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A Strategy to Apply a Graded Approach to a New Research Reactor I&C Design

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1. Introduction

A project for the development of a new research reactor (NRR) was launched by KAERI in 2012. It has two purposes: 1) providing a facility for radioisotope production, neutron transmutation doping, and semiconductor wafer doping, and 2) obtaining a standard model for exporting a research reactor (RR). The instrumentation and control (I&C) design should reveal an appropriate architecture for the NRR export. The adoption of a graded approach (GA) was taken into account to design the I&C and architecture.

Although the GA for RRs is currently under development by the IAEA, it has been recommended and applied in many areas of nuclear facilities. The Canadian Nuclear Safety Commission allows for the use of a GA for RRs to meet the safety requirements [1]. Germany applied the GA to a decommissioning project [2]. It categorized the level of complexity of the decommissioning project using the GA. In the case of 10 C.F.R. Part 830 § 830.7, a contractor must use a GA to implement the requirements of the part, document the basis of the GA used, and submit that document to U.S. DOE [3]. It mentions that a challenge is the inconsistent application of GA on DOE programs. RG-1.176 states that graded quality assurance brings benefits of resource allocation based on the safety significance of the items [4]. The U.S. NRC also applied the GA to decommissioning small facilities [5]. The NASA published a handbook for risk-informed decision making that is conducted using a GA [6]. ISA-TR67.04.09-2005 supplements ANSI/ISA–S67.04.01–2000 and ISA-RP67.04.02-2000 in determining the setpoint using a GA [7].

The GA is defined as a risk-informed approach that, without compromising safety, allows safety requirements to be implemented in such a way that the level of design, analysis, and documentation are commensurate with the potential risks of the reactor [1].

The IAEA is developing a GA through DS351 and has recommended applying it to a reactor design according to power and hazarding level. Owing to the wide range of RR utilization, the safety requirements for RRs may not be required to be applied to every RR in the same way [8]. DS351 also states that the way in which the requirements are demonstrated to be met for a multipurpose and high power RR might be very different from the way in which the requirements are demonstrated to be met for a RR with very low power and very low associated radiological hazards to the facility staff, the public, and the environment. The GA should not compromise safety or waive the safety requirements.

The GA is not a quantitative method but rather a qualitative method to determine the scope and level of application of the safety requirements to the design of a RR. It adopts a systematic approach and engineering judgment for the determination. The GA is applicable in all stages of the RR lifetime. Any grading during the lifetime should ensure that safety functions are maintained and that there are no radiological hazards to the operators and public. The grading activities should be based on a safety analysis, regulatory requirements, and engineering judgment [8]. In DS351, the GA activities consist of two steps: 1) categorizing a facility into a range of the highest to the lowest risk, which is an initial grading of the facility, and 2) grading the system, structure, and components important to safety, which is a more detailed grading of the facility. As an example of the GA, fewer inspections and hold points for a 100 kW RR than those for a 5 MW RR can be determined.

For the application of the GA to the I&C design of an RR, Rahman proposed the GA to develop the digital MMIS (Man-Machine Interface System) for RRs regarding cyber security, software V&V, and human factors engineering [9]. However, it did not show the specific design decisions. Suh presented the overall I&C architecture for the NRR, but it has a lack of rationale for the design decision making [10]. This paper presents a strategy to make a design decision for NRR I&C systems. According to the characteristics and safety analysis of the NRR, the proper design level should be determined to avoid an over design.

2. The Strategy of the GA Application

A procedure of the GA for the NRR I&C design is shown in Fig. 1. A strategy to perform each step of the procedure is presented in this paper.

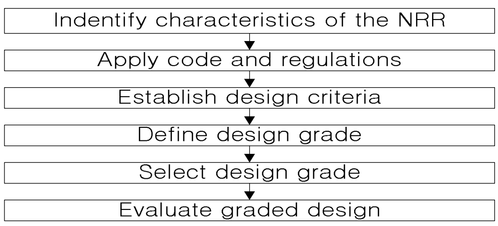

Fig. 1. A Procedure of the GA for the NRR I&C

To identify the characteristics of the NRR, NS-R-4 mentioned that most RRs have a small potential for hazards to the public compared with power reactors, but they may pose a greater potential for hazards to operators [11]. The characteristics of the NRR are as follows:

- Not for generating electric power but for utilizing neutrons

- Open tank in pool type reactor
- Not operating under sub-cooled pressure but under atmospheric pressure
- Spreading many manual switches for a reactor trip over reactor buildings
- Less expensive for one reactor trip
- Frequent access to reactor during operation
- Loading neutron utilization equipment into a reactor during the operation
- Relatively simple and easy start-up operation

Table 1. Postulated Initiating Events for the NRR

Category	Postulated Initiating Events selected from NS-R-4
Loss of electric power	Loss of normal electric power
Reactivity induced accident	Startup accident
	Control rod drive failure or system failure
	Influence by experiments and experimental devices
Loss of primary coolant system (PCS) flow	PCS pump failure
	PCS pump seizure
	Fuel channel blockage
	Coolant reduction due to core bypass
Loss of heat sink	Failure of pumps in shutdown cooling system
Loss of coolant	Rupture of the primary coolant boundary (Dt/4 size rupture)
	Rupture of the primary coolant boundary (10inch pipe rupture)
	Rupture of the primary coolant boundary (Guillotine rupture)

To apply code and regulations to the RR, NS-R-4 [11] is a mandatory code. Some of the code and regulations applied to power reactors are selected to ensure safety. NUREG-1537 [12] and IAEA-TECDOC-973 [13] can be a reference for the design. To establish design criteria, the I&C design is based on digital technologies. Defense-in-depth and diversity, single failure criteria, and independence are selected as basic design criteria. The I&C should be able to mitigate the events in Table 1 as excerpted from [11].

Table 2. I&C Classification According to Countries

ORGANIZATIONS AND/OR COUNTRIES	CLASSIFICATION			
IAEA	Systems Important to Safety		Systems not important to safety	
	Safety system	Safety related system		
IEC	Category A	Category B	Category C	Unclassified
France	1E	2E	IFC/NC	
European Utilities Requirements (EUR)	F1A (Automatic)	F1B (Automatic and Manual)	F2	Not Classified
UK	Category 1	Category 2	Not classified	
USA	1E	Non-nuclear safety		

To define the design grade, the classification and grade should first be distinguished. The classification is to define the safety level of the I&C in a reactor design and is differently defined in Table 2 as excerpted from [14]. The NRR I&C design follows Korea's classification, such as I, II and III defined in [15]. The grade is to define the design level at the safety level according to the characteristics of the reactors. For example, the design level of a reactor protection system for a pressurized water reactor and an open pool type reactor should be different.

3. Conclusions

The GA application was surveyed and a procedure of the GA for the NRR I&C design was presented. The rationale of the I&C design will be derived using the GA. The I&C will be designed to mitigate the postulated initiating events of the NRR. The way of grading, selecting, and evaluating the I&C design will be presented next after a further study.

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