November 26~30, 2017
Hwabaek International Convention Center (HICO)
Gyeongju, Korea

ISOFIC 2017

International Symposium on Future I&C for Nuclear Power Plants

www.isofic.org

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Contents

Foreword

Man Gyun Na

General Chair

Chosun University Korea

Welcome to ISOFIC 2017 in Gyeongju, Capital City of Silla Kingdom

As the general chair of this symposium, I'm honorably pleased to have you in Gyeongju for ISOFIC 2017 (International Symposium on Future I&C for Nuclear Power Plants). The first ISOFIC symposium was held in Seoul, 2002, going back in time. This symposium is the sixth since then. All ISOFIC symposiums except one time in China were held in Korea. The last symposium was held in Jeju Island, Korea. In that symposium, about 130 papers were presented. About 100 papers will be presented in this symposium. The number of papers is a little reduced due to the unstable Korean peninsula security situation by the nuclear bomb and ICBM tests in North Korea.

Nuclear power supplies about 15% of the total power demand globally. As of August 2017, a total of 447 nuclear power plants are in operation in 30 countries, and 58 reactors are under construction in 15 countries. In Korea, KORI Unit 1, the first nuclear power plant started to generate the electricity in 1978. 40 years have passed since then and now, Korea is operating 24 nuclear power plants and constructing 5 new nuclear power plants domestically and 4 nuclear power plants in UAE (United Arab Emirates).

New Korean government declared anti-nuclear power plant policy this year. The first operated KORI Unit 1 nuclear plant in Korea was shut down permanently this year. The permanent shutdown of KORI unit 1 NPP was determined in the former government, of course. The construction of Shin-Kori Units 5 and 6 nuclear power plants to be being constructed in Korea had been suspended for three months. But the construction was resumed after an ad hoc committee (so called, public discussion panel) recommended the construction of Shin-Kori Units 5 and 6 nuclear power plants. The pro- and anti-groups of nuclear power plants is debating violently in Korea. I strongly desire that nuclear power will be sustained and increased against global climate change.

Korea had developed APR1400(Advanced Power Reactor 1400) in 2002 and the advanced type of NPP, APR+ in 2012. Also, Korea developed MMIS(Man-Machine Interface System) in 2008 and applied it into Shin-Hanul Unit 1 & 2 Nuclear Power Plants that is to be operated soon. The research in the nuclear I&C become more and more active recently in many countries. Recently, the nuclear I&C research using the core technologies of 4-th generation industry revolution is attracting interests. Especially, the research on autonomous operation, artificial intelligence application, cyber security, FPGA based controller, defense-in-depth and multi-channel communication, and human factors engineering are actively being conducted. ISOFIC 2017 is handling these topics.

The symposium starts on November 26 and has a number of regular sessions until November 29. We have three fantastic plenary talks from Prof. J. Wesley Hines, The University of Tennessee, Knoxville; Prof. Takeshi Matsuoka, Utsunomiya University, Japan and Mr. Ian Jung, US NRC. Also, we have three keynote speeches from Dr. Kook-Hun Kim, Doosan Heavy Industry & Construction; Prof. Akio Gofuku, Okayama University, Japan; Dr. Chang-Hwoi Kim, KAERI. On November 27 we will have a banquet at the first floor lobby of HICO. Finally, on November 30 there will be a technical tour at low and intermediate level radioactive waste (LILW) disposal facility in Gyeongju.

I hope you will have a great pleasure and comfortable time in Gyeongju city. Gyeongju was the capital of the ancient kingdom of Silla (57 BC – 935 AD). A number of archaeological sites and cultural properties remain in this city. Among such historical treasures, Seokguram grotto, Bulguksa temple, Gyeongju Historic Areas and Yangdong Folk Village are designated as World Heritage Sites by UNESCO. The major historical sites have helped Gyeongju become one of the most popular tourist destinations in Korea. I recommend you to tour major historical sites during your stay.

Lastly, I deeply appreciate the financial support from Doosan Heavy Industry & Construction Company, KAERI, and KEPCO-ENC.

November 27, 2017

General Chair of ISOFIC 2017

Welcoming Remarks

Hark Rho Kim

Honorary Chair

President
Korean Nuclear Society

Principal Researcher
Korea Atomic Energy
Research Institute

Dr. Kim received BS and MS in Seoul National University, and Ph.D. in Nuclear Engineering, KAIST(Korea Advanced Institute of Science and Technology) in 1994. Dr. Kim has been worked for KAERI(Korea Atomic Energy Research Institute) more than thirty years in developing research reactors, small modular reactors, and advanced reactors. During his career, he drove the team for SMART development project and led the KAERI-Daewoo Consortium for JRTR(Jordan Research and Training Reactor) project. Until recently, he has been a senior vice president at KAERI.

Dr. Kim is now taking a role of the president of Korean Nuclear Society and making all his efforts into the peaceful and sustainable nuclear energy. He is also the member of Korean representative in the GIF(Generation IV International Forum) policy group and the vice chairman on external collaboration.

Congratulatory Remarks

Soon Heung Chang

President
Handong Global University

Chairman
Advisory Committee
on Nuclear Safety and
Security, Nuclear Safety
and Security Commission

Dr. Chang received Ph.D. in Nuclear Engineering, Massachusetts Institute of Technology in 1981 and was a professor and a provost of KAIST(Korea Advanced Institute of Science and Technology). Dr. Chang has been devoted more than thirty years of his life to nuclear engineering education in KAIST. In terms of his profession, Dr. Chang served as a member of International Nuclear Safety Advisory Group of IAEA and a commissioner of Korea Nuclear Safety Commission as well. Also, he was one of the five members of the International Advisory Experts of the Investigation Committee for the Fukushima Accident. In 2014, he was inaugurated as President of Handong Global University located in Pohang, South Korea. Dr. Chang has put his educational emphasis on creative project based on whole-person education and it has impacted on the field of higher education in the world.

History

Following the success of the first conference in 2002 in Seoul, Korea, the international conference, ISOFIC (International Symposium on Future I&C for Nuclear Power Plants), has been held every three years. We are proud to announce that ISOFIC 2017 will be held November 26-30, 2017 at Hwabaek International Convention Center (HICO) in Gyeongju, Korea. The ISOFIC 2017 promotes academic and practical information exchanges mainly on the topics of innovative Instrumentation and Control (I&C), and human system interface technologies.

While nuclear I&C has recently made significant advances through technological innovation, many challenges remain such as communication between safety and non-safety systems, software verification and validation (V&V), the choice between spatially fixed or screen-switchable indicators, availability of information during severe accidents, I&C architecture against common mode failures, and cyber threats and security. There are concerns that these challenges may cause delays in the construction process of nuclear power plants.

However, we believe that challenges also bring opportunities. ISOFIC 2017 will contribute to worldwide nuclear I&C society by further connecting designers, manufacturers, operators, regulators, and researchers and sharing solutions towards advancing nuclear I&C technologies and early settlement of digital I&C technology within the nuclear.

Research Themes

The subject areas of ISOFIC 2017 are as follows:

- Methods of Sensing, Processing, and Communication
- Surveillance, Diagnostics, and Prognostics
- Robotics & Automatic Remote Technologies
- Wireless Technologies in Nuclear Applications
- Modernization of I&C and Control Room
- Future I&C technologies for Nuclear Applications
- Cyber Security
- Safety Critical Software Development and Qualification
- Cognitive Systems Engineering for Process Control
- Human Factors/Human Reliability Assessment
- System Simulation Technologies
- System Reliability and Risk
- MFM and Safety Culture
- Testing and Maintenance

Organization

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Conference Venue & Meeting Rooms

Hwabaek International Convention Center (HICO)

- Location: 507, Bomun-ro, Gyeongju-si, Gyeongsangbuk-do, Republic of Korea
- Website: http://www.crowncity.kr/hico/en/main/main.do

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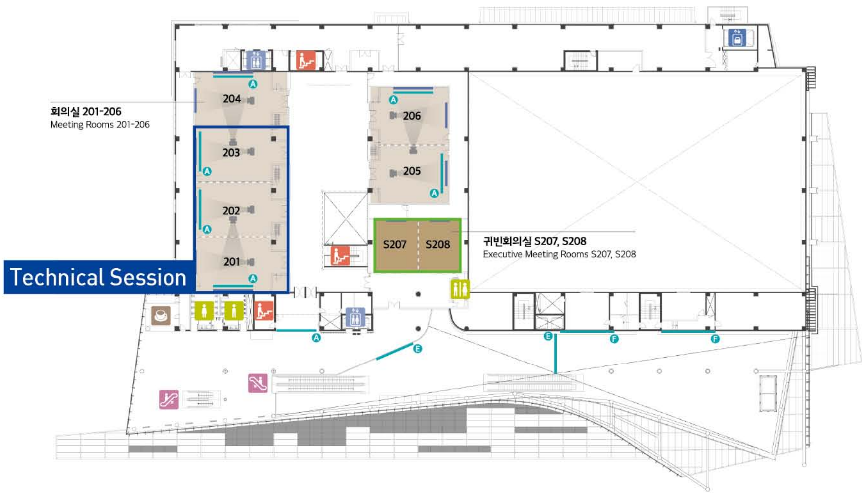

- 계단 stairs
- 승강기 elevator
- 화물용 엘리베이터 freight elevator
- 에스컬레이터 escalator
- 화장실 restroom
- 카페테리아 cafeteria
- 스크린 screen
- 현수막 바텐 banner batten
- 빔프로젝터 beam projector

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Plenary Talk 1

"An Integrated Information Architecture for Equipment Lifecycle Prognostics and Reliability Improvement"

November 27, 2017, Monday, 09:00~09:30

Prof. J. Wesley Hines

Head of Nuclear Engineering Department, University of Tennessee

American Nuclear Society Fellow

Nuclear system on-line-monitoring is becoming a crucial component of improving safety, reliability, and profitability. The Holy Grail is the development prognostic methodologies to accurately predict the Remaining Useful Life (RUL) of a system or component for predictive maintenance and effective risk mitigation. Calculating precise RUL estimates requires both system specific maintenance information and performance data to develop representative lifecycle models. Current conventional prognostic methods focus on process data and do not utilize maintenance data to directly influence the modeling and data analysis.

Conventional Lifecycle Prognostics is a term used when the RUL is seamlessly predicted from beginning of component life (BOL) to end of component life (EOL). When a component is put into use, the only information available may be past failure times, and the predicted failure distribution can be estimated with reliability methods such as Weibull Analysis (Type I). As the component operates, it begins to consume its available life. This life consumption may be a function of system stresses, and the failure distribution should be updated (Type II). When degradation becomes apparent, this information can be used to again improve the failure distribution estimate (Type III) of the specific component.

However, as parts begin to degrade and components fail, maintenance personnel are responsible for making repairs and replacements and recording these actions in a computerized maintenance management system (CMMS). These maintenance practices may bring the component or system to an "as good as new", "as good as used", or some other type of renewal process. The specific equipment maintenance actions impact future system degradation and this effect should be exploited to improve prognostic capabilities.

Early results of the integration of past maintenance information with conventional lifecycle prognostics indicates that maintenance specific models produce significantly lower prediction error and model uncertainty. This serves as a proof of concept for investigation into more effective ways to utilize maintenance data in prognostic modeling, while also emphasizing the importance of digital maintenance records in industry. The information systems of the future should integrate reliability data, performance data, and maintenance data to better predict remaining useful life of specific components and systems during operation.

Plenary Talk 2

"Fukushima Nuclear Power Plant Accidents in the Viewpoint of PSA"

November 27, 2017, Monday, 09:30~10:00

Prof. Takeshi Matsuoka

Visiting Professor, College of Nuclear Science and Technology, Harbin Engineering University

Lecturer, Center for Fundamental Education, Utsunomiya University

Member, Science Council of Japan

On 11th March, 2011, most sever nuclear power plant accidents in the history have been occurred at the Fukushima site due to the massive earthquake and subsequent large Tsunami. As the results of the loss of all AC power or station blackout, the reactor cores have melted down in Unit 1 through Unit 3. The authors, the Science Council of Japan, have continued the discussions what were happened at the Fukushima accident, and investigated new evidences, and also made hearing to TEPCO's technical members. Based on these activities, the core damage probabilities for Units 1 through 3 have been evaluated by event tree analyses under the condition of Tsunami attack. The core damage probability at 168 hours (7 days) after the Tsunami are obtained as 0.71(Unit 1) and 0.12 (Units 2 and 3). The discussions are made for the discrepancy between the analysis results and the actual situation, that is, all the three reactor cores have melted down. We have also pointed out some factors for the background causes of the Fukushima Nuclear Power Plant Accidents.

Plenary Talk 3

"A Global Perspective on the Future of Instrumentation and Control"

November 27, 2017, Monday, 10:00~10:30

Mr. Ian Jung

Chief of the Instrumentation, Controls, and Electrical Engineering Branch

Division of engineering, Office of Nuclear Regulatory Research

U.S. Nuclear Regulatory Commission

Digital technology for instrumentation and control applications has evolved over time and contributed positively to the operation and safety of nuclear power plants. The benefits of the digital technology are tremendous including accuracy, absence of drift, capability to store data, diagnostics and testing capabilities, improved human-machine interfaces, and others. However, there have been challenges along with the benefits of the digital technology from the viewpoints of safety and security as well as licensing and implementation. This presentation will provide a historical perspective, the ongoing challenges, and future opportunities and considerations for the safe, secure and efficient deployment of evolving digital/other technologies for the instrumentation and control systems of nuclear power plants. It will discuss topics such as licensing, safety, security, risk-informed/performance-based approach, technology neutrality, and harmonization.

Keynote Speech 1

"Nuclear I&C: Issues and Way to Go"

November 28, 2017, Tuesday, 09:00~09:40

Dr. Kook Hun Kim

Vice President

Head of Nuclear I&C BU

Doosan Heavy Indstries & Construction

Member, National Academy of Engineering of Korea

The first ISOFIC was launched in 2002 to let know Korea's Nuclear I&C R&D program, KNICS. At that time, Korea was not able to supply I&C system for nuclear plant and so almost every PWR's I&C system had been supplied by Westinghouse.

And many I&C experts didn't know that Korea has launched KNICS project. So to organize the first ISOFIC, many of the participants were invited with some expense support. However, ISOFIC settled down very quickly and firmly. As you see participants are from all over the world and quality of papers are excellent. Also ISOFIC is providing a ground for academic and technology discussions, exchange of knowledge and not often but sometimes business related topics. I believe ISOFIC's role in developing future I&C technology and finally contribute to improving NPP's safety.

Along with the growth of ISOFIC, Korea I&C team lead by Doosan has supplied for total I&C system for SHN 1&2 and preparing for SKN 5&6, both are APR 1400. Also supplied 14 units of control rod control system and preparing for new 8 units.

Since 1990's, digital I&C has provided a very efficient tool for NPP's improved safe and user-friendliness. However digital I&C has made s new issue, for example diversity, V&V, human factor engineering and even to the cognition level. And recently wireless, big data and ICT, sometimes related with digital innovation, new areas for digital I&C is waiting for us.

To prepare and tackle to the new issues for NPP's safety and nuclear energy economics is our role.

Keynote Speech 2

"Co-operator as an Intelligent Operator Support System for Resilient Operation of NPPs"

November 28, 2017, Tuesday, 13:30~14:10

Prof. Akio Gofuku

Division of Medical Bioengineering, Graduate School of Natural, Science and Technology, Okayama University

Lessons learned from the Fukushima Daiichi accident revealed various weak points in the design and operation of nuclear power plants at the time. The lessons include the improvement of education and training of nuclear personnel and non-technical trainings such as crew resource management that is applied in all airlines are being introduced in Japanese utilities under the concept of resilience engineering. Although a human can exerts great performance by well-organized training and education, the performance usually degrades in an emergency situation. Considering the advancement of computer technology and artificial intelligence, it is a promising way to develop intelligent systems to support nuclear personnel.

To support the activities of human operators in main control rooms of engineering plants, a concept of co-operator is proposed. A co-operator is an intelligent and integrated system and supports human operators like a co-pilot in aircraft control. It monitors plant condition through plant instrumentation systems and supports the situation awareness and counter action planning of human operators by mutually interacting with them under the condition that they make a final decision.

In the speech, the concept and necessary sub-systems of the co-operator are introduced. Then, a counter action planning technique in an emergency pant situation is presented as an important sub-system of the co-operator. The technique utilizes a qualitative reasoning based on a functional model constructed in the framework of Multilevel Flow Modelling (MFM) and quantitative information can be generated by a combination of numerical simulation package. In addition, an interactive display technique of the information generated by the co-operator is presented.

Keynote Speech 3

"Current Status of an Information and Communication Technologies for Nuclear Power Plants"

November 29, 2017, Wednesday, 09:00~09:40

Dr. Chang Hwoi Kim

Director

Nuclear ICT Research Division,
Korea Atomic Energy Research Institute

The 4th Industrial Revolution brings diverse changes in overall industries worldwide. In accordance with the recent changes in the nuclear industry policies in the Republic of Korea, researches using ICT technologies are planned for nuclear safety enhancement. In detail, projects for the development of technologies including artificial intelligence-based autonomous operation and human error removal technology, ICT-based decommissioning process automation technology, big data-based failure monitoring and operation optimization technology, and robot technology for quick responses to nuclear accidents, are planned as government lead projects. This presentation introduces the results of past major I&C researches, the status of present researches, and the status of ICT-based ongoing and planned researches. It is expected to produce greater outcomes for nuclear safety through the joint efforts of the organizations performing ICT-related researches.

