August 24~28, 2014

International Convention Center (ICC) Jeju, Jeju Island, Korea

ISOFIC/ISSNP 2014

International Symposium on Future I&C for Nuclear Power Plants

International Symposium on Symbiotic Nuclear Power Systems

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Contents

ISOFIC/ISSNP 2014

International Symposium on Future I&C for Nuclear Power Plants
International Symposium on Symbiotic Nuclear Power Systems

Foreword

Kook Hun Kim, Ph.D.

Senior Vice President, Nuclear I&C, Doosan Heavy Industry & Construction

Academian, National Academy of Engineering of Korea

General Chair of ISOFIC/ISSNP 2014

Welcome to ISOFIC/ISSNP 2014 in Jeju Island.

As the general chair of this symposium, I'm honorably pleased to have you in Jeju for ISOFIC/ISSNP 2014 (International Symposium on Future I&C for Nuclear Power Plants / International Symposium on Symbiotic Nuclear Power Systems).

As a stable base load source for electrical power grids, nuclear power supplies about 15% of the total power demand globally. As of June 2013, a total of 437 nuclear power plants are in operation in 30 countries, and 68 reactors are under construction in 14 countries. Currently, more than 20% of the electric power demand in 14 countries is supplied by the nuclear power generation. As for the share of power generation, nuclear power source accounts for more than 70% of the total electricity generation in France, followed by the Slovak Republic, Ukraine, Belgium, and Hungary (more than 40%); Korea, Sweden, Switzerland, and Slovenia (more than 30%); the United States, United Kingdom, and Russia (less than 20%).

In China, 28 units of nuclear power plants are under construction at 4 sites. The Chinese government plans to increase the capacity of nuclear power generation from 13.8GW (2%) to 86GW (6%) by 2020. However, it changed its plan to increase the generation down to 70~75GW due to lately awareness in safety, and lack of necessary equipment, facilities and human resources.
Korea officially started to generate the electricity via nuclear power with the KORI-1 nuclear power plant in 1978. 35 years passed after then and now, Korea is operating 23 nuclear power plants and building 5 new nuclear power plants domestically and 4 nuclear power plants in UAE(United Arab Emirates). Korea is working not only for successful completion in UAE but also actively promote to award the next nuclear plant contracts in other countries such as Saudi Arabia, South Africa, and Vietnam. Korea had also developed APR1400(Advanced Power Reactor 1400) through continuous technical research and is now working further to develop the advanced type of NPP, APR+. Korea completed the localized technique development of MMIS(Man-Machine Interface System) and RCP(Reactor Coolant Pump), which was installed with the imported products from foreign countries in the last precedent NPP until Shin-Kori NPP unit 3 & 4, and applied them into Shin-Hanul Nuclear Power Plants, Unit 1 & 2.
The research in the nuclear I&C become more and more active recently in many countries. With the increased interest and consideration of digitally upgrading

the analog I&C system in operating nuclear power plants and the newly constructed nuclear power plants, the latest digital technologies such as cyber security, FPGA based controller, defense-in-depth and multi-channel communication are actively researched and some of them are already installed and operated in the NPP. This ISOFIC2014 will be handling these topics.

To remind you the past step, I would like to show the history of ISOFIC. The first ISOFIC (International Symposium On the Future I&C for Nuclear Power Plants) was held in Seoul, 2002, co-hosted by KNICS(Korea Nuclear Instrumentation & Control System) R&D Center and Core University Program Japan-Korea. 160 attendees, including 23 foreign experts from 7 countries (Japan, USA, Russia, China, Norway, Finland, Brazil), presented 49 papers. The second ISOFIC was held in Tongyeong, Korea, 2005, co-hosted by KNICS R&D Center and IAEA. The purpose of the symposium was to internationally promote domestic nuclear I&C technology, to cooperate and exchange technology information with foreign experts and to cooperate with IAEA. 51 papers were presented in the symposium and the number of attendees and technology topics were eminently increased compared with ISOFIC2002, which could make the KNICS project more acknowledged worldwide. The necessity of nuclear I&C localization in Korea had been clarified through "Nuclear I&C commercialization and future direction" panel discussion composed of VIPs from Ministry of Science and Technology, KHNP(Korea Hydro and Nuclear Power), KERI(Korea Electro-technology Research Institute) and several presidents of other companies. The third ISOFIC was held in Harbin, China, 2008 and the fourth ISOFIC was held in Daejeon, Korea, 2011.

As for the another symposium, ISSNP's history, the first ISSNP was held in Tsuruga, Japan, 2007, with the topics of safety and risk studies from social, environmental, economic and other general nuclear engineering integrated aspects of energy systems. The second and third ISSNP were held in Harbin, China in 2008, and Daejeon, Korea in 2011, respectively. I hope you can have a great pleasure and comfortable time in this symposium place, Jeju Island. Jeju is a very beautiful island that was selected as one of the "New 7 wonders of nature worldwide". Not only Mt. Halla, which can be viewed anywhere in the island, but also other superb natural sceneries can help you fully fascinated. I would like to suggest confidently that the sun rise from Mt. Sung-san will be the best memorable moment in your life. I hope that you can enjoy the natural wonders of Jeju Island, during your stay.

July 31, 2014

General Chair of ISOFIC/ISSNP 2014

History

Following the successes of the first ISOFIC (International Symposium on Future I&C for Nuclear Power Plants) in 2002 in Seoul, Republic of Korea and the first ISSNP (International Symposium on Symbiotic Nuclear Power Systems) in 2007, Tsuruga, Japan, the joint international conference of ISOFIC/ISSNP 2014 will be held during August 24~28 2014 at International Convention Center Jeju (ICC Jeju) in Jeju, Republic of Korea. The ISOFIC/ISSNP 2014 is organized to promote academic exchanges of the topics of mainly instrumentation and control (I&C), human machine interface (HMI) technologies, and symbiosis of technology in nuclear industries.

Research Themes

The categories of the conference including the general nuclear engineering are as follows :

- **ISOFIC (I&C)**

  Sensors, modern control, diagnostics and surveillance, digital upgrades, software V&V, cyber security, safety and reliability of digital systems, risk and safety evaluation, etc.

- **ISOFIC (HMI)**

  Human factors engineering, human performance, human reliability assessment, control room design, operator support systems, etc.

- **ISSNP**

  Safety and risk studies from social, environmental and economic aspects, other general nuclear engineering (ex. Reactor physics, thermal-hydraulics, reactor core and plant behavior, nuclear fuel behavior, etc.) and integrated aspects of energy systems (ex. Multipurpose utilization of nuclear energy, nuclear fuel cycle, plant decommissioning, comparative study of nuclear energy with other energy technologies, etc.)

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Organizers

- Organized by Korean Nuclear Society
- Sponsored by Korea Atomic Energy Research Institute, Doosan Heavy Industries & Construction, and Korea Advanced Institute of Science & Technology

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Conference Venue & Meeting Rooms

International Convention Center Jeju (ICC Jeju)

Location 224 Jungmungwangwang-ro, Seogwipo, Jeju-do, 697-858, Korea

Website www.iccjeju.co.kr

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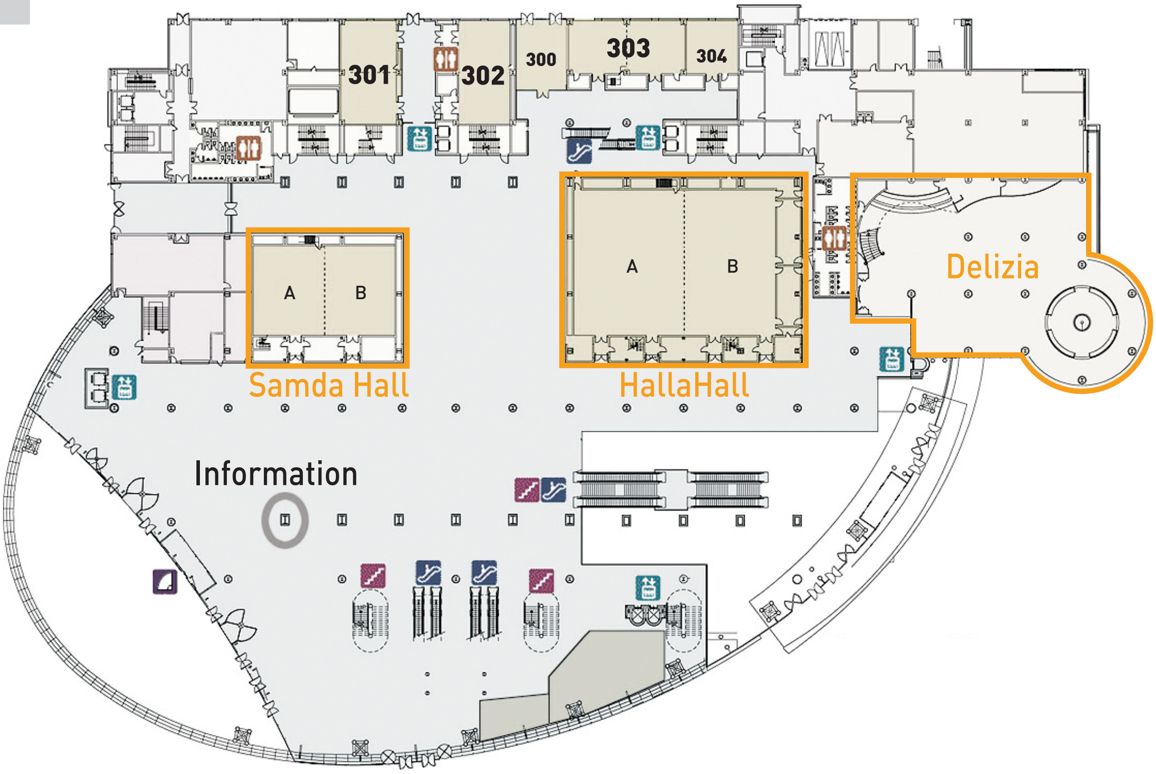

Invited Talk 1

August 25, 2014, Monday, 09:20~09:50 (Samda Hall)

Jong Kyun Park

Director

Division of Nuclear Power, Department of Nuclear Energy

International Atomic Energy Agency

IAEA's Perspectives on Global Nuclear Power – Opportunities and Challenges

Invited Talk 2

August 25, 2014, Monday, 09:50~10:30 (Samda Hall)

Ian Jung

Chief of Instrumentation, Controls and Electronics Engineering Branch 2

Division of Engineering Office of New Reactors

U.S. Nuclear Regulatory Commission, USA

Charting the Course Ahead - Perspectives on the Instrumentation and Control for Future Nuclear Power Plants

As the global landscape of the design and construction of nuclear power plants continues to evolve, an objective reflection of where we are as an instrumentation and control (I&C) discipline is warranted. The reflection will help us either confirm whether we are on the right course or chart a new or different course. Observations of the past and lessons learned can show us much information for us to ponder upon and, more importantly, to take specific actions. In particular, the introduction of digital technology and the ways it has thus far been applied in safety-critical applications such as nuclear power plants warrant a systematic, critical assessment to chart the course ahead. This speech is intended to provide the audience with my assessment of how I&C has evolved and been applied in nuclear power plants and my perspectives on the future of I&C. It will also lay out the vision of the future I&C framework and some of the key elements that must be in place to realize, or at least to move toward, the vision.

Keynote Speech

Current Status and Prospect of Nuclear Power

August 25, 2014, Monday, 13:30~14:10 (Samda Hall)

Jong-Tae Seo

Executive Vice President
KEPCO E&C, Korea

"What should we do in order for the "Nuclear Power" to meet Global Energy Needs?" is the main theme. As the nuclear is one of the clean and valuable options to the global energy demand, the future of nuclear industry will be discussed based on the current status of nuclear power both in Korea and the world. Especially, the nuclear safety in post-Fukushima era as well the challenging issues regarding instrumentation and control area will be discussed. Without ensured safety, there will be no role for nuclear in meeting global energy needs. Also, without a close international cooperation and collaboration, development of future safe and robust I&C system might be difficult. The global nuclear community has to rise again by responding successfully to challenges we are facing after Fukushima disaster under the strong technical leadership of experts.

Current Status of Nuclear Power Development in China

August 26, 2014, Tuesday, 09:00~09:40 (Samda Hall)

Feng Gao

Vice Dean of Design Institute of China Nuclear Power Design Co. Ltd

China General Nuclear Power Group, China

The presentation will provide the following information about nuclear development in China:

- Nuclear power development strategy
- China development China nuclear power planning
- Chinese nuclear power project approval
- China nuclear power development of the three group general situation
- Companies involved
- Chinese projects in plan and projects in progress
- China nuclear power development

Keynote Speech

On-site Structural Health Monitoring for Nuclear Power Plants and Its Application to Reliability Assessment

August 26, 2014, Tuesday, 13:30~14:10 (Samda Hall)

Fumio Kojima

Professor
Organization of Advanced Science and Technology
Kobe University, Japan

Structural health monitoring (SHM) is a key component for keeping safety operation of nuclear power plants. Severe accidents have been often caused by variety of material degradations due to the long-term service of structured materials. SHM plays an essential role in preventing catastrophic events at the course of material aging. It involves the broad concept of assessing ongoing and in-service performance of structures using variety of measurements. Those elements include sensors in structures, data acquisition, data management, data interpretation, diagnosis, etc. In the first part of this talk, typical case studies for critical damages are overviewed and I will discuss how SHM for structural integrity could work. Secondly, several efforts on practical implementations of SHM are demonstrated using various monitoring techniques, such as electromagnetic nondestructive testing, ultrasonic testing, etc. The final part of this talk is devoted to the recent critical issues on SHM associated with the reliability assessment of SHM.

The Challenges of Licensing Modern Instrumentation and Control Systems - A Canadian Perspective

August 27, 2014, Wednesday, 09:00~09:40 (Samda Hall)

Gregory Lamarre

Director SED/DAS
CNSC, CCSN
Canada

This presentation will focus on some of the key challenges being faced by the Canadian Nuclear Safety Commission (CNSC) in conducting ongoing licensing reviews of modern instrumentation and control systems associated with both new designs and system modifications of existing plants. With the rapid evolution of I&C designs, modern regulations and guidance need to be both clear and adaptive enough to ensure that high overall levels of safety and performance are assured regardless of the particular I&C design. Post Fukushima lessons learned also need to be effectively brought into the regulatory framework and appropriately considered by both the operators and regulators in their decision-making. There is an important role to be played by international organisations such as MDEP in the harmonization of requirements as these relate to the fundamental safety principles for modern I&C systems. Finally, the emergence of highly sophisticated and constantly evolving cyber security threats pose a very serious challenge to the continued safe and secure operation of these modern highly interconnected I&C systems.

Program at a Glance

August 24, 2014 (Sunday)	
17:00-19:30	Registration starting at 17:00
18:00-19:30	Welcoming Reception (Ocean View)

	Samda Hall	Room #301	Room #302	Room #303
August 25, 2014 (Monday)				
08:00-18:00	Registration starting at 08:00			
09:00-09:20	**Opening Ceremony (Samda Hall)** **Opening Remarks** Dr. Kook Hun Kim (General Chair) **Welcoming Address** Dr. Moon Hee Chang (President-elect, Korean Nuclear Society)			Chair: Man Cheol Kim (Chung-Ang Univ.)
09:20-09:50	Invited Talk 1: Jong Kyun Park (IAEA)			Chair: Poong Hyun Seong (KAIST)
09:50-10:30	Invited Talk 2: Ian Jung (NRC, USA)			
	Photo time			
10:30-10:40	Coffee Break			
10:40-12:00	ISOFIC(HMI)#1: Advanced Human Machine Interface	ISOFIC(I&C)#1: I&C Circuit Design	ISOFIC(I&C)#2: FPGA Applications	ISSNP#1: Nuclear Safety
12:00-13:30	Lunch Break			
13:30-14:10	Keynote Speech: Jong Tae Seo (KEPCO E&C, Korea) Samda Hall			Chair: Man Gyun Na (Chosun Univ.)
14:10-14:20	Coffee Break			
14:20-16:00	ISOFIC(HMI)#2: Main Control Room	ISOFIC(I&C)#3: Operation and Control 1	ISOFIC(I&C)#4: Plant Diagnostics	ISSNP#2: Thermal Hydraulic
16:00-16:20	Coffee Break			
16:20-17:40	ISOFIC(HMI)#3: Safety Culture	ISOFIC(I&C)#5: Operation and Control 2	ISOFIC(I&C)#6: Cyber Security 1	ISSNP#3: Reactor Design
18:00-20:00	Banquet (Ocean View) Performance “BaramYuHei” by Hana Art			
August 26, 2014 (Tuesday)				
08:00-18:20	Registration			
09:00-09:40	**Keynote Speech: Feng Gao (CNPDC, China) Samda Hall**			Chair: Yang Ming (HEU)
09:40-10:00	Coffee Break			
10:00-12:00	ISOFIC(HMI)#4: Alarm and Diagnostic System	ISOFIC(I&C)#7: Risk Assessment	ISOFIC(I&C)#8: Cyber Security 2 (Special Session)	ISSNP#4: Advances in Functional Modeling Method
12:00-13:30	Lunch Break			
13:30-14:10	Keynote Speech: Fumio Kojima (Kobe Univ., Japan) Samda Hall			Chair: Akio Gofuku (Okayama Univ.)
14:10-14:20	Coffee Break			
14:20-16:00	ISOFIC(HMI)#5: Support for Operators and Field Workers	ISOFIC(I&C)#9: Software Safety 1	ISOFIC(I&C)#10: Severe Accident Monitoring	ISSNP#5: Fukushima Accident Analysis from Different Aspects
16:00-16:20	Coffee Break			
16:20-18:20	ISOFIC(HMI)#6: Human Reliability Analysis and Human Errors	ISOFIC(I&C)#11: Software Safety 2	ISOFIC(I&C)#12: Wireless Applications	ISSNP#6: Will written Procedures Solve Everything?
August 27, 2014 (Wednesday)				
08:00-18:00	Registration			
09:00-09:40	**Keynote Speech: Gregory Lamarre (CNSC, Canada) Samda Hall**			Chair: Hyun Gook Kang (KAIST)
09:40-10:00	Coffee Break			
10:00-12:00	ISOFIC(HMI)#7: Human Factors Engineering	ISOFIC(I&C)#13: Component Monitoring and Prognostics	ISOFIC(I&C)#14: Sensor Technology	ISSNP#7: System Analysis
12:00-14:00	Lunch Break & Best Paper Award			Chair: Gyunyoung HEO (KHU)

August 28, 2014 (Thursday)	
09:00-16:30	Technical Tour (Smart Grid & Wind Power in Jeju)

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TECHNICAL SESSIONS

International Symposium on Future I&C for Nuclear Power Plants
International Symposium on Symbiotic Nuclear Power Systems

ISOFIC(I&C) #1: I&C Circuit Design

Room #301

Session Chair: Shigeru Kanemoto (Aizu Univ.), Jang-Yeol Kim (KAERI)

Development of Digital I&C System using C4ISR Architect Framework

Jae Cheon JUNG (1) and Phamle QUANG (2)
(1) KEPCO International Nuclear Graduate School, Korea
(2) Ninh Thuan Nuclear Power Project Management Board, Vietnam

The architect framework for the digital I&C system is presented in this work. With rapid changes in digital I&C technology, there is a strong need to provide uniform methods to describe the system functions and their performance in context with the physical configuration and logical behavior. C4ISR framework would provide the process and method for the digital system in that it allows the three different views of operational, systems and services, and technical standards. Therefore, stakeholders can share information that is related to the system interfaces, the actions or activities that those components perform, and rules or constraints for those activities from the initial state of system development. As a result, the lifecycle cost and development time for the digital I&C system can also be optimized. These benefits can be obtained by introducing views and products to reveal the logical, behavioral, and performance characteristics of the architecture. To prove this approach, the plant protection system (PPS) is chosen and the measure of effectiveness (MOE) is evaluated. An MOE of PPS is introduced as: functional effectiveness, performance effectiveness, and interoperability effectiveness..

