main feedwater (MFW)	9.40E-01
Transients with MFW initially available	7.30E-00
Non-recoverable loss of direct current (DC) Bus A	5.00E-03
Non-recoverable loss of DC Bus B	5.00E-03
Steam generator tube rupture	1.00E-02
Large loss of coolant accident (LOCA), 15.24-73.66 cm	5.00E-04
Medium LOCA, 5.08-15.24 cm	1.00E-03
Small LOCA, 1.27-5.08 cm	1.00E-03
Very small LOCA, less than 1.27 cm	1.30E-02
Interfacing LOCA	1.60E-06

5.2 Risk Consideration

Once the hazards have been identified, severity and likelihood ratings must be addressed. Hence, risk consideration is achieved by two steps. Step 1 is to specify the severity ratings of the hazards. Severity is a subjective issue. The question is how to decide in which level of severity, the injury will be classified. To solve this problem, we have classified the severity (S) of injury or consequence index (C), by specifying the level of employee’s inability, in association with the duration that the employee is absent from his work, according to the obligations of Guidelines for Hazard Identification, Risk Assessment and Risk Control (Department of Occupational Safety and Health, 2008). The results are included in Table 3. According to this standard, severity ratings or consequence index (C) are evaluated for each hazard as shown in Table 4.

Table 3
Definition of the injury’s severity gradation

Severity of harm factor (S) or consequence index (C)	Severity	Effects of hazard sources
5	Catastrophic	Death
4	Fatal	Permanent inefficiency
3	Serious	Absence from the work and return with health problem
2	Minor	Absence from the work and return after full recovery
1	Negligible	No one human injury

Table 4
Consequence index (C) for each hazard

Initiating Event	5	4	3	2	1
Loss of offsite power			/		
Transient with loss of MFW	/				
Transient with MFW initially available		/			
Non-recoverable loss of DC Bus A			/		
Non-recoverable loss of DC Bus B			/		
Steam generator tube rupture	/				
Large LOCA, 15.24 - 73.66 cm			/		
Medium LOCA, 5.08 - 15.24 cm			/		
Small LOCA, 1.27 - 5.08 cm		/			
Very small LOCA, less than 1.27 cm		/			
Interfacing LOCA				/	

Table 5
Gradation of the likelihood ratings (L) in association with the potential hazard sources characteristics and its event frequency

Probability factor or likelihood ratings (L)	Characteristics of hazard sources	Frequency of events occurring
5	Unavoidable	1 Event during a time period of t<10^4h
4	Probability slightly greater than 50%	1 Event during a time period of 10^4<t<10^6h
3	Almost improbable (or remote)	1 Event during a time period of 10^6<t<9,900,000h
2	Improbable	1 Event during a time period of 9,900,000<t<10^7h
1	Impossible	1 Event during a time period of t>10^7h

Table 6
Likelihood rating (L) for each hazard

Initiating event	Mean annual frequency (year-1)	Frequency (year)	Frequency (hour)	Likelihood Rating (L)
loss of offsite power	7.70E-02	1.30E+01	1.14E+05	4
Transient with loss of MFW	9.40E-01	1.06E+00	9.32E+03	5
Transient with MFW initially available	7.30E-00	1.37E-01	1.20E+03	5
Non-recoverable loss of DC Bus A	5.00E-03	2.00E+02	1.75E+06	3
Non-recoverable loss of DC Bus B	5.00E-03	2.00E+02	1.75E+06	3
Steam generator tube rupture	1.00E-02	1.00E+02	8.76E+05	4
Large LOCA, 15.24-73.66 cm	5.00E-04	2.00E+03	1.75E+07	3
Medium LOCA, 5.08-15.24 cm	1.00E-03	1.00E+03	8.76E+06	3
Small LOCA, 1.27-5.08 cm	1.00E-03	1.00E+03	8.76E+06	3
Very small LOCA, less than 1.27 cm	1.30E-02	7.69E+01	6.74E+05	3
Interfacing LOCA	1.60E-06	6.25E+05	5.48E+09	1