Program at a Glance

	Room #201	Room #202	Room #203
	November 26, 2017 (Sunday)		
17:00~19:30	Registration		
18:00~19:30	Welcoming Reception : Hilton Hotel Pine Room (1st Floor)		
	November 27, 2017 (Monday)		
08:00~18:00	Registration		
08:30-09:00	Opening Ceremony (HICO 1st Grand Hall) Opening Remarks : Man Gyun Na (General Chair) Welcoming Address : Hark Rho KIM (President, Korean Nuclear Society) (Honorary Chair) Congratulatory Address : Soon Heung CHANG (President, Handong Global University)		
09:00~09:30	Plenary Talk 1 : Prof. J. Wesley Hines (University of Tennessee, Knoxville) Advanced Prognostics Using Both Process and Maintenance Data		
09:30~10:00	Plenary Talk 2 : Prof. Takeshi Matsuoka (Utsunomiya University) Fukushima Nuclear Power Plant Accidents in the Viewpoint of PSA		
10:00-10:30	Plenary Talk 3 : Mr. Ian Jung (US Nuclear Regulatory Commission) Introduction of Embedded Digital Devices – What Are They Telling Us?		
	Photo time		
10:30~10:40	Coffee Break		
10:40~12:20	Methods of Sensing, Processing, and Communication	Modernization of I&C and Control Room	Cognitive Systems Engineering for Process Control
12:20~14:00	Lunch Break		
14:00~15:40	Surveillance, Diagnostics, and Prognostics Ⅰ	Future I&C Technologies for Nuclear Applications Ⅰ	Human Factors/Human Reliability Assessment
15:40~16:00	Coffee Break		
16:00~17:40	Surveillance, Diagnostics, and Prognostics Ⅱ	Future I&C Technologies for Nuclear Applications Ⅱ	System Simulation Technologies
18:00~20:00	Banquet (HICO 101~103)		
	November 28, 2017 (Tuesday)		
08:00~18:00	Registration		
09:00~09:40	Keynote Speech : Dr. Kook-Hun Kim (Doosan Heavy Industries & Construction) Nuclear I&C: Issues and Way to Go		
09:40~10:00	Coffee Break		
10:00~12:00	Robotics & Automatic Remote Technologies Ⅰ	Special Session:Cyber Security Ⅰ	System Reliability and Risk
12:00~13:30	Lunch Break		
13:30~14:10	Keynote Speech : Prof. Akio Gofuku (Okayama University) Co-operator as an Intelligent Operator Support System for Resilient Operation of NPPs		
14:10~14:20	Coffee Break		
14:20~16:00	Robotics & Automatic Remote Technologies Ⅱ	Special Session:Cyber Security Ⅱ	Special Session:MFM & Safety Culture Ⅰ
16:00~16:20	Coffee Break		
16:20~18:20	Robotics & Automatic Remote Technologies Ⅲ	Special Session:Cyber Security Ⅲ	Special Session:MFM & Safety Culture Ⅱ
	November 29, 2017 (Wednesday)		
08:00~18:00	Registration		
09:00~09:40	Keynote Speech : Dr. Chang Hwoi Kim (Korea Atomic Energy Research Institute) Current Status of an Information and Communication Technologies for Nuclear Power Plants		
09:40~10:00	Coffee Break		
10:00~12:00	Wireless Technologies in Nuclear Applications	Safety Critical Software Development and Qualification	Testing and Maintenance
12:00~14:00	Lunch & Best Paper Award (HICO 1st Grand Hall)		
	November 30, 2017 (Thursday)		
08:30~13:00	Technical Tour (Korea Radioactive Waste Agency (KORAD))		

Schedule

Monday, November 27, 2017 10:40-12:20

Methods of Sensing, Processing, and Communication

Chair: Wes HINES (University of Tennessee), Inkoo HWANG (KAERI)

10:40 AM	**Accuracy Review on Long Wired RTD Instrumentation Circuits,** Inkoo HWANG, Jungtack KIM, Sub HUR (KAERI)
11:00 AM	**Implementation of Post-Accident Monitoring Systems (PAMS) in Ukrainian NPP's,** Taras TKACH (NNEGC "Energoatom")
11:20 AM	**Flow Rate Measurement Deviation Analysis considering Process Fluid Density Inconsistency for Differential Pressure Transmitter,** Eunsuk OH, Byung Rae KIM, Seog Hwan JEONG (KEPCO E&C)
11:40 AM	**A Modified Parity Space Averaging Technique for Online Calibration of Redundant Sensors in Nuclear Reactors,** Moath KASSIM, Gyunyoung HEO (Kyung Hee University)
12:00 PM	**Experience with Long Term Performance of Safety Related Pressure and Differential Pressure Transmitters in Ukrainian Nuclear Power Plants,** Sergiy LEBEDYNSKYY, Volodymyr LEBEDYNSKIY (Private JSC "Manometr-Kharkiv")

Modernization of I&C and Control Room

Chair: Jamie COBLE (University of Tennessee), TBD

10:40 AM	**An Overview of the Progress in Research and Development in Advanced Instrumentation, Information, and Control Systems Technologies to Support Light Water Reactor Sustainability,** Bruce P. HALLBERT, Kenneth D. THOMAS (INL)
11:00 AM	**Modernization of NPP's Safety I&C - Challenges and Solutions,** Bernd RUDOLF (AREVA GmbH)
11:20 AM	**Modernization of Nuclear Power Plant's Instrumentation and Control Systems on the Basis of FPGA Technology,** Ievgenii BAKHMACH, Olexandr SIORA (Research and Production Corporation Radiy), Anton ANDRASHOV (Radics LLC), Vyacheslav KHARCHENKO, Andriy KOVALENKO (Centre for Safety Infrastructure-Oriented Research and Analysis)
11:40 AM	**Integrated Human and Organizational Factors-oriented Design and Development Method,** Nicolas HENRY, Renaud AUBIN (EDF R&D)
12:00 PM	**Complexity Analysis of an FPGA-Based ESF-CCS,** Joyce MAYAKA, Jae Cheon JUNG (KINGS)

Cognitive Systems Engineering for Process Control

Chair: Jonghyun KIM (Chosun University), TBD

10:40 AM	**Computerized Procedure Interface for Nuclear Power Plant,** Yeonsub JUNG (KHNP)
11:00 AM	**Application of Ecological Interface Design in NPP Operator Support System,** Alexey ANOKHIN (JSC "Rusatom Automated Control Systems"), Alexey IVKIN (Obninsk Institute for Nuclear Power Engineering), Sergey DOROKHOVICH (Simulation Systems Ltd.)
11:20 AM	**Application of Petri Nets for Formalization of NPP I&C Functional Design,** Alexey CHERNYAEV, Elena ALONTSEVA, Alexey ANOKHIN (JSC "Rusatom Automated Control Systems")
11:40 AM	**Suggestion of a RNN-based Plant Diagnosis System for Extreme Situations in Nuclear Power Plants,** Seongkeun KANG, Poong Hyun SEONG (KAIST)

Monday, November 27, 2017 14:00-15:40

Surveillance, Diagnostics, and Prognostics I

Chair: Zhe DONG (Tsinghua University), Man Gyun NA (Chosun University)

2:00 PM	Sensor Selection Based On Boolean Network, Zhe DONG, Yifei PAN, Xiaojin HUANG (Tsinghua University)
2:20 PM	Monitoring the Status of Safety Functions using LSTM, Jaemin YANG, Jonghyun KIM (Chosun University)
2:40 PM	AREVA's Spent Fuel Pool Level Measurement Solutions, Nicolas THILLOSEN, Sergio ESTEVEZ HERNANDEZ, Ryan REYNOLDS (AREVA GmbH)
3:00 PM	Prediction of LOCA Break Position and Size Using MSVM, Ju Hyun BACK , Kwae Hwan YOO, Young Do KOO, Man Gyun NA (Chosun University)
3:20 PM	New PRODIAG Algorithm and Acceptance Test, Young Soo PARK, Richard VILIM (ANL)

Future I&C technologies for Nuclear Applications I

Chair: Bruce P. HALLBERT (INL), Hyun Gook KANG (RPI)

2:00 PM	IAEA Activities in the field of Instrumentation and Control Engineering, Janos EILER (IAEA)
2:20 PM	Complementing Renewable Energy Production with Small Modular Reactors, Richard BISSON, Jamie COBLE, Kevin TOMSOVIC (University of Tennessee)
2:40 PM	Classification of Abnormal Conditions: A Data-driven Aid for the Selection of Abnormal Operating Procedures in NPPs, Ibrahim AHMED, Sanghwa LEE, Gyunyoung HEO (Kyung Hee University)
3:00 PM	Application of Multi-objective Particle Swarm Optimization in Condenser Control System Parameters Tuning, Yiliang LI, Zhi CHEN (Nuclear Power Institute of China)
3:20 PM	Fiber Optic Cable for NPP Harsh Environment, Zuzana KONE ČNÁ (Czech Technical University in Prague), Vít PLAČEK, Petr HAVRÁNEK (ÚJV Řež,)

Human Factors / Human Reliability Assessment

Chair: Nicolas HENRY (EDF R&D), Yeonsub JUNG (KHNP)

2:00 PM	Autonomous Algorithm for Safety Function State of Nuclear Power Plant by Using LSTM, Daeil LEE, Jonghyun KIM (Chosun University)
2:20 PM	Modeling the Resilience of Severe Accident Management Organizations Using AHP, Jooyoung PARK (Chosun University), Ji-tae KIM (KINS), Jonghyun KIM (Chosun University)
2:40 PM	Sunburst Hierarchical Visualization Technique-based Navigation Support Interface for Information Processing System (IPS) in Nuclear Power Plants, Seung Min LEE, Gwi Sook JANG, Gee Yong PARK (KAERI)
3:00 PM	The Development of Regulation Guideline Manual Regarding Beyond Design Basis Accident and Severe Accident, Ji-Yoon HAN, Dong-Jin KIM, Ji-Tae KIM, Yun-Hyung CHUNG (KINS)
3:20 PM	Safety Assessment Framework for the Nuclear Decommissioning, HyungJun KIM, Seung Jun LEE (UNIST)

Monday, November 27, 2017 16:00-17:40

Surveillance, Diagnostics, and Prognostics Ⅱ

Chair: Jae Cheon JUNG (KINGS), TBD

4:00 PM	Development of a Smart Support System for Diagnosing Severe Accidents in Nuclear Power Plants, Kwae Hwan YOO, Ju Hyun BACK, Man Gyun NA (Chosun University), Seop HUR, Hyeon Min KIM (KAERI)
4:20 PM	Estimation of Cutter Wear of a Milling Machine Using a Support Vector Regression Method, Young Do KOO, Man Gyun NA (Chosun University), Jung-Taek KIM (KAERI)
4:40 PM	Towards Extracting 3-D Structural Representations of AGR Core Fuel Channels from 2-D In-Core Inspection Videos, Kristofer LAW, Graeme WEST, Paul MURRAY (University of Strathclyde), Chris LYNCH (EDF Energy Generation)
5:00 PM	Development of an Improved Data-Driven Diagnostic Platform for Process Plants: Case Study of Feedwater Heater Leakage, Gayeon HA, Ibrahim AHMED, Gyunyoung HEO (Kyung Hee University)
5:20 PM	An Ingenious Pressure Surveillance Algorithm to Detect CO2 Ingress Accidents in a Sodium-cooled Fast Reactor, Dong -Won LIM, Jaehyuk EOH, Ji-Young JEONG (KAERI)

Future I&C technologies for Nuclear Applications Ⅱ

Chair: Janos EILER (IAEA), Chang-hwoi KIM (KAERI)

4:00 PM	Determination of Allowable Setpoint for Safety Instrument in Consideration of Uncertainty and Confidence Level, Sanghoon BAE, Young-ki KIM, Chang-hwoi KIM (KAERI)
4:20 PM	Analysis of MEMS Based Earthquake Instrument for Nuclear Power Plant, Md. Mehedi HASAN, Jae Cheon JUNG (KINGS)
4:40 PM	VHDL Verification of FPGA based ESF-CCS for Nuclear Power Plant I&C System, Restu MAERANI, Jae Cheon JUNG (KINGS)
5:00 PM	Case For The Adoption Of FPGA Technology In The Implementation And Replacement Of Equipment And Systems In Nuclear Power Plants, Mark Joseph BURZYNSKI (SunPort SA)

System Simulation Technologies

Chair: Takeshi MATSUOKA (Utsunomiya University), Kee-Choon KWON (KAERI)

4:00 PM	Improvement of Wolsung Simulator including Severe Accident Analysis Models, Munsoo KIM, Yeonsub JUNG (KHNP)
4:20 PM	The Design and Implementation of Refuelling Machine Simulator Control System Based on FPGA, Zhijun HE, Peng ZHANG (China Nuclear Power Engineering CO., LTD)
4:40 PM	The Analysis and Simulation Study of the Control System for the Floating Nuclear Power Plant ACP100S, Zhi CHEN, Kai YOU, Tao LongTAO (Nuclear Power Institute of China)
5:00 PM	Multi-unit Small Modular Reactor Control System Experimental Platform Design, Xinyu WEI, Junyan QING, Fuyu ZHAO (Xi'an Jiaotong University)

Tuesday, November 28, 2017 10:00-12:00

Robotics & Automatic Remote Technologies I

Chair: Mitch PRYOR (University of Texas), Youngsoo CHOI (KAERI)

10:00 AM	**Novel UAV and UGV Platforms for Physical Interaction with the Environment in support of Nuclear D&D Operations,** Richard M. VOYLES, David CAPPELLERI, Shoushuai MOU (Purdue University), Howie CHOSET (Carnegie Mellon University), Robert BEAN (Purdue University), Rodrigo RIMANDO (U. S. Department of Energy)
10:20 AM	**Robotic Demonstrations Conducted at DOE Portsmouth Facility,** Wendell H. CHUN (University of Colorado Denver), Rodrigo V. RIMANDO (U.S. Department of Energy)
10:40 AM	**Derivation of Robot Mission for Nuclear Emergency Response,** Young Soo CHOI, Sung Uk LEE, Jai Wan CHO, Kyung Min JEONG (KAERI)
11:00 AM	**A Path Planning Algorithm for a Mobile Robot for Steam Generator Inspection in Nuclear Power Plants,** Kyungmin JEONG, Sun Young NOH, Youngsoo CHOI (KAERI)
11:20 AM	**Recent Works on Emergency Response Robots at Nuclear Robotics Laboratory of KAERI,** Ji Sup YOON, Jai Wan CHO, Youngsoo CHOI, Kyung-min JEONG, Jongwon PARK (KAERI)

Special Session: Cyber Security I

Chair: Eric LEMOINE (CNSC), Chul Hwan JUNG (CNSC)

10:00 AM	**Using Virtual and Augmented Reality to Improve Cyber Security and Physical Protection of Nuclear Material and Nuclear Facilities,** Scott GODWIN, Samuel CLEMENTS, Doug MacDONALD, Nick CRAMER, Rick REINSCHE, Shawn VanDYKE (PNNL)
10:20 AM	**Logging and Monitoring Parameters for Cyber Security Events of Digital I&C,** Jae-Gu SONG, Jung-Woon LEE, JunYoung SON and Cheol-Kwon LEE (KAERI), Paul SMITH (AIT)
10:40 AM	**Trustworthy Computer Security Incident Response for Nuclear Facilities,** Mislav FINDRIK, Ivo FRIEDBERG, Ewa PIATKOWSKA, Paul SMITH (AIT), Jae-Gu SONG (KAERI)
11:00 AM	**Development of a Quantitative Method for Evaluating Security Controls Based on Intrusion Tolerant Concept: Consideration of Adverse Effects,** Chanyoung LEE, Poong Hyun SEONG (KAIST)
11:20 AM	**Three-level Deep Packet Inspection for I&C systems of NPPs,** Jianghai LI, Xionghua SHENG, Shuanglin JIANG, Xiaojin HUANG (Tsinghua University)
11:40 AM	**A Graded Approach for Cyber Security Evaluation of Nuclear I&C System with Bayesian Update,** Jinsoo SHIN, Gyunyoung HEO (Kyung Hee University), Hanseong SON (Joongbu University)

System Reliability and Risk

Chair: Wu Jie (Institute of Nuclear Energy Safety Technology), Seung Jun LEE (UNIST)

10:00 AM	**An Approach To Assess The Impact Of Instrumentation With An Embedded Digital Device,** Richard T. WOOD, Tanner G. JACOBI, Dan C. FLOYD (University of Tennessee)
10:20 AM	**Analysis on Accident Sequences of SGTR Accident Considering the Status of Safety Valves,** Jaehyun HAM (KAIST), Hyun Gook KANG (RPI)
10:40 AM	**Uncertainty Characterization For Dynamic Risk Assessment,** Robby CHRISTIAN, Hyun Gook KANG (RPI)
11:00 AM	**A Coordination Review of AVR Limiter and Protective Function in Excitation System for Reliable Power System Operation in NPP,** Ji-Kyung PARK, Hong-Seok JANG and Young-Sik CHO (KINS)
11:20 AM	**An Estimation of the Effectiveness of an Hybrid-SIT System under SGTR Accident,** In Seop JEON, Hyun Gook KANG (RPI)
11:40 AM	**Analysis of operator available time for responding to accident situations in a digitalized main control room,** Ji Suk KIM, Eun Seo SO, Seung Hoon CHAE, Jae Seon HA, Eun Jin JEONG, Man Cheol KIM (Chung-Ang University)

Tuesday, November 28, 2017 14:20-16:00

Robotics & Automatic Remote Technologies II

Chair: Richard M. VOYLES (Purdue University), Kyungmin JEONG (KAERI)

2:20 PM	Generating Survey Plans for Autonomous Robots using Source and Instrumentation Data, Robert Blake ANDERSON, Mitch PRYOR, Sheldon LANDSBERGER (University of Texas at Austin)
2:40 PM	Dynamic Analysis of an In-Vessel Transfer System in Prototype Gen-IV Sodium-cooled Fast Reactor, Youn-Hee KWON, Dong-Won LIM, Sung-Kyun KIM (KAERI)
3:00 PM	Augmented Teleoperation for D&D, Young Soo PARK (ANL), Joohee KIM (Illinois Institute of Technology), Byung Seon CHOI (KAERI)
3:20 PM	Direct Lidar Odometry for a Rotating Multi-beam Lidar, Taewon KIM, Youngsoo CHOI (KAERI)
3:40 PM	A Design of Robust Control Algorithm for a Decommissioning Hydraulic Manipulator, Myoung Ho KIM, Sung Uk LEE (KAERI)

Special Session: Cyber Security II

Chair: Scott GODWIN (PNNL), Gyunyoung HEO (Kyung Hee University)

2:20 PM	Development of A Prototype FPGA based Security Module to Control Data Communication Network Access, Mohamed Abdallah EL-AKRAT, Jae Cheon JUNG (KINGS)
2:40 PM	Cyber Informed Engineering, Robert Stephen ANDERSON (INL)
3:00 PM	Risk Assessment of Operator Errors Induced by Cyber-Attacks on Nuclear Power Plants, Jong Woo PARK, Seung Jun LEE (UNIST)
3:20 PM	Implementation of Cyber Security for Computerized Operator Support System of Nuclear Facilities, Wei ZHENG , Lei MAO (Shanghai Nuclear Engineering Research and Design Institute)
3:40 PM	Identification of Critical Digital Assets for Nuclear Instrumentation System in Research Reactor, Sungmoon JOO, Sang Mun SEO, Yong–Suk SUH (KAERI)