Automatic Testbed Establishing System for Small Digital Devices

Jang-Yeol KIM, Kwang-Seop SON, Se-Woo CHEON and Young-Jun LEE
Korea Atomic Energy Research Institute, Korea

Software testing phase involves component test, software integration test, software-hardware integration test and system test. This paper is focused on a system test. The system test checks the functionality and performance. To qualify as the safety-critical software, the exhaustive testing is very important. If there are many test cases, it still remains as a challenge. System testing of software or hardware is conducted on completeness, correctness, consistency and traceability to evaluate the Code and Standard compliance with specified requirements. System testing falls within the scope of black box testing, and require no knowledge of the inner design of the code or logic. In this study, we established an automatic testbed for small digital devices. The testbed can perform not only on manual mode, but also on automatic mode. We performed the functional test on a small digital device using our testbed. The result from manually testing the device was same as the result from the automated testbed we established.

Designing Bus Structure for Digital Controller of Nuclear Power Plant

Dongil LEE, Gangmin PARK, Myeongkyun LEE, Kwanwoo YOO and Donghwa YUN
PONUTech Co., Ltd., Korea

In this paper, the bus structure design for the digital controller of nuclear power plant is proposed. The most important portion is the bus in the digital controller. A bus is responsible for transmission data among the various I/O board, Processor board and Communication board. An existing bus is a parallel bus structure such as the VME bus (VERSA Module Euro bus) and an extended SRAM (Static Random Access Memory) Controller. CPU processing speed and communication speed is faster than before, but the parallel bus has a speed limit. Because it is the physically lines shared and the weakness about reflected wave. That is, using the parallel bus has been the cause what is the performance degradation of system. To solve the parallel bus problem, the bus has been developed by serial. The configuration of transceiver logic was simplified by not share the bus that all boards and processor boards was connected to the point-to-point. Serial bus was configured using physically LVDS (Low Voltage Differential Signaling). The speed limit of bus was broken by the LVDS what is consisted of a strong against the reflected-wave of signal, noise and etc. Each channel of bus has a different speed mode and is able to set a required transmission way. Different transmission ways of each channel are able to transmit data to match the required response time of system. The LVDS speed has about basically 1Gbps but the bus was composed with less than 50Mbps considering the environmental characteristics of nuclear. It was shown that the configuration of the serial bus was the performance and the reliability more improving than parallel bus.

Highly Reliable NPP Instrumentation Using Constant Voltage Feedback Circuits

Seung J. YOO, Bo H. CHOI, Ji H. KIM and Chun T. RIM
KAIST, Korea

A highly reliable nuclear power plant (NPP) instrumentation using constant voltage feedback circuits is proposed. Contrary to conventional NPP instrumentation, two operational amplifiers are used at auxiliary building to supply constant DC voltage across the potentiometer or wheatstone bridge type sensors, such as resistance temperature detectors (RTD) and strain gauges. The proposed constant voltage feedback circuits maintain its output voltage as constant regardless of the length of lead wire from the auxiliary building to the sensors. A detail analysis of the proposed feedback circuits and design procedures including the internal resistance and parasitic LC components of lead wire are presented. A prototype with lumped RLC values for modeling lead wires is fabricated and experimentally verified to supply constant 10V up to 200m distance under 0.8% error. Due to its versatile characteristics with cost effective structure, the proposed scheme can be generally extended to pressure meters and water-level recorders to guarantee robust measurements without conventional current transducers under severe accidents.

ISOFIC(I&C) #2: FPGA Applications

Room #302

Session Chair: Steve Yang (Doosan HFC), Moon Jae Choi (KEPCO E&C)

FPGA-based I&C Applications in NPPs Modernization Projects: Case Study

Anton A. ANDRASHOV, Vladimir V. SKLYAR and Alexander A. SIORA

Research and Production Corporation Radiy, Ukraine

Instrumentation and Control (I&C) systems represent one of several most important parts of each Nuclear Power Plant (NPP). I&C modernization projects are performed in the context and to support the overall NPP goals, objectives, and internal and external commitments. The goals and objectives of NPPs are defined substantially by the utilities long-term and short-term business plans. FPGA-based I&C applications may contribute to successful implementation of NPPs I&C modernization projects.

FPGAs Emulate Microprocessors – A Successful Case for HFC NPP Digital I&C Upgrade

Allen HSU , Ivan CHOW , Carl REESE , Jong KIM and Steve YANG

Doosan HF Controls Corp., USA

Field Programmable Gate Arrays (FPGAs), as programmable logic devices (PLDs) have gained a great deal of interests for implementing safety I&C applications in nuclear power plants (NPPs) largely owing to the FPGAs' potential advantage over the currently more common microprocessor-based digital I&C applications. First of all, FPGAs have adequate capabilities for most digital I&C applications in NPPs. Secondly, FPGAs provide products with longer lifetime, improve testability, and reduce the drift which occurs in analog-based systems, from hardware perspective. Thirdly, FPGAs, from software perspective, can be made simpler, less reliant on complex software such as operating systems, which should make FPGAs easier to qualify for nuclear safety applications. Fourthly, FPGAs are less vulnerable to cyber attacks when FPGAs implement the I&C systems that do not contain high-level, general purpose software that may be easily subjected to malicious modifications. Finally, FPGAs can bring cost reduction in an I&C digital upgrade because FPGAs can provide simpler licensing process than microprocessor-based digital I&C, and FPGAs can be implemented more efficiently.

This paper will present one successful case for YGN Unit 3&4 I&C upgrade using FPGA-based components to replace the obsolete Intel 8085 Microprocessor-based controllers. In this case, FPGAs emulated the process of the existing microprocessors and interpreted the execution of CPU processing. More than 160 of the FPGA-based SBC-01 controllers replacing the Intel 8085 Microprocessor-based Printed Circuit Boards have been installed and running successfully for safety I&C applications over the last five years. In this upgrade, the new FPGA-based controller board SBC-01 emulated the functions of Intel 8085 microprocessor correctly. It is a successful and cost-effective upgrade.

In this paper, lifecycle design and implementation process and rigorous V&V activities that were used in the upgrade and that are consistent with US NRC regulatory requirements and industrial standards are presented. Then hardware and software requirements including interfaces for replacing the existing microprocessor-based controllers with FPGA-based components are described. Next, HFC

FPGA design philosophy and architecture for emulating Intel 8085 Microprocessor is described. Finally, the experience, benefits, and recommendations using FPGAs to emulate microprocessors for applications in the nuclear safety digital I&C upgrades as well as for new builds are summarized and discussed.

Field Programmable Gate Array-based I&C Safety System

Hyun-Jeong KIM, Koh-Eun KIM, Young-Geul KIM,
Jong-Soo KWON, Woong-Seock CHOI, Se-Do SOHN
and Seung-Min BAEK
KEPCO E&C, Korea

Programmable Logic Controller (PLC)-based I&C safety system used in the operating nuclear power plants has the disadvantages of the Common Cause Failure (CCF), high maintenance costs and quick obsolescence, and then it is necessary to develop the other platform to replace the PLC. The Field Programmable Gate Array (FPGA)-based Instrument & Control (I&C) safety system is safer and more economical than Programmable Logic Controller (PLC)-based I&C safety system. Therefore, in the future, FPGA-based I&C safety system will be able to replace the PLC-based I&C safety system in the operating and the new nuclear power plants to get benefited from its safety and economic advantage. FPGA-based I&C safety system shall be implemented and verified by applying the related requirements to perform the safety function.

This paper deals with the requirements, implementation, verification and related software tools to design the FPGA-based I&C safety system in the nuclear power plant.

ISOFIC(I&C) #3: Operation and Control 1

Room #301

Session Chair: Janos Eiler (IAEA), Peiwei Sun (Xi'an Jiaotong Univ.)

IAEA Activities in the Area of NPP I&C Engineering

Janos EILER
International Atomic Energy Agency

The IAEA is the world's center of cooperation in the nuclear field. It was set up in 1957 as the world's "Atoms for Peace" organization within the United Nations family. The Agency works with its Member States and multiple partners worldwide to promote safe, secure and peaceful nuclear technologies.
The presentation covers three major topics:

- Global nuclear power situation;
- Short introduction of the IAEA;
- Activities in the area of NPP I&C engineering.

The first part of the presentation provides key data on the actual situation of nuclear power generation, including data on operating and shot-down plants, as well as reactors being under construction. The age of currently operating reactors is analyzed to understand the importance of I&C system modernization for the safe and long-term operation of these plants.
The second part provides a short summary on the history of IAEA and its major areas of assistance to global nuclear power.
The third, main part of the presentation describes in details the IAEA work related to instrumentation and control engineering. All major activities are discussed starting with a short introduction and accompanied with practical examples. The areas covered are:

- Technical Working Group on Nuclear Power Plant Instrumentation and Control;
- IERICS Review Missions;
- Meetings, Workshops, Conferences;
- Coordinated Research Projects;
- Technical Cooperation Activities; and
- Publications.

The information discussed in the main part of the presentation will be very useful for all colleagues involved in NPP I&C system research and development, design, manufacture, installation, commissioning, operations, and maintenance. The audiences will gain a comprehensive knowledge on how to find information and publications at the IAEA, how to participate in IAEA activities, and what the key program elements are in the I&C field for the period of 2014-2017.

Advanced Instrumentation, Information, and Control Systems Technologies Research in Support of Light Water Reactors

Bruce P. HALLBERT and Kenneth THOMAS
Idaho National Laboratory, USA

The Advanced Instrumentation, Information, and Control (II&C) Systems Technologies Pathway conducts targeted research and development (R&D) to address aging and reliability concerns with the legacy instrumentation and control and related information systems of the U.S. operating light water reactor (LWR) fleet. This work involves two major goals: (1) to ensure that legacy analog II&C systems are not life-limiting issues for the LWR fleet, and (2) to implement digital II&C technology in a manner that enables broad innovation and business improvement in the nuclear power plant operating model. Resolving long-term operational concerns with the II&C systems contributes to the long-term sustainability of the LWR fleet, which is vital to the nation's energy and environmental security.

Study on Load Follow Operation of CPR1000 without Boron Adjustment

Xinyu WEI, Li WANG and Fuyu ZHAO
Xian Jiaotong University, China

Due to development of electric power system, the proportion of nuclear power and its needs for Load follow operation have become large. Then there would be higher requirements to the power control ability of the nuclear power system.
The project with China's own brands and using the advanced pressurized water reactor (APWR) nuclear power technology, CPR1000, adopts a control method called Mode G, using G banks, N banks, R bank and soluble boron to perform the load follow control. The common Point Model, widely used in conventional core modeling, cannot successfully research the axial power distribution. But in this paper, a two-node dynamic function model was built, taking CPR1000 as an example, considering the coupling coefficient and mutual influence. The transient parameter values were obtained by the steady-state calculation of single channel with its original structure parameter and operation parameter. Then the core system simulation of CPR1000 was modeled. In order to carry out the load follow operation without the boron adjustment, known as BTP mode, the control banks were regrouped to control the average temperature by M banks and axial offset by AO bank. The last, the optimal control implementation was connected to the core simulation platform to operate with the variable daily load. The feasibility and effectiveness of the control policy and the simulation system are validated by the calculation results via one-dimensional in axial computer code APOC.
Results of computations performed for a reference reactor were present, giving the possibility that the optimal control policy could stretches the capability of the reactor to follow an average daily load curve. The paper demonstrated in principle that the BTP mode was feasible for CPR1000.

LPV Modeling and Control of Canadian Supercritical Water-cooled Reactors

Peiwei SUN
Xi'an Jiaotong University, China

Canadian direct-cycle supercritical water-cooled reactor (SCWR) is a pressure-tube type SCWR under development in Canada. The coolant flow rate is required to be maintained in a correct proportion to the reactor power output so that the main steam temperature can be kept at a constant proportion to the reactor power output so that the main steam temperature can be kept at a constant. As a result, the gain and the time constant of the system vary significantly as the power level changes and Canadian SCWR is a highly nonlinear system. To deal with the high degree of nonlinearity, gain scheduling control can be applied. But the stability is only achieved at selected operating points, it cannot be guaranteed for all operating region in theory. What's more, there are abrupt changes when the controller are switching. The control performance can be deteriorated. In this paper, a linear parameter varying (LPV) method is proposed to solve such problems. Linear dynamic models are obtained through Jacobian linearization at six operating points. The power level is chosen as the varying parameter. An LPV model is derived through fitting these linear dynamic models and it is verified with the nonlinear model. $H\infty$ theory is applied to design the LPV controller. Through wide range simulation, the performance of the LPV controller is verified.

Remote Monitoring and Instrumentation Strategies for Integral Reactors

Belle R. UPADHYAYA, , Matthew R. LISH, Ryan A. TARVER and J. Wesley HINES
University of Tennessee, USA

The University of Tennessee is engaged in research and development projects related to instrumentation and controls for small modular reactors (SMR) and integral pressurized water reactors (iPWR). The approach incorporates the deployment of physics-based models for control design and parameter estimation, development of noncontact sensors for flow monitoring, and placement of sensors to maximize fault detection and isolation. The results of research and development illustrate the feasibility of sensor location in space-constrained environment. Major issues and challenges in I&C design are addressed.

ISOFIC(I&C) #4: Plant Diagnostics

Room #302

Session Chair: Akio Gofuku (Okayama Univ.), Gyunyoung Heo (KHU)

An Integration Technique of Diagnos Results in Hybrid-type Diagnostic Systems

Kazuma TAKATA, Akio GOFUKU and Taro SUGIHARA
Okayama University, Japan

A hybrid diagnostic system that a final diagnostic result is given by integrating the results from subsystems is one of promising ways to simulate decision making under uncertainly. An important topic to be considered in developing a hybrid diagnostic system is how to integrate the dagnostic results from subsystems. An integration technique was proposed, where two sets of values called confidence values and trust values were used in the integration of diagnostic results by subsystems. This paper examines the applicability of the integration technique by conducting several simulations under the condition that subsystems diagnose plant condition by random processes. The applicability of the determination of trust values for subsystems is confirmed by several simulations by changing diagnostic performances of subsystems.

Study for the Design Method of Multi-agent Diagnostic System to Improve Diagnostic Performance for Similar Abnormality

Hirotsugu MINOWA and Akio GOFUKU
Okayama University, Japan

Accidents on industrial plants cause large loss on human, economic, social credibility. In recent, studies of diagnostic methods using techniques of machine learning such as support vector machine is expected to detect the occurrence of abnormality in a plant early and correctly. There were reported that these diagnostic machines has high accuracy to diagnose the operating state of industrial plant under mono abnormality occurrence.

But the each diagnostic machine on the multi-agent diagnostic system may misdiagnose similar abnormalities as a same abnormality if abnormalities to diagnose increases. That causes that a single diagnostic machine may show higher diagnostic performance than one of multi-agent diagnostic system because decision-making considering with misdiagnosis is difficult.

Therefore, we study the design method for multi-agent diagnostic system to diagnose similar abnormality correctly. This method aimed to realize automatic generation of diagnostic system where the generation process and location of diagnostic machines are optimized to diagnose correctly the similar abnormalities which are evaluated from the similarity of process signals by statistical method.

This paper explains our design method and reports the result evaluated our method applied to the process data of the fast-breeder reactor Monju.

Semi-Supervised Classification for Fault Diagnosis in Nuclear Power Plants

Jianping MA and Jin JIANG
University of Western Ontario, Canada

Pattern classification methods have become important tools for fault diagnosis in industrial systems. However, it is normally difficult to obtain reliable labeled data to train a supervised pattern classification model for applications in a nuclear power plant (NPP). However, unlabeled data easily become available through increased deployment of supervisory, control, and data acquisition (SCADA) systems. In

this paper, a fault diagnosis scheme based on semi-supervised classification (SSC) method is developed with specific applications for NPP. In this scheme, newly measured plant data are treated as unlabeled data. They are integrated with selected labeled data to train a SSC model which is then used to estimate labels of the new data. Compared to exclusive supervised approaches, the proposed scheme requires significantly less number of labeled data to train a classifier. Furthermore, it is shown that higher degree of uncertainties in the labeled data can be tolerated. The developed scheme has been validated using the data generated from a desktop NPP simulator and also from a physical NPP simulator using a graph-based SSC algorithm. Two case studies have been used in the validation process. In the first case study, three faults have been simulated on the desktop simulator. These faults have all been classified successfully with only four labeled data points per fault case. In the second case, six types of fault are simulated on the physical NPP simulator. All faults have been successfully diagnosed. The results have demonstrated that SSC is a promising tool for fault diagnosis.

Case-based Reasoning Diagnostic Technique Based on Multi-attribute Similarity

Makoto TAKAHASHI (1) and Akio GOFUKU (2)
(1) Tohoku University, Japan
(2) Okayama University, Japan

Case-based diagnostic technique has been developed based on the multi-attribute similarity. Specific feature of the developed system is to use multiple attributes of process signals for similarity evaluation to retrieve a similar case stored in a case base. The present technique has been applied to the measurement data from Monju with some simulated anomalies. The results of numerical experiments showed that the present technique can be utilizes as one of the methods for a hybrid-type diagnosis system.

Simulation of Core Support Barrel Vibration Monitoring Using Ex-Core Neutron Noise Analysis and Fuzzy Logic Algorithm

Robby CHRISTIAN (1), Seon Ho SONG (2) and Hyun Gook KANG (1)
(1) KAIST, Korea
(2) Korea Institute of Nuclear Safety, Korea

The application of ex-core Neutron Noise Analysis (NNA) to monitor the vibration characteristics of a reactor Core Support Barrel (CSB) was studied. Ex-core flux data was obtained using a non-analog Monte Carlo neutron transport method in a simulated CSB model. The implicit capture and Russian Roulette technique was optimized through a sensitivity study to simulate the neutron transport. A combination of two-dimensional and three-dimensional beam and shell mode vibration of CSB was modelled. Parallel processing was employed to reduce the simulation time. An NNA module was developed to analyze the ex-core flux data based on its time variation, Normalized Power Spectral Density (NPSD), Normalized Cross-Power Spectral Density (NCPSD), Coherence and phase differences. The data was then analyzed with a fuzzy logic module to determine the vibration characteristics.
The ex-core flux signal fluctuation was directly proportional to the CSB's vibration observed at 8 and 15 Hz in the beam mode vibration, and at 8 Hz in the shell mode vibration. The Coherence result between flux pairs was unity at the vibration peak frequencies. A set of out-of-phase and in-phase unique pattern of phase differences was observed for each of the vibration models. The fuzzy logic module successfully recognized the correct vibration frequencies, modes, orders, directions, and phase differences within 4.1 miliseconds for the three-dimensional beam and shell mode vibrations.

ISOFIC(I&C) #5: Operation and Control 2

Room #301

Session Chair: Bruce Hallbert (INL), Jae Cheon Jung (KINGS)

Establishment of Scenarios for a Hybrid SIT Operation with an Active System

In Seop JEON and Hyun Gook KANG
KAIST, Korea

After the Fukushima accident, passive nuclear safety was strengthened to prevent the recurrence of a severe accident. A hybrid safety injection tank (H-SIT) is suggested as one of the passive systems used to prevent a severe accident. Therefore, the development of an operation strategy for a H-SIT with an active system is required for increasing the level of safety. In this study, a master logic diagram, difference analysis, matching process, and accident case classification are performed to determine the proper parameters. Then, the conditions that require the use of a H-SIT are determined using a decision process. The operation strategy analysis indicates that a H-SIT can mitigate five failures, namely, the safety injection pump (SIP), passive auxiliary feedwater system (PAFS), depressurization system, shutdown cooling pump (SCP) and recirculation system failures and this strategy also indicates that each scenario has its own pressure range in which the H-SIT can be used.