Step 2 is to specify the likelihood ratings (L) of that hazards. The occurrence of injury or damage (or the likelihood of hazard sources' occurring) may depend on several factors related to the actual interaction of the employee with a hazard source and also to the energy transferred during this interaction. The likelihood or probability of occurrence of hazards is based on the mean annual frequency given in Table 2. Table 5 shows the compression of gradation of the likelihood ratings (L) in association with the potential hazard sources characteristics and its event frequency and the compression is based on Department of Occupational Safety and Health (2008). Based on this standard, likelihood ratings (L) for each hazard is identified in Table 6.

5.3 Quantified Risk Evaluation and Safety-related Decision Making

Methods of quantified risk evaluation need to be as precise as possible to differentiate the risk level of various activities. Quantified measurement techniques enable risk assessors to scale their appreciation of the severity of the short and long term consequences of accidents and the factors that influence the occurrence of an accident scenario (van Duijne et al., 2008). In our case, we will implement the decision matrix risk assessment (DMRA) technique. The combination of a consequence or severity and likelihood range, gives us an estimate of risk, the product of the consequence index (C) and likelihood index (L) provides a measure of risk (R_{DM}) which is expressed by the relation:

$$R_{DM} = C \cdot L \quad (1)$$

By using this expression, a quantitative risk value, R_{DM} is calculated. Eventually, the technique is consummated by the construction of the risk matrix shown in Figure 11 and the decision-making table shown in Figure 13. In the final step, the risk value is tabulated in Table 7 and risk assessment decision for each hazard is done according to the decision-making table. Based on Table 7, the risk assessment decision had shown that 3 initiating events on high level, 9 initiating events on medium level and 1 initiating events on low level. The initiating events on high level must immediately take control on risk to avoid accidents from happen.

Table 7
Risk value and risk assessment decision for each hazard

Initiating event or hazard sources	C	L	R_{DM}	Risk assessment decision
loss of offsite power	3	4	12	Medium
Transient with loss of MFW	5	5	25	High
Transient with MFW initially available	4	5	20	High
Non-recoverable loss of DC Bus A	3	3	9	Medium
Non-recoverable loss of DC Bus B	3	3	9	Medium
Steam generator tube rupture	5	4	20	High
Large LOCA, 15.24-73.66 cm	3	3	9	Medium
Medium LOCA, 5.08-15.24 cm	3	3	9	Medium
Small LOCA, 1.27-5.08 cm	4	3	12	Medium
Very small LOCA, less than 1.27 cm	4	3	12	Medium
Interfacing LOCA	2	1	2	Low

	Severity (S)				
Likelihood (L)	1	2	3	4	5
5	5	10	15	20	25
4	4	8	12	16	20
3	3	6	9	12	15
2	2	4	6	8	10
1	1	2	3	4	5

Fig. 13. The risk matrix

RISK	DESCRIPTION	ACTION
15 - 25	HIGH	A HIGH risk requires **immediate** action to control the hazard as detailed in the hierarchy of control. Actions taken must be documented on the risk assessment form including date for completion.
5 - 12	MEDIUM	A MEDIUM risk requires a planned approach to controlling the hazard and applies temporary measure if required. Actions taken must be documented on the risk assessment form including date for completion.
1 - 4	LOW	A risk identified as LOW may be considered as acceptable and further reduction may not be necessary. However, if the risk can be resolved quickly and efficiently, control measures should be implemented and recorded.

Fig.14. The decision-making table

6. Conclusion

The safety feature of the mobile nuclear reactor plays an important role to design the reactor. The safety analysis using the probabilistic risk analysis and deterministic analysis should be considered more to decrease the probability of accident to occur. By having a lot of research on the safety analysis, the design of mobile nuclear power reactor can be achieved to be commercialized for many purposes such as emergency or isolated area.

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