Special Session: MFM and Safety Culture I

Chair: Morten LIND (DTU), Poong Hyun SEONG (KAIST)

2:20 PM	Knowledge Acquisition and Strategies for Multilevel Flow Modelling, Morten LIND (DTU)
2:40 PM	Enhanced Reasoning with Multilevel Flow Modeling Based on Time-to-detect and Time-to-effect Concepts, Seung Geun KIM, Poong Hyun SEONG (KAIST)
3:00 PM	Identifying Causality from Alarm Observations, Denis KIRCHHÜBEL, Xinxin ZHANG, Morten LIND, Ole RAVN (DTU)
3:20 PM	Barrier Identification by Functional Modeling of a Nuclear Power System, Jing WU, Morten LIND, Xinxin ZHANG (DTU), Pardhasaradhi KARNATI (ELDOR Technology AS)
3:40 PM	Reasoning about Cause-effect through Control Functions in Multilevel Flow Modelling, Xinxin ZHANG, Morten LIND (DTU)

Tuesday, November 28, 2017 16:20-18:20

Robotics & Automatic Remote Technologies Ⅲ

Chair: Wendell H. CHUN (University of Colorado), Young Soo PARK (ANL)

4:20 PM	Design of Emergency Response Robot Platform K-R2D2, Sun Young NOH, Kyungmin JEONG (KAERI)
4:40 PM	Robotic Technologies for Nuclear Remediation, Test and Evaluation, Knowledge Management, and Student Training, Leonel E. LAGOS, Dwayne MCDANIEL, Himanshu UPADHYAY, Ravi GUDAVALLI, Joseph SINICROPE, Peggy SHOFFNER (Florida International University)
5:00 PM	Air-ground Collaborative System for Nuclear Accident Monitoring, Jongwon PARK, Young-Soo CHOI (KAERI)
5:20 PM	Test of RGI system under Dense Aerosol Environments, Jai Wan CHO, Young Soo CHOI, Kyung Min JEONG (KAERI)
5:40 PM	Development of a Remotely Controlled Robot and Tool for Maintaining Tasks in Nuclear Facilities, Hocheol SHIN, Yeonggeol BEA, You Rack CHOI, Seung Ho JUNG (KAERI)

Special Session: Cyber Security Ⅲ

Chair: Robert Stephen ANDERSON (INL), Cheol-Kwon LEE (KAERI)

4:20 PM	Status of Canadian Cyber Security Regulatory Framework and Implementation at Nuclear Facilities, Chul Hwan JUNG, Eric LEMOINE (CNSC)
4:40 PM	Cyber Security Assessment Methodology of Critical Digital Asset in Nuclear Power Plant, Ieck Chae EUOM, Sung Cheol KIM, Joo Hyoung LEE (KEPCO KDN)
5:00 PM	Security Management of Virtualised Supervisory I&C Systems in Nuclear Facilities, M. HEWES, N. HOWARTH, C. HUNT, A. NOONAN (Australian Nuclear Science and Technology Organisation)

Special Session: MFM and Safety Culture Ⅱ

Chair: Andreas BYE (OECD HRP), Akio GOFUKU (Okayama University)

4:20 PM	Accident Management of the Station Blackout at BWR by Using Multilevel Flow Modeling, Mengchu SONG, Akio GOFUKU (Okayama University)
4:40 PM	Modelling and Validating a Deoiling Hydrocylone for Fault Diagnosis using Multilevel Flow Modeling, Emil Krabbe NIELSEN, Mads Valentin BRAM, Jérôme FRUTIGER, Gürkan SIN, Morten LIND (DTU)
5:00 PM	An Overview of the MFM Suite for Diagnostic and Prognostic Reasoning of Industrial Process Plants, Harald P-J THUNEM (Institute for Energy Technology)
5:20 PM	Development of an Evaluation Method for Nuclear Safety Culture Competency using Social Network Analysis, Sang Min HAN, Poong Hyun SEONG (KAIST)
5:40 PM	What We Have Learned so Far About the Importance of MTO in Control Room Design, Andreas BYE (OECD HRP)
6:00 PM	Review of Emergency Operating Guidelines from Nuclear Safety Culture Perspectives, Ho Bin YIM, Jae Min PARK, Chang Gyun LEE, Myung Hoon LEE, Jae Young HUH, Gyu Cheon LEE (KEPCO E&C)

Wednesday, November 29, 2017 10:00-12:00

Wireless Technologies in Nuclear Applications

Chair: Leonel E. LAGOS (Florida International University), Jung-Soo KOH (KINS)

10:00 AM	**A Study on Electromagnetic Compatibility to Adopt Wireless Technology in Nuclear Power Plants,** Dong-Jin LEE, Jaeyul CHOO, Hyun Shin PARK, Youngdoo KANG, Youngsik CHO (KINS)
10:20 AM	**Application Methodology of Wireless Communication Technology for Nuclear Power Plants,** Taejin KIM, Myunghoon AHN, Jongsoo KWON, Joohwan LEE (KEPCO E&C)
10:40 AM	**Electromagnetic evaluation for precaution against electromagnetic interference in nuclear power plants,** Jaeyul CHOO, Dong-Jin LEE, Hyung Tae KIM, Daehee KIM, Youngdoo KANG, Hyun Shin PARK, Youngsik CHO (KINS)

Safety Critical Software Development and Qualification

Chair: Ian JUNG (USNRC), Man Cheol KIM (Chung-Ang University)

10:00 AM	**FBDScenaGen+: GA-based High-Quality Scenario Generator for FBD Simulation,** Eui-Sub KIM, Sejin JUNG, Junbeom YOO (Konkuk University), Young Jun LEE, Jang Soo LEE (KAERI)
10:20 AM	**A Framework for the Safety Assurance of Safety Software in Nuclear Power Plants,** Kee-Choon KWON, Jang-Soo LEE (KAERI), Eunkyoung JEE (KAIST)
10:40 AM	**One--Step - Logic Automatic Translation For FPGA Applications,** Allen HSU, Steve YANG (Doosan HF Controls Corp.)
11:00 AM	**Development of Software Testing Environment for Safety-critical Software Reliability Quantification,** Sang Hun LEE (RPI), Seung Jun LEE (UNIST), Jinkyun PARK (KAERI), Eun-chan LEE (KHNP), RPI (Hyun Gook KANG)

Testing and Maintenance

Chair: Richard T. WOOD (University of Tennessee), TBD

10:00 AM	**Contribution of Electronic Circuit Simulation to Maintain and Exploit Perpetuated I&C Systems in Nuclear Power Plant,** Alain OURGHANLIAN (EDF Lab Chatou)
10:20 AM	**An Improved Response Time Test Methodology for the Plant Protection System and Engineered Safety Feature - Component Control System,** Chang Jae LEE, Jae Hee YUN (KEPCO E&C)
10:40 AM	**Development of an Information Reference System using Reconstruction Models of Nuclear Power Plants,** Yuki HARAZONO (Kyoto University), Taro KIMURA (SoftBank Corp.), Hirotake ISHII, Hiroshi SHIMODA (Kyoto University), Yuya KOUDA (JAEA)
11:00 AM	**Model Based Sensor Parameter Estimation and Smart Calibration Scheme,** Mujtaba MUJAHID, Ahmed YAR, Talha AZFAR (PAEC)
11:20 AM	**Updated Electromagnetic Compatibility Guidance For Nuclear Power Plant Instrumentation And Control Systems,** Richard T. WOOD, David M. DAWOOD (University of Tennessee)
11:40 AM	**Equipment Testing for Severe Accident Conditions,** Vít PLAČEK (ÚJV Řež), Zuzana KONEČNÁ (FEE – Czech Technical University in Prague)

Technical Sessions

Monday, November 27, 2017 10:40-12:20

1. Methods of Sensing, Processing, and Communication

Room #201
Session Chair: Wes HINES (University of Tennessee), Inkoo HWANG (KAERI)

Accuracy Review on Long Wired RTD Instrumentation Circuits

Inkoo HWANG, Jungtack KIM, Sub HUR

Division of Nuclear ICT Research, Korea Atomic Energy Research Institute, Daejeon, Korea

This paper presents a review on the several measurement circuits alleviating the influence of signal line wire resistance in the RTD applications in a plant instrumentation and control system. A 3-wired bridge circuit, a 3- or 4-wired current source circuit, or a usage of a transmitter is the typical solution which is widely adopted in instrumentation and control industries. The amounts of accuracy errors in different measuring techniques were calculated and compared each other in a practical installed situation. Other pros and cons of each circuit to be considered for I&C designers were also reviewed.

Implementation of Post-Accident Monitoring Systems (PAMS) in Ukrainian NPP's

Taras TKACH

NNEGC "Energoatom", Kyiv, Ukraine

Short historical background of PAMS implementation is introduced. The general order of safety related systems modification, the development of the engineering solution and the main concept of PAMS construction are shown. Works on implementation of PAMS in Ukrainian NPP's are represented and perspectives of I&C application are discussed.

Flow Rate Measurement Deviation Analysis considering Process Fluid Density Inconsistency for Differential Pressure Transmitter

Eunsuk OH, Byung Rae KIM, Seog Hwan JEONG

I&C System Engineering Dept., KEPCO E&C, 111, Daedeok-daero 989 Beon-Gil, Yuseong-gu, Daejeon, 34057, Republic of Korea

During cold hydro test for a nuclear power plant, a possible process measurement deviation was found that flow rate may be indicated lower than the rated flow. The previous analysis had been performed to identify the root cause, and as a result of the analysis, the exemption of high static line pressure correction to differential pressure (DP) transmitters was one of the major deviation factors.
Additionally, it was identified that the process fluid density for the test was not same as the normal operating process fluid density. This paper presents considerations, such as process fluid density compensations, to be incorporated in the process flow measurement due to the process fluid density variations which may occur in the cold hydro test stage. The process fluid density deviations may be induced by fluid type difference between the test and the actual operation, and by temperature and pressure variations during the test and the actual operation, thereafter, flow rate indication decreased by 1.21%.

A Modified Parity Space Averaging Technique for Online Calibration of Redundant Sensors in Nuclear Reactors

Moath KASSIM, Gyunyoung HEO

Department of Nuclear Engineering, Kyung Hee Univ., 17104, Yongin-si, Gyeonggi-do, Republic of Korea

Redundant sensors are usually used in nuclear reactors to measure critical variables and estimate their averaged time-dependent for maintaining safety and reliability of the reactor. Non-healthy sensors can badly influence the estimation result of the process variable. As online condition monitoring was introduced to enhance the reliability and maintainability of reactors, diagnosing the performance of redundant sensors online for the purpose of maintenance has become with high importance. Cross Calibration (CC) method is widely used to detect the anomaly of any sensor's readings among the redundant group. CC is a method that performs online averaging of redundant signals generating possible highly accurate estimation of the process variable and then compares each sensor signal with this estimate. Parity Space Averaging (PSA) technique is one of the averaging techniques used in CC method, it is used to weight the redundant signals based on their error band consistency. PSA assigns high weight to the signals that have shared bands, giving them weights regarding how many bands they share, and excluding the inconsistent signal from the averaging calculation by giving very low weight. EPRI has applied the parity space averaging in the Instrument Calibration and Monitoring Program (ICMP), thus, to enhance this technique, in this paper three methods are introduced for improving the PSA applied in the ICMP. The first was to add another consistency factor (so called Trend consistency *TC*) to consider to preserve the edge which can be a characteristic behavior or a real equipment fault of the process parameter. The second method proposed to replace the error band weighting factor (W^a) and the band consistency factor (C) by a weighting factor based on distance (W^d) and *TC* weighting factor, and the third method was to only replace W^d by W^a and apply it along with the *TC* and C . The redundant sensors underwent a preprocessing technique called Cross Moving Median (CMM) which can deal with noise, outliers, and missing data. Research reactor data sets were used to perform the validation of this method; four redundant hydrogen pressure transmitter signals (from S#1 to S#4) were obtained from the cold neutron source facility. Results regarding the ±3σ band showed that 2nd & 3rd modified approaches have rescannable improvement to the PSA technique.

Experience with Long Term Performance of Safety Related Pressure and Differential Pressure Transmitters in Ukrainian Nuclear Power Plants

Sergiy LEBEDYNSKYY, Volodymyr LEBEDYNSKIY

Private JSC "Manometr-Kharkiv", Lizy Chaikinoi Lane 17, Kharkiv 61052, Ukraine

Since 2001 Ukrainian nuclear power plants were reequipped with pressure transmitters "Safir" which were developed by Private JSC "Manometr-Kharkiv" in the late 90ties. At the moment more than twenty five thousand transmitters are in operation at 15 NPP units in Ukraine. According to current NPP safety requirements a line of specialized solutions for spent fuel pool level measurement and for presure measurement during accident situations: LOCA, HELB, harsh accident with operations conditions up to 250 °C, 160 kGy/h, 2,88 MGy TID gamma radiation were developed.

Monday, November 27, 2017 14:00-15:40

2. Surveillance, Diagnostics, and Prognostics I

Room #201
Session Chair: Zhe DONG (Tsinghua University), Man Gyun NA (Chosun University)

Sensor Selection Based On Boolean Network

Zhe DONG, Yifei PAN, Xiaojin HUANG

Institute of Nuclear and New Energy Technology, Collaborative Innovation Centre of Advanced Nuclear Energy Technology, Key Laboratory of Advanced Reactor Engineering and Safety of Ministry of Education, Tsinghua University, Beijing 100084, China

Process fault diagnosis strategies rely heavily on various types of sensors for temperature, pressure, concentration and etc. Due to the redundancy of the sensors in process systems such as chemical and nuclear plants, sensor selection schemes can deeply influence the diagnostic efficiency. In this paper, a Boolean network with its linear representation is proposed for describing the fault propagation among sensors, the sufficient conditions for both fault detectability and discriminability are given, and a sensor selection method for fault detection and discrimination is then proposed.
Finally, the theoretic result is applied to realize the diagnosis oriented sensor selection for a nuclear steam supply system, which not only verifies the feasibility but also show the implementation steps of theoretic results.

Monitoring the Status of Safety Functions using LSTM

Jaemin YANG, Jonghyun KIM

Department of Nuclear Engineering, Chosun University, 309 Pilmun-daero, Dong-gu, Gwangju 501-759, Republic of Korea

Human errors is one of the major factor for events that can aggravate the plant safety. Therefore, in case of abnormal or emergent situation, monitoring safety functions or diagnosis of Nuclear Power Plant (NPP) states are burdensome in spite of necessity. This study attempts to develop algorithms for monitoring the status of safety functions and diagnosis of accident. Therefore, it is expected that this approach can be applied to diagnose the overall NPP states.

AREVA's Spent Fuel Pool Level Measurement Solutions

Nicolas THILLOSEN[1], Sergio ESTEVEZ HERNANDEZ[1], Ryan REYNOLDS[2]

[1]AREVA GmbH, Henri-Dunant-Straße 50, 91058 Erlangen, Germany

[2]AREVA Inc. 3315 Old Forest Road, 24501 Lynchburg VA, Unites states

Reliable Monitoring of the water level in Spent-Fuel-Pool (SFP), also under degraded ambient conditions, is an essential function from the safety engineering point of view, and has become increased attention recently, as a consequence of the Fukushima accident. The proven and innovative solutions by AREVA NP focus on the fulfillment of the different customer requirements and provide complementary solutions for New Builds and modernization projects worldwide with Augmented Quality (AQ) as well as 1E-qualified severe accident resistant solutions fulfilling post-Fukushima and latest safety authorities requirements.

Prediction of LOCA Break Position and Size Using MSVM

Ju Hyun BACK, Kwae Hwan YOO, Young Do KOO, Man Gyun NA

Department of Nuclear Engineering, Chosun University, 309 Pilmun-daero, Dong-gu, Gwangju 501-759, Republic of Korea

Nuclear power plants (NPPs) consist of very large complex systems. If accidents happen in NPPs, operators will try to find out abnormal plant states by observing the temporal trends of some important parameters. In this regard, the objective of this study is to identify the accidents when the accidents happen in NPPs. In this study, the loss of coolant accidents (LOCAs) were identified and their break sizes were predicted using the multi-connected support vector machine (MSVM) model. The optimal parameter values of the MSVM model are obtained using genetic algorithms (GAs). The proposed algorithm uses the short time-integrated simulated sensor signals after the reactor trip. The results show that the MSVM model can predict the break position and size of the LOCAs accurately. Therefore, the LOCA identification and the accurate prediction of break size are useful for NPP operators when they try to manage LOCA accidents at NPPs.

New PRODIAG Algorithm and Acceptance Test

Young Soo PARK, Richard VILIM

Nuclear Engineering Division, Argonne National Laboratory, Lemont, IL, USA

This paper presents the improvement of an automated-reasoning computer program for nuclear power plant diagnosis, namely PRODIAG. PRODIAG, first developed at Argonne National Laboratory, is a physics-based fault diagnosis system which is plant configuration independent. To further enhance the code extensibility and maintenance, code upgrade has been made incorporating a modern object-oriented programming language, and an open-source automated reasoning engine. Verification test was also performed on a series of simulated faulty power plant operation data.

Monday, November 27, 2017 16:00-17:40

3. Surveillance, Diagnostics, and Prognostics Ⅱ

Room #201
Session Chair: Jae Cheon JUNG (KINGS), TBD

Development of a Smart Support System for Diagnosing Severe Accidents in Nuclear Power Plants

Kwae Hwan YOO[1], Ju Hyun BACK[1], Man Gyun NA[1], Seop HUR[2], Hyeon Min KIM[2]

[1]*Department of Nuclear Engineering, CHOSUN University, 309 Pilmun-daero, Dong-gu, Gwangju 501-759, Republic of Korea*
[2]*Korea Atomic Energy Research Institute, 111 Daedeok-daero, 989 Beon-gil, Yuseoung-gu, Daejeon, Republic of Korea*

Recently, human error has rarely (although it is not often) occurred during the power generation of nuclear power plants (NPPs). For this reason, many countries are conducting researches on the smart support systems of NPPs. Smart support systems can help decisions of operators in severe accident occurrence. In this study, a smart support system was developed to predict the core uncovery time, reactor vessel failure time, and containment failure time. Also, through this system, operator can predict the accident scenario, accident location and accident information in advance. In addition, it is possible to decide the integrity of the instrument and predict the life of the instrument. The data was obtained by simulating severe accident scenarios for the Optimized Power Reactor 1000 (OPR1000) using modular accident analysis program (MAAP) code. The prediction of the accident scenario, accident location and accident information is conducted using artificial intelligence (AI) methods.