Instrument Fault Detection Sensitivity of an Empirical Model under Accident Condition in NPPs

Jae Hwan Kim, Seop HUR, Se Woo CHEON and Jung Taek KIM
Korea Atomic Energy Research Institute, Korea

After the recent accident in Fukushima, Japan, it has been proven that we cannot obtain fully reliable information from instruments during severe accident conditions. Although the reactor core really melted down, the RV water level indicator showed a more optimistic value than the actual conditions. Accordingly, plant operators were under the misunderstanding that the core was not exposed. This caused confusion for the incident response. Therefore, it is necessary to be equipped with a function that informs operators of the status of the instrument integrity in real time. If plant operators verify that the instruments are working properly during accident conditions, they able to make safer decisions. In an effort to solve this problem, we considered an empirical model using a Process Equipment Monitoring (PEM) tool as a method of instrument diagnosis in a nuclear power plant.

A Test Model in a RF Anechoic Chamber for the Application of Wi-Fi Communication in Korean Operating NPPs

Yongsik KIM, Minseok KIM, Hosun RYU, Songhae YE and Gwangdae LEE
KHNP Central Research Institute, Korea

The objective of this study is to make a test model and confirm its effectiveness in a radio frequency (RF) anechoic chamber before conducting a field test in Korean operating NPPs for use of Wi-Fi communication technology. This paper is focused on electromagnetic/radio-frequency interference (EMI/RFI) issue and discusses a methodology and its test result for overcoming that issue.
Whenever wireless communication is performed between an access point (AP) and a smart phone, EMI/RFI problem always happens around those devices. It is necessary to decide how many wireless devices local workers will use and select what facilities and systems to protect from EMI/RFI, which are so-called EMI/RFI sensitive equipment. The number of wireless devices was decided as many as possible in the area where those devices could be used, and some sensitive equipment that shall not malfunction under electromagnetic environment were chosen. The test bed which considered above mentioned conditions was constructed and an experiment was carried out inside a radio-frequency anechoic chamber.
Comparing with the allowable operating envelopes for electromagnetic level from RG-1.180, each maximum level of the test results acquired from a RF anechoic chamber is not over the limit even in case of considering the maximum local workers' usage. This result shows that it is highly likely that Wi-Fi communication can be used without any problem if sensitive equipment has observed the electromagnetic susceptibility limit of RG-1.180.

Application of Logic Alarm Cause Tracking System to Shinhanul 1&2 Nuclear Power Plants

Jung Taek KIM (1), Se Woo CHUN (1), Sang Jung LEE (2) and Sung Pil LYU (3)
(1) Korea Atomic Energy Research Institute, Korea
(2) Chungnam National University, Korea
(3) Semyung University, Korea

After the TMI accident, many alarm reduction systems and diagnostic systems have been studied to reduce nuisance alarms and to detect the causes of an abnormal state. These systems provide an operator with information on significant alarms or causes of an abnormal state for an operator to identify that state. In this paper, an operator-aid system, Logic Alarm Cause Tracking System (LogACTs), is proposed for tracking the logics of an alarm, finding the causes of an alarm, displaying the highlighted alarm procedure related to the causes, and suppressing and filtering nuisance alarms due to the physical or logical connections between components or systems in an abnormal state. The system can be used by an operator to identify the detailed causes of an alarm without checking all the causes of the candidates by alarms. The proposed system will be applied to a Korean Standard Nuclear Power Plant of a PWR, ShinHanul 1&2 Nuclear Power Plant.

ISOFIC(I&C) #6: Cyber Security 1

Room #302

Session Chair: Rizwan Uddin (Univ. of Illinois), Chul Hwan Jung (CNSC)

Analysis of Operational and Management Cybersecurity Controls for Nuclear Facilities

Jin Seok OH and Jae Cheol RYOU
Chungnam National University, Korea

U.S. NRC developed this RG 5.71 by tailoring the baseline security controls described in NIST Special Publication 800-53 "Recommended Security Controls for Federal Information Systems and Organizations" to provide an acceptable method to comply with the 10 CFR 73.54. The purpose of this publication is to provide guidelines for selecting and specifying security controls for information systems. In this paper, we are going to analyze and compare the NRC RG 5.71 and the NIST SP800-53, in particular, for operational security controls and management security controls. If RG 5.71 omits the specific security control that is included in SP800-53, we would review that omitting is adequate or not. If RG 5.71 includes the specific security control that is not included in SP800-53, we would also review the rationale. And we are going to consider some security controls to strengthen cybersecurity of nuclear facilities.

Methodology for Applying Cyber Security Risk Evaluation from BN Model to PSA Model

Jinsoo SHIN (1), Gyunyoung HEO (1), Hyun Gook KANG (2) and Hanseong SON (3)
(1) Kyung Hee University, Korea
(2) KAIST, Korea
(3) Joongbu University, Korea

There are several advantages to use digital equipment such as cost, convenience, and availability. It is inevitable to use the digital I&C equipment replaced analog. Nuclear facilities have already started applying the digital system to I&C system. However, the nuclear facilities also have to change I&C system even though it is difficult to use digital equipment due to high level of safety, irradiation embrittlement, and cyber security. A cyber security which is one of important concerns to use digital equipment can affect the whole integrity of nuclear facilities. For instance, cyber-attack occurred to nuclear facilities such as the SQL slammer worm, stuxnet, DUQU, and flame. The regulatory authorities have published many regulatory requirement documents such as U.S. NRC Regulatory Guide 5.71, 1.152, IAEA guide NSS-17, IEEE Standard 7-4.3.2, and KINS Regulatory Guide 08.22.
One of the important problem of cyber security research for nuclear facilities is difficulty to obtain the data through the penetration experiments. Therefore, we make cyber security risk evaluation model with Bayesian network (BN) for nuclear reactor protection system (RPS), which is one of the safety-critical systems to trip the reactor when the accident is happened to the facilities. BN can be used for overcoming these problems. We propose a method to apply BN cyber security model to probabilistic safety assessment (PSA) model, which had been used for safety assessment of system, structure and components of facility. The proposed method will be able to provide the insight of safety as well as cyber risk to the facility.

A Case Study on Cyber Security Program for the Programmable Logic Controller of APR-1400

S.H. SONG (1), M.S. LEE (2), T.H. KIM (2), C.H. PARK (3), S.P. PARK (4), and H.S.KIM (5)
(1) Korea University, Korea
(2) Formal Works Inc., Korea
(3) LINE Corp., Japan
(4) AhnLab, Inc., Korea
(5) Sejong University, Korea

As instrumentation and control (I&C) systems for modern Nuclear Power Plants (NPPs) have been digitalized to cope with their growing complexity, the cyber-security has become an important issue. To protect the I&C systems adequately from cyber threats, such as Stuxnet [1]that attacked Iran's nuclear facilities, regulations of many countries require a cyber-security program covering all the lifecycle phases of the system development, from the concept to the retirement. This paper presents a case study of cyber-security program that has been performed during the development of the programmable logic controller (PLC) for modern NPPs of Korea. In the case study, a cyber-security plan, including technical, management, and operational controls, was established through a security risk assessment. Cyber-security activities, such as development of security functions and periodic inspections, were conducted according to the plan: the security functions were applied to the PLC as the technical controls, and periodic inspections and audits were held to check the security of the development environment, as the management and operational controls. A final penetration test was conducted to inspect all the security problems that had been issued during the development.The case study has shown that the systematic cyber-security program detected and removed the vulnerabilities of the target system, which could not be found otherwise, enhancing the cyber-security of the system.

Digital I&C and Cyber Security for Nuclear Power Plants

Yongkyu AN, Zbigniew KALBARCZYK, William SANDERS, Calogero SOLLIMA and Rizwan-uddin
University of Illinois, U.S.A

Efforts are underway to switch existing nuclear power plants (NPPs) based on analog to digital control systems, while digital I&C systems are expected to be used in all new NPPs. Digital I&C is expected to solve obsolescence problem of analog components, and improve safety and performance. However, before analog systems can be switched to digital, much work needs to be done to ensure nuclear-grade safety and reliability of these completely new or hybrid systems. The goals of this research project are to identify possible faults and to evaluate resiliency of safety-critical digital I&C systems destined to be used in NPPs. A test bed is being developed for this purpose. This test bed consists of a model of a reactor, a digital controller, and associated communication links. The digital controller has a Triple-Modular Redundant (TMR) architecture to ensure continuous availability of the controller. A simple real time NPP simulator has been developed in LabVIEW using the point kinetics equation for the core, and models for a pressurizer, and a pump. The model for the primary loop is fairly complete, while that for the secondary loop is still in progress. The NPP simulator and the TMR controller, with its associated application program, have been assembled, and communications between them are established. The test bed also includes a set of specialized fault/error injectors to inject different types of faults/errors including both transients and permanents. A fault injection module is developed in LabVIEW in order to simulate hardware failures. The module contains a fault list manager (FLM), a fault injection manager (FIM), and a result analyzer. FLM picks a fault type and fault location at random from the pre-generated list of fault locations and types, and communicates this information to FIM, which injects faults into the system. The paper describes the details of the test bed, the associated fault injection system, and the results of the fault injection studies being carried out using this test bed for digital I&C for NPPs. Potential future use of the test bed includes cyber security tests of digital I&C systems for NPPs, stability analysis of NPP test bed connected to a simulator of the electric grid, and human machine interface and human factor engineering studies of newly developed control rooms for NPPs.

ISOFIC(I&C) #7: Risk Assessment

Room #301

Session Chair: Alain Ourghanlian (EDF), Hyun Gook Kang (KAIST)

Dependability Evaluation of Advanced Diverse Protection System

Yang Gyun OH (1), Yoon Hee LEE (1), Se Do SOHN (1), Seung Min BAEK (1) and Sang Jeong LEE (2)
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For the mitigation of anticipated transients without scram (ATWS) as well as common cause failure (CCF) within the plant protection system (PPS) and the emergency safety feature – component control system (ESF-CCS), the diverse protection system (DPS) has been designed by KEPCO Engineering & Construction Company (KEPCO E&C). Recently KEPCO E&C has developed the advanced diverse protection system (ADPS), which has four redundant channels, in an attempt to enhance a fault-tolerant capability of the system. For the evaluation of overall system improvement effects of the ADPS compared with the DPS, the dependability evaluation results are described herein. For all dependability attributes, this paper suggests a practical dependability evaluation method which uses quantitative dependability scores and indices. An overall dependability evaluation index (DEI) for the ADPS is evaluated with the average value of reliability/ security/maintainability/safety indices (i.e., RID, SID, MID, and SID') for dependability. The evaluation results show that the DEI value of ADPS can be improved by approximately 23% compared with that of the DPS, thanks to its fault-tolerant system architecture, software design changes, and external interface design features. Several suggestions have been made, in this paper, of an overall quantitative dependability evaluation method for the nuclear instrumentation and control (I&C) systems including the DPS and ADPS, and the usefulness of dependability evaluation on nuclear I&C systems has been confirmed.

Improvement of PSA Models Using Monitoring and Prognostics

Gyunyoung HEO, Yoon-Suk CHANG and Hyungdae KIM
Kyung Hee University, Korea

Probabilistic Safety Assessment (PSA) has performed a significant role for quantitative decision-making by finding design and operational vulnerability and evaluating cost-benefit in improving such weak points. Especially, it has been widely used as the core methodology for Risk-Informed Applications (RIAs). Even though the nature of PSA seeks realistic results, there are still 'conservative' aspects. The sources for the conservatism come from the assumption of safety analysis and the estimation of failure frequency. Surveillance, Diagnosis, and Prognosis (SDP) utilizing massive database and information technology is worthwhile to be highlighted in terms of the capability of alleviating the conservatism in the conventional PSA. This paper provides enabling techniques to concretize the method to provide time- and condition-dependent risk by integrating a conventional PSA model with condition monitoring and prognostics techniques. We will discuss how to integrate the results with frequency of initiating events (IEs) and failure probability of basic events (BEs). Two illustrative examples will be introduced: (1) how the failure probability of a passive system can be evaluated under different plant conditions and (2) how the IE frequency for Steam Generator Tube Rupture (SGTR) can be updated in terms of operating time. We expect that the proposed PSA model can take a role of annunciator to show the variation of Core Damage Frequency (CDF) in terms of time and operational conditions.

Effect Analysis of Faults in Digital I&C Systems of Nuclear Power Plants

Seung Jun LEE (1), Wondea JUNG (1) and Man Cheol KIM (2)
(1) Korea Atomic Energy Research Institute, Korea
(2) Chung-Ang University, Korea

A reliability analysis of digital instrumentation and control (I&C) systems in nuclear power plants has been introduced as one of the important elements of a probabilistic safety assessment because of the unique characteristics of digital I&C systems. Digital I&C systems have various features distinguishable from those of analog I&C systems such as software and fault-tolerant techniques. In this work, the faults in a digital I&C system were analyzed and a model for representing the effects of the faults was developed. First, the effects of the faults in a system were analyzed using fault injection experiments. A software-implemented fault injection technique in which faults can be injected into the memory was used based on the assumption that all faults in a system are reflected in the faults in the memory. In the experiments, the effect of a fault on the system output was observed. In addition, the success or failure in detecting the fault by fault-tolerant functions included in the system was identified. Second, a fault tree model for representing that a fault is propagated to the system output was developed. With the model, it can be identified how a fault is propagated to the output or why a fault is not detected by fault-tolerant techniques. Based on the analysis results of the proposed method, it is possible to not only evaluate the system reliability but also identify weak points of fault-tolerant techniques by identifying undetected faults. The results can be reflected in the designs to improve the capability of fault-tolerant techniques.

Dependability Assessment by Static Analysis of Software Important to Nuclear Power Plant Safety

Alain OURGHANLIAN
EDF Research and Development, Chatou, France

We describe a practical experimentation of safety assessment of safety-critical software used in Nuclear Power Plants. To enhance the credibility of safety assessments and to optimize safety justification costs, Electricité de France (EDF) investigates the use of methods and tools for source code semantic analysis, to obtain indisputable evidence and help assessors focus on the most critical issues.

EDF has been using the PolySpace tool for more than 10 years. Today, new industrial tools, based on the same formal approach, Abstract Interpretation, are available. Practical experimentation with these new tools shows that the precision obtained on one of our shutdown systems software is very significantly improved.

In a first part, we present the analysis principles of the tools used in our experimentation. In a second part, we present the main characteristics of protection-system software, and why these characteristics are well adapted for the new analysis tools. In the last part, we present an overview of the results and the limitation of the tools

Implementation of Prognostics for Operation Optimization of Research Reactors

Rahman Khalil UR, Myoung-Suk KANG and Gyunyoung HEO
Kyung Hee University, Korea

The optimization of operation especially maintenance and surveillance of various components and systems of research reactor using prognostic have been emphasized in this study to save cost and time while keeping safety and reliability high. Potential candidates of research reactor, on which prognostic can be implemented, are categorized into category I and category II systems and components based on the working during the operation of research reactor. Category I enlists the vital power system (DC battery life assessment), Instrumentation & Control (I&C systems) and similar ones while valves (pneumatic/motor operated), pumps and likewise are grouped in category II. The work approach for prognostic has been proposed. It has been found, based on literature survey, that prognostic is highly relevant for category I & II systems and components for their surveillance scheduling, because it can reduce cost and downtime, and maintain high level of safety and reliability.

Comparison of Failure Analysis and Operating Experiences of Digital Control Systems

Eun-Chan LEE and Tae-Young SHIN
Korea Hydro & Nuclear Power Co., Ltd, Korea

This study focuses on digital control systems that have the same functions but different designs. Some differences and common points between these two digital control systems are analyzed in terms of vulnerabilities in plant operation. In addition, this study confirms why unexpected outcomes can occur through a comparison of the system failure experiences with the analytic results of FMEA and FTA. This evaluation demonstrates that the digital system may have vulnerable components whose single failures can cause plant transients even if the system has a redundant structure according to its system design

ISOFIC(I&C) #8: Cyber Security 2(Special Session)

Room #302

Session Chair: Robert Anderson (INL), Cheol Kwon Lee (KAERI)

Development of a Cybersecurity Test-bed for NPP I&C systems

C.K. LEE, J.G. SONG, K.S. HAN, G.Y. PARK and J.W. LEE
Korea Atomic Energy Research Institute, Korea

Security concerns about Nuclear Power Plants(NPPs) systems were generally related to protecting against physical attacks. However, since a decade ago, there has been a growing recognition that I&C and digital control systems are vulnerable to cyber attacks from sophisticated methods and it has become very hard to guarantee the safety of I&C systems in nuclear facilities from highly intelligent cyber attacks like Stuxnet virus. It is also anticipated that the possibility of cyber attacks like APT(Advanced persistence Threats) and their impacts will be increased as information technology advances.

Now KAERI is planning to prevent cyber incidents from occurring in nuclear facilities through developing countermeasures such as cyber security devices and related technologies. However, the possibility of cyber incidents grows and the techniques of cyber attack evolve continuously regardless of the countermeasures applied to I&C systems. This fact leads us to develop a test-bed for conducting vulnerability assessments, network scans, and penetration tests, instead of performing these on real NPP I&C systems. In this research we designed an ESF-CCS test-bed for a variety of cyber security tests. Several considerations have contributed to the establishment of test-bed, which are the use of: 1) same equipment and devices, 2) same application softwares, 3) same control functions for related components, 4) same interfaces between components, and 5) monitoring and acquiring of the status of system and components which function recording, storing, recalling and displaying inputs to and output from the test-bed including real-time operational data.

The purposes of ESF-CCS test-bed are: 1) configuring the test environment against the possible internal and external cyber threats, 2) identifying the possibility whether previously known malicious code can impact a specific nuclear power facilities, 3) analyzing the effects and assessing the risks of nuclear power plants, provided that cyber attacks occurred, 4) performing the penetration tests targeting at NPP I&C in-service and under construction, and reporting the security issues and their potential impacts of cyber attacks on nuclear power facilities, 5) determining the feasibility of a particular set of cyber attack vectors, and 6) identifying the higher or lower vulnerabilities that could result from improper system configuration, operational weaknesses, and known/unknown hardware or software flaws.

In addition, the design details of cyber security monitoring for the test-bed and security data logging, will be introduced.

A Tool for I&C System Architecture Design: the French Connexion Cluster

Christophe POIRIER (1), Siwar KRIAA (1), François PEBAY-PEYROULA (2), Chokri MRAIDHA (3), and Valérie ZILLE (4)
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(2) Atos Worldgrid, Grenoble, France
(3) CEA, LIST, France
(4) Areva NP, Paris La Defense, France

In this paper we present a model-based engineering framework covering the I&C design process from requirements to the design of I&C architectures. This framework based on Increment and Papyrus open source tools provides an extensible modeling environment as well as traceability and verification tools.

Cyber Security Controls Selection for NPP I&C Assets

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Recently, various control equpment and communication protocols have been applied to nuclear. power plants. Also, the design characteristics of nuclear I&C systems are becoming more complex and the possibility of cyber-attacks using malicious code has been increased. In Korea, regulatory body have required cyber security plan for nuclear I&C system. Also, all I&C system and equipment must be classified according to cyber security level and technical, operational and management security controls must be provided based on each level [1, 2]. It is necessary to determine the best set of security controls for NPP I&C system [3].
In our research, selection process of security control which can be used for I&C system has developed. Figure 1 shows the overall structure of the selection process. The process includes 3 steps which are selection of security control, implementation of security control and evaluation of security control. The methodology which has been developed by this research might be used for establish, implement, evaluate the security controls for protecting nuclear I&C system from cyber-attacks.

Canadian Regulatory Framework of Cyber Security for Nuclear Power Plants and the CSA N290.7 Standard

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Canadian Nuclear Safety Commission, Canada

Cyber security has been an emerging issue to all stakeholders of nuclear power plants (NPPs) including design authorities, operating utilities and regulatory bodies. This area is of particular interest to those who work for I&C systems designs and operations. To regulate or support the cyber security activities of NPPs, the IAEA, regulatory bodies and associated standards organizations have developed or have been developing regulatory governance documents, guidance documents or standards for the cyber security of NPPs.
The Canadian Nuclear Safety Commission (CNSC) has developed regulatory requirements for the cyber security program and I&C design of NPPs. The requirements can be found in the CNSC REGDOC-2.5.2. The Canadian Standards Association (CSA) has developed the N290.7 "Cyber security for nuclear power plants and small reactor facilities." This new CSA standard is developed based on the experiences of Canadian nuclear industries and regulatory activities of the CNSC, and is aligned with the general consensus of international practices. This standard will be issued at the end of 2014. The CNSC is planning to refer to the standard in its regulation of cyber security measures for the detailed requirements of cyber security for NPPs.
This presentation provides the status of the Canadian

regulatory framework for cyber security of nuclear power plants as well as the key structures and requirements of the N290.7. The background issues in developing the N290.7 are discussed, such as the identification and classification of cyber essential assets, elements of the cyber security program, the lifecycle approach, and the selection and application rules of cyber security controls. Applications of the N290.7 standard to I&C systems designs and operations for cyber security are also discussed.

Stairway to the Secure Nuclear Power Plant - How to Build Its Technology Roadmap

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National Security Research Institute of ETRI, Korea

Nuclear power plants (NPPs) play an important role in electricity generation. Nuclear power provides about 74.8 percent of electricity production in France, 51 percent in Belgium, and 30.4 percent in South Korea according to the report of World Nuclear Association (May 2013, www.world-nuclear.org). Despite of world'shuge dependence on nuclear power, as we have learned from Fukushima, the safety of NPP facilities cannot be empathized enough. For the sake of the secure NPPs, the security technologies should be identified to protect them. It cloud be possible that NPPs adopt existing security techniques as they are, or a new security technique might be necessary to build more secure NPPs. After listing up all (existing or new) security techniques and prioritize them, it is essentia l to draw the technology roadmap for the next step: what should we do tomorrow? The steps of building the technology roadmap are as follows:

1) Identify what components are in the NPP facility and analyze what threats exist in each component.
2) List up the requirement of NPP's cyber-physical security(CPS). In this step, the architect can refer several standards and regulatory guides (e.g. RG 5.71 [1]). The history of NPP CPS related documents are well de scribed in the introduction of Song, et al. [2].
3) Collect all existing security techniques for IT systems.
4) Couple the threats at NPPs (step 1) with the requirements of NPP CPS (step 2).
5) Link the requirements of NPP CPS (step 2) with the existing security techniques (step 3).
6) Make things tidy. At this step , it is possible to acquire which threats of NPP CSP need which security techniques.
7) Check if the chosen security techniques are applicable to the target NPP components. Find necessary but non-existent security techniques if possible.
8) Put the final securit y techniques in order.
9) Build the technology roadmap according to the result from step 8.

The contribution is that this paper provides the methodology of building the technology roadmap for secure nuclear power plants and several corresponding examples in detail.

Nuclear Instrumentation and Control Cyber Testbed Considerations – Lessons Learned

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Identifying and understanding digital instrumentation and control (I&C) cyber vulnerabilities within nuclear power plants and other nuclear facilities is critical if nation states desire to operate nuclear facilities safely, reliably, and securely. To demonstrate objective evidence that cyber vulnerabilities have been adequately identified and mitigated, a testbed representing a facility's critical nuclear equipment must be replicated. Idaho National Laboratory (INL) has built and operated similar testbeds for common critical infrastructure I&C for over 10 years. This experience developing, operating, and maintaining an I&C testbed in support of research identifying cyber vulnerabilities has led the Korean Atomic Energy Research Institute of the Republic of Korea to solicit the experiences of INL to help mitigate problems early in the design, development, operation, and maintenance of a similar testbed. The following information will discuss I&C testbed lessons learned and the impact of these experiences to KAERI.

ISOFIC(I&C) #9: Software Safety 1

Room #301

Session Chair: Janne Valkonen (VTT), Jang Soo Lee (KAERI)

Review of Software Reliability Assessment Methodologies for Digital I&C Software of Nuclear Power Plants

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Korea Atomic Energy Research Institute, Korea

Digital instrumentation and control (I&C) systems are increasingly being applied to current nuclear power plants (NPPs) due to its advantages; zero drift, advanced data calculation capacity, and design flexibility. Accordingly, safety issues of software that is main part of the digital I&C system have been raised. As with hardware components, the software failure in NPPs could lead to a large disaster, therefore failure rate test and reliability assessment of software should be properly performed, and after that adopted in NPPs. However, the reliability assessment of the software is quite different with that of hardware, owing to the nature difference between software and hardware. The one of the most different thing is that the software failures arising from design faults as "error crystal", whereas the hardware failures are caused by deficiencies in design, production, and maintenance. For this reason, software reliability assessment has been focused on the optimal release time considering the economy. However, the safety goal and public acceptance of the NPPs is so distinctive with other industries that the software in NPPs is dependent on reliability quantitative value rather than economy. The safety goal of NPPs compared to other industries is exceptionally high, so conventional methodologies on software reliability assessment already used in other industries could not adjust to safety goal of NPPs. Thus, the new reliability assessment methodology of the software of digital I&C on NPPs need to be developed. In this paper, existing software reliability assessment methodologies are reviewed to obtain the pros and cons of them, and then to assess the usefulness of each method to software of NPPs.

Model Checking for Licensing Support in the Finnish Nuclear Industry

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(2) Fortum Power and Heat, Finland

This paper examines how model checking can be used to support the qualification of digital I&C software in nuclear power plants, in a way that is consistent with regulatory demands – specifically, the common position of seven European nuclear regulators and authorised technical support organisations. As a practical example, we discuss the third-party review service provided by VTT for the power company Fortum in the I&C renewal project of the Loviisa plant in southern Finland.

An Effective Application Process for Code Coverage Analysis

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This paper describes a method and procedure to create test cases for code coverage analysis as a part of module testing of Plant Protection System (PPS) software for Shin-Hanul Nuclear Power Plants Units 1&2. The test cases were generated through two step processes. In the

first step, the test cases were generated to achieve 100% coverage for models designed by SCADE. In the second step, the test cases that had been generated at the first step were used to measure coverage for source code and added to achieve 100% coverage when not satisfied with 100% coverage criteria. These two step processes contributed to effective creation of test cases and saving of the time and cost for code coverage analysis.

Determination of RPS Software Test Cases Based on Input Profile Considering Multiple Initiating Events

Mohammad. KHALAQUZZAMAN (1), Jaehyun CHO (1), Seung Jun LEE (1), Man Cheol KIM (2) and Wondea JUNG (1)
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(2) Chung-Ang University, Korea

Probabilistic safety analysis (PSA) of digital reactor protection system (RPS) has been increasingly receiving attention of PSA researchers for the years because of the ambiguity to figure out software failure mechanism. Many approaches have been developed for software reliability estimation. Input-profile-based testing is one of the approaches for software failure probability quantification [1, 2, 3] and is often used by developers to ensure the functionality of the most used functions of a system. RPS software needs to be tested for ensuring the capability of software to perform desired functions on demand. In this regard, it is necessary to prepare the test input spaces for performing software tests effectively. This study has been carried out to identify software test cases and the ranges of the deviation of process parameters at pre-trip condition which is processed by RPS software for generation of trip signals. This work was carried out considering the reactor trip parameters in both primary and secondary systems of PWR, and presents an overview of the selection of RPS software test cases for different process parameters. The number of test cases for different safety parameters could vary due to the occurrences of simultaneous multiple initiating events. A plant process parameter deviates from the normal ranges after occurrence of initiating event and the rate of the change of the deviation could be much faster for simultaneous occurrences of multiple events, depending upon the relationship between process parameters of the system. The input profile of RPS software must be prepared to make sure the functional capability and also for reliability assessment of RPS. Thus, the consideration of multiple initiating event occurrences to determine the software test cases is important. In this regard, case studies have been performed considering different scenarios – main-steam line breaks with loss of feedwater supply, and loss of reactor coolant pump with loss of feedwater supply. This study is expected to be helpful for developing test cases of RPS software.

Comparison of Hazard Analysis Requirements for Instrumentation and Control System of Nuclear Power Plants

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(2) Konkuk University, Korea

A hazard, in general, is defined as "potential for harm." In this paper, the scope of "harm" is limited to the loss of a safety function in a Nuclear Power Plant (NPP). The Hazard Analysis (HA) of an Instrumentation and Control (I&C) systems is to identify the relationship from the logical faults, error, and failure of I&C systems to the physical harm of the nuclear power plant, and also to find the impact of the external hazard, e.g., tsunami, of the nuclear power plant to the I&C systems. This paper includes the survey of the existing hazard analysis requirements in the nuclear industries. The purpose of the paper is to compare the HA requirements in various international standards in unclear domain, specifically the safety requirements and guidance for the instrumentation and control system for the nuclear power plant from IAEA, IEC, IEEE, and NRC.

ISOFIC(I&C) #10: Severe accident Monitoring

Room #302

Session Chair: Seop Hur (KAERI), Yeong Cheol Shin (KHNP CRI)

Post-Accident Monitoring System for Severe Accidents

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To cope with a severe accident such as Fukushima Nuclear power plants, fully independent monitoring system separated (isolated) from the conventional instrumentation and control system is needed. Also, a remote control room which is movable and usable at a distant location is needed for safe plant control and monitoring in emergency. Korea Atomic Energy Research Institute (KAERI) has started a new project to cope with these problems.

Identification of Safety System Malfunction using Correlation Visualization

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(2) Korea Atomic Energy Research Institute, Korea

I&C systems should provide the reliable information particularly when an accident occurs and operators should also take action to mitigate the accident by identifying the cause of accident. However, incorrect information might be provided due to malfunction or failure of safety system caused by an accident. Accordingly, operators get confused and it made difficult for them to take appropriate actions. If they fails to grasp the exact information of progress then it might lead to spread of accident.

Thus, we propose the idea that enables to identify safety system malfunctions or failures in such environment impossible to guarantee of safety system. For identifying the malfunctions or failures of a safety system, the correlation visualization which shows linear relationship between parameters is performed. The correlation visualization results show different images with various scenarios, so it determines which system has problem to operate. Therefore we can identify the safety system operability by the comparison of the correlation visualization results. Furthermore, we can determine specifically which parameter shows abnormal signal using correlation visualization after identifying the operability of safety system. The methodology can determine the situation by viewing the overall signals and include the result robustly though incorrect signal input to the indictor.

Proposed methodology is demonstrated through case

study by specifying SBLOCA as an initiating event that is judged greater contribution of severe accident. Various accident data are obtained according to operability of safety system and confirmed a feasibility of idea by applying to case study.

Overall reliability of I&C system can be improved using suggested methodology. As a result, this research can be helpful in accident response and management if safety system malfunctioned when an accident occurred.

Development of a Severe Accident Mitigation Support with Speediness and Credibility

S. HUR, J.C.PARK, J.G. CHOI, J.T.KIM, and C.H.KIM
Korea Atomic Energy Research Institute, Korea

Since the Fukushima Accident, severe accident management strategies have been widely re-examining globally. The severe accident mitigation guidelines are expected to be reinforced through adding on the countermeasures against site extended damages.

It is difficult to develop the specific procedures for severe accident mitigation because of complexity of the severe accident phenomena. In order to perform any mitigation action, the operating staffs should confirm the plant accident status and determine the proper mitigation functions and mitigation paths through the process limitation check, component availability check, and expected adverse effect investigation. For that reason, it has trouble to perform a prompt mitigation action under the current mitigation strategy. Another concern during severe accidents is to identify the accident status with information credibility. If some information from process instruments is false, it is possible that the staffs have faulty determination and wrong action.

This study suggests a methodology of severe accident mitigation support with speediness and credibility. Using this methodology, the severe accident is automatically identified based on the information credibility check. And then, proper mitigation function, available mitigation routes, and an optimal mitigation path are automatically suggested.

The basic logic of the information credibility is based on environmental evaluation, historical evaluation and some conventional methods such as redundancy and diversity comparison of instruments. To identify the available mitigation routes, availability of paths and components, source status, process limitation, expected adverse effect, and mitigation capability of the path are automatically were evaluated. Among the available routes, the optimal mitigation path was finally suggested based on the path priority criteria and physical relationship.

In-Core-Instrumentation Methods for 3-Dimensional Distribution Information of Reactor Core Temperatures and Melt-down

Yeong Cheol SHIN (1), Myoung Eun, Chae (2) and Sung Jin KIM (2)
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(2) Woojin Inc., Korea

The tsunami-induced nuclear accident at Japanese Fukushima power plants in March 2011 has revealed some weaknesses in the severe accident monitoring system. The plant instrumentation did not provide utility, safety experts, and government officials with adequate and reliable information. The information on the reactor core damage and coolability is critical for making decisions correctly as well as in a timely manner during the course of the mitigation of severe accidents. Current Pressurized Water Reactor (PWR)s have an In-Core-Instrumentation (ICI) system that measures the temperature distribution of the top surface (i.e. Core Exit Temperatures) of the reactor core mainly to indicate when to begin Severe Accident Mitigation Guidelines (SAMG). This design concept giving only the core exit temperature has limitations in terms of sufficiency as well as availability of the information necessary for diagnosis on the status of the degraded core and the effectiveness of the measures taken as mitigation strategies. The reactor core exit temperatures are not sufficient to support the assessment of the degree of the core damage and the location of the molten core debris and recognition whether the core damage progresses on or it is mitigated. The ICI location being at the top of the reactor core also makes the ICI thermocouples vulnerable to melt-down because the upper part of the reactor core uncovers first, thereby melt down at the early stage of the accident. This means that direct indication of reactor core temperature will be lost and unavailable during the later stages of severe accident. To address the aforementioned weaknesses of the current ICIs, it is necessary to develop a new ICI system that provides information that is more expanded and more reliable for accident mitigation. With the enhanced information available, the SAMG can be prepared in more refined and effective way based on the direct and suitable indication of status of damages and the 3-dimensional temperature profile of the core rather than guesses and assumptions. Furthermore, this goal needs to be achieved economically and with minimal changes to current design of reactor and its instrumentation that has been proven and well established through many years of operation. In this paper, methods for a new ICI system to provide three-dimensional view of the reactor core temperatures and melt-down are introduced.

Prediction of Hydrogen Concentration in Containment at Severe Accidents Using FNN

Dong Yeong KIM, Ju Hyun KIM, Kwae Hwan YOO and Man Gyun NA
Chosun University, Korea

Recently, severe accidents of the nuclear power plants (NPPs) have become globally an impending concern. In this paper, severe accidents were analyzed based on OPR1000. The increase of the hydrogen concentration in severe accidents is one of the factors threatening the integrity of the containment. It was determined that a method using fuzzy neural network (FNN) has been developed for predicting the hydrogen concentration. And the FNN model was verified based on the NPPs simulation data acquired by using MAAP4 code. It is expected that the containment can be kept safely because the hydrogen concentration can be predicted well at the beginning of the real accident.

ISOFIC(I&C) #11: Software Safety 2

Room #301

Session Chair: Arndt Lindner (isTec), Jun Beom Yoo (Konkuk Univ.)

Analysis Method of Common Cause Failure on Non-safety Digital Control System

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The effects of common cause failure on safety digital instrumentation and control system had been considered in defense in depth analysis with safety analysis method. However, the effects of common cause failure on non-safety digital instrumentation and control system also should be evaluated. The common cause failure can be included in credible failure on the non-safety system. In the I&C architecture of nuclear power plant, many design feature has been applied for the functional integrity of control system. One of that is segmentation. Segmentation defenses the propagation of faults in the I&C architecture. Some of effects from common cause failure also can be limited by segmentation. Therefore, in this paper there are two type of failure mode, one is failures in one control group which is segmented, and the other is failures in multiple control group because that the segmentation cannot defense all effects from common cause failure. For each type, the worst failure scenario is needed to be determined, so the analysis method has been proposed in this paper. The evaluation can be qualitative when there is sufficient justification that the effects are bounded in previous safety analysis. When it is not bounded in previous safety analysis, additional analysis should be done with conservative assumptions method of previous safety analysis or best estimation method with realistic assumptions.

Reliability Quantification Method for Safety Critical Software Based on a Finite Test Set

Sung Min SHIN (1), Hee Eun KIM (1), Seung Jun LEE (2) and Hyun Gook KANG (1)
(1) KAIST, Korea
(2) Korea Atomic Energy Research Institute, Korea

Software inside of digitalized system have very important role because it may cause irreversible consequence and affect the whole system as common cause failure. However, test-based reliability quantification method for some safety critical software has limitations caused by difficulties in developing input sets as a form of trajectory which is series of successive values of variables. To address these limitations, this study proposed another method which conduct the test using combination of single values of variables. To substitute the trajectory form of input using combination of variables, the possible range of each variable should be identified. For this purpose, assigned range of each variable, logical relations between variables, plant dynamics under certain situation, and characteristics of obtaining information of digital device are considered. A feasibility of the proposed method was confirmed through an application to the Reactor Protection System (RPS) software trip logic.

Software Reliability Estimation of the Reactor Protection System for Lungmen Nuclear Power Station

Jing Ya WANG and Hwai Pwu CHOU
National Tsing Hua University, Taiwan

In this paper, a software reliability estimation method is applied

to estimate the software reliability of the reactor protection system (RPS) for Lungmen ABWR. In order to estimate the software failure probability, a flow network model of software is constructed. The total number of xecutions and the execution time of each software statement are obtained, and the reliability of each statement is obtained. During the test, the one-time test scenario follows a Bernoulli distribution and the multiple-test scenarios follow a binomial distribution. The software reliability of the digital trip module (DTM) and the trip logic unit (TLU) of the RPS of Lungmen ABWR can then be estimated. The results show that the RPS software has a good reliability.

Coded Calculation for Floating Point Values in Safety I&C - Implementation and Experiences

Arndt LINDNER, Christian GERST and Andreas MÖLLEKEN
TÜV Rheinland ISTec-GmbH, Germany

The paper describes a methodology to detect erroneous floating point calculations in digital safety I&C during run time. The methodology has the potential to detect processor failures as well as memory failures. It is based on the extension of the normally used algebra to the complex number plain. In the complex number plain a set of sub-algebras is defined. The sub-algebras are characterized by a subset of valid numbers, the decision criteria for validity of a number and appropriate modified operations (addition, subtraction, multiplication, division). In case of a failure of the processor or the memory, the calculation in any of the sub-algebras will result in complex numbers that are not element of the set of elements of the sub-algebra. This is detected by the given criteria. The theoretical background of the methodology was already presented at the NPIC&HMIT conference in San Diego in 2012. The paper presents the extension of the methodology to logical functions and the implementation in a real I&C platform. The results of practical tests are given. This includes tests of calculation overhead and detection of typical failures. Additionally experiences regarding floating point precision are provided.

HARMONICS — EU FP7 Project on the Reliability and Safety Assessment of Modern Nuclear I&C Software

Janne VALKONEN (1), Sofia GUERRA (2), Robin BLOOMFIELD (2), Nguyen THUY (3), Josef MÄRTZ (4), Bo LIWÅNG (5) and Jari HÄMÄLÄINEN (1)
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The reliability of computer-based systems implementing safety functions is a critical issue for the modernization and construction of nuclear power plants, in particular because software can usually not be proven to be entirely free of defects. The differences in regulation and safety justification principles between different countries restrict efficient co-operation and hinder the emergence of widely accepted best practices. This paper gives an introduction to an EU FP7 project HARMONICS (Harmonised Assessment of Reliability of Modern Nuclear I&C Software, 2011-2014) which has an overall objective to ensure that the nuclear industry has well founded and up-to-date methods and data for assessing software of computer-based safety systems.

An Integrated Software Development Framework for PLC & FPGA based Digital I&Cs

Junbeom YOO (1), Eui-Sub KIM (1), Dong Ah LEE (1) and Jong-Gyun CHOI (2)
(1) Konkuk Univeristy, Korea
(2) Korea Atomic Energy Research Insitute, Korea

NuDE 2.0 (Nuclear Development Environment) is a model-based software development environment for safety- critical digital systems in nuclear power plants. It makes possible to develop PLC-based systems as well as FPGA-based systems simultaneously from the same requirement or design specifications. The case study showed that the NuDE 2.0 can be adopted as an effective method of bridging the gap between the existing PLC and upcoming FPGA-based developments as well as a means of gaining diversity.