Estimation of Cutter Wear of a Milling Machine Using a Support Vector Regression Method

Young Do KOO[1], Man Gyun NA[1], Jung-Taek KIM[2]

[1]*Department of Nuclear Engineering, Chosun University, 309 Pilmun-daero Dong-gu, Gwangju 61452, Republic of Korea*
[2]*Nuclear ICT Research Division, Korea Atomic Energy Research Institute, 989-111 Daedeok-daero Yuseong-gu, Daejeon 34039, Republic of Korea*

The integrity of various machineries and equipment, which are used in several industrial fields, can be considered as a factor determining safety and efficiency such as productivity and economy. In the sense, prognostics and health management (PHM) techniques are used to estimate the machine condition and life for effective maintenance and risk minimization. It is known that PHM has already been applied in a wide range of industrial fields such as automotive industry, aeronautics industry, several energy industries, and military.

PHM denotes diagnostics that monitors conditions of the mechanical system or device and detects its failure symptoms, prognostics of remaining useful life (RUL), and effective health management using sensors.

However, in a case of nuclear power plants (NPPs) that consist of large architecture and complex internal structures, it is known that the actual failure data of its system and device are difficult to be obtained, compared to other industrial fields. Thus, since the cutter in a milling machine can be considered as a rotor such as a pump and a turbine in a NPP, the cutter wear data from PHM 2010 society conference data challenge were used in an effort to study PHM technologies for NPPs.

In this study, support vector regression (SVR) as a data-driven approach was used to estimate a total of 6 cutter wear of a milling machine. The basic concept of SVR is to map the input data into a high-dimensional space by nonlinear mapping to solve a linear regression problem in this space. Among the cutters, 3 actual cutter wear data were compared with estimated values obtained from the SVR method using the signals from dynamometer, accelerometers, and acoustic emission sensors built in the experimental device. Consequently, the proposed method can accurately estimate the degree of cutter wear in the experimental device and it is expected that SVR has a capability to estimate wear of the rotating machines in NPPs in the future.

Towards Extracting 3-D Structural Representations of AGR Core Fuel Channels from 2-D In-Core Inspection Videos

Kristofer LAW[1], Graeme WEST[1], Paul MURRAY[1], Chris LYNCH[2]

[1]*Department of Electronic and Electrical Engineering, University of Strathclyde, 99 George Street, Glasgow G1 1RD, United Kingdom*

[2]*EDF Energy Generation, Barnett Way, Gloucester GL4 3RS, United Kingdom*

Remote Visual Inspection (RVI) of Advanced Gas-cooled Reactor (AGR) nuclear power stations allows engineers to gain an understanding of the AGR graphite core health by investigating the incorporated fuel channels. During planned, periodic outages, video footage of the pre-selected fuel channels is acquired using specialist inspection tools and is subsequently taken offline for further analysis using visualization techniques. Current methods of visualization however provide limited structural information due to the loss of depth information as a direct result of the image acquisition process. This paper introduces a new bespoke 3-D reconstruction framework to recover lost depth information to produce 3-D point cloud reconstructions of fuel channels from inspection videos. We also present here a new, lab based, experimental rig setup with which we effectively captured data under lab controlled conditions to verify our 3-D reconstruction algorithms. Our proposed method is tested on 2-D in-core inspection videos in addition to the footage captured within laboratory conditions and outperforms state-of-the-art incremental reconstruction frameworks whilst producing a more representative 3-D point cloud for improved in-core visualization.

Development of an Improved Data-Driven Diagnostic Platform for Process Plants: Case Study of Feedwater Heater Leakage

Gayeon HA, Ibrahim AHMED, Gyunyoung HEO

Department of Nuclear Engineering, Kyung Hee University, Yongin-si, 17104, South Korea

The degradation conditions of process plant components such as Feedwater Heater (FWH) can have a significant effect on the performance of the overall plant efficiency depending on the associated functions of those components. In this regard, there is need for a more robust equipment condition assessment technique that will provide a diagnostic system to detect equipment failures early enough to warn operators in advance of impending failures and to inform maintenance personnel about the nature of the failure. Several data-driven techniques to diagnose the degradation conditions of process plant components have been proposed. However, the actual performance of the data-driven methods when applied to the real-time environment after training depends largely on the quality of data used for training, as actual plant data are noisy. This paper seeks to improve the quality of data used for degradation assessment by introducing a bilateral filter. Bilateral filter is used as a smoothing and prediction technique to preserve the edges in the signal variables in order to eliminate the noise for effective signal predictions. Bilateral filters are applied to both training data and test data with dynamic filter bandwidths in order to extract multiple features thereby make it possible to extract the interest and relevant features from the data and to generate different patterns for each filter bandwidth. That is, multiple features can be obtained. Then, we developed a diagnostic model using Support Vector Machine (SVM) as a classification method. With the diagnostic platform created, the proposed methodology has been validated by applying it to internal leakage diagnosis of Feedwater Heater (FWH) in Nuclear Power Plants (NPPs).

An Ingenious Pressure Surveillance Algorithm to Detect CO_2 Ingress Accidents in a Sodium-cooled Fast Reactor

Dong-Won LIM, Jaehyuk EOH, Ji-Young JEONG

SFR System Design Div., Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, 34057, Korea

The supercritical CO_2 Brayton cycle employed in Sodium-cooled Fast Reactors (SFRs) is more advantageous than a conventional Rankine cycle energy conversion system. One of the benefits is an enhanced plant safety since potential reactions of CO_2 with liquid sodium have been reported to be less stringent than a sodium-water reaction (SWR) anticipated in the Rankine cycle, and CO_2 reactions also take more chemical reaction time than an SWR. Contradictorily, the reaction characteristics of CO_2 require a scrupulous plant operation. Moderate chemical interactions between CO_2 and liquid sodium imply that detecting CO_2 ingress accidents (unlike conspicuous physical indications such as noisy wastage due to SWR) makes it hard to detect at its early stage. In other words, the plant can run for an extended period of time until the rupture disk bursts, letting a damaged sodium-to-CO_2 heat exchanger degrade further. To detect CO_2 ingress accidents, this paper proposes an ingenious approach to compare the pressure measurements in real time of two identical heat exchangers, which are in a typical SFR configuration. The approach was originated from the ideas that the CO_2 ingress - a source of pressure transient in a loop - occurs owing to a crack at the pressure boundary wall of a sodium-to-CO_2 heat exchanger, a certain self-recovery of structural damage does not happen over time, and probabilistically, an accident occurs at only one component out of two at the first place. This pressure surveillance algorithm by which the threshold and decision time parameters can be determined is based on probabilistic performance analysis by setting false alarm and true detection rates. The decision time is chosen to be suf.ciently long to allow a particular ingress level can be detected even in the presence of asymmetric performances of the two heat exchangers. Finally, the proposed algorithm was developed with a simplified mass and energy transfer (SMET) model, and verified with experimental data obtained from a water mock-up test. The results show that 99.99% detection probability can be achieved for 30 cc/sec air ingress, which is equivalent to 0.12 g/sec CO_2 ingress, using a detection time of less than 70 seconds, limiting down to 0.1% false alarms due to sensor noise.

Tuesday, November 28, 2017 10:00-12:00

4. Robotics & Automatic Remote Technologies I

Room #201
Session Chair: Mitch PRYOR (University of Texas), Youngsoo CHOI (KAERI)

Novel UAV and UGV Platforms for Physical Interaction with the Environment in support of Nuclear D&D Operations

Richard M. VOYLES[1], David CAPPELLERI[2], Shoushuai MOU[2], Howie CHOSET[3], Robert BEAN[2], Rodrigo RIMANDO[4]

[1]*Polytechnic Institute, Purdue University, West Lafayette, IN, USA, American*

[2]*College of Engineering, Purdue University, West Lafayette, IN, USA, American*

[3]*Robotics Institute, Carnegie Mellon University, Pittsburgh, PA, USA, American*

[4]*Department of Energy/Environmental Management, Washington, DC, USA, American*

The current state-of-the-art in decontamination and decommissioning (D&D) remains largely based on manual operations in personal protective equipment (PPE) with human-operated equipment augmenting human abilities, as needed. Increasingly, robotic or remote-piloted equipment, such as back hoes and bull dozers, have been making inroads into conventional construction projects, and these implements are beginning to appear in D&D projects in the nuclear industry. With the potential benefits to worker safety and reduced risk of exposure, we have been working with a broader array of robotic equipment to bring these derived benefits earlier in the process to the realm of inspection and assessment of potentially contaminated facilities.

In this paper, we describe recent demonstrations at the Portsmouth Gaseous Diffusion Plant-and their proposed extensions to more difficult problems-during a U.S. DOE "Science of Safety" exercise to explore new technologies and new applications for nuclear D&D. We first explore the ramifications of a new class of unmanned aerial vehicle (UAV) designed for physical interaction with the environment that permits new levels of fidelity in tasks such as remote swabbing of potentially contaminated surfaces and inspection and sealing of enclosures. We also describe the advanced mobility possible with modular, snake-like unmanned ground vehicles (UGV) and hybrid combinations for advanced inspection and characterization.

The Dexterous Hexrotor, BoomCopter, and Tiltrotor VTOL are example UAVs capable of hovering in place, as well as horizontal flight. With varying degrees of precision, these flying vehicles can apply controllable forces to the environment for a large variety of useful tasks, while still being able to cover significant distances between physical interactions, even with limited battery life. The three distinct flying morphologies represent tradeoffs between the precision of contact forces and the range of coverage with a single battery charge. Example applications of contact-based swabbing of lightly contaminated surfaces; opening, inspecting and sealing electrical enclosures; and inspection of very large structures are examined. This robotic vehicles can keep humans out of hazardous environments while extending their ability to see and feel.

Likewise, the MOTHERSHIP and CMU Snake robots are serpentine, ground-based robots that can explore confined spaces and other unknown or challenging spaces. Modular in design, these robots can be configured for a variety of tasks, leveraging operator skills acquired in one task for other tasks.

They can also be combined to capitalize on the heterogeneous nature of the size scales, so composite mechanisms can be purpose-built to fit the task at hand.

Robotic Demonstrations Conducted at DOE Portsmouth Facility

Wendell H. CHUN[1], Rodrigo V. RIMANDO[2]

[1]*School of Engineering and Applied Science, University of Colorado Denver, Colorado, USA*

[2]*Office of Environmental Management, U.S. Department of Energy, Washington DC, USA*

The U.S. Department of Energy's (DOE) next major decommissioning project is at its Portsmouth Facility in Piketon, Ohio. This plant was one of three large gaseous diffusion plants in the United States that was initially constructed to produce enriched uranium, a source used by commercial nuclear reactors from all around the world. The DOE Science of Safety (SOS) program is chartered with the smart infusion and integration of scientific and technological advancements into the routines of workers to enhance worker health and safety and to reduce the federal government's liability of its nuclear legacy cleanup. The specific purpose for the SOS Robotics Challenge in 2016 at the Portsmouth Gaseous Diffusion Plant was to have workers and operators demonstrate EM-Mission relevant, novel, adequately mature, robotic and related enabling technologies onsite and to increase awareness and garner support of EM stakeholders, appropriators, and Congress on opportunities of mission-relevant robotic technologies.

In August of 2016, we conducted a week of demonstrations at Portsmouth with a goal of connecting the user community with a wide range of robots/robot technologies, and earn the support of the many stakeholders involved. This demonstration was able to showcase the selected technologies in a realistic environment. The twelve demonstrations were used to train future operators, and thus gaining valuable lessons learned. The knowledge and information gathered that week enabled future planning and development, leading to improved operations for upcoming DOE tasks. Each demonstration was evaluated to understand its known risks using the DOE Technology Readiness Levels. And the range of different types of demonstrations were very broad and representative of robotic technologies in general. Some demonstrations would require further development while demonstrations that featured inspection tasks with a teleoperated platform performed very well, as expected based on the maturity of these types of robots. The technology was primarily off the shelf from leading universities from around the country.

At DOE, robots as *tools* to be used by *trained* operators to do their jobs better, safer, and more efficiently. The users were representative of the local steel workers union whose future job is the decontamination and decommissioning (D&D) of the Portsmouth site. The common issues with robots in a nuclear environment are communication (whether tethered versus wireless), and command and control with respect to human operators (referred to this as teleoperation, and is one end of a continuous spectrum up to robot systems that are automatic and autonomous). In this paper, we expand on each of the twelve demonstrations, and relate them to the needs of the Portsmouth site. This technology represented a baseline of general robotic capabilities that would feed into a technology roadmap that is currently in development at DOE.

Derivation of Robot Mission for Nuclear Emergency Response

Young Soo CHOI, Sung Uk LEE, Jai Wan CHO,
Kyung Min JEONG

Nuclear Robot & Diagnosis Team, KAERI, 111 Daedeok-daero 989 beon-gil Yuseong-gu Daejon, 34057, Korea

In an accident situation which is expected to be extreme conditions such as high temperature, high pressure, and high radioactivity, human workers are restricted under such conditions. Robot, which is relatively free of restrictions on environmental conditions, can be applied to prevent and mitigate events more quickly and stably in abnormal and emergency state. In order to develop robot responding to nuclear events, it is necessary to determine the robot's mission that can contribute to the safety of the power plant. Through the review of the abnormal operation procedures, emergency operation procedures and severe accident management guidelines, we have drawn up robot missions. The criteria of robot missions are field work needs and anticipated human threats. The extracted 20 missions are listed and some of important mission scenarios are analyzed to develop robot system. And testbed will be constructed for the verification of robot performance on the basis of the mission.

A Path Planning Algorithm for a Mobile Robot for Steam Generator Inspection in Nuclear Power Plants

Kyungmin JEONG, Sun Young NOH, Youngsoo CHOI

Korea Atomic Energy Research Institute, Daejeon, Korea

The reliability and performance of a steam generator (SG) is one of the serious concerns in the operation of pressurized water nuclear power plants. Because of high levels of radiation, robotic systems have been used to inspect and repair SG tubes. This paper introduces a mobile robotic system that positions the inspection and repair tools while hanging down from the tube sheets where the tubes are fixed. In the comparison that the end point of typical serial manipulators used in those areas are controlled by analytically calculating the each joint angle from the inverse kinematics equations, the motion control of the end point of the mobile robot is not so easy problem because there are some restrictions for them. Usually the direction of the motion of the mobile robot should be selected from the finite number of orientation that depends on the arrangement of the tubes and the distance of movement at one time is also depending on the pitch of the tubes and the stroke length of the joint of the robot. Thus the path planning of the mobile robot is essentially a searching problem for the motion tree that represents the state of the robot in its node. This paper suggests an optimized path planning algorithm and provides some simulation results for the steam generator of APR1400 nuclear power plants.

Recent Works on Emergency Response Robots at Nuclear Robotics Laboratory of KAERI

Ji Sup YOON, Jai Wan CHO, Youngsoo CHOI,
Kyung-min JEONG, Jongwon PARK

Nuclear Robotics Lab, KAERI, 111, Daedeok-daero 989, Yusung-gu, Daejeon, 34057, Korea

Following the Fukushima accident, the importance of safety and emergency preparedness of nuclear power plants (NPPs) has been increasingly emphasized. In 2012, the Nuclear Robotics Laboratory (NRL) at Korea Atomic Energy Research Institute (KAERI) initiated research on an unmanned emergency response robotics system. This research is fully funded by the government and is aimed at providing a practical means which countermeasure the initial accident stages of NPPs.

Considering that the robotic systems that tried to mitigate the damage caused by the Fukushima accident did not work adequately, the robotic system to be developed should be robust, be supportive to human operators, and perform unmanned operations in a remote manner. In this research, three key features are emphasized considering the lessons learned from the Fukushima accident: the mobility of robotic systems, remote monitoring capability, and reliability of the sensing modules and mobile platforms. For mobility, commercially available mobile platforms such as an all-terrain vehicle (ATV) and a forklift were adopted and modified for remote operation. For remote monitoring, a range-gated image (RGI) camera system was introduced for obtaining visual information in an invisible environment due to dense fog inside a reactor confinement building. For reliability, radiation hardened electronics were investigated and adopted. In this paper, the recent works on this system are briefly introduced.

Tuesday, November 28, 2017 14:20-16:00

5. Robotics & Automatic Remote Technologies Ⅱ

Room #201
Session Chair: Richard M. VOYLES (Purdue University), Kyungmin JEONG (KAERI)

Generating Survey Plans for Autonomous Robots using Source and Instrumentation Data

Robert Blake ANDERSON, Mitch PRYOR, Sheldon LANDSBERGER

Department of Mechanical Engineering, University of Texas at Austin

This paper reviews components for generating survey plans for a robotic system at UT Austin used autonomously survey safety travel corridors, laboratory spaces, and other areas that require routine radiation surveys. Survey results can be archived and compared to previous surveys in order to detect either changes in the field or new hot spots that exceed a given threshold. The system already utilizes dynamic obstacle avoidance and a robust supervisory controller accounting for closed doors, temporary obstacles, and battery limitations, etc. In order to automate survey planning, the software must automatically select sensors, survey resolution, survey locations, and counting time necessary in order to generate meaningful survey data in a reasonable amount of time. In our approach, the survey points span a rectangular grid pattern with spacing either set by the operator, designated by survey requirements, or optimized with respect to confidence levels and/or survey time. The system considers counting statistics in its data collection function that tracks the count uncertainty and guarantees confidence limits given by a Poisson distribution specified by the task. Previous iterations of the system have been deployed in U.S. Department of Energy environments, and newly developed automated planning algorithms are tested at U.T. Austin to evaluate effectiveness and feasibility.

Dynamic Analysis of an In-Vessel Transfer System in Prototype Gen-IV Sodium-cooled Fast Reactor

Youn-Hee KWON[1], Dong-Won LIM[2], Sung-Kyun KIM[3]

[1]SFR System Design Div., Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, 34057, Korea

A Sodium-cooled Fast Reactor (SFR) as a generation IV (Gen-IV) reactor design employs one of the most advanced refueling methods. These refueling systems are mostly controlled automatically using state-of-the-art sensors and actuators. The in-vessel fuel transfer system (IVTS) refers to a fuel handling machine and driving mechanism in the reactor vessel, and IVTS is the most critical in that accurate positioning and stable driving make the reactor operation reliable for a few decades. Sub-components that IVTS comprises are only the moving parts in the vessel, and dynamic analysis of the IVTS components is of great importance for the successful SFR design and operation. This paper considers IVTS dynamic simulations of a Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR). For PGSFR IVTS, double rotating plugs (DRP) and a fixed-arm charge machine (FACM) are deployed for the refueling system. This paper proposes an efficient modeling approach to reflect the full 3D dynamic behavior of PGSFR IVTS. The modeling task engages in formulating a standard robot manipulator problem set for PGSFR IVTS. Based on the modeling with a coordinate system, its kinematics analysis yielded strategic path planning. A simplified dynamic simulation model was used as a spring-mass-damper system. The simulation method proposed reflected hydrodynamic forces without employing the fluid domain. The parametric study on an angular velocity of up to 7 rpm of each actuator was carried out through a dynamic simulation. The motion induced vibration was also analyzed to set up a safe refueling procedure. The study concluded that a rotational speed of up to approximately 2 rpm of all rotational components can be allowed based on the tip displacement analysis.

Augmented Teleoperation for D&D

Young Soo PARK[1], Joohee KIM[2], Byung Seon CHOI[3]

[1]Nuclear Engineering Division, Argonne National Laboratory, Lemont, IL, USA

[2]Department of Electrical and Computer Engineering, Illinois Institute of Technology, Chicago, IL, USA

[3]Korea Atomic Energy Research Institute, Daejeon, Republic of Korea

This paper presents a development of a new teleoperation method to enhance the performance of telerobotic operations for nuclear facility decontamination and decommissioning (D&D). It focuses on enhancement of human-robot interface based on introduction of virtual reality (VR) and augmented reality (AR) technologies which can provide immersive artificial environment for the operator to interact with. In this regard, the key technology innovation is realized by 3D sensing and reconstruction and 'virtual fixtures'. The development will allow to use simple and robust robotic systems for complex and dexterous D&D task operations.

Direct Lidar Odometry for a Rotating Multi-beam Lidar

Taewon KIM, Youngsoo CHOI

Nuclear Robot.Diagnosis Laboratory, Korea Atomic Energy Research Institute, Daejeon, Korea

In this paper, we propose a novel lidar odometry algorithm. It uses only a rotating multi-beam lidar without any other sensors, but is fast, precise and robust. Existing state-of-the-art methods use feature matching or finding the nearest neighbor to calculate cost between two point clouds. However, the proposed method calculates the cost directly using cylindrical projection instead of feature matching and finding the nearest neighbor. This is an extension of the direct method to a rotating multi-beam lidar. We also propose reliability weighting. When the pose estimation is finished, the reliability weight is calculated between the two cylinder images and used as the initial weight of the next step. This reduces the effects of moving objects and occlusion. Finally, we evaluate the algorithm in the KITTI odometry benchmark.

A Design of Robust Control Algorithm for a Decommissioning Hydraulic Manipulator

Myoung Ho KIM, Sung Uk LEE

Nuclear Robot·Dignosis Laboratory, Korea Atomic Energy Research Institute, 111 Daedeok-Daero 989 Beon-Gil Yuseong-Gu Daejeon, 34057, KOREA

In the decommissioning process of the nuclear power plant, robotic systems that can work on hazard environment instead of human are essential due to high levels of radiation. Especially, nuclear facilities are large and complex, and also the operation of handling heavy-duty objects is needed during dismantling process. Thus, a hydraulic manipulator was developed to save time and cost for decommissioning nuclear power plant. However, hydraulic systems have many uncertainties and robust controller is necessary for precise control. Most existing robust controllers require acceleration information to cancel uncertainties and improve performance. Acceleration information that is obtained by numerical differentiation is very noise and difficult to obtain directly. Therefore, a method that is robust and does not use acceleration information is required. In this paper, we studied about simple robust control algorithm without acceleration information for decommissioning hydraulic manipulator. In addition, the simulation and experiment are carried out to validate control performance.

Tuesday, November 28, 2017 16:20-18:20

6. Robotics & Automatic Remote Technologies Ⅲ

Room #201
Session Chair: Wendell H. CHUN (University of Colorado), Young Soo PARK (ANL)

Design of Emergency Response Robot Platform K-R2D2

Sun Young NOH, Kyungmin JEONG

Nuclear Robot · Diagnosis Laboratory, Korea Atomic Energy Research Institute, Daejeon, Korea

This paper presents a mechanical and electrical design of tripedal mobile platform for emergency response in man-made environments. Previously, we described design concepts of K-R2D2, which has 14 degrees of freedom (11 for active motions and 3 for passive motions), 120cm tall and weighs 93kg. It has three legs which are linearly extendable along each length to change the mass center during walking. The proposed leg mechanisms is modeled for gait locomotion based on walking up the stairs and each leg is connected with its foot by a variable stiffness ankle which runs free or is rigidly fixed. This paper also explains the robot controller architecture which is easily reconfigurable.

Robotic Technologies for Nuclear Remediation, Test and Evaluation, Knowledge Management, and Student Training

Leonel E. LAGOS, Dwayne MCDANIEL, Himanshu UPADHYAY, Ravi GUDAVALLI Joseph SINICROPE, Peggy SHOFFNER

Applied Research Center, Florida International University, 10555 West Flagler Street, Suite 2100, Miami, FL 33174, USA

For over 20 years, the Applied Research Center (ARC) at Florida International University (FIU) has conducted university based fundamental and applied research for the U.S. Department of Energy's Office of Environmental Management (DOE EM) in the area of deactivation and decommissioning (D&D), high level waste (HLW) processing, technology innovation, development, testing/evaluation, and deployment; knowledge management; and STEM student training. Recent research with innovative robotic/remote systems include the development of a mini rover system and a pneumatic pipe crawler developed for the inspection of double shell tanks (DST) at DOE's Hanford Site in east Washington State. Other innovative remote system developments have included the design and prototyping of a wall decontamination system and the test and evaluation of a wall climbing remote system patented by International Climbing Machines (ICM). In addition, FIU is currently working to identify cross-cutting robotic technologies being developed for other applications that could potentially be used in support of hot cell and glove box D&D activities. The test and evaluation of D&D technologies includes investigating the effectiveness of commercially available intumescent coatings to enhance the fire resiliency of fixatives and facilities in support of D&D projects facing potential fire and/or extreme heat conditions; developing a to-scale hot cell testbed for testing, evaluating, and demonstrating technologies; and evaluating an advanced fogging agent developed at Idaho National laboratory (INL) for fixing loose radioactive contamination as well as for knocking down airborne particulates. To collect, maintain, preserve and disseminate the D&D knowledge base from DOE and around the world, FIU developed an online web-based system, the D&D Knowledge Management Information Tool (D&D KM-IT), to capture the knowledge, experience, and lessons learned from past and present D&D projects. The D&D KM-IT currently hosts information on over 1,300 D&D technologies, including over 521 robotic and remote systems. Finally, the scientific research being performed at ARC provides real-world hands-on training for FIU STEM students through the DOE Fellows Program, preparing the workforce of tomorrow to meet the challenges of the nation's largest environmental cleanup mission.

Air-ground Collaborative System for Nuclear Accident Monitoring

Jongwon PARK, Young-Soo CHOI

Korea Atomic Energy Research Institute, Daedeok-daero 989-111, Yuseong-gu, Daejeon, 04535, Republic of Korea

Nuclear accidents involve radioactivity leakage, which can cause a catastrophic environmental disaster. Therefore, the emergency response to nuclear accidents is crucial in the early stages. Generally, nuclear power plants are huge and consists of many auxiliary buildings. In addition, the individual buildings are high, large and complicated. Therefore, it is practically difficult to acquire information of the accident with the existing robot systems in a quick manner. In this study, we propose a joint operation system of unmanned ground vehicle (UGV) and unmanned aerial vehicle (UAV) to perform rapid ground to aerial accident monitoring.

Test of the RGI (range-gated imaging) system under Dense Aerosol (Fog) Environments

Jai Wan CHO, Young Soo CHOI, Kyung Min JEONG

Robotics & Component Diagnostics Dept., Korea Atomic Energy Research Institute, Daejeon, Korea

A RGI (range-gated imaging) system, which can be used to eliminate backscatter in strong scattering (dense aerosol) environments, is based on two high speed technologies. It uses high power, ultra-short pulse laser (350ps, 40uJ) as the light source. And open the optical gate of the ICCD camera with micro-channel plate (MCP) image intensifier in a very short time (about 200ps) while the laser pulses reflected by the object is coming back to the ICCD camera. Using this range-gated imaging technology, the effect of scattered light can be reduced and clear image is obtained.

In this paper, the test results of the RGI (range-gated imaging) system under dense aerosol (fog) environments, which simulates the severe accident situation of the nuclear power plant, are described. To verify the monitoring performance of the RGI system under dense fog environment, we made the test facility (2.5m × 2.5m × 15m, WHD). The fogs are injected inside the test facility until the concentration of the fog is saturated. At the same fog concentration conditions, we compared the monitoring performances of the RGI system and color CCD camera.

Development of a Remotely Controlled Robot and Tool for Maintaining Tasks in Nuclear Facilities

Hocheol SHIN, Yeonggeol BEA, You Rack CHOI, Seung Ho JUNG

Nuclear Robot and Diagnosis Team, Korea Atomic Energy Institute, 989-111 Daedeok-daero, Yuseong-gu, Daejeon, 305-353, Korea

We are developing a remotely controlled mobile robot for maintaining tasks in nuclear facilities, which consists of a mobile platform, a telescopic mast, and a dual arm slave robot with a working tool. As the remote target tasks, we set up a remote manipulation of the manual operation mechanism of the nuclear fuel changer of the heavy water NPP and remote pipe cutting/welding, which may be necessary in the case of an emergency or dismantling of the NPP. To effectively perform remote tasks, we designed the architecture of an integrating program. The integrating program has a system component control module, a virtual guide implementing module, and egocentric remote control algorithms. We also developed tools to perform the target remote tasks.

Wednesday, November 29, 2017 10:00-12:00

7. Wireless Technologies in Nuclear Applications

Room #201
Session Chair: Leonel E. LAGOS (Florida International University), Jung-Soo KOH (KINS)

A Study on Electromagnetic Compatibility to Adopt Wireless Technology in Nuclear Power Plants

Dong-Jin LEE, Jaeyul CHOO, Hyun Shin PARK, Youngdoo KANG, Youngsik CHO

Korea Institute of Nuclear Safety, Daejeon, Korea

Recently, increasing numbers of wireless technologies have been deployed in industrial environments such as nuclear power plants (NPPs). Adoption of wireless technologies has proved benefits in many industries in terms of saving cable costs and installation time as well as increased flexibility of information gathering through temporary sensor deployment. Nevertheless, wireless technologies are not currently deployed in nuclear industry at large due to some serious issues. The electromagnetic compatibility (EMC) between the wireless devices and the existing plant instrumentation and control systems is major issue in deploying wireless technologies in NPPs. In this paper, Electromagnetic Interference and Radio Frequency Interference (EMI/RFI) in Regulatory Guide 1.180 rev.1 are observed and other regulation standards (MIL-STD 461E, IEC 61000-4) which are endorsed in Regulatory Guide 1.180 rev.1 is also examined. It also provides EMC and EMI/RFI characteristics when transceivers such as portable and fixed wireless devices are used in free-space and multi-path environment.

Application Methodology of Wireless Communication Technology for Nuclear Power Plants

Taejin KIM, Myunghoon AHN, Jongsoo KWON, Joohwan LEE

I&C department, KEPCO E&C, 111 Daedeok-daero 989beon-gil, Yuseong-gu, 34057, Republic of Korea

Wireless communication technologies have been continuously studied due to their flexibility and cost efficiency in nuclear industry. In order to apply wireless technologies into nuclear power plants (NPPs) for communication purpose, electromagnetic Interference (EMI) with the installed instrumentation and control (I&C) equipment in NPPs should be considered. In this paper, we have investigated characteristics of various wireless technologies and proposed a Terrestrial Trunked Radio (TETRA) technology as an adequate wireless one for NPPs through rigorous review. Also, we suggest an applicable method to apply TETRA technology into NPPs by minimizing EMI on the existing equipment. Upon investigation of the I&C equipment in the Shin-Kori NPP Unit 3 (SKN3) which is the first-of-a-kind (FOAK) plant of advanced pressurized water nuclear reactor 1400 MW electricity (Advanced Power Reactor 1400, APR1400), the standards of RS103 in MIL-STD-461E and IEC 61000-4-3 are used for EMI testing. Based on the regulatory position in Reg. Guide 1.180, the exclusion zone is presented by calculation for the equipment qualified by RS103. To set up the exclusion zone of the equipment qualified by IEC 61000-4-3, the intensities of interference to the existing equipment for the test signals of RS103 and IEC 61000-4-3 are compared by analyzing the electric field intensities and the bandwidths of the test signal's spectrums. Moreover, electromagnetic waves from the wireless devices are simulated to compare the electric field intensities between free space and cabinet-installed environment.

Electromagnetic Evaluation for Precaution against Electromagnetic Interference in Nuclear Power Plants

Jaeyul CHOO, Dong-Jin LEE, Hyung Tae KIM,
Daehee KIM, Youngdoo KANG, Hyun Shin PARK,
Youngsik CHO

Department of Instrument, Control, and Electrical Systems, KINS, 34142, Daejeon, Korea

This paper represents the electromagnetic evaluation to avoid electromagnetic interference problems in a nuclear power plant using a mode-matching method. Using the mode-matching method in conjunction with the separation of variables, the superposition principle, and the Fourier transform, we perform the electromagnetic interpretation for three postulated topics: an open cabinet containing the digital instrumentation and control (I&C) modules when it is exposed to an external electromagnetic source, the axially-ruptured coaxial cable with multiple dielectrics, and the shorted cable trays between lateral walls. The examined results provide us with the useful information for avoiding the electromagnetic interference problem for the postulated cases.

Monday, November 27, 2017 10:40-12:20

8. Modernization of I&C and Control Room

Room #202
Session Chair: Jamie COBLE (University of Tennessee), TBD

An Overview of the Progress in Research and Development in Advanced Instrumentation, Information, and Control Systems Technologies to Support Light Water Reactor Sustainability

Bruce P. HALLBERT, Kenneth D. THOMAS
Idaho National Laboratory

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 (NPP) 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.

Achieving a set of long term sustainable plant II&C systems that are based on modern digital technologies is being accomplished largely by modernization via individual refurbishment projects. The outcome of many refurbishments will lead to a hybrid mixture of analog and digital technologies many plants and even main control rooms. Operators and maintainers of II&C systems will, for an extended duration, require competencies with both types of technologies. This represents one approach to longterm asset management that also allows plant personnel to become familiar with newer digital systems as they gradually replace analog devices.

A long-term strategic approach to plant needs and modernization opportunities has been developed with owner-operators of US nuclear power generation, in order to establish a plan for research and development that enables technology development and long-term modernization within the existing light water reactor fleet. A series of research and development projects are being executed that enable the development and deployment of new II&C technologies in existing nuclear plants. Through the Light Water Reactor Sustainability (LWRS) program, individual utilities and plants support these projects and are also able to directly leverage the results of research projects.

Modernization of NPP's Safety I&C-Challenges and Solutions

Bernd RUDOLF

AREVA GmbH, Henry-Dunant Str. 50, 91058 Erlangen, Germany

A substantial share of the world's fleet of nuclear power plants reached already a service life were modernization of the I&C systems needs to be considered. Such I&C modernization projects present various challenges for plant owners as well as for suppliers. In the past more than 10 years, multiple modernizations of safety I&C systems based on digital systems have been successfully implemented or are currently under execution, in addition to the current new builds that are equipped with digital I&C. Nevertheless, concerns regarding modernization based on digital I&C still exist or even seem to increase.

As a leading supplier of safety I&C modernization solutions AREVA NP has implemented a wide range of project types from small scale to comprehensive, and with highest qualification requirements, including its digital safety I&C platform TELEPERM® XS.

The presentation provides an overview of the challenges, approaches and solutions of I&C system modernizations based on selected examples.

Information will be shared related to modernization drivers, definition of modernization scope, project constraints, the selection of the I&C platform, the technical solution, the project implementation, the cooperation between the parties as well as the success factors Main key success factors are, of course, a well-defined perimeter of the modernization project, proven engineering solutions, a suitable and modular I&C platform, suppliers' competence and strong teamwork between operator and supplier.

This said I&C modernization based on digital technology is a proven and sustainable investment in a nuclear power plant's safe and reliable operation.

Modernization of Nuclear Power Plant's Instrumentation and Control Systems on the Basis of FPGA Technology

Ievgenii BAKHMACH[1], Olexandr SIORA[1], Anton ANDRASHOV[2], Vyacheslav KHARCHENKO[3], Andriy KOVALENKO[3]

[1]Research and Production Corporation Radiy, 29 Academician Tamm str., Kropyvnytskiy, 25009, Ukraine

[2]Radics LLC, 29 Academician Tamm str., Kropyvnytskiy, 25009, Ukraine

[3]Centre for Safety Infrastructure-Oriented Research and Analysis, 37 Astronomicheskaya str., Kharkiv, 61085, Ukraine

The paper describes possible scope and conditions for implementation of Nuclear Power Plant's (NPP's) Instrumentation and Control (I&C) systems modernization, and discusses approaches for modernization projects. It also proposes comprehensive description of the strategy for modernization projects implementation, describes different project roles, including designers, project managers, plant managers, operators, maintainers, and regulators.

Integrated Human and Organizational Factors-oriented Design and Development Method

Nicolas HENRY, Renaud AUBIN

EDF/R&D PRISME Dept., I&C Group, Chatou, 78401, France

Safety-related industries have been interested in Human and Organizational Factors (HOF) for a long time. HOF integration in design and development projects is now widely recognized as a key factor in the success of such projects in delivering plants or systems that can be operated safely and efficiently. However, even the most comprehensive of HOF design approaches are based on the hypothesis that the design, development and evaluation phases of a project are sequential and focus their efforts on the design and evaluation phases.

Within the same period, the software development industry has seen the emergence of agile software development methods. Their goal is to produce software that meets the users' requirements without a very precise or comprehensive knowledge of these requirements at the beginning of the project. They promote an iterative, incremental and evolutionary approach and make operational proposals about project management and development team organization and practices so as to be able to integrate new or modified users' requirements during the project lifecycle.

The study presented in this paper is based on the hypothesis that these two approaches are complementary and can be combined to propose an integrated HOF.oriented design and development method. It includes a theoretical analysis of both approaches and feedback from several development projects led by EDF Research & Development using this innovative approach.

Complexity Analysis of an FPGA-Based ESF-CCS

Joyce MAYAKA, Jae Cheon JUNG

Department of NPP Engineering, KEPCO International Nuclear Graduate School, Ulsan, Korea

In the recent past, there has been a lot of research into the use of Field Programmable Gate Arrays (FPGA) in the safety systems of Nuclear Power Plants. This interest is driven by, among other factors, the search for a platform that is relatively easy to verify, validate, and ultimately license. Current Instrumentation and Control (I&C) Systems are based on Programmable Logic Controllers (PLC), which due to their software structure, have inherent complexity. Complexity results in difficult and costly verification and validation (V&V), safety justification and system maintenance. The difficulty experienced in V&V is due to the possibility of latent errors in the software, as well as the inability to separate support functions from the primary control functions. The Engineered Safety Features Component Control System (ESF-CCS) serves to actuate the safety components in the event of an accident. The primary purpose of the ESF-CCS is to prevent the release of radioactive material, as well as to prevent damage to the core. This study proposes a design for the ESF-CCS based on the FPGA architecture and is aimed at a reduction in system complexity. The reduction in complexity is achieved by the use of flat hardware logic and the separation of logically independent functions. Following this, complexity analysis of the developed system is carried out in order to facilitate the comparison of the FPGA-based system to the PLC-based system.

Monday, November 27, 2017 14:00-15:40

9. Future I&C technologies for Nuclear Applications I

Room #202
Session Chair: Bruce P. HALLBERT (INL), Hyun Gook KANG (RPI)

IAEA Activities in the field of Instrumentation and Control Engineering

Janos EILER

International Atomic Energy Agency (IAEA)

Complementing Renewable Energy Production with Small Modular Reactors

Richard BISSON[1], Jamie COBLE[1], Kevin TOMSOVIC[2]

[1]*Department of Nuclear Engineering, University of Tennessee, 1004 Estabrook Dr, Knoxville, TN, 37996, USA*

[2]*Department of Electrical Engineering and Computer Science, University of Tennessee, 1520 Middle Dr, Knoxville, TN, 37996, USA*

As inherently intermittent sources of renewable energy (such as wind farms) more fully penetrate the energy grid, peaking power is largely being supplied by carbon-emitting natural gas turbines. These gas turbines are favored due to their fast response from shutdown to full power. However, significant greenhouse gas emissions could be avoided if these plants were replaced with carbon-neutral nuclear facilities to provide peaking power to complement renewable generation and meet overall power demand. There is a great deal of previous work regarding reactor power shaping with control rod movement for both currently operating nuclear power plants and proposed plants, but the literature on load-following to meet less predictable, more rapidly varying power demand is less comprehensive. The Westinghouse International Reactor Innovative and Secure (IRIS) small modular reactor (SMR) is used as a candidate reactor design for modeling, simulation, and control studies. The nodal IRIS model includes the primary system and steam generator; simple assumptions and correlation models are currently used for the balance of plant. Nuclear energy generation is described by the point reactor kinetics equations with six neutron precursor groups; currently, only temperature-based reactivity feedback terms are included, but power-based effects (e.g., xenon buildup) are being integrated. The control scheme for the power peaking operation of the IRIS iPWR model would ultimately lead to the development of real operational mechanisms and principles in a grid with significant renewables capacity. The 350 MWe IRIS reactor is coupled with a roughly 100 MWe-capacity wind farm to evaluate the capability of the IRIS reactor to respond to quickly fluctuating power demand to provide power peaking and reserve power. The results of grid simulations show that fast response is possible, but system output is persistently lower than grid demand. New control strategies, including a supervisory control scheme, are being developed to improve plant response.

Classification of Abnormal Conditions: A Data-driven Aid for the Selection of Abnormal Operating Procedures in NPPs

Ibrahim AHMED, Sanghwa LEE, Gyunyoung HEO

Nuclear Engineering Department, Kyung Hee University, 1732 Deogyeong-daero, Giheung-gu, Yongin-si, Gyeonggi-do 17104, Korea

Abnormal Operating Procedures (AOPs) are provided in nuclear power plants (NPPs), which are procedures that specify operator actions to restore normal operating conditions following a transient or abnormalities. The selection of the appropriate AOPs is decided by the combination of multiple alarms which need to be analyzed by the NPP operator to identify the actual abnormality that occurred. However, multiple alarms from multiple systems often occur at the same time during an incident, making it difficult for the operator to select a correct response efficiently in a time-critical situation. Furthermore, quick system recovery to normal condition before the trip is important since any delay may result to a condition that will degenerate to the emergency situation thereby leading to the unwanted shutdown of the plant. Therefore, considering the fact that plant recovery to normal condition before reactor trip occurs is paramount and the operator's state of condition in time-critical situations in case of multiple alarms analysis, the benefits of an aid to assist operator in knowing the actual plant condition in time thereby selecting the appropriate response procedures cannot be overemphasized. In addition, owing to the fact that operator depends only on the alarm systems in selecting AOPs, the symptoms from process parameters are also significant in selecting proper AOPs. In this regards, we proposed a data-driven based pattern detection and classification method that concatenates the symptoms from the process sensors (analog signals) and the alarm signals (digital signals) together for effective identification and classification of abnormities in the event of abnormal situations in NPP. The proposed method is validated using simulation analog data from the Multi-dimensional Analysis for Reactor Safety (MARS-KS) simulator and artificially generated alarm digital data for the case of abnormalities concerning steam generator tube in NPP. The proposed method employing four classification models, Linear Discriminant Analysis (LDA), Classification and Regression Tree (CART), Support Vector Machine (SVM), and Random Forest (RF), are trained, and their performance are evaluated on the test set. The proposed method utilizing RF model performed excellently with 100% performance on the test set over the proposed method utilizing other classification models. The excellent results obtained from the case study suggest that the proposed model is a promising approach for aiding the selection of AOPs in the event of abnormal conditions and minimizing the operators' burden in identifying the actual plant status during abnormal situations.

Application of Multi-objective Particle Swarm Optimization in Condenser Control System Parameters Tuning

Yiliang LI, Zhi CHEN

Nuclear Power Design and Research Sub-institute, Nuclear Power Institute of China, China

The control system of condenser includes three mutually coupled subsystems: PID control system of water level, pressure, and condensate sub-cooling degree. For the situation of complex coupling among systems during PID parameter tuning, a multi-objective particle swarm optimization (MOPSO) based on Pareto optimality is applied. This method provides a way to tune PID parameters of every coupled control system without decoupling, which can avoid uncertainty and influence of coupling. This method takes dynamic performances as the optimization objectives, and selects MOPSO to obtain Pareto-optimal solution set. The MOPSO combines particle swarm optimization (PSO) with Pareto dominance relationship for a multi-objective optimization. The algorithm of MOPSO adopts a comprehensive learning mechanism to update optimal set, which improves precision and diversity of the solutions [1]. The simulation result proves that the MOPSO can tune PID parameters of coupled control systems better than classical engineering tuning method.

Fiber Optic Cable for NPP Harsh Environment

Zuzana KONEČNÁ[2], Vít PLAČEK[1], Petr HAVRÁNEK[1]

[1]ÚJV Řež, a. s., Hlavní 130, Řež, Husinec, Czech Republic

[2]Dept. of Electrotechnology, Czech Technical University in Prague, Technická 2, Praha, Czech Republic

The good experience with optical communication and its advantages in industrial applications have lead to introduction of optical communication technology into more conservative areas, such as the nuclear and aerospace industries. The number of optical cables in nuclear power plants has been increasing. Fiber optic cables are commonly used at nuclear power plants in I&C systems but they are usually used in mild environments. For example, the Czech nuclear power plant in Dukovany, a PWR type with four 440 MW units, has 1330 installed optical cables. All the optic cables are situated in mild environments, i.e. without radiation. Nevertheless, currently, the number of applications in harsh environments with radiation is increasing. Application of safety related fiber optic cables in harsh environments of nuclear power plants (NPPs) requires their qualification. The international standard for qualification IEEE 1682: Standard for Qualifying Fiber Optic Cables, Connections, and Optical Fiber Splices for use in Safety Systems in Nuclear Power Generating Stations – has existed since 2011.

One of the most prevalent effects of radiation exposure is an increase of signal attenuation (signal loss). This is a result of fiber darkening due to radiation exposure and it plays a very important role in application of fiber optics in radiation environment. After the irradiation the fiber optics goes through a "recovery" process during which the optical properties improve again. The extent of recovery can be affected among others by time and temperature after the radiation aging. Tested simplex and breakout cables with MM fibers increase their attenuation during irradiation. Nevertheless, during next 30 days of recovery process after irradiation, the attenuation decreased as expected. Quite different was the situation for jelly filled loose tube SM cable. Attenuation decreased (recovered) 3 days after the irradiation. After this short period, the trend changed and attenuation increased to a value well above the attenuation just after the irradiation, see red line in Figure. Well above the acceptance criterion for NPP application. A lot of experiments were carried out to explain such an unexpected behavior. Different samples were irradiated and measured the recovery process; e.g. only SM fibers with different dopants, tubes filled with jelly as well as dry loose tubes (without jelly), cables made from the tested tubes and different tubes material. The presentation will describe all the experiments and bring explanation for this unexpected property.

Monday, November 27, 2017 16:00-17:40

10. Future I&C technologies for Nuclear Applications Ⅱ

Room #202
Session Chair: Janos EILER (IAEA), Chang-hwoi KIM (KAERI)

Determination of Allowable Setpoint for Safety Instrument in Consideration of Uncertainty and Confidence Level

Sanghoon BAE[1], and Young-ki, KIM[2], Chang-hwoi, KIM[1]

[1]*Division of Nuclear ICT research, Korea Atomic Energy Research Institute, 111, Daedeok-Daero, Yuseong-Gu, Daejeon, 34057, Republic of Korea*

[2]*Division of Research Reactor System Design, Korea Atomic Energy Research Institute, 111, Daedeok-Daero, Yuseong-Gu, Daejeon, 34057, Republic of Korea*

This paper is intended to determine a more practical allowable trip setpoint for safety I&C channels in research reactors considering not only measurement uncertainty but also the confidence level. For the safety channel, with an instrument loop, it is mandatory to have the allowable setpoint in accordance with the relevant standard in order to take swift safety actions such as a reactor shutdown. At this time, uncertainty for the instrument loop should be identified in a proper manner considering many conditions. In order to have a reliable setpoint, first we should figure out the detailed conditions of every sample test with the number of tests to find the coverage factor. Second, uncertainty of each component should be defined except for terms related to the overall error, and uncertainty of the total loop should be calculated with an adequate margin. In this paper, the two step process in determination of the setpoint for the safety channels is highlighted.

Analysis of MEMS Based Earthquake Instrument for Nuclear Power Plant

Md. Mehedi HASAN, Jae Cheon JUNG

Department of NPP, KEPCO International Nuclear Graduate School (KINGS), 658-91 Haemajj-ro, Seosaeng-myeon, Ulju-gun, Ulsan 689-882, Republic of Korea

This work explores an analysis of Micro Electro Mechanical System (MEMS) earthquake instrument which can be used for seismic monitoring in the nuclear power plant. Conventional geophone based seismic instruments suffer significant reduction in recorded velocity-domain amplitudes below their natural frequency. MEMS accelerometers can record to fractions of 1 Hz without any relative reduction in acceleration amplitudes. Firstly, mathematical transfer function models of conventional geophone, MEMS, servo acceleration with feedback loop, and force balanced acceleration using pendulum type accelerometer are elicited. Then the dynamic behaviors of those instruments are assessed due to the effects of transfer function on frequency contents using MATLAB program. MEMS based digital accelerometers provide a broadband linear response (DC to 800 Hz) and very low distortion. A new MEMS based earthquake instrumentation consisting FPGA data processing system is conceptually designed, and synthesized. Ground motion information recorded by the seismic measuring instrument can be observed promptly after an earthquake as Operating Basis Earthquake (OBE), and Safe Shutdown Earthquake (SSE) levels at nuclear power plant to avoid seismic issues. The system can ensure more availability of the plant confirming integrity of the structure, system, and components.

VHDL Verification of FPGA based ESF-CCS for Nuclear Power Plant I&C System

Restu MAERANI[1], Jae Cheon JUNG[2]

[1]Department of NPP Engineering, KINGS, Ulsan, 45014, Indonesia

[2]Department of NPP Engineering, KINGS, Ulsan, 45014, Republic of Korea

Verification becomes the focus of activities during the integration phase of design life cycle in the development of the system. Verification methods that will not take much cost and time should be properly selected, accordance with the Measurement of Effectiveness (MOEs) need. Verification is one phase that must be done after completing the implementation process. Since Instrumentation & Control (I&C) system has a role as a very crucial to the control protection system in Nuclear Power Plant (NPP), then software verification is very essential and shall to be achieved for safety critical issue in system level. According to IEEE 1076-2008 standard, VHDL is a language that is easy to read by machines and humans; and make it easier for process development, verification, synthesis and testing for hardware reliability in the design. Because this design uses VHDL code for Field Programmable Gate Array (FPGA) based Engineered Safety features – Component Control System (ESF-CCS) and by referring to the NUREG/CR-7006 during VHDL verification on behavioral simulation process, it should be equivalent with the post layout simulation. Furthermore, Vivado will be used as the VHDL verifier, where the VHDL code itself is created, in order to simplify the process of verification with this design life cycle phase on re-engineering process. By using this methodology, the testing process will automatically can be represented as one of verification process from several software verification methods that can be developed.

Case For The Adoption Of FPGA Technology In The Implementation And Replacement Of Equipment And Systems In Nuclear Power Plants

Mark Joseph BURZYNSKI

SunPort SA, Avenue de l'Université 24, CH-1005 Lausanne, Switzerland

Approximately 40 percent of the world's 450 nuclear reactors have some digital I&C upgrades of safety related systems. The 60 reactors currently under construction will incorporate some form of digital instrumentation and control (I&C) technology. The paper recognizes that analog instrumentation is obsolete, equipment and parts are difficult to obtain in the market, are expensive and not necessarily more reliable than digital.

Main goals of regulators and industry are to achieve safe, reliable and cost-efficient operation of reactor fleets. Use of digital technology is essential to achieve improved equipment and plant reliability. Field programmable gate array (FPGA) digital technology also provides ways to minimize impact of obsolescence and reduce plant operating costs. Digital technology allows designers to create very reliable applications thus reducing initiating events and minimizing spurious reactor scrams/trips at a significantly lower cost than analog based applications.

The various concerns associated with the use of digital technology in nuclear power plants are identified. These concerns include difficulties in quantifying software reliability, lack of universally accepted methodologies to address reliability, cyber security issues, and the perception that digital I&C systems are more prone to adverse effects from common cause failures.

The case is made for the adoption of FPGA technology. Solutions to the concerns expressed by industry and regulators are identified and organized for each objective (i.e., reliability improvement, obsolescence management, and cost reduction). The benefits that FPGA technology provide to improve the solutions are also identified. Additional questions worth asking when pursuing FPGA solutions for nuclear power plant projects are also identified for further consideration.

Tuesday, November 28, 2017 10:00-12:00

11. Cyber Security I

Room #202
Session Chair: Eric LEMOINE (CNSC), Chul Hwan JUNG (CNSC)

Using Virtual and Augmented Reality to Improve Cyber Security and Physical Protection of Nuclear Material and Nuclear Facilities

Scott GODWIN, Samuel CLEMENTS, Doug MACDONALD, Nick CRAMER, Rick REINSCHE, Shawn VANDYKE

Pacific Northwest National Laboratory

For decades physical protection specialists have performed vulnerability assessments to identify attack vectors that attackers can use to either sabotage nuclear facilities or steal nuclear material. They then build protection systems to defend potential pathways. Over the years these systems have become more interconnected for better detection, assessment, and response but in so doing have developed a reliance on digital connectivity that may be vulnerable to cyber-attack. It is difficult to quantify the level of risk associated with these integrated digital systems against a blended attack. Physical enabled cyber-attacks or cyber enabled physical attacks take advantage of weaknesses inherent in both the cyber and physical domains to diminish defenses, and could provide attack paths that are not appropriately protected. This paper discusses how the use of virtual and augmented reality systems can aid in understanding the blended attack and help develop robust protection mechanisms against the advanced cyber/physical adversary. Virtual and augmented reality complement traditional tabletop analysis and software based modeling and simulation tools in use today.

This paper addresses the virtual and augmented reality work being conducted and expands upon the work done for the IAEA Computer Security Conference in June 2015. Currently, the system effectiveness of cyber-physical protection systems is evaluated and measured using table top exercises, Monte Carlo simulations, and uniquely developed vulnerability assessment software packages. The results are interpreted by subject matter experts and implemented by highly trained security practitioners. Decision makers and other stakeholders must trust in the experts' judgement, but may not truly understand all of the intricacies or the justifications for the recommended changes. The training techniques used to show each of these individuals the interdependencies of today's modern interconnected systems, at the level they need for understanding, varies widely and has been met with varying degrees of success. These techniques range from classic classroom instruction, hands-on performance based repetition, and high-level briefings to show the "art of the possible". It's been proven that there is a bigger impact on the security professionals when the material is made relevant to their situation, and they can experience it firsthand. A recent attempt at modernizing the delivery method for cyber-physical awareness training involved a representative model of a nuclear power plant in which a guide walked a user through a blended physical and cyber security scenario. The model was tremendously successful with thousands viewing it and hundreds of conversations on the challenges of today's interconnected cyber and physical security systems. The model was subsequently used to brief multiple ambassadors, displayed at additional conferences and used in multiple training exercises. One drawback of this diorama is its size and weight. It is fragile, cumbersome, time consuming and expensive to ship the model to each venue, and valuable opportunities have been lost as a result. Additionally the model was built around one possible cyber-physical attack scenario and can only be used for that specific pathway.

Virtual and augmented reality can provide many of the benefits of the physical model, and can take awareness and instructional training to the next logical step. It is immersive, realistic, and lets the participants truly experience the scenario they are being shown. It engages multiple senses and helps users experience the cyber-physical interplay in much more depth and realism than classroom training, formal presentations, or table top analyses ever could. Additionally virtual and augmented reality systems are compact enough that they are man portable and can be checked as luggage. Multiple systems can be transported at a fraction of the cost of the physical model. Additionally, the scenarios can be easily tweaked and modified to meet the end user's needs. This includes tailoring the granularity of the training to the expected technical level of the audience, and highlighting specific challenges they may be facing. The use of virtual and augmented reality can provide enhanced and customized training and awareness that is not achievable using today's standard training methods.

This paper will compare and contrast traditional classroom and performance based training, tabletop analyses, software modeling and simulation, and static physical models against virtual and augmented reality and highlight lessons learned from research and development efforts to build training and awareness exercises in virtual worlds.

Logging and Monitoring Parameters for Cyber Security Events of Digital I&C

Jae-Gu SONG[1], Jung-Woon LEE[1], JunYoung SON[1] , Cheol-Kwon LEE[1], Paul SMITH[2]

[1]Nuclear ICT Research Division, Korea Atomic Energy Research Institute, Daejeon, Korea

[2]Center for Digital Safety and Security, AIT Austrian Institute of Technology, Vienna, Austria

Instrumentation and control (I&C) systems in nuclear power plants have been digitalized for the purpose of maintenance and precise operation. This digitalization, however, brings out issues related to cyber security [1, 2].

In order to keep the system from cyber adversary activities, it is necessary to identify parameters related to cyber incidents in digital I&C systems. Cyber incident detection is the process of monitoring the events occurring in a digital system or network and analyzing them for signs of possible incidents [3]. Hence, parameters related to cyber incidents are crucial for the detection of possible cyber incidents.

However, it is more difficult to detect and analyze security related incidents in the digital I&C systems than in the general IT systems. In general, I&C systems have various diagnostic information necessary for maintaining a safe status, but the information has not been selected in consideration of the identification of cyber-attacks.

In this paper, we describe the major considerations for making a list of candidate logging and monitoring parameters that can be used for cyber security diagnosis of the digital I&C systems. The candidate logging and monitoring parameters could be classified in more detail through an activity of mapping cyber-attack techniques and scenarios to the parameters.

Trustworthy Computer Security Incident Response for Nuclear Facilities

Mislav FINDRIK[1], Ivo FRIEDBERG[1], Ewa PIATKOWSKA[1], Paul SMITH[1], Jae-Gu SONG[2]

[1]Center for Digital Safety and Security, AIT Austrian Institute of Technology, Donau-City-Straße 1, Vienna 1220, Austria

[2]I&C Human Factors Research Division, Korea Atomic Energy Research Institute, Daejeon, Korea

New forms of advanced computer security threat are targeting critical infrastructures, including nuclear facilities. These threats use sophisticated and stealthy methods to target a specific infrastructure, with the aim of causing operational consequences. For nuclear facilities, this could involve compromising Instrumentation and Control (I&C) systems that underpin nuclear security and safety functions. In this context, effective and rapid incident response is necessary to mitigate and contain the potential effects of a cyber-attack. Incident response includes a detection and analysis phase, wherein incidents are identified, and their effects are understood. This phase involves reasoning about the state of systems, i.e., whether they are compromised or not, based on potentially unreliable sources of information. In this paper, we motivate and present a high-level architecture to support reasoning for incident response, based on unreliable detection capabilities. With an example nuclear-relevant scenario, we indicate how its reasoning component can be realized with the use of Evidential Networks – a graph structure that represents knowledge about a target domain, and supports inference using unreliable information sources.

Development of a Quantitative Method for Evaluating Security Controls Based on Intrusion Tolerant Concept: Consideration of Adverse Effects

Chanyoung LEE, Poong Hyun SEONG

Department of Nuclear and Quantum Engineering, KAIST, Daejeon, Korea

The introduction of digital techniques has brought up a new issue of cyber security; concerns are continuously growing in the nuclear industry. Considering that security techniques to be applied were not considered together when I&C systems were developed, it is necessary to analyze not only security enhancement, but also the influence of applied security techniques on the target system. In this study, a quantitative method for performing such evaluating is developed. The method can provide information including the degree to which security level can be improved and how the reliability of existing systems will be affected. In addition, a two-dimensional index called the *Hybrid Security Index* (HSI), which includes an *Incompatibility Index* and a *Performance Index* is developed. Validity of the suggested method was proven by conducting a case study.

Three-level Deep Packet Inspection for I&C systems of NPPs

Jianghai LI[1], Xionghua SHENG[1], Shuanglin JIANG[2], Xiaojin HUANG[1]

[1]Institute of Nuclear and New Energy Technology, Collaborative Innovation Centre of Advanced Nuclear Energy Technology, Key Laboratory of Advanced Reactor Engineering and Safety of Ministry of Education, Tsinghua University, Beijing, China

More and more instrumentation and control (I&C) systems of nuclear power plants (NPPs) employ the common Internet protocol TCP/IP or its variants as their networking protocols. To monitor traffic flow of the I&C network for cyber security, an intrusion detection system (IDS) is necessary. The traditional IDS for information technology (IT) systems merely checks the header of the IP packet which contains the source/destination IP address and the port number. It is effective against the intruders who use only the internet security techniques. However, to detect intrusions by the intruders who are familiar with industrial control protocols such as Modbus, S7, the traditional IDS is not enough. Because it cannot detect malicious code in the data portion of IP packets which uses control protocols to launch an attack. The deeper packet inspection (DPI) on the data portion of IP packets is required to detect such more advanced attacks.

We category intruders into three classes based on the information and techniques that they can use. They are IT hackers, I&C hackers, and NPP hackers. Different hackers are able to make different kinds of malicious data packets on I&C network. In order to detect attempts of the three kinds of intruders, a three-level deep packet inspection method is proposed. The information on the specific control protocol of I&C systems and on equipment and processes of NPP is employed to implement the three-level DPI.

To verify the three-level DPI method, a demonstration security box is built which adopt the programmable logic component (PLC) Siemens S7-200 as its control system. The typical intrusions by the three classes of intruders are realized on attack server of the security box. The three-level DPI method is implemented in the detection module of the security box.

A Graded Approach for Cyber Security Evaluation of Nuclear I&C System with Bayesian Update

Jinsoo SHIN[1], Gyunyoung HEO[1], Hanseong SON[2]

[1]*Department of Nuclear Engineering, Kyung Hee University, Yongin-si, 17104, South Korea*

[2]*Department of Game Engineering, Joongbu University, Geumsan-gun, 32713, South Korea*

Since digital technology was introduced for instrumentation and control (I&C) system in nuclear power plants (NPPs), cyber security has become one of the safety issues. Actually, the Davis Besse nuclear power plant in Ohio was infected by the SQL Slammer worm in January 2003 and nuclear facilities in Iran have been targeted by cyber-attacks, including the one known as "stuxnet" in 2010. In this regard, the regulatory agencies have published the guides or standards. These regulatory guides include enormous range for cyber security in NPPs. For this reason, it is difficult to determine whether the users such as licensee, security manager, and regulator, truly and consistently comply with the regulatory guides for cyber security. In order to overcome these problems, in this study, we proposed the cyber security evaluation methodology and develop the cyber security evaluation model with Bayesian belief networks (BBN) to help users to apply the regulatory guide for cyber security. The cyber security evaluation model consists of the architecture model and activity-quality model. The architecture model is made up of I&C architecture, malicious activity, and mitigation measure. Cyber-attack is initiated by an attacker performing some malicious activities on the target. This is accomplished by the malicious activities that penetrate all of the mitigation measures existing in the target of attack and assist the attacker to get what they want from the attack target. In addition to the factors such as I&C architecture, malicious activity, and mitigation measure, the check-list of the regulatory guide, for instance, cyber security technical standard, was added as an activity-quality model. The architecture model and activity-quality model for cyber security in NPPs are integrated into a single cyber security evaluation model with BBN and this model can be evaluated quantitatively in terms of the degree of cyber security. BBN can facilitate to model a complex system such as the I&C system under cyber-attack in NPPs. In addition, posterior information can be obtained through back propagation through Bayesian update with the prior information of the model, so that it is possible to perform various scenario analyses. The goals of this study are as follows: 1) to propose a cyber security evaluation methodology that reflects the cyber security regulatory standards and I&C architecture for NPPs, 2) to develop a cyber security evaluation model that can be quantitatively evaluated by applying the proposed methodology, and 3) to conduct the case studies on cyber security evaluation of NPPs using the developed model.

Tuesday, November 28, 2017 14:20-16:00

12. Cyber Security Ⅱ

Room #202
Session Chair: Scott GODWIN (PNNL), Gyunyoung HEO (Kyung Hee University)

Development of A Prototype FPGA based Security Module to Control Data Communication Network Access

Mohamed Abdallah EL-AKRAT, Jae Cheon JUNG

Department Of NPP Engineering, KEPCO International Nuclear Graduate School, Ulsan, Korea

In this paper, field programmable gate array (FPGA) based Security Module is presented, this module run an application of encryption using hardware. Using FPGA technology in industrial systems, especially in nuclear power plants, strongly present in I&C application, which considered as state-of-the-art in NPP I&C systems; although there are several challenging problems in the area of cybersecurity assurance for complex FPGA-based I&C systems consideration of all the possible vulnerabilities. In this work, a prototype FPGA based security module is developed to control network security for mitigating man in the middle (MITM) attacks, using commercial FPGA to verify and validate the designed hardwired security module to assure data confidentiality, and integrity of data communication system in APR 1400 nuclear power plant, model-based system engineering approach is applied to analysis system requirements, enhanced function flow block diagram (EFFBD) is created and simulated by using CORE9 university edition to compare between the current system and the developed module , HDL code is developed using ALDEC Design Suite as a programming tools and to run System synthesis and implementation for performance simulation and design .

Cyber Informed Engineering

Robert Stephen ANDERSON

Idaho National Laboratory, Idaho Falls, ID, US

Securing nuclear Instrumentation and Control (I&C) in an ever-changing cyber threat landscape requires more than a dedicated team of cybersecurity professionals. Traditional static defense mechanisms like airgaps and reliance on obscure protocols and access mechanisms are not sufficient for in-depth defense in an always-connected, information-rich cyber environment. Though technical solutions exist to protect availability, integrity and confidentiality of industrial systems, these solutions typically secure external system boundaries and not the underlying digital system's engineering design. Training engineering personnel in cybersecurity or training information technology specialists in engineering is expensive and often ineffective at addressing systemic vulnerabilities in large and complex digital systems.

INL has developed a framework for bridging the gap between engineering design and cybersecurity to identify cyber vulnerabilities at the earliest stages in the system development life cycle and apply both engineering solutions and information technology to minimize the cyber-attack surface across the entire system engineering process. This methodology focuses on aiding engineering staff who traditionally envision, plan, design, implement, operate, and maintain such systems to understand cyber risk (without becoming cyber experts), and to integrate the subject matter expertise of cybersecurity specialists.

In this presentation, INL will present principles of the Cyber Informed Engineering (CIE) methodology and describe how they can be implemented and integrated. A full technical report on Cyber Informed Engineering, including an application aid and an assessment aid will be made available to all interested.

Risk Assessment of Operator Errors Induced by Cyber-Attacks on Nuclear Power Plants

Jong Woo PARK, Seung Jun LEE

Department of Nuclear Engineering, UNIST, 50, UNIST-gil, Ulsan, 44919, Korea

A cyber-attack has emerged one of the new dangerous threats as the analog instrument and controls (I&Cs) has been replaced to digital I&Cs of nuclear power plants (NPPs). Since an NPP is one of the safety critical infrastructures, a cyber-attack on safety or non-safety systems can cause a serious consequence by initiating dangerous events and causing failures of important mitigating systems. However, it is hard to predict cyber-attacks on NPPs, because they are conducted with intention and there are too many possible cases. To develop a cyber security strategy, it is required to analyze the risk of a cyber-attack, but there is no evaluation method or model for assessing the risk of cyber-attacks on an NPP. In this paper, a risk assessment method for cyber-attacks was proposed based on probabilistic safety assessment (PSA) which is the most widely used for risk assessment in nuclear field. Of lots of possible attack scenarios, this work is focused on the risk of human errors induced by cyber-attack.

Usually only error of omission (EOO) is considered in a PSA model, however more various types of human errors could be caused due to cyber-attacks by injecting wrong information and blocking information. In the cyber-attack risk assessment, not only EOO but also error of commission (EOC) should be considered. In case of TMI-2 accident, the human operators turned off the safety injection system due to the wrong information of safety depressurization valve. This kind of situation could be intentionally made by cyber-attacks. To develop a PSA model for cyber-attacks, new basic events for EOCs which have serious impacts on the plant safety were identified and the risk by them were quantitatively evaluated. For case studies, potential cyber-attack scenarios to cause EOOs and EOCs were evaluated using the developed model.

Implementation of Cyber Security for Computerized Operator Support System of Nuclear Facilities

Wei ZHENG , Lei MAO

Shanghai Nuclear Engineering Research and Design Institute, China

Computerized operator support systems are deployed within Plant Monitor and Control Level, include computer-based procedure system, alarm management system and nuclear performance calculation system, and provide decision support for operators during all plant conditions, including normal condition and accident condition. Cyber security attack will adversely affect the reliability, availability, confidentiality and integrity of computerized operator support system, leading to deny access to system, services or data, and adversely impact the operation of systems, networks, and associated equipment. So computerized operator support system should be adequately protected against cyber attacks.

The paper will describe a software security development Life Cycle method while enhancing verification and validation (V&V) during development and incorporating the cyber security considerations into the software development process from design phase to test phase. The life cycle of cyber security plan and activities of implementation of cyber security for computerized operator support system of nuclear facilities, including concepts phase, requirements analysis phase, design phase, implementation phase and test phase are explained in detail. Requirements development and V&V process concerning cyber security methods including account management, access control, session management, authorization and password management, log and audit, communication security, hardware configuration, software and service and fault tolerance mechanism are implemented in computerized operator support system. Cyber security test for computerized operator support system will also be described in this paper, including black box testing, penetration testing and vulnerability scan and etc. It also explains experiences resolving security vulnerabilities of the system and summarizes considerations and experiences in the development process.

Identification of Critical Digital Assets for Nuclear Instrumentation System in Research Reactor

Sungmoon JOO[1], Sang Mun SEO[2], Yong–Suk SUH[2]

[1]*Decommissioning Technology Research Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero 989Beon-gil, Daejeon 34057, Korea*

[2]*Research Reactor System Design Division, Korea Atomic Energy Research Institute, 111 Daedeok-daero 989Beon-gil, Daejeon 34057, Korea*

Identification of critical digital assets is the first step for cyber security in nuclear facilities. Among many systems in nuclear reactor instrumentation and control system, nuclear instrumentation system is unique in the sense that it can be considered as a single system or a collection of several sub-systems. This paper presents a preliminary result of the identification of critical digital assets in an open-pool type research reactor. Keyword: Cyber Security, Critical Digital Assets, Nuclear Instrumentation System, Research Reactor.

Tuesday, November 28, 2017 16:20-18:20

13. Cyber Security Ⅲ

Room #202
Session Chair: Robert Stephen ANDERSON (INL), Cheol-Kwon LEE (KAERI)

Status of Canadian Cyber Security Regulatory Framework and Implementation at Nuclear Facilities

Chul Hwan JUNG, Eric LEMOINE

Systems Engineering Division, Canadian Nuclear Safety Commission, Ottawa, Canada

Cyber security has been an emerging issue for key stakeholders of Nuclear Power Plants (NPPs), including design authorities, operating utilities and regulatory bodies for the continued safe and secure operation of NPPs and other nuclear facilities.

The Canadian Nuclear Safety Commission (CNSC) is the nuclear regulatory authority that has the sole responsibility of oversight of cyber security in the Canadian nuclear industry. The CNSC developed cyber security regulatory requirements for Design of NPPs in 2014. The requirements can be referred to in CNSC REGDOC-2.5.2. Additionally in 2015, the Canadian Standards Association (CSA) issued the CSA Standard N290.7-14 "Cyber Security for Nuclear Power Plants and Small Reactor Facilities." The CNSC has referred this standard in its regulation of cyber security for operating NPPs and small reactor facilities.

This presentation provides the overall status of the Canadian regulatory framework for cyber security, the status of licensee's implementation efforts for compliance with N290.7-14, and the roadmap development activities by the CNSC for cyber security at non-NPP facilities will also be discussed.

Cyber Security Assessment Methodology of Critical Digital Asset in Nuclear Power Plant

Ieck Chae EUOM, Sung Cheol KIM, Joo Hyoung LEE

Security Consulting Team, KEPCO KDN, Naju, Korea

Nuclear Power Plant Operators have approached the problem of cyber security by simply attempting to apply nation's committed catalog of cyber security requirements to every Critical Digital Asset under evaluation, which can number into the hundreds. This current approach does not provide guidance on how to assess a given requirement with a security method that effectively takes Critical Digital Asset. This paper analyzes Cyber Security Assessment Methodology about Industrial Control Systems. And then give an efficient methodology. It approaches the Regulations of KINAC/RS-015 from a technical vulnerability point of view, where any given Critical Digital Asset can be assessed for vulnerabilities

Security Management of Virtualised Supervisory I&C Systems in Nuclear Facilities

Hewes, M., Howarth, N., Hunt, C., Noonan, A.

Australian Nuclear Science and Technology Organisation (ANSTO), Lucas Heights, Australia

We review the functionality provided by virtualisation platforms, available to Nuclear I&C operators from a cyber security management perspective.

Wednesday, November 29, 2017 10:00-12:00

14. Safety Critical Software Development and Qualification

Room #202
Session Chair: Ian JUNG (USNRC), Man Cheol KIM (Chung-Ang University)

FBDScenaGen+: GA-based High-Quality Scenario Generator for FBD Simulation

Eui-Sub KIM[1], Sejin JUNG[1], Junbeom YOO[1], Young Jun LEE[2] , Jang Soo LEE[2]

[1]*Computer Science and Engineering, Konkuk University, Seoul, Korea*

[2]*I&C.HF Division, KAERI, Daejeon, Korea*

Simulation plays an important role in verifying the functionality of FBD programs. When we develop a safety-level PLC-based digital I&Cs in NPP, the simulation quality is important to demonstrate functionally correct/safe. The simulation quality is derived from the scenario quality based on test coverages such as structural code coverage and fault coverage. This paper introduce a tool, FBDScenaGen+, which generates a set of high-quality scenarios based on FBD structural code coverage. We apply genetic algorithm (GA) with the tool in order to increase the quality. The case study will show a feasibility and effectiveness of the proposed method and tools.

A Framework for the Safety Assurance of Safety Software in Nuclear Power Plants

Kee-Choon KWON[1], Jang-Soo LEE[1] , Eunkyoung JEE[2]

[1]*Nuclear ICT Research Div., Korea Atomic Energy Research Institute, Daejeon, Rep. of Korea*

[2]*School of Computing, Korea Advanced Institute of Science and Technology, Daejeon, Rep. of Korea*

When developing and conducting verification and validation of the safety software in nuclear power plants to receive a license from the regulatory body, it is difficult to judge the safety and dependability of the development, implementation, and validation activities by simply reading and reviewing the documentation. A systematic evaluation technique is necessary to determine whether particular software safety assurance activity defects are at acceptable levels. In this study, we apply a safety case methodology to assess the level and depth of the results of the development and validation performed by the manufacturer to target a bi-stable processor of a digital reactor protection system, and analyze the evaluation results. Also, we assess the hazard analysis techniques to measure the applicability and compare them according to the software development life cycle. We proposed a new framework by applying a modified hazard analysis, including a safety case methodology. We confirmed that it is possible to effectively supplement the existing safety demonstration method.

One-Step--Logic Automatic Translation For FPGA Applications

Allen HSU, Steve YANG

Doosan HF Controls Corp., 1624 W. Crosby Road, #124, Carrollton, TX 75006, USA

Field Programmable Gate Arrays (FPGAs), as programmable logic devices (PLDs) have gained interests for implementing safety Instrumentation and Control (I&C) applications in nuclear power plants (NPPs) owing to the FPGAs' potential advantage over the currently more common microprocessor-based digital I&C applications. Generic I&C platforms using FPGA have been developed for safety applications such as Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS). The RPS and ESFAS implementation starts with plant specific requirements and licensing basis, which are translated into system requirements specification and followed by the system architectural design. The system design and architecture implementation is then realized via the generic FPGA platform, where the RPS and ESFAS applications logics are executed.

To execute the RPS and ESFAS applications logics, logic drawings or diagrams are first translated into Hardware Description Languages (HDLs). Then, using an electronic design automation tool, a technology-mapped netlist is generated. This process is called synthesis, in which the HDL or schematic design is translated to logic gates, memory units, registers and connections. The netlist can then be implemented by the FPGA manufacturer's proprietary software to fit to the actual FPGA architecture. This includes translation, map and place-and-route processes. The designer as well as V&V engineers will verify the map, place-and-route results via timing analysis, simulation, and other verification methodologies. Once the design and verification process is complete, the programming file generated (also using the FPGA manufacturer's proprietary software) is employed to (re)configure the FPGA. This file is transferred to the FPGA via a serial interface (JTAG).

The process of translating logic drawings into the HDLs is laborious and errors prone, especially for a system with a great number of inputs and outputs. It is therefore desired to automate the translation process, and achieve errors-free in implementing and executing safety logic algorithms. For this reason, One-Step software tool for FPGA applications has been developed. One-Step not only automates the logic translations but also creates dynamic images of a CAD (Computer-Aided Design) drawing for display on the workstations so that logic drawings can be dynamically monitored with the system on-line. The One-Step tool has been evaluated in accordance with the relevant industry standards (i.e., guidance as provided in IEEE Std 7-4.3.2-2003).

Development of Software Testing Environment for Safety-critical Software Reliability Quantification

Sang Hun LEE[1], Seung Jun LEE[2], Jinkyun PARK[3], Eun-chan LEE[4], Hyun Gook KANG[1]

[1]*Department of Mechanical, Aerospace and Nuclear Engineering department, Rensselaer Polytechnic Institute (RPI), 110 8th street Troy NY USA, 12180, Republic of Korea*

[2]*School of Mechanical, Aerospace and Nuclear Engineering, Ulsan National Institute of Science and Technology (UNIST), 50 UNIST-gil Ulsan Republic of Korea, 44919, Republic of Korea*

[3]*Integrated Safety Assessment Division, Korea Atomic Energy Research Institute (KAERI), 111 Daedeok-daero 989beon-gil Yuseong-gu, Daejeon Republic of Korea, 34057, Republic of Korea*

[4]*Korea Hydro & Nuclear Power Co., Ltd., 1655 Bulguk-ro Gyeongju-si Gyeongsangbuk-do Republic of Korea, 38120, Republic of Korea*

Recently, software has been used within nuclear power plants (NPPs) to digitalize many instrumentation and control (I&C) systems. To guarantee the safety of the NPP, the software reliability of the safety-critical systems must be properly quantified. In this study, the input-profile-based software test method using simulation-based software test-bed is proposed. The simulation-based software test-bed was developed by emulating the microprocessor architecture (CPU register, memory) of a programmable logic controller (PLC) used in NPP safety-critical applications and capturing its behavior at each machine instruction line. The software test cases which represents the possible states of software input and internal variables were developed in consideration of the digital signal processing of the safety-critical PLC as well as the plant thermo-hydraulics data in case of NPP accident. The effectiveness of the proposed safety-critical software test method is demonstrated via a case study for the KNICS RPS BP processor trip logic. Compared with the existing software testing methods, the proposed method can effectively generate the software test sets required for a software exhaustive testing, and reduce the software testing time by avoiding the repeated test for the same software input. Furthermore, the method can be employed to quantify the software reliability of NPP safety-critical I&C applications, and ensure the safe operation of the NPP.

Monday, November 27, 2017 10:40-12:20

15. Cognitive Systems Engineering for Process Control

Room #203
Session Chair: Jonghyun KIM (Chosun University), TBD

Computerized Procedure Interface for Nuclear Power Plant

Yeonsub JUNG

Central Research Institute, KHNP, Daejeon, Korea

Computerized procedure system for nuclear power plant has been designed and applied to power plants. User interface is based on both flowchart and logic tree. Both interfaces are integrated in Flowlogic Diagram.

Application of Ecological Interface Design in NPP Operator Support System

Alexey ANOKHIN[1], Alexey IVKIN[2], Sergey DOROKHOVICH[3]

[1]*JSC "Rusatom Automated Control Systems", 25, Ferganskaya str., Moscow, 109507, Russian Federation*

[2]*Obninsk Institute for Nuclear Power Engineering of the National Research Nuclear University "MEPhI", 1, Studgorodok, Obninsk, Kaluga Region, 249040, Russian Federation*

[3]*Simulation Systems Ltd., 133, Lenin str., Obninsk, 249035, Kaluga Region, Russian Federation*

Most publications confirm that ecological interface is a very effective tool supporting operators in recognition of complex unusual situations and in decision making. The present paper describes the experience of implementation of ecological interface concept for visualization of material balance in drum-separator of RBMK-type NPPs. Functional analysis of the domain area was carried out, which revealed main factors and contributors to the balance. The proposed ecological display was designed so that to facilitate execution of the most complicated cognitive operations, such as comparison, summarizing, prediction, etc. The experimental series carried out at NPPs demonstrated considerable reduction of operators' mental load, time of reaction and error rate.

Application of Petri Nets for Formalization of NPP I&C Functional Design

Alexey CHERNYAEV, Elena ALONTSEVA, Alexey ANOKHIN

Design Division, JSC "Rusatom Automated Control Systems", 25 Pherganskaya str., Moscow, 109507, Russian Federation

The paper describes the experience gathered from functional modelling of the process system using the Petri net formalism. The Petri net modification used in this study combines multilevel prioritized Petri Net with inhibitory arcs allowing formalization of hierarchical function-oriented description of a process system, which is a part of I&C functional design process. The hierarchy contains four levels, namely goals, abstract functions, process functions and equipment. Simple process system was modelled in order to analyze an adequacy of the selected Petri net modification. The modelling process includes building of abstract / process functions and equipment hierarchy followed by building of Petri net and realization / testing of this net using special software. The reduced function and equipment state space assuming only two states (function is fulfilled / not fulfilled, equipment is switched on / off) was used for the purpose of the present study. The experience acquired during the study allows to conclude that application of Petri net provides development of strictly formalized structural functional model which can be a subject for farther analysis using full scope or engineering simulator.

Suggestion of a RNN-based Plant Diagnosis System for Extreme Situations in Nuclear Power Plants

Seongkeun KANG, Poong Hyun SEONG

Dept. of Nuclear and Quantum Engineering, KAIST, Daejeon, Korea

The safe operation of nuclear power plants (NPPs) is a common goal for all nuclear industries. NPPs sometimes face with tiny events or rarely encounter accidents which are more severe than events. But events and accidents will not largely spread by well-activating safety systems and by correctly diagnosing and controlling the plants by the operators. Nevertheless, accidents including severe accidents have occurred only once in a while which caused by combining problems of human errors, natural disaster, failure of systems, etc. Therefore, the researches of preventing and mitigating those accidents have to keep going on future.

In this paper, we focused on diagnosing the initiating events or accidents in order to help main control room (MCR) operators and technical support center (TSC) members. In addition, we considered not only diagnosing the initiating events or accident but also giving the information about malfunctioning instrumentations in real time by comparing signal trends of each instrumentation to MCR operators and TSC members. We designed the recurrent neural network (RNN)-based plant diagnosis system for giving information to MCR operators and TSC members and also we conducted the case study to verify the suggested system. Moreover, we compared the suggested RNN-based system with traditional rule-based expert system.

Monday, November 27, 2017 14:00-15:40

16. Human Factors/Human Reliability Assessment

Room #203
Session Chair: Nicolas HENRY (EDF R&D), Yeonsub JUNG (KHNP)

Autonomous Algorithm for Safety Function State of Nuclear Power Plant by Using LSTM

Daeil LEE, Jonghyun KIM

Department of Nuclear Engineering, Chosun University, Gwangju, Korea

With the improvement of computer performance and the emergence of cutting-edge artificial intelligence (AI) algorithms, the autonomous operation based on AI is being applied to many industries. An autonomous algorithm is a higher level of concept than conventional automatic operation in nuclear power plants. In order to achieve an autonomous operation, the autonomous algorithm needs to include the superior function level to monitor, control and diagnose automated subsystems, and AI algorithms need to be suitable to make these superior functions. The artificial neural network (ANN), which is one of the AI approaches, can solve problems about the dynamic system that include the non-linear input and output values. Safety systems of nuclear power plants (NPPs) have non-linear values, and are controlled through a combination between the automation systems based on conventional controller and the manual control by operator. It means that the automation level of current NPPs corresponds to the shared control that is not autonomous control. This study aims at improving the safety system of automation level from the shared control to the autonomous control. This study suggests a model of safety systems in NPPs by using function based hierarchical framework, and autonomous algorithm to control and diagnose the safety function state by using the Long Short Term Memory (LSTM) that is one of the recurrent neural network (RNN) methods.

Modeling the Resilience of Severe Accident Management Organizations Using AHP

Jooyoung PARK[1], Ji-tae KIM[2], Jonghyun KIM[1]

[1]Department of Nuclear Engineering, Chosun University, 309 Pilmun-daero, Dong-gu, Gwangju 501-709, Republic of Korea

[2]Department of Instrument, Control & Electrical Engineering, Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon 305-338, Republic of Korea

Resilience can be defined as the intrinsic ability of a system to adjust its functioning prior to, during, or following changes and disturbances, so that it can sustain required operations under both expected and unexpected conditions. The concept of resilience in the organization of NPPs has been highlighted since the Fukushima Daiichi Nuclear Power Plant accident. An IAEA report addresses that a resilient organization is one that quickly realizes deviation from normal operations and has the ability to make even the toughest and least popular decisions and to manage the margins in which it can manoeuver.
This study attempts to model the resilience of severe accident management organizations, based on the author's previous research. First, a qualitative model of the resilience was developed for the organizational factors by reviewing emergency response plans in Korean NPP. Then, a quantitative model for entire severe accident management organizations has been developed by using the Analytic Hierarchy Process (AHP) method. For performing this method, several experts who are working on implementing, regulating or researching the severe accident management have participated in collecting the expertise on the relative importance of attributes and elements. Finally, a few simulations using the System Dynamics were conducted to discuss which factors have the most influence on resilience.

Sunburst Hierarchical Visualization Technique-based Navigation Support Interface for Information Processing System (IPS) in Nuclear Power Plants

Seung Min LEE[1], Gwi Sook JANG[2], Gee Yong PARK[2]

[1]Korea Atomic Energy Research Institute, 989-111, Daedeok-daero, Yuseong-gu, Daejeon, Korea

[2]Nuclear ICT Research Division, Korea Atomic Energy Research Institute, 989-111, Daedeok-daero, Yuseong-gu, Daejeon, Korea

Analogue-based information display panels in Main Control Room (MCR) have been changed to digitalized visual display units (VDUs). The VDU provides enormous amount of information and digital-based various functions to support human operators' monitoring and control tasks. However, to display lots of information on the limited VDU screen generates a major secondary task, that is a navigating, and it induces the degradation of human performance. To support the navigating tasks for layered display systems, the sunburst visualization technique, which is proven that it is useful for conducting navigating tasks, is applied to describe the whole hierarchical structure in small space. In this work, Sunburst hierarchy visualization technique-based navigating support system (SUNNI) is suggested. It is expected that the suggested system helps human operators to get rapid understanding and the formation of a mental model for the information display structure, to support page movement function, to decrease working time, and finally to improve the human performance.

The Development of Regulation Guideline Manual Regarding Beyond Design Basis Accident and Severe Accident

Ji-Yoon HAN, Dong-Jin KIM, Ji-Tae KIM, Yun-Hyung CHUNG

Department of I&C Systems, Korea Institute of Nuclear Safety(KINS), 62 Gweahak-ro, Yuseong-gu, Deajeon k, 34142, Republic of Korea

After Fukushima accident, many countries have analyzed Fukushima accident and established action plans for evaluation and development of emergency response facilities and emergency response organization for further improvements to nuclear safety and emergency protection. In Korea, the stress test for 2 units of operating nuclear power plant(NPP) was fulfilled as Fukushima follow-up measures. Reflecting the lessons learnt from Fukushima accident and the experiences of stress tests in Korea, the issues of human and organizational factors under beyond design basis accident (BDBA) and severe accident (SA) conditions should be addressed systematically during the design process of a new NPP under construction. According to the Nuclear Safety Act (NSA) revised and promulgated on June 22, 2015, operating licensees or applicants are required to submit an accident management plan (AMP) which describes the organizational responsibilities, equipment, and procedures or guidelines for implementing the accident management strategy. According to new Nuclear Safety and Security Commission (NSSC) rules, the AMP should be developed to cover accident including BDBA and SA. And human factors aspects also should be considered in AMP. In this matter, Korea Institute of Nuclear Safety (KINS), which is a technical support organization, developed the regulatory guideline supplement the existing regulatory guidelines related to human factors engineering program. It presents additional requirements to address HOF(Human and Organizational Factors) issues related to BDBA and SA conditions during the design process of a new NPP under construction. And for helping practical and comprehensive application of regulatory guideline for new NPP, the handbook has been developed. In this paper, the method and consideration for each human factors element of regulatory guideline are researched based on emergency response facilities and organization in BDBA and SA.

Safety Assessment Framework for the Nuclear Decommissioning

Hyung Jun KIM, Seung Jun LEE

Nuclear Science and Engineering, UNIST, 112, UNIST-gil 50, Ulsan, 44919, Korea

This study presents and analyzes safety assessment frameworks for nuclear decommissioning process. The safety assessment process for decommissioning will minimize the risks associated with radioactive materials by providing guidance. The guidance ensures safety of workers and the public through the identification of potential risks and mitigating consequences of potential risks. The safety assessment framework is to derive the potential risks of decommissioning nuclear plants and possible accidents of decommissioning activities. It is divided into 4 steps as a safety assessment framework. "Hazard Identification" is the first step of the safety assessment framework. Basically, it is necessary to analyze the initial events that may occur during the decommissioning nuclear process. Step 2 is 'Hazard Screening'. At this stage, a potential exposure routes, could harm workers during the nuclear decommissioning process, should be considered. In order to derive the hazard about decommissioning, human error analysis through HAZOP (Hazard and Operability) and instrumental error analysis through FMEA (Failure Mode & Effect Analysis) are performed considering the paths of the exposure. Step 3 is 'Identification of Scenarios'. This step`s purpose is that making a list of accident scenarios using the hazard derived in the second stage. Step 4 is 'Hazard Analysis'. The final step is to develop probabilistic models for accident scenarios and to calculate worker exposure in accident scenarios to draw preventive and additional actions to reduce consequences [1, 2].

Monday, November 27, 2017 16:00-17:40

17. System Simulation Technologies

Room #203
Session Chair: Takeshi MATSUOKA (Utsunomiya University), Kee-Choon KWON (KAERI)

Improvement of Wolsung Simulator including Severe Accident Analysis Models

Munsoo KIM, Yeonsub JUNG

Nuclear System Safety Laboratory, KHNP Central Research Institute, 70, 1312 beon-gil, Yuseong-daero, Yuseong-gu, Daejeon, Korea

KHNP is promoting the Wolsung simulator development project that is to adapt severe accident analysis models in the simulator. This project will build a new full scope simulator of a CANDU plant specific to the Wolsung unit 1 plant using L3M Orchid® simulation environment.
For the severe accidents trainings, the Wolsung simulator will include severe accident simulation module using the MAAP4-CANDU. In the early part of a transient, MAAP4 may run slower than real time and much faster than real time at other times. Those problems can be caused instability of simulation system. Wolsung simulator hard ware have built in L3M and for completion of the simulator, the simulator will be validated against a number of acceptance test procedures including normal evolutions, steady state, transient and malfunction tests, severe accidents cases, spare requirements.

The Design and Implementation of Refuelling Machine Simulator Control System Based on FPGA

Zhijun HE, Peng ZHANG

I&C Division, China Nuclear Power Engineering CO., LTD