ISOFIC(I&C) #12: Wireless Applications

Room #302

Session Chair: Jin Jiang (Univ. of West Ontario), Chang Hwoi Kim (KAERI)

Technical Survey on Applications of Wireless Sensor Networks in Nuclear Power Plants

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(2) University of Electronic Science and Technology, China
(3) AMS Technology Center, USA

Even though, there is no yet general consensus on using wireless technologies in nuclear power plants, the use of wireless sensor network in nuclear power plants (NPPs) has been investigated recently by several industrial and research organizations. The topics of interests include (1) potential interaction of wireless sensor networks with the sensitive protection equipment, (2) radiation damage of the electronics on board sensor notes, (3) optimal placement of relay nodes, and (4) potential applications, such as dose monitoring, and equipment condition monitoring . Several wireless sensor networks have been deployed in the NPPs on a trial basis to perform these tasks. Researchers have also proposed novel techniques to safely deploy wireless sensor networks in NPPs. In addition, several organizations are working to specify standards for WSN in NPP. This paper has reviewed the current state of the art of wireless sensor networks in NPPs.

Challenges and Prospects of Equipment Health Monitoring with Wireless Sensor Network in Nuclear Power Plants

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(2) University of Western Ontario, Canada
(3) AMS Technology Center, USA

A wireless sensor network (WSN) system can offer tremendous benefits to equipment condition monitoring in newly-built and/or refurbished nuclear power plants (NPPs). However, it has not been widely accepted so far because of the following requirements by the NPP operators i) electromagnetic (EM) emissions from the wireless transceivers must not interfere with the functionalities of the sensitive safety and protection systems in the plant, ii) a WSN must perform reliably in the presence of high levels of EM interference from devices such as relays and motor driven pumps, and ionizing radiation sources, 3) dependable WSN performan in harsh industrial environments that are cluttered with cable trays, piping, valves, pumps, motors, and concrete and steel structures, and iv) strict compliance with the nuclear regulatory guideline on EM emissions by the wireless devices. This paper will address the key issues associated with the deployment of WSN for equipment condition monitoring in NPPs. Some promising WSN technologies that can be used in NPP applications are also discussed.

Radiation Resistance Test of Wireless Sensor Node and the Radiation Shielding Calculation

Liqian LI (1), Quan WANG (2), Changjian DENG (3), Quincy ALEXANDER (1), Bhaskar SUR (1), Dongyi CHEN (3) and JIN Jiang (2)
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(2) University of Western Ontario, Canada
(3) The University of Electronic Science and Technology, China

A wireless sensor network (WSN) is being developed for nuclear power plants. Amongst others, ionizing radiation resistance is one essential requirement for WSN to be successful. This paper documents the work done in Chalk River Laboratories of Atomic Energy of Canada Limited (AECL) to test the resistance to neutron

and gamma radiation of some WSN nodes. The recorded dose limit that the nodes can withstand before being damaged by the radiation is compared with the radiation environment inside a typical CANDU (CANada Deuterium and Uranium) power plant reactor building. MCNP calculation is used to extend the limited work range of some measurement instruments. Shielding effects of cadmium and lead to neutron and gamma radiations are also analyzed with the help of MCNP. The shielding calculation can be a reference for the node case design when high dose rate or accidental condition (like Fukushima) is to be considered.

Experimental Evaluation of Wireless Communication Channels under Radiation Environment

Quan WANG (1), Liquan LI (2), Jin JIANG (1), Ataul BARI (1), Changjian DENG (3), Bhaskar SUR (1), and Dongyi CHEN (3)
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(2) Atomic Energy of Canada Limited, Canada
(3) The University of Electronic Science and Technology, China

Deployment of wireless systems in nuclear power plants has attracted a lot of attention recently. However, before wireless systems can be installed in a nuclear power plant, it is necessary to evaluate the effect of radiation environment on electromagnetic wave which is the communication media for all radio wave based wireless systems. This is particular important if the wireless systems are expected to work in a harsh and radioactive environment following a severe accident. This paper presents some results of an experiment for evaluating the effect of radiation on electromagnetic wave. The experiments involve placing transmitter antenna and receiver antenna in a hot cell with variable strength of radiation to study the attenuation effects of the radioactive media. The results indicate that radiation does not effect on the electromagnetic wave propagation. This fact should be considered during the design and deployment wireless systems in a potentially radioactive environment.

Impact of EM Emissions from WSN Nodes on Sensitive Protection Equipment in NPP – an Experimental Characterization

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One of the major issues that prevents wireless sensor networks (WSNs) from being widely deployed in nuclear power plants (NPPs) is that the EM emission from WSN devices can potentially interfere with the sensitive measurement equipment in the safety and protection systems of the plant. In practice, unintentional NPP shut down due to the use of wireless devices in the plant has been reported. This paper discusses the results of experiments carried out in Chalk River Laboratories of Atomic Energy of Canada Limited (AECL) to study the impact of EM emissions from standard WSN modules on some of the most sensitive instruments found in NPPs. Characteristics of a WSN that will lead to a safe deployment in a NPP have also been discussed.

Performance Evaluation of Terrestrial Emergency Communication System in NPPs

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(2) KAIST, Korea

The Fukushima accident induced by the great earthquake and tsunami reveals the vulnerability of I&C System. In the severe environment, the normal I&C system did not work properly and results in false information about the internal situation in NPP. Eventually the accident was not properly handled at the early stage. Therefore advanced emergency response system using a wireless channel is necessary to cope with the severe accident. In this paper, we introduce the ERS consisting of the HMS and MCS the ECS linking the HMS with MCS and the performance requirement of the ECS is analyzed. The ECS satisfying the requirement is designed conceptually and the performance of the ECS is evaluated through analysis and simulator. The terrestrial communication system is designed based on the IEEE 802.11. Analyzed performance results prove that the performance requirement can be sufficiently achieved. But if the scalability of data capacity is considered later, use of the advanced 802.11 standard such as 802.11n and multiple signal paths between the HMS and MCS are necessary.

ISOFIC(I&C) #13: Component Monitoring and Prognostics

Room #301

Session Chair: **J. Wesley Hines (UTK), Jung Taek Kim (KAERI)**

Lifecycle Prognostic Model Development and Initial Application Results

Brien JEFFRIES, J. Wesley HINES, Alan NAM, Michael SHARP and Belle UPADHYAYA
The University of Tennessee, USA

In order to obtain more accurate Remaining Useful Life (RUL) estimates based on empirical modeling, a Lifecycle Prognostics algorithm was developed that integrates various prognostic models. These models can be categorized into three types based on the type of data they process. The application of multiple models takes advantage of the most useful information available as the system or component operates through its lifecycle. The Lifecycle Prognostics is applied to an impeller test bed, and the initial results serve as a proof of concept.

Summary of Nuclear Power Plants Prognostics and Health Management Report: PNNL-21515

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The recent changes in oil and gas production, as well as wider geo-political events are causing the economics for electricity generation to currently be in a state of some turbulence in the United States (USA). In spite of this, nuclear power continues to meet about a fifth of the electricity needs in the USA. Currently, three separate thrusts to ensure safe and economical nuclear power development to give energy security are being pursued in the USA: (i) longer term operation for the legacy fleet, from 40–60 and possibly 60–80 years; (ii) four near-term new nuclear plants with a 60-year design life; and (iii) small modular reactors (SMR) design certification, which are expected to employ light water reactor technology, at least in the medium term. Within these activities, attention is turning to enhanced methods for plant component and structural health management. The state of the art in prognostic and health management systems for nuclear power plants was recently reviewed and presented in a report (PNNL-21515); this paper summarizes the key findings of that review.

Advanced Machine Learning Algorithm Application for Rotating Machine Health Monitoring

Shigeru KANEMOTO, Masaya WATANABE (1) and Noritaka YUSA (2)
(1) The University of Aizu, Japan
(2) Tohoku University, Japan

The present paper tries to evaluate the applicability of conventional sound analysis techniques and modern machine learning algorithms to rotating machine health monitoring. These techniques include support vector machine, deep leaning neural network, etc. The inner ring defect and misalignment anomaly sound data measured by a rotating machine mockup test facility are used to verify the above various kinds of algorithms. Although we cannot find remarkable difference of anomaly discrimination performance, some methods give us the very interesting eigen patterns corresponding to normal and abnormal states. These results will be useful for future more sensitive and robust anomaly monitoring technology.

Fuzzy Logic Approach to Diagnosis of Feedwater Heater Performance Degradation

Yeon Kwan KANG, Hyeonmin KIM, Gyunyoung HEO (1) and Seok Yoon SONG (2)
(1) Kyung Hee University, Korea
(2) KHNP, Korea

Since failure in, damage to, and performance degradation of power generation components in operation under harsh environment of high pressure and high temperature may cause both economic and human loss at power plants, highly reliable operation and control of these components are necessary. Therefore, a systematic method of diagnosing the condition of these components in its early stages is required. There have been many researches related to the diagnosis of these components, but our group developed an approach using a regression model[1] and diagnosis table[2], specializing in diagnosis relating to thermal efficiency degradation of power plant. However, there was a difficulty in applying the method using the regression model to power plants with different operating conditions because the model was sensitive to value. In case of the method that uses diagnosis table, it was difficult to find the level at which each performance degradation factor had an effect on the components. Therefore, fuzzy logic was introduced in order to diagnose performance degradation using both qualitative and quantitative results obtained from the components' operation data. The model makes performance degradation assessment using various performance degradation variables according to the input rule constructed based on fuzzy logic. The purpose of the model is to help the operator diagnose performance degradation of components of power plants.

This paper makes an analysis of power plant feedwater heater by using fuzzy logic. Feedwater heater is one of the core components that regulate life-cycle of a power plant. Performance degradation has a direct effect on power generation efficiency. It is not easy to observe performance degradation of feedwater heater. However, on the other hand, troubles such as tube leakage may bring simultaneous damage to the tube bundle and therefore it is the object of concern in economic aspect. This study explains the process of diagnosing and verifying typical failure mode of feedwater heater such as high drain level, low shell-side pressure, tube-side plugging and water box plate defect based on fuzzy logic approach and simulation model.

Development of an Integrity Evaluation Method for Safety-Critical Components of NPPs using MFM

Young Gyu NO and Poong Hyun SEONG
KAIST, Korea

The maintenance strategy of NPPs has been changed from time-based maintenance to condition-based maintenance. In this study, the safety-critical components included in safety-critical system can be usually divided into motor-driven valves and pumps. Among the various pumps included in safety-critical component, high pressure safety injection pump was selected and the failure modes of safety-critical components are analyzed to develop the monitoring technology. Also, this study developed the integrity evaluation method using MFM. This is a methodology in means-end and part-whole way, for automatic real time fault diagnosis of power plant process failure. This study shows MFM based fault diagnosis approach for the HPSI pump. Therefore, the possibility of using MFM for the integrity evaluation for HPSI pump in confirmed through this study.

Vibration Signal Analysis of Main Coolant Pump Flywheel Based on Hilbert-Huang Transform

Meiru LIU, Hong XIA, Bin LI and Yang YANG
Harbin Engineering University, China

In this paper, a 3D model for the dynamic analysis of flywheel based on finite element method is presented. The static structure analysis for the model provides stress & strain distribution cloud charts. The modal analysis provides the basis of dynamic analysis due to its ability to obtain the natural frequencies and the vibration-made vectors of the first 10 orders. The results show the main faults are attrition and crack that also indicated the locations and patterns of faults. Fault dynamic simulation has been performed to gain the vibration signals of flywheel under different faults' condition. In this paper, we attempt to present an algorithm for the application of Hilbert-Huang transform (HHT) for the analysis of flywheel vibration.This simulation indicated that the proposed flywheel vibration signal analysis method performs well, so that the method can be used for the detection of attrition and crack fault diagnosis in reactor main coolant pump.

ISOFIC(I&C) #14: Sensor Technology

Room #302

Session Chair: Fumio Kojima (Kobe Univ.), Chun Taek Rim (KAIST)

Improvement of the Seismic Alarm and the Spent Fuel Pool Instrumentation post Fukushima Dai-Ichi Accident

HanCheol LEE, Choon-Young JOO, Kyung-In SHIN and Jong-Jae CHOI
KEPCO E&C, Korea

After the Fukushima Dai-Ichi accident, various basic design features were revisited for the existing and new nuclear power plants. One of the topics was associated with the defense against the earthquakes. Among the variety of seismic designs of the plant systems, the design of the seismic alarm and the improvement of the spent fuel pool instrumentation for the conventional plant and MMIS-based plants in Korean are introduced in this paper.
It briefly describes what has been requested or recommended by the US NRC and it introduces the enhancement designed for the following topics for SKN 1&2 plant and the SKN 3&4 plants, respectively.

- Seismically qualified annunciator for seismic alarm
- Improvement of the spent fuel pool (SFP) instrumentation

Mode Control of Guided Wave in Magnetic Hollow Cylinder Using ElectroMagnetic Acoustic Transducer Array

Akinori FURUSAWA, Fumio KOJIMA and Atsushi MORIKAWA
KOBE University, Japan

The aim of this work presented here is to demonstrate the method for exciting and receiving torsional and longitudinal mode guided wave with Electromagnetic Acoustic Transducer (EMAT) ring array. First of all, three-dimensional guided wave simulator is developed in order to analyze propagation of the guided wave numerically. Finite Difference Time Domain (FDTD) method is used for the simulator. To enhance the accuracy of the surface wave, Zero Stress Formulation (ZSF) is used for implementation of the free boundary condition. Second, the guided wave testing systems using EMAT ring array are provided: one is for torsional mode guided wave and the other is for longitudinal mode. Finally, experimental and numerical results are compared and discussed. The results of experiments and simulation agree with well, showing potential of the EMAT ring array for mode controllable guided wave transmitter and receiver.

Measurement of Wall-thinning Defect in Pipeline for Circulation System of Nuclear Power Plant using Shearography

Chan-geun KANG, Hyun-ho KIM, Hyun-chul JUNG, Man-gyun NA and Kyeong-suk KIM
Chosun University, Korea

Shearography was developed by Leendertz based on the concept to derive derivative for deformation after configuring proper optical interferometry. Where defect exists at rigid object, external force will create stress concentration. Rigid body deformation does not involve change in strain rate, and shearography method will be highly suitable to measure the defect in object, and it has less influence from disturbance. Due to these advantages, it has been widely used as non-destructive analysis technique. In this paper, shearography is used to measure wall-thinning defect according to internal temperature change in pipeline for circulation system of nuclear power plant. 2.5 inch pipeline specimen is prepared, and wall-thinning depth is 50% and 75%, respectively. Using pipeline

circulation system, internal temperature of pipeline specimen to 50 ~ 200 ^{o}C is changed 50^{o}C, and blower is used to quench the specimen for measuring defect shape. With the experiment results, 50%, 75% of the depth of the defect is possible to measure the size and shape. Thus, the actual circulatory system can be measured by applying the wall thinning defects.

Evaluation of Defects Detection of Loop System in the Pipe using Lock-in Infrared Thermography

S. C. KIM, H. C. JUNG, T. H. CHOI, H. I. JUNG and K. S. KIM
Chosun University, Korea

The pipes of nuclear power plant could be thinned by the corrosion and fatigue and the defect could lead to a big accident. For this reason, the effective non-destructive testing method is necessary. [1] This perform research of angle rated defect detection conditions and nuclear power plant piping defect detection by Lock-In Infrared Thermography technique. Defects were processed according to change for wall-thinning length, Circumference orientation angle and wall-thinning depth. In the used equipment IR camera and two halogen lamps, whose full power capacitany is 1 kW, halogen lamps and target pipe's distance fixed 2m. To analysis of the experimental results ensure for the temperature distribution data, by this data measure for defect length.

Effect of Lock-in Frequency on Wall-Thinned Defects Detection Using IR Thermography

Kwae Hwan YOO, Ju Hyun KIM, Man Gyun NA, Jin Weon KIM, Hyun Chul JUNG and Kyeong Suk KIM
Chosun University, Korea

Recently, various inspection techniques for improving the safety of nuclear power plants (NPPs) are being studied. Wall-thinned defect of the pipe are a major cause of reducing the NPP integrity. The purpose of this study was to detect the wall-thinned defects of Nuclear Power Plant (NPP) pipes using the lock-in infrared (IR) thermography method. When using the technique of lock-in IR thermography to detect wall-thinned defects of the pipe, it is very important to select the appropriate lock-in frequency. In this study, we applied a cooling and heating method for detecting wall-thinned defects of the pipe of NPPs

Feasibility of Johnson Noise Thermometry Based on Digital Signal Processing Techniques

Inkoo HWANG (1) and Yang-Mo KIM (2)
(1) Korea Atomic Energy Research Institute, Korea
(2) Chung-Nam University, Korea

This paper presents an implementation strategy of noise thermometry based on a digital signal processing technique and demonstrates its feasibilities. A key factor in its development is how to extract the small thermal noise signal from other noises, for example, random noise from amplifiers and continuous electromagnetic interference from the environment. The proposed system consists of two identical amplifiers and uses a cross correlation function to cancel the random noise of the amplifiers. Then, the external interference noises are eliminated by discriminating the difference in the peaks between the thermal signal and external noise. The gain of the amplifiers is estimated by injecting an already known pilot signal. The experimental simulation results of signal processing methods have demonstrated that the proposed approach is an effective method in eliminating an external noise signal and performing gain correction for development of the thermometry.

ISOFIC(HMI) #1: Advanced Human Machine Interface

Samda Hall

Session Chair: Ian Jung (NRC), Yeon Sub Jung (KHNP CRI)

Human Machine Interface (HMI) Developments in HAMMLAB

Håkan SVENGREN, Lars HURLEN and Christer NIHLWING
Institute for Energy Technology (IFE), Norway

In this ongoing project a complete set of 30 inch operator work-station screens and two common large screens have been developed and implemented based on experience from a number of lab experiments and usability tests in HAMMLAB (Halden Man-Machine Laboratory). A state-based alarm system is developed as an add-on to the normal alarm list to reduce the amount of alarms when protection signals are activated. A computerized procedure system is included to improve team transparency, reduce the time to perform procedures and to minimize the risk for erroneous operations. A full evaluation of the design is planned to be conducted spring 2015.

Design of a Human Machine Interface for a Reliability Monitoring System of Nuclear Power Plants

Yuxin ZHANG, Ming YANG, and Shengyuan YAN
Harbin Engineering University, Harbin, China

This paper presents the design of a Reliability Monitoring System (RMS) for nuclear power plant which was newly developed by authors. The RMS, combining a dynamic reliability analysis system with an online fault management system, can assist technical personnel with their daily tasks such as system configuration management, maintenance plan making and state monitoring from a point of view that degradation of equipment performance, bad plant configurations and incorrect actions will decrease the probability of systems performing designed functions. For utilizing the RMS, technical personnel will monitor the plant operational state, conform the fault diagnosis, input the intended operating or maintenance actions, and monitor the probability changes of systems through a graphical HMI. A HMI evaluation experiment was conducted using an eye tracking system and a full scale simulator of PWR NPP. Visual data was recorded and analyzed after the experiment. The experiment results are presented and the HMI design problems revealed by the evaluation experiment are discussed.

HMI Design of MIRROR PLANT for Safe Operation and Application to Vinyl Acetate Monomer Process

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Yokogawa Electric Corporation, Japan

Dynamic plant simulators have always been used off-line for operator training and control loop design prior to the plant construction phase. Here, we propose on-line use of a dynamic plant simulator for the development of new plant operation. The developed MIRROR PLANT is an on-line dynamic plant simulator that can perfectly simulate dynamic plant behavior, and can also be used to forecast future plant behavior by making the computer run the simulation faster than real-time. Using the estimated and forecast data, the plant operator can detect abnormal situations in the plant. Before activating an alarm from the conventional control system, the operator will be able to perform proactive operation to maintain safety. In this paper, we propose a new human–machine interface (HMI) design to realize proactive operation and discuss application of the HMI to the vinyl acetate monomer process as an example of MIRROR PLANT.

Objective Threshold for Distinguishing Complicated Tasks

Jinkyun PARK and Wondea JUNG
Korea Atomic Energy Research Institute, Korea

Estimating the likelihood of human error in a reliable manner is really important for enhancing the safety of a large process control system such as Nuclear Power Plants (NPPs). In this regard, from the point of view of Probabilistic Safety Assessment (PSA), various kinds of Human Reliability Analysis (HRA) methods have been used for several decades in order to systematically evaluate the effect of human error on the safety of NPPs. However, one of the recurrence issues is to determine the level of an important Performance Shaping Factor (PSF) by using a clear and objective manner with respect to the context of a given task. Unfortunately, there is no such criterion for a certain PSF such as the complexity of a task. For this reason, in this study, an objective criterion that is helpful for identifying a complicated task is suggested based on the Task Complexity (TACOM) measure. To this end, subjective difficulty scores rated by high speed train drivers are collected. After that, subjective difficulty scores are compared with the associated TACOM scores being quantified based on tasks to be conducted by high speed train drivers. As a result, it is expected that high speed train drivers feel a significant difficulty when they are faced with tasks of which the TACOM scores are greater than 4.2. Since TACOM measure is a kind of general tool to quantify the complexity of tasks to be done by human operators, it is promising to conclude that this value can be regarded as a common threshold representing what a complicated task is.

ISOFIC(HMI) #2: Main Control Room

Samda Hall

Session Chair: Leena Norros (VTT), Dhong Hoon Lee (KINS)

Main Control Room Upgrade for Kori Unit 1 in Korea

Jaetaeg HA and Moonjae CHOI
KEPCO E&C, Korea

Kori Unit 1 is a 30 years old nuclear power plant and its MCR and MCB was upgraded based on the latest Human Factors Engineering (HFE) principles. The objectives of applying the Human Factors Engineering (HFE) principles are to minimize the human errors and to enhance the safe operation of the plant. In order to systematically incorporate the HFE design principles into the Human System Interface (HSI) design, HFE Program Plan (HFEPP) for Kori Unit 1 was developed and the plan provided an overview of the HSI design process along with detailed methods and results.

The upgrade includes addition of Bypassed and Inoperable Status Indication System (BISI) and the replacement of the conventional MMI devices such as hardwired handswitches, recorders and indicators with new advanced control and display devices using VDUs (Video Display Units). The VDUs significantly improve the effectiveness and efficiency of the monitoring function.

Plant Monitoring System (PMS) and Plant Annunciator System (PAS) were upgraded also by replacing the outdated systems with advanced digital systems with future expansion capability. In addition, the MCR related equipment and/or facilities were replaced or improved. Some of these include the enhancement of MCR interior designs for better working environment, dimmable ceiling lighting, aesthetically pleasing décor of ceiling, wall and floor as well as ergonomically improved operator consoles.

Conducting Multistage HFE Validations - Constructing Systems Usability Case

Jari LAARNI, Paula SAVIOJA, Leena NORROS, Marja LIINASUO, Hannu KARVONEN, Mikael WAHLSTRÖM (1), Leena SALO (2)
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(2) Fortum, Finland

This paper describes how independent stepwise Human Factors Engineering (HFE) validations have been conducted in Fortum Loviisa power plant control room modernization project. We discuss the challenges of HFE verif ication and validation in a project which is realized in multiple phases, stretches over several years in time, and is in tight coupling with automation modernization. These challenges eventually lead to the need of developing a new approach to control HFE validation: conducting validations in a s tepwise manner. In our sub-system validation approach a particular sub-system of the control room are always the main focus of testing but simultaneously we also analyse the overall operational concept and its possible development needs.

An Investigation for Arranging the Video Display An Investigation for Arranging the Video Display Unit Information in a Main Control Room of Advanced Nuclear Power Plants

Chong-Cheng HSU and Chih-Wei YANG
Institute of Nuclear Energy Research, Taiwan

Current digital instrumentation and control (I&C) and main control room (MCR) technology has extended the capability of integrating information from numerous plant systems and transmitting needed information to operations personnel in a timely manner that could not be envisioned when previous generation plants were designed and built.

A MCR operator can complete all necessary operating actions on the video display unit (VDU). It is extremely flexible and convenient for operators to select and to control the system display on the screen. However, a high degree of digitalization has some risks. For example, in nuclear power plants, failures in the instrumentation and control devices could stop the operation of the plant. Human factors engineering (HFE) approaches would be a manner to solve this problem.

Under HFE considerations, there exists "population stereotype" for operation. That is, the operator is used to operating a specific display on the specific VDU for operation. Under emergency conditions, there is possibility that the operator will response with this habit population stereotype, and not be aware that the current situation has already changed.

Accordingly, the advanced nuclear power plant should establish the MCR VDU configuration plan to meet the consistent teamwork goal under normal operation, transient and accident conditions. On the other hand, the advanced nuclear power plant should establish the human factors verification and validation (HF V&V) plan of the MCR VDU configuration to verify and validate the configuration of the MCR VDUs, and to ensure that the MCR VDU configuration allows the operator shift to meet the HFE consideration and the consistent teamwork goal under normal operation, transient and accident conditions.

This paper is one of the HF V&V plans of the MCR VDU configuration of the advanced nuclear power plant. The purpose of this study is to confirm whether the VDU configuration meets the human factors principles and the consistent operation under the various plant operation conditions. The research results are gathered by a subjective rating scale which is done by twenty-seven participants.

There are some significantly statistical results. The conclusion from these results include (1) the highest consent degree of operators for satisfaction of the MCR VDU configuration is in normal condition; (2) under the HFE consideration, the operators have positive response for the present MCR VDU configuration; (3) under the consistent operation, the operators have positive response for satisfaction of the MCR VDU configuration.

Development of an Optimization Method for Determining Automation Rate in Nuclear Power Plants

Seung Min LEE (1), Jong Hyun KIM (2) and Poong Hyun SEONG (1)
(1) KAIST, Korea
(2) KEPCO International Nuclear Graduate School, Korea

Since automation was introduced in various industrial fields, it has been known that automation provides positive effects like greater efficiency and fewer human errors, and negative effect defined as out-of-the-loop (OOTL). Thus, before introducing automation in nuclear field, the estimation of the positive and negative effects of automation on human operators should be conducted.
In this paper, by focusing on CPS, the optimization method to find an appropriate proportion of automation is suggested by integrating the suggested cognitive automation rate and the concepts of the level of ostracism. The cognitive automation rate estimation method was suggested to express the reduced amount of human cognitive loads, and the level of ostracism was suggested to express the difficulty in obtaining information from the automation system and increased uncertainty of human operators' diagnosis. The maximized proportion of automation that maintains the high level of attention for monitoring the situation is derived by an experiment, and the automation rate is estimated by the suggested automation rate estimation method. It is expected to derive an appropriate inclusion proportion of the automation system avoiding the OOTL problem and having maximum efficacy at the same time.

A Study on the Application of the Display Sharing Function In the APR1400 MCR

Sung Kon KANG, Chan Ho SUNG, No Kyu SEONG and Yeon Sub JUNG
KHNP Central Research Institute, Korea

APR1400 digital Main Control Room (MCR) has been applied to Korean Nuclear Power Plants since Shin-Kori 3&4. APR1400 MCR has the advantages to reduce the physical and mental workload and to increase situation awareness through integrated information of Compact Operator Console, Computerized Procedure System (CPS) and Large Display Panel (LDP). But although digital MCR has many advantages, it has a weak point that operators cannot easily oversee other operator's operational behavior like soft control actions. This environment makes it difficult to perform concurrent and independent verification from the viewpoint of human error prevention. This paper presents the application method and design of display sharing function through combination using Large Display Panel (LDP) and Information Flat Panel Display (IFPD) of operator console for more reliable operation than before. Because display sharing function is applied within the Information Processing System (IPS), it is required to maintain the system requirement of the IPS such as the CPU and network load. For quality assurance, display sharing function will be developed in accordance with Human Factors Engineering program (NUREG-0711) and Software Program Manual (SPM). We expect that this technology strengthen the reliable operation of the APR1400.

ISOFIC(HMI) #3: Safety Culture

Samda Hall

Session Chair: Yeon Sub Jung (KHNP CRI), Jong Hyun Kim (KINGS)

Insight and Lessons Learned on Organizational Factors and Safety Culture from the Review of Human Error-related Events of NPPs in Korea

Ji Tae KIM, Dhong Hoon LEE and Young Sung CHOI
Korea Institute of Nuclear Safety, Korea

Event investigation is one of the key means of enhancing nuclear safety deriving effective measures and preventing recurrences. However, it is difficult to analyze organizational factors and safety culture. This paper tries to review human error-related events from perspectives of organizational factors and safety culture, and to derive insights and lessons learned in developing the regulatory infrastructure of plant oversight on safety culture.

Development of Safety Culture Assessment Strategy for Korean NPP

Jung Hwan PARK and Jong Hyun KIM
KEPCO International Nuclear Graduate School, Korea

This paper aims at 1) developing the requirements for a method to evaluate the operational safety culture, 2) evaluating currently available methods based on the requirements, and 3) suggesting a method to evaluate and improve the operational safety culture for Korean nuclear power plants. This paper reviews the widely-used methods to assess safety culture for NPPs and their basis. Then, this paper develops the requirements for the method to evaluate operational safety culture for Korean NPPs. Based on these requirements, Korean Safety Culture Indicators (KSCI) and evaluation measures are also suggested. Finally this paper proposes the guidelines to develop improvements to safety culture from the evaluation results.

Development of a New Safety Culture Assessment Method for Nuclear Power Plants (NPPs)

Sangmin HAN and Poong Hyun SEONG
KAIST, Korea

This study is conducted to suggest a new safety culture assessment method in nuclear power plants. Criteria with various existing safety culture analysis methods are united, and reliability analysis methods are applied. The concept of the most representative methods, Fault Tree Analysis (FTA) and Failure Mode and Effect Analysis (FMEA), are adopted to assess safety culture. Through this application, it is expected that the suggested method will bring results with convenience and objectiveness.

Analysis on Isolation Condenser Operation by Fukushima Daiichi Unit 1 Operators

Man Cheol KIM
Chung-Ang University, Korea

Fukushima Daiichi nuclear accident resulted in the core damage in three reactors and the release of considerable amount of radioactive material to the environment, not to mention significant social impact and anti-nuclear atmosphere all around the world. This paper provides a review of the findings related to shift operators' operation of the isolation condenser in Unit 1 to examine shift operators' response to the situation. Based on the review of the findings, a situation assessment model was developed to analyze shift operators' understanding on whether core cooling was successfully performed in Unit 1 through the operation of isolation condenser. It was found that lack of information could be one of the main causes for the failure in core cooling by the IC in Unit 1. It is also recommended that the differences in the mathematical model for the situation assessment and that of the real operator need to be further investigated.

ISOFIC(HMI) #4: Alarm and Diagnostic System

Samda Hall

Session Chair: Yangping Zhou (Tsinghua Univ.), Poong Hyun Seong (KAIST)

Supporting Operators to Diagnose Abnormal Situations from Annunciated Alarms

Yochan KIM and Wondea JUNG
Korea Atomic Energy Research Institute, Korea

To support operators in the diagnosis of abnormal operating procedures (AOPs), we developed a decision support system for an OPR-1000 type power plant. This system aids operators who identify abnormal situations from annunciated alarms using three functions: an AOP flowchart, AOP search, and alarm simulation. This paper introduces the developed system, compares the characteristics of the functions in the system, and discusses the strength of this approach compared with other previous research. It is expected that the advanced functions may elevate the performance and reliability of operators who manage abnormal situations.

Adavanced Alarm System Design for APR1400

Ho-sic HAM, In-soo KANG, Jeong-heung BANG and Jong-jae CHOI
KEPCO E&C, Korea

Video Display Unit (VDU) for alarm provided to the APR 1400 main control room is realized by the digitalized VDU and enhances the operational convenience and plant safety compared to the existing conventional type. Alarm design among the Man-Machine Interface (MMI) in the main control room of the APR 1400 is very important in the aspect of the operational convenience. This paper describes the major functions of the alarm design applied to the construction of the Korean Nuclear Power Plants.

Development of Alarm Cause Analysis Tools for Nuclear Power Plants

Sung-Pil LYU, Tae-gon YOO, Moo-ryong KIM (1) and Eun-Ju KIM (2)
(1) Semyung University, Korea
(2) Korea Atomic Energy Research Institute, Korea

For alarm-cause analysis, pieces of information such as the associative relationships between related alarms and system equipment and their states are essential. To obtain such information, search and analysis of many documents such as alarm response procedure and system operation procedures are necessary. This process requires many experts' time and efforts and may incur human errors while searching and analyzing such documents.
To obtain the cause of alarm from the related information, the causal relationships between alarms and system equipment should be analyzed from the contents of such procedures. The causal relationships between alarms and system equipment can be found through analysis of the hierarchical structures of documents and the relationships between sentence and paragraphs. In this paper, an alarm cause analysis tool that provides experts information of the causal relationship between related alarms and system equipment through the systematic classification and the automation of interpretations of sentences or words within alarm response procedure and operation procedures is proposed.

Experiments for Evaluating Application of Bayesian Inference to Situation Awareness of Human Operators in NPPs

Seung Geun KIM and Poong Hyun SEONG
KAIST, Korea

Bayesian methodology has been used widely used in various research fields. It is a method of inference using Bayes' rule to update the estimation of probability for the certain hypothesis when additional evidences are acquired. According to the current researches[1], malfunction of nuclear power plant can be detected by using this Bayesian inference which consistently piles up the newly incoming data and updates its estimation. However, those researches are based on the assumption that people are doing like computer perfectly, which can be criticized and may cause a problem in real world application. Studies in cognitive psychology indicates that when the amount of information becomes larger, people can't save the whole data because people have limited memory capacity which is well known as working memory[2], and also they have attention problem. The purpose of this paper is to consider the psychological factors and confirm how much this working memory and attention will affect the resulted estimation based on the Bayesian inference. To confirm this, experiment on human is needed, and the tool of experiment is Compact Nuclear Simulator (CNS).

Research on Composite Real-time Fault Diagnosis Based on MFM and its Application to HTR-PM

Yong ZHANG, Yang Ping ZHOU, Lei SHI, Yu jie DONG and Xiao jin HUANG
INET, Tsinghua University, China

Two conceptual improvements of the fault diagnosis method based on Multilevel Flow Model (MFM) developed by Lind Morten are introduced to achieve the goal of composite real-time fault diagnosis. The first defines a new concept type of MFM, time, so that the temporal failure information of a system can be better presented and recorded. The second one describes an accessible approach to obtain reliable mixed fault diagnosis result by use of causal dependency matrix, rather than the traditional causal dependency graph, and the analysis of multiple root cause superposition. Then the improved method is applied to HTR-PM (High Temperature Gas-cooled Reactor Pebble-bed Module) and it is expected that it will have a good ability to detect and diagnose accidents timely before reactor trip.

Severe Accident Occurrence Time Prediction by using Support Vector Classification and Support Vector Regression

Seung Geun KIM and Poong Hyun SEONG
Korea Advanced Institute of Science and Technology, Korea

The occurrence of severe accident in Fukushima reminded that getting information about severe accident scenario is important in order to achieve ultimate goal of NPPs. In this research, methods for severe accident occurrence time prediction in various LOCA scenarios by using support vector classification and support vector regression were proposed. Break location and size identifications for prior step were properly conducted and time when maximum core temperature exceeds 1200 degree Celsius, reactor vessel failure time and containment failure time were predicted with acceptable errors.

ISOFIC(HMI) #5: Support for Operators and Field Workers

Samda Hall

Session Chair: Francois Dionis (EdF), Seung-Min Baek (KEPCO E&C)

Representing Operational Knowledge of PWR Plant by Using Multilevel Flow Modelling

Xinxin ZHANG (1), Morten LIND (1), Sten Bay JØ RGENSEN (1), Niels JENSEN (1) and Ole RAVN (2)
(1) Technical University of Denmark, Denmark
(2) Safepark Consultancy, Denmark

The aim of this paper is to explore the capability of representing operational knowledge by using Multilevel Flow Modelling (MFM) methodology. The paper demonstrates how the operational knowledge can be inserted into the MFM models and be used to evaluate the plant state, identify the current situation and support operational decisions. This paper will provide a general MFM model of the primary side in a standard Westinghouse Pressurized Water Reactor (PWR) system including sub-systems of Reactor Coolant System, Rod Control System, Chemical and Volume Control System, emergency heat removal systems. And the sub-systems' functions will be decomposed into sub-models according to different operational situations. An operational model will be developed based on the operating procedure by using MFM symbols and this model can be used to implement coordination rules for organize the utilization of different MFM models in different situation. Combining the operational model and different process models, MFM can be used to identify plant situation.

A Conceptual Application for Computer-based Procedures for Handheld Devices

Linda Sofie LUNDE-HANSSEN
Institute for Energy Technology, Norway

This paper describes the concepts and proposed design principles for an application for computer-based procedures (CBPs) for field operators in the nuclear domain (so-called handheld procedures). The concept is focused on the field operators' work with procedures and the communication and coordination between field operators and control room operators. The goal is to overcome challenges with shared situation awareness (SA) in a distributed team by providing effective and usable information design. An iterative design method and user-centred design is used for tailoring the concept to the context of field operations. The resulting concept supports the execution of procedures where close collaboration is needed between control room and field operations, e.g. where particular procedure steps are executed from remote control points and others from the control room. The resulting conceptual application for CBPs on handheld devices is developed for mitigating the SA challenges and designing for usability and ease of use.

Industrial Personal Computer Based Display for Nuclear Safety System

Ji-Hyeon KIM, Aram KIM, Jung-Hee JO, Ki-Beom KIM, Sung-Hyun CHEON, Joo-Hyun CHO, Se-Do SOHN and Seung-Min BAEK
KEPCO-E&C, Korea

The safety display of nuclear system has been classified as important to safety (SIL:Safety Integrity Level 3). These days the regulatory agencies are imposing more strict safety requirements for digital safety display system. To satisfy these requirements, it is necessary to develop a safety-critical (SIL 4) grade safety display system.

This paper proposes industrial personal computer based safety display system with safety grade operating system and safety grade display methods. The description consists of three parts, the background, the safety requirements and the proposed safety display system design.

The hardware platform is designed using commercially available off-the-shelf processor board with backplane bus. The operating system is customized for nuclear safety display application. The display unit is designed adopting two improvement features, i.e., one is to provide two separate processors for main computer and display device using serial communication, and the other is to use Digital Visual Interface between main computer and display device. In this case the main computer uses minimized graphic functions for safety display.

The display design is at the conceptual phase, and there are several open areas to be concreted for a solid system.

The main purpose of this paper is to describe and suggest a methodology to develop a safety-critical display system and the descriptions are focused on the safety requirement point of view.

Integrated Solution for Field Operation

Renaud AUBIN and François DIONIS
EDF/R&D, France

This document presents our approach to design and to implement mobile applications for field operations. Internal on-field studies yield to the fact that the value added by mobile solutions is correlated with the easiness of their integration with each other and with the underlying information systems. Moreover, the fast-growing mobile market brings new concepts to the mass and industrial applications design can benefit from these. As a consequence, a simple components-based approach has been applied to design and develop mobile applications for field operations and on-site experiments of the resulting applications have been conducted. Four applications have already been developed for tagouts, lineups, intelligent padlocks management and human performance tools.

A Development Method of Mobile Computerized Procedure System for the Cooperation among Field Workers and Main Control Room Operators in Korean Nuclear Power Plants

Sungjin LEE, Nokyu SEONG and Yeonsub JUNG
KHNP Central Research Institute, Korea

Human errors can occur during the test and maintenance of steam generator, safety injection system and other various systems and devices in nuclear power plants (NPPs). Most of human errors can be improved by the human error prevention techniques such as self-check, peer-check, concurrent verification and etc. Another important technique is to share work information among main control room (MCR) operators and field workers. Various field service automation tools have been developed with recent information technology in many countries [1]. APR1400 computerized procedure system (CPS) has been developed for the MCR operators of Shin-Kori 3&4 units. Especially, the concurrent verification support design is applied in the construction project of Shin-Hanul 1&2 CPS [2]. It is expected that the proposed mobile CPS can enhance the reduction of human errors by supporting human error prevention techniques and information sharing. This paper describes the technical issues of the mobile CPS (mobile CPS) in the initial development stage. Based on the design of APR1400, CRI CPS has been developed and operated for SKN 3&4 HFE V&V and license test for the MCR operating staff. Therefore the mobile CPS will be developed by upgrading the CRI CPS with improved features.

ISOFIC(HMI) #6: Human Reliability Analysis and Human Errors

Samda Hall

Session Chair: Steven Arndt (NRC), Man Cheol Kim (Chung-Ang Univ.)

A Conceptual Framework of Human Reliability Analysis for Execution Human Error in NPP Advanced MCRs

Inseok JANG (1), Ar Ryum KIM (1), Wondea JUNG (2) and Poong Hyun SEONG (1)
(1) KAIST, Korea
(2) Korea Atomic Energy Research Institute, Korea

The operation environment of Main Control Rooms (MCRs) in Nuclear Power Plants (NPPs) has changed with the adoption of new human-system interfaces that are based on computer-based technologies. The MCRs that include these digital and computer technologies, such as large display panels, computerized procedures, and soft controls, are called Advanced MCRs. Among the many features of Advanced MCRs, soft controls are a particularly important feature because the operation action in NPP Advanced MCRs is performed by soft control. Using soft controls such as mouse control, and touch screens, operators can select a specific screen, then choose the controller, and finally manipulate the given devices.
Due to the different interfaces between soft control and hardwired conventional type control, different human error probabilities and a new Human Reliability Analysis (HRA) framework should be considered in the HRA for advanced MCRs. In other words, new human error modes should be considered for interface management tasks such as navigation tasks, and icon (device) selection tasks in monitors and a new framework of HRA method taking these newly generated human error modes into account should be considered. In this paper, a conceptual framework for a HRA method for the evaluation of soft control execution human error in advanced MCRs is suggested by analyzing soft control tasks.

An Implementation of Operational Experience Analysis for Addressing Human Reliability Analysis Issues in Nuclear Power Plants

Chih-Wei YANG and Hui-Wen HUANG
Institute of Nuclear Energy Research, Taiwan

Human reliability analysis (HRA) is an integral part of probabilistic risk assessments (PRAs). Although various approaches and methods have been proposed since the first HRA was performed almost four decades ago, the technology associated with HRA is still not fully developed. The limitations of the existing HRA approaches become particularly apparent when the role of the human is examined in the context of nuclear power plants (NPPs). HRA approaches in the cognitive perspective try to take into consideration the operator, the system and their interactions. Cognitive models can help in analyzing human mental processes that can lead to error. This study documents the implementation of operating experience analysis in nuclear domains and describes the future improvement of HRA approaches. This review provides a summary of the HRA literature in order to the field of HRA approaches. Researchers may have knowledge of the capability of the tools and an understanding of their strengths and weaknesses in variety types of nuclear reactors.

Consideration on HRA Implementation during LPSD Operation

Ar Ryum KIM (1), Jaewhan KIM (2), Inseok JANG (1) and Poong Hyun SEONG (1)
(1) Korea Advanced Institute of Science and Technology, Korea
(2) Korea Atomic Energy Research Institute, Korea

During low power and shutdown operation, it has been pointed out that the important of human actions is significantly increased. Because automatic control may be disabled, most control room annunciation titles indicate alarm status, and procedures are insufficient, human operators plays a much larger role during outages and in lower power state. In this regards, in order to reduce human errors and secure nuclear power plant safety, it is necessary to identify and estimate human errors during LPSD operations. However, many researchers have argued that there is no comprehensive LPSD human reliability analysis (HRA) method so far. In this study, we reviewed and implemented the existing HRA methods during LPSD operations: Korean standard HRA (K-HRA), Simplified plant analysis risk HRA (SPAR-H), Caused based decision tree (CBDT), and human cognitive reliability/ operator reliability experiments (HCR/ORE). We reviewed HRA methods based on several reports related to the unique aspects of human operator during LPSD operation and HRA requirements. In addition, the existing HRA methods were implemented during various plant operating states (POS) of LPSD operation. Loss of shutdown cooling system (SCS) was selected as initiating event. As a result of review and implementation of HRA methods, we derived the some limitations of the existing HRA methods and related procedure during loss of SCS event.

Identification of Human-induced Initiating Event in the Low & Shutdown Operation by using CESA Method

Yongchan KIM and Jonghyun KIM
KEPCO International Nuclear Graduate School, Korea

This paper suggests a procedure to identify human-induced initiating events during low and shutdown state in Nuclear Power Plant (NPP). Human-induced initiating events, also called Category B actions in human reliability analysis (HRA), are operator actions that may lead directly to initiating events either by themselves or in combination with equipment failures. Most of conventional probabilistic safety analyses (PSAs) typically assume that the frequency of initiating events also includes the probability of human-induced initiating events. However, some regulatory documents require Category B actions to be specifically analyzed and quantified in the PSA. In addition, a NUREG report also addresses that an explicit modeling of Category B actions could potentially lead to important insights for human performance on safety. However, there is no standard procedure to identify Category B actions which are either recommended by regulations or widely used in the PSA. This paper develops a systematic procedure to identify the Category B actions for the shutdown and low power. The procedure includes several steps to derive operator actions that may lead to initiating events in the low and shutdown stage. Those steps are the selection of initiating events to be analyzed, the selection of systems or components, the screening of unlikely operating actions, and quantification of initiating events. The procedure also suggests the detailed activity of each step such as the information required, screening rules, and output of steps. Finally, the applicability of the suggested approach is also investigated to show its feasibility.

An Empirical Investigation into Causality of Unsafe Act and Recovery during EOP Simulation

Sun Yeong CHOI and Wondea JUNG
Korea Atomic Energy Research Institute, Korea

A data collection worksheet and guideline to collect HRA (Human Reliability Analysis) data with simulator data sources were developed for the HRA data handbook project by KAERI. Using the data worksheet, simulator data were collected and analyzed for an HRA qualitative database. The purpose of this paper is to define the causalities of operators' UAs (Unsafe Acts) ending in an inappropriate component manipulation and recovery during an EOP (Emergency Operating Procedure) operation, and to show some results for the causality from a case study. The reason we suggest the causality of an UA is because an inappropriate manipulation during an EOP operation can be resulted by the causality among operators in an MCR (Main Control Room). Therefore, a 'causality' data field was inserted into the data worksheet to identify the real initiator, and related operators for an inappropriate component manipulation. With this 'causality' data field, an HRA analyzer can establish who caused an UA (or a recovery) and who was involved in the process. They can also calculate the HEP (Human Error Probability) grouped by the initiator if they are interested in the HEP by the initiator.

ISOFIC(HMI) #7: Human Factors Engineering

Samda Hall

Session Chair: Leena Salo (Fortum), Sa-Kil Kim (KAERI)

Interactive Virtual Reactor and Control Room for Education and Training at Universities and Nuclear Power Plants

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(2) Toshiba Corporation, Japan

Efficient and effective education and training of nuclear engineering students and nuclear workers are critical for the safe operation and maintenance of nuclear power plants. With an eye toward this need, we have focused on the development of 3D models of virtual labs for education, training as well as to conduct virtual experiments. These virtual labs, that are expected to supplement currently available resources, and have the potential to reduce the cost of education and training, are most easily developed on game-engine platforms. We report some recent extensions to the virtual model of the University of Illinois TRIGA reactor.

A Human Factors Study on an Information Visualization System for Nuclear Power Plants Decommissioning Engineering

Chih-Wei YANG, and Li-Chen YANG
Institute of Nuclear Energy Research, Taiwan

Most nuclear power plants (NPPs) in the world have an operating life of up to 40 years. The utility should prepare a comprehensive decommissioning plan with purpose to document and to display how decommissioning activities can be safely performed. In the past, most studies related to NPPs decommissioning planning put emphasis on technical issues, little attention have been given to human factors in decommissioning activities. In fact, human factors are a critical factor to successful NPPs decommissioning. NPPs decommissioning will face potential risks. These risks include not only dismantling and moving large equipment but also treating with the radioactive materials. Using information visualization system, such as virtual reality (VR) technology, for staff training can improve decommissioning work safety and economy. Therefore, this study presents a study using VR to solve real world problems in the nuclear plant decommissioning. Then appropriate cases for introducing VR systems are summarized and future prospects are given. This study assesses availability and performance of the work training system by using heuristic evaluation and actual experiment. In the result, block type of radiation visibility was found relatively better both in performance and person's preference than other types. The results presented in this paper illustrate the VR applications a NPP decommissioning perspective.

Visual Fatigue Evaluation: Improvement of Reflected Glare on Touch Screen for Nuclear Power Plant

Kang-Hung LIU (1), Chih-Wei Yang (1), Sheue-Ling HWANG (2) and Jin-Liang LIOU (3)

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(3) Taiwan Power Company, R.O.C.

The purpose of this research is to evaluate the visual fatigue of operators caused by glare problems in the main control room of nuclear power plant. Within the limitation in the main control room, reflectors were set under the light source which generates reflected glare on touch screens. Through avoiding the light directly shines on touch screens, reflected glare were eliminated. This research matched up the setting process of reflectors, evaluated the visual fatigue of operators, and collected user's opinions before reflector setting, after the first setting, and after the second setting. The design of reflectors could refer the result of evaluations and the collection of opinions. Nevertheless, the improvement of reflected glare on touch screens could be verified by this evaluations. The result showed that setting reflectors under the light source could eliminate reflected glare effectively, and the visual fatigue was reduced both on subject and object evaluations. However, the setting direction of reflectors has potential effect on operators' visual fatigue, so the real setting of reflectors still need to be evaluated completely. The near point accommodation could reflect the effect of visual fatigue caused by changes of lighting environment. Thus, the verification of new lighting environment according to the near point accommodation is suggested.

Development of a HFE Program for an Operating NPP: Balancing between ExistingDesign Practices and Human Factors Standards

Leena SALO (1) and Paula SAVIOJA (2)

(1) Nuclear and Thermal Power, Fotum, Finland

(2) VTT Technical Research Centre of Finland, Finland

This paper describes HFE program development project conducted at a Finnish power company Fortum. The aim of developing a formal HFE program was to improve integration of human factors issues in design of technical systems and to systematically document the HFE process of the company. As Fortum has a long tradition of designing control room solutions, the starting point of the HFE program development was the existing own design practices. On the other hand, the aim was to create a program which would comply with international guidelines such as NUREG-0711. The program development was conducted by tracing the HFE design practices in an on-going I&C modernization project. This empirical work was carried out by interviews of designers and other HFE key stakeholders. After the explication of the current practices, the gaps, overlaps and differences in relation to the international standards and guidelines were identified. Based on an analysis of current practices and guidelines and standards a new HFE process model was created. The design process model can be followed in modifications which concern systems with human user interfaces of any kind. The model consists of five separate phases which comply with the general engineering design process model utilized at the company. The HFE program is intended to be both a practical guide on how to take human factors issues into consideration in the design of NPP systems and also a tool for the management of HFE activities.

A System Development of a Human Factors Assessment for Human Performance in Nuclear Power Plants

Yeonju OH, Yonghee LEE, Tongil JANG and Sakil KIM
Korea Atomic Energy Research Institute, Korea

The aim of this paper is to set up and develop an assessment system for human factors performance by erroneous response characteristic of operators in nuclear power plants (NPPs). There are new undefined human factors issues such as an uneasy emotion, unpredictable threatening situation, mental workload, and unsafe act to react faster, make better decisions although introduce an advanced main control room because of hierarchical information structural. These human factors issues could be not resolved from human reliability assessment which evaluates the probability of a human error occurring throughout the completion of a specific task. This paper provides assessment guide of human factors issues as an experimental mythology especially, presents an assessment case of measurement and analysis from neurophysiology approach. It would be the most objective psycho-physiological research technique on human performance for a qualitative analysis considering the workload, usability, and safety aspects. This paper can be an important trial to experimentally assessment human performance and it can be used as an index for recognition and a method to apply human factors engineering V&V, which is required as a mandatory element of human factor engineering program plant for a NPP design.

Future Direction of the Instrumentation and Control System for Security of Nuclear Facilities

Woo Jin KIM, Ho Bin YIM and Jae Kwang KIM
Korea Institute of Nuclear Nonproliferation and Control, Korea

Instrumentation and control (I&C) systems are pervasively used as a vital component in modern industries. Nuclear facilities, such as nuclear power plants (NPPs), originally use I&C systems for plant status monitoring, processes control, and many other purposes. After some events that raised security concerns, application areas of I&C systems have been expanded to physical protection of nuclear material and facilities. As nuclear policies over the world are strengthening security issues [1], the future direction of roles and technical requirements of security related I&C systems is described: An introduction of I&C systems, especially digitalized I&C systems, to security of nuclear facilities requires many careful considerations, such as system integration, verification and validation (V/V), etc. Institute of Nuclear Nonproliferation and Control (KINAC) established "International Nuclear Nonproliferation and Security Academy, INSA" in 2014. One of the main achievements of INSA is test-bed implementation for technical criteria development of nuclear facilities' physical protection systems (PPSs) as well as for education & training of those systems. The test bed was modified and improved more suitably from the previous version to modern PPSs including state-of-the-art I&C technologies [2]. KINAC is confident in the new test bed to become a fundamental technical basis of security related I&C systems in near future.

ISSNP #1: Nuclear Safety

Room #303

Session Chair: Fumiya Tanabe (Sociotechnical Systems Safety Research Institute), Hyun Gook Kang (KAIST)

An Analysis of Effect of Break-up Timing on the Necessity of a Feed-and-Bleed Operation in the Case of TLOFW with LOCA

Bo Gyung KIM (1), Ho Joon YOON (2), Sang Ho KIM (1) and Hyun Gook KANG (1)
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A Feed-and-bleed (F&B) operation is a process to cool the reactor by the primary side directly. If adequate residual heat removal through the secondary side is not available, the heat can be removed from the RCS by F&B operation [1]. A total loss of feedwater (TLOFW) accident is used to represent an accident involving the failure of cooling by the secondary cooling system. Even if the secondary cooling system fails, the RCS can be cooled by F&B transients when a loss of coolant accident (LOCA) with a TLOFW accident occurs. During an F&B transient, the RCS has a residual heat removal mechanism. If the break size is large, an F&B transient continuously occurs if the SIS is available [2]. If the break size is small to sufficiently decrease

the RCS pressure, the SIS cannot inject the coolant, causing the F&B transient to terminate. After the termination of the F&B transient, the residual heat cannot be removed, and the necessity of an F&B operation increases.

The operators may hesitate to initiate F&B operation if a clear cue is not provided, since its initiation implies the radioactive coolant releases into the containment. Therefore, the necessity of F&B operation is needed to be identified. The factors affected the necessity of F&B operation are the availability of the safety injection system and safety depressurization system, water inventory in the primary and secondary cooling systems, break size in a loss-of-coolant accident, and time of accident occurrence. The necessity of F&B operation can be changed according to different timing of break-up despite same break size. Moreover, different timing of break-up makes the operators more complicated. To identify effect of timing of break-up, a thermohydraulic analysis was performed using the MARS code. This study is expected to provide a useful guideline to identify the necessity of an F&B operation under combined accident.

Evaluation of Plant Operational States with the Consideration of Loop Structures under Accident Conditions

Takeshi MATSUOKA
Harbin Engineering University, China
Utsunomiya University, Japan

Nuclear power plants have logical loop structures in their system configuration. The paper explains the method to solve loop structure in reliability analysis. As examples of loop structured systems, RCIC and HPCI of BWR are taken up and analyzed under station blackout accident condition. The analysis result shows an important role of loop structure under sever accidents. For the evaluation of the safety of nuclear power plants, it is necessary to accurately evaluate loop structure's reliability.

First Application of SIMMER-III to Local Fuel Coolant Interactions in a Simulated Molten Fuel Pool

Song Bai CHENG, Ken-ichi MATSUBA, Mikio ISOZAKI, Kenji KAMIYAMA, Tohru SUZUKI and Yoshiharu TOBITA
Japan Atomic Energy Agency, Japan

Studies on local fuel-coolant interactions (FCI) in a molten pool are important for the analyses of severe accidents that could occur for sodium-cooled fast reactors (SFRs) [1]. To clarify the mechanisms underlying this interaction, a series of simulated experiments was conducted at the Japan Atomic Energy Agency by delivering a given quantity of water into a molten fuel pool formed with a low-melting-point alloy [1]. From the preliminary experimental analyses [1], it has been recognized that with the increasing of water volume, a limited pressure buildup can be observed for both the melt and cover gas regions at a given melt and water temperature conditions. In this study, to further understand this interaction, SIMMER-III, an advanced fast reactor safety analysis code [2], is utilized for analyses. It is found that the SIMMER-III code can comparatively well reproduce the transient pressure tendencies measured from experiments. In addition, from comparative analyses between different cases, the observed limited pressurization characteristics from experiments can be confirmed as well. To achieve a deeper and more systematic understanding on local FCIs, with the ongoing of experimental analyses, more numerical analyses over various conditions, such as difference in water subcooling, melt temperature as well as water release site, have been also planned.

ISSNP #2: Thermal Hydraulic

Room #303

Session Chair: Takeshi Matsuoka (Utsunomiya Univ.), Sung-Jin Lee (Doosan)

Study on Sensor fault Diagnosis in nu on PCA

Fei XU and Minjun PENG
Harbin Engineering University, China

To solve the fault diagnosis of the safety-related sensors in nuclear power plants, a monitoring model for several important sensors is established based on principal component analysis (PCA). Sensor fault detection, sensor identification and faulty sensor signals reconstruction can be achieved by calculating the squared prediction error, sensor validity index and reconstruction index. Employing the data from a real-time full-scope simulator of nuclear power plants, the monitoring model is proved effectively to detect the complete failure fault, fixed bias fault, drifting fault, and precision degradation fault. The simulation result illustrates the effectiveness of the model for the sensor fault diagnosis and recovery.

Heat Transfer Analysis of the Passive Residual Heat Removal Heat Exchanger

Wenwen ZHANG, Wenxi TIAN, Guanghui SU and Suizheng QIU
Xi'an Jiaotong University, China

In the present study, thermal-hydraulics characteristics of AP1000 passive residual heat removal heat exchanger (PRHR-HX) at initial operating stage were analyzed based on the porous media models. The data predicated by RELAP5 under the condition of the station blackout was employed as the inlet flow rate and temperature boundary of CFD calculation. The heat transfer from the primary side coolant to the in-containment refueling water storage tank (IRWST) side fluid was calculated in a three-dimensional geometry during iterations, and the distributed resistances were added into the C-type tube bundle regions. Three-dimensional distributions of velocity and temperature in the IRWST were calculated by the CFD code ANSYS FLUENT. The primary temperature, heat transfer coefficients of two sides and the heat transfer were obtained using the coupled heat transfer between the primary side and the IRWST side. The simulation results indicated that the water temperature rises gradually which leads to a thermal stratification phenomenon in the tank and the heat transfer capability decreases with an increase of water temperature. The present results indicated that the method containing coupled heat transfer from the primary side fluid to IRWST side fluid and porous media model is a suitable approach to study the transient thermal-hydraulics of PRHR/IRWST system.

Study on Natural Circulation Transition Process for IP200

Sun LIN, Minjun PENG and Genglei XIA
Harbin Engineering University, China

Natural circulation in a nuclear reactor helps to achieve passive heat removal capability which can enhance the inherent safety of the reactor. In this paper, ideal steady-state programming control strategy (to keep average temperature of coolant and steam pressure constant) is applied for the normal operation of Integrated PWR-200 (IP200). RELAP5/MOD4.0 is applied to analyze the transient characteristics and the key factors of the transition process from forced circulation to natural circulation. The factors mainly consist of the reactor power level, the resistance and rotational inertia of main coolant pumps (MCPs) and whether to isolate the MCPs by bypass loops.

ISSNP #3: Reactor Design

Room #303

Session Chair: Ming Yang (HEU), Gee Yong Park (KAERI)

Comparative Study of Porosity on a Coating Method of Silicon Carbide

DongHee LEE, Kwangheon PARK and Seonho NOH
Kyung Hee University, Korea

Because of the Fukushima accident, a reduction of violent reactions between steam and nuclear fuel claddings should be required when a severe accident has occurred. In particular, the development of accident-resistant nuclear fuel is required as a measure to prevent hydrogen explosions correlated to violent reactions. Silicon Carbide-SiC coating on claddings may be a good option; however, a SiC-composite protective layer on cladding seems better to protect against severe accidents and may be in steady states.

In this research, two types of SiC coating methods were tried: composites using a precursor under a low-temperature process and a formula of composites using supercritical CO_2. Zry-4 was used as a specimen, and three types of specimens were prepared. The first specimen was coated using only the PIP method, the second specimen was coated using the PIP method and supercritical CO_2 method and the third specimen was coated using the PIP method once and the supercritical CO_2 method twice. As a result, the PIP method had a bigger porosity than the other coating methods. The PIP method + supercritical CO_2 method had a smaller porosity than the PIP method. The PIP method + supercritical method(2×) had some porosity but not as much as the other methods.

Optimization of Pressurizer Based on Genetic-simplex Algorithm

Cheng WANG, Chang-qi YAN and Jian-junWANG
Harbin Engineering University, China

Pressurizer is one of the key components in nuclear power system. It's important to control the dimension in the design of pressurizer through optimization techniques. In this work, a mathematic model of a vertical electrical heating pressurizer is established. A new genetic-simplex algorithm (GSA) that combines genetic algorithm and simplex algorithm is developed to enhance the searching ability, and the comparison between modified algorithm and the original one is conducted by calculating the benchmark function. Furthermore, the optimization design of pressurizer, taking minimization of volume and net weight as objectives, is carried out considering thermal-hydraulic and geometric constraints through GSA. The sensitivities of some parameters, which may influence the volume and the net weight of pressurizer, are also analyzed.

The results show that the mathematical model is agreeable for the pressurizer. It is also shown that the new algorithm is more effective than the traditional genetic algorithm. The optimization design shows obvious validity and can provide guidance for real engineering design.

Development of Research Reactor Simulator and Its Application to Dynamic Test-bed

Kee-Choon KWON, Jae-Chang PARK, Seung-Wook LEE, Dane BAANG and Sung Won BAE
Korea Atomic Energy Research Institute, Korea

We developed HANARO and Jordan Research and Training Reactor (JRTR) real-time simulator for operating staff training. The main purpose of this simulator is operator training, but we modified this simulator as a dynamic test-bed to test the reactor regulating system in HANARO or JRTR before installation. The simulator configuration is divided into hardware and software. The simulator hardware consists of host computer, 6 operator stations, network switch, and large display panel. The simulator software is divided into three major parts : a mathematical modeling module, which executes the plant dynamic modeling program in real-time; an instructor station module that manages user instructions; and a human machine interface(HMI) module. The developed research reactors are installed in Korea Atomic Energy Research Institute nuclear training center for reactor operator training. To use simulator as a dynamic test-bed, the reactor regulating system modeling software of the simulator replaced by hardware controller and the simulator and target controller are interfaced with hard-wired and network-based interface.

Investigation on the MOC with A Linear Source APProximation Scheme in Three-dimensional Assembly

Chenglin ZHU and Xinrong CAO
Harbin Engineering University, China

Method of characteristics (MOC) for solving neutron transport equation has already become one of the fundamental methods for lattice calculation of nuclear design code system. At present, MOC has three schemes to deal with the neutron source of the transport equation: the flat source approximation of the step characteristics (SC) scheme, the diamond difference (DD) scheme and the linear source (LS) characteristics scheme. The MOC for SC scheme and DD scheme need large storage space and long computing time when they are used to calculate large-scale three-dimensional neutron transport problems. In this paper, a LS scheme and its correction for negative source distribution were developed and added to DRAGON code. This new scheme was compared with the SC scheme and DD scheme which had been applied in this code. As an open source code, DRAGON could solve three-dimensional assembly with MOC method. Detailed calculation is conducted on two-dimensional VVER-1000 assembly under three schemes of MOC. The numerical results indicate that coarse mesh could be used in the LS scheme with the same accuracy. And the LS scheme applied in DRAGON is effective and expected results are achieved. Then three-dimensional cell problem and VVER-1000 assembly are calculated with LS scheme and SC scheme. The results show that less memory and shorter computational time are employed in LS scheme compared with SC scheme. It is concluded that by using LS scheme, DRAGON is able to calculate large-scale three-dimensional problems with less storage space and shorter computing time.

ISSNP #4: Advances in Functional Modeling Method

Room #303

Session Chair: Leena Norros (VTT)

Development of a Functional Platform for System Reliability Monitoring of Nuclear Power Plants

Ming YANG, Zhijian ZHANG and Hidekazu YOSHIKAWA
Harbin Engineering University, China

This paper presents MFM builder, a platform based on Multilevel Flow Modeling (MFM), which provides a graphical interface for modeling functions of complex artificial systems such as nuclear power plant with emphasizing the designed purposes of systems. Several algorithms based on MFM have been developed for dynamic system reliability analysis, fault diagnosis and quantitative software reliability analysis. A Reliability Monitoring System (RMS) of PWR nuclear power plant was developed by integrating above algorithms. Experiments by connecting RMS with a full scale PWR simulator showed that it took 16 seconds for RMS calculating the reliability changes over time of safety-related systems according to given system configurations in the 31 days by one computer run. The proposed reliability monitoring system can be used not only offline as a reliability analysis tool to assist the plant maintenance staffs in maintenance plan making, but also online as a operator support system to assist the operators in Main Control Room (MCR) in their various tasks such as configuration management, fault diagnosis and operational decision making.

Applying Functional Modeling for Accident Management of Nuclear Power Plant

Morten LIND (1,2) and Xinxin ZHANG (1)
(1) Technical University of Denmark, Denmark
(2) Harbin Engineering University, China

The paper investigate applications of functional modeling for accident management in complex industrial plant with special reference to nuclear power production. Main applications for information sharing among decision makers and decision support are identified. An overview of Multilevel Flow Modeling is given and a detailed presentation of the foundational means-end concepts is presented and the conditions for proper use in modelling accidents are identified. It is shown that Multilevel Flow Modeling can be used for modelling and reasoning about design basis accidents. Its possible role for information sharing and decision support in accidents beyond design basis is also indicated. A modelling example demonstrating the application of Multilevel Flow Modelling and reasoning for a PWR LOCA is presented.

ISSNP #5: Fukushima Accident Analysis from Different Aspects

Room #303

Session Chair: Hidekazu Yoshikawa (HEU)

Post-facta Analyses of Fukushima Accident and Lessons Learned

Fumiya TANABE
Sociotechnical Systems Safety Research Institute, Japan

Independent analyses have been performed of the core melt behavior of the Unit 1, Unit 2 and Unit 3 reactors of Fukushima Daiichi Nuclear Power Station on 11-15 March 2011. The analyses are based on a phenomenological methodology with measured data investigation and a simple physical model calculation. Estimated are time variation of core water level, core material temperature and hydrogen generation rate. The analyses have revealed characteristics of accident process of each reactor. In the case of Unit 2 reactor, the calculated result suggests little hydrogen generation because of no steam generation in the core for zirconium-steam reaction during fuel damage process. It could be the reason of no hydrogen explosion in the Unit 2 reactor building. Analyses have been performed also on the core material behavior in another chaotic period of 19-31 March 2011, and it resulted in a re-melt hypothesis that core material in each reactor should have melted again due to shortage of cooling water. The hypothesis is consistent with many observed features of radioactive materials dispersion into the environment.

Estimation of Dynamic Behavior of Nuclear Power Plant System Sate under Severe Accident Conditions

Takeshi MATSUOKA
Harbin Engineering University, China
Utsunomiya University, Japan

Dynamic behavior of nuclear power plant system has been evaluated under accident conditions. Analyses have been performed by the GO-FLOW methodology, which supports the function of "reliability monitor", a main part of the risk monitor now being developed at Harbin Engineering University (HEU). A hypothetical sequence of accident conditions has been settled based on the Fukushima daiichi accident. Success probabilities of system operation have been obtained with the growth of accident. The present analyses have shown that dynamic behavior of nuclear power plant under accident conditions will be easily obtained by the "reliability monitor".

ISSNP #6: Will Written Procedures Solve Everything?

Room #303

Session Chair: Morten Lind (Tech Univ. of Denmark)

Integrated Functional Modeling Method for NPP plant DiD Risk Monitor and Its Application for Conventional PWR

Hidekazu YOSHIKAWA, Ming YANG and Zhijian ZHANG
Harbin Engineering University, Japan

The development of a new risk monitor system is introduced in this paper, which can be applied not only to severe accident prevention in daily operation but also to serve as to mitigate the radiological hazard just after severe accident happens and long term management of post-severe accident consequences. The summary of the fundamental method is summarized on how to configure the Plant Defense in-Depth (Did) Risk Monitor by object-oriented software system based on functional modeling approach. Following the authors' preceding preliminary study for AP1000, the way of realizing the proposed method of configuring the plant Did risk monitor was investigated for a safety-enhanced Japanese PWR design to meet with the tight anti-severe accident requirements set by national regulation in Japan after Fukushima Daiichi accident. The result of this example practice of the presented preliminary study for Japanese PWR was for the level 4 of the Did in case of beyond design basis accident, that is, loss of all AC power + RCP seal LOCA, against the former case of AP1000 for level 3 Did in case of large LOCA.

Can Proceduralization Support Coping with the Unexpected?

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Operations of safety critical industries unquestionably require a diversity of technical and organizational control measures to increase stability and predictability of the complex sociotechnical systems. Nevertheless, experiences from recent severe accidents and results of safety research have questioned the effectiveness of the prevailing safety management strategy that mainly relies on standardization and designed-in defenses. This paper discusses the identified need to balance between stability and flexibility in a concrete safety issue, i.e., proceduralization.

The main research problem of our study is whether procedure guided practice can offer sufficient support for flexibility of operating activity. We shall frame our study with the help of a model that explains different aspects of procedures. We then elaborate how these different aspects were considered empirically in our 3-phase study. In the first study we interviewed 62 main control room operators and asked how they consider procedures to support balancing. In the second study we observed in detail 12 NPP operator crews' activity in a simulated loss-of-coolant accident. In a third study we inquired 5 procedure designers about their conceptions concerning procedure guidance in operator work. Drawing on either interview or behavioral data we analyzed the personnel's stance to the flexibility and stability balancing, and how the conceptions portray in the practices of procedure usage.

Our results demonstrate that the operators are aware of the need for balancing flexibility and stability and consider

successful balancing to represent “good” professional action. In actual action many operators, however, tend towards more straightforward following of procedures. Designers also see the capability for balancing stability and flexibility as a key operator competence but describe actual acting simply as procedure-following. According to the documents of the nuclear community, procedure-following is the ideal to be emphasized. The paper will be finished by discussing what new insights our results would provide for developing training of procedure usage and for the design of procedures.

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Room #303

Session Chair: **Hwai-Pwu Chou(National Tsing Hua Univ., Taiwan), Miao Chi (HEU)**

The Research on Operation Strategy of Nuclear Power Plant with Multi-reactors

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In this paper, the operation characteristics and control strategy of nuclear power plant (NPP) with multi-modular pressurized water reactors (PWR) were researched through simulation. The main objective of this research was to ensure the coordinated operation and satisfy the convenience of turbine-generator and reactor's load adjustment in NPP with multi-reactors (MR). According to the operation characteristics of MR-NPP, the operation and control strategy was proposed, which was "the average allocation of load for each reactor" and "maintaining average temperature of coolant at a constant". The control system was designed based the operation and control strategy. In order to research the operation characteristics and control strategy of MR-NPP, the paper established the transient analysis model which included the reactors and thermal hydraulic models, turbine model, could simulate and analyze on different operating conditions such as load reducing, load rising. Based on the proposed operation and control strategy and simulation models, the paper verified and validated the operation strategy and control system through load reducing, load r ising. The results of research simulation showed that the operation strategy was feasible and can make the MR-NPP running safely as well as steadily on different operating conditions.

Comparative Health Risk Assessment of CdTe Solar PV System and Nuclear Power Plant

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In terms of national energy policy decision-making process, several key factors, including low production cost, negligible risk or impact to environment and population around the facility, must be considered. The purpose of this paper is to assess the public health risk in case of postulated nuclear power plant and CdTe solar PV system accident and compare the estimated public health risk. Both systems release toxic materials to the environment which adversely affect nearby population by exposure from the inhalation and ingestion of the toxic material transported via air. By simulating the airborne transport of released toxic material using Gaussian plume model and modeling exposure pathways to nearby population, average individual health risk is assessed and public health risk per power capacity of each system is compared. The result shows that the average public health risk per power capacity of NPP is less than the case of solar PV system. This implies that NPP has lower risk in terms of public health risk in case of severe accident while it can be used as more reliable energy source than renewable energy source so that NPP would take priority over other renewable energy sources in terms of national energy policy.

Development of Nuclear Spent fuel Maritime Transportation Scenario

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Spent fuel transportation of South Korea is to be conducted through near sea because it is able to ship a large amount of the spent fuel far from the public comparing to overland transportation. The maritime transportation is expected to be increased and its risk has to be assessed. For the risk assessment, this study utilizes the probabilistic safety assessment (PSA) method and the notions of the combined event. Risk assessment of maritime transportation of spent fuel is not well developed in comparison with overland transportation. For the assessment, first, the transportation scenario should be developed and categorized. Categories are assorted into the locations, release aspects and exposure aspects. This study deals with accident that happens on voyage and concentrated on ship-ship collision. The collision accident scenario is generated with event tree analysis. The scenario will be exploited for the maritime transportation risk model which includes consequence and accident probability.

A Plant Damage State Early Warning System

Chih-Yao HSIEH and Hwai-Pwu CHOU
National Tsing Hua University, Hsinchu, Taiwan

In case of a severe accident, operators need to follow the emergency operating procedures (EOPS) to limit the damage. In order to assist operators to face a lot of Plant Damage States (PDS) suddenly, we try to predict and identify the Plant Damage State (PDS) for early warning and decision making. In this study, Containment Event Tree (CET) is used in this event-oriented approach to help severe accident management. The Taipower Lungmen nuclear power station (LNPS), an advanced boiling water reactor, is chosen for case study. The LNPS full scope engineering simulator is used to generate the testing data for method development.

Research on Integration of NPP Operational Safety Management Performance Systems

Miao CHI and Liping SHI
Harbin Engineering University, China

The operational safety management of Nuclear Power Plants demands systematic planning and integrated control. NPPs are following the well-developed safety indicator systems proposed by IAEA Operational Safety Performance Indicator Programme, NRC Reactor Oversight Process or the other institutions. Integration of the systems is proposed to benefiting from the advantages of both systems and avoiding improper application into the real world. The authors analyzed the possibility and necessity for system integration, and propose an indicator system integrating method.

Research on Operation and Control Strategy of 600MW PWR in Load Follow

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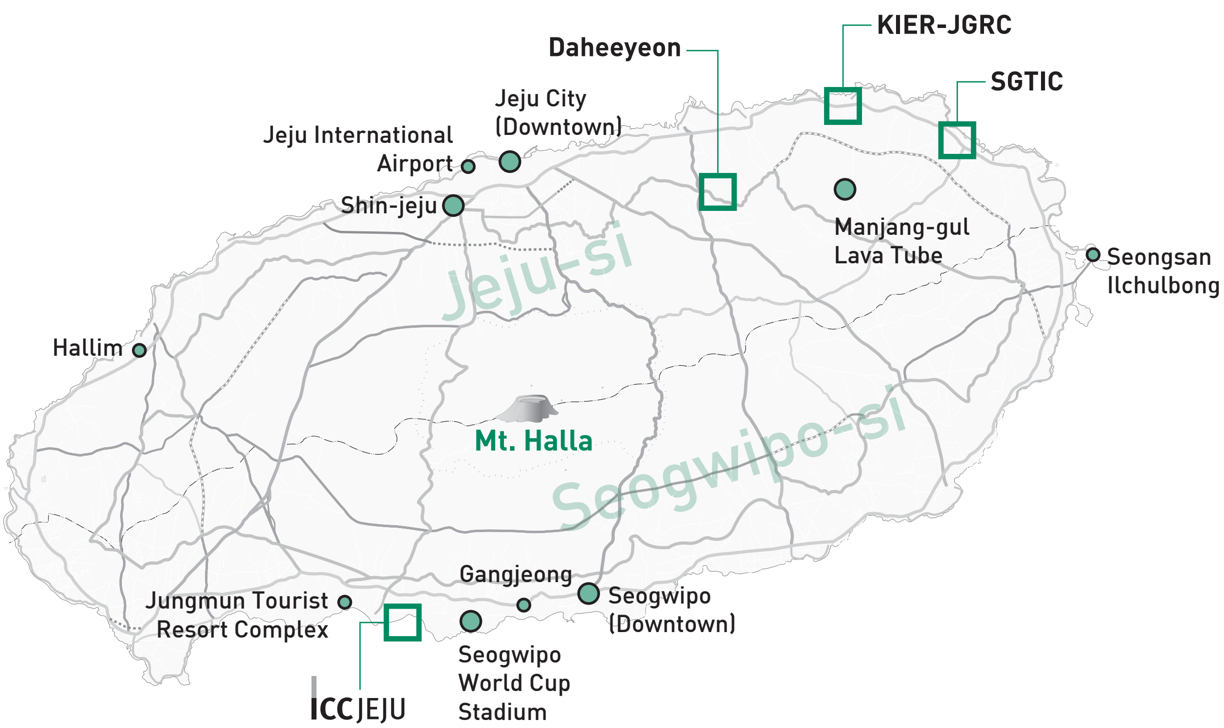
600MW Pressurized Water Reactor (PWR) is designed to operate in Constant Axial Offset Control (CAOC) strategy with base load originally. By calculations over a typical load follow scenario "12-3-6-3"(100-50-100%FP) via the CASMO-4E&SIMULATE-3 package, values of core operating parameter have been examined. With the progress of the nuclear power industry, advanced reactors are considered to have a good performance in load follow, economy and flexibility. Under the premise of fuel loading and structural dimensions unchanged, two independent control rod groups M and AO are used in 600MW pressurized water reactor to provide fine control of both the core reactivity and axial power distribution, which is named " Improved G strategy ." The influences of different control rod distributions, composition materials, and overlap steps had in power changes have been examined in a comparative study to choose the optimal one.Then we simulate a range of load follow scenarios of the redesigned 600MW core without adjusting soluble boron concentration in the begin, middle and end of first cycle. This paper additionally demonstrated the moderator temperature coefficient and shutdown margin values of the reactor in Improved G strategy to compare with the thermal safety design criteria. It ' s demonstrated that adequate djustment of control rod groups enable the core to perform load follow through Improved G strategy in 80% of cycle and save a large volume of liquid effluent particularly toward the end of cycle.

Technical Tour

Time	Place	Technical Tour	Remarks
09:00	ICC Jeju	Together & Departure	
09:00~10:30	Transfer	ICC Jeju to Korea Institute of Energy Research-Jeju Global Research Center	
10:30~11:20	KIER JGRC	Tour Outcome Diffusion Building	http://jeju.kier.re.kr
		Salinity Gradient Generation	
		Sea Wind Power	
11:20	KIER JGRC	Take a Picture	
11:20~11:40	Transfer	KIER JGRC to Smart Grid Test-bed Information Center	
11:40~12:40	SGTIC	Tour SG Center	http://www.kepco.co.kr/sg
12:40~13:00	Transfer	SG Center to Daheeyeon	Restaurant
13:00~15:00	Daheeyeon	Lunch	
		Tea Museum & Gotjawal Cave	
15:00~16:30	Transfer	Daheeyeon to Airport & ICC Jeju	

Jeju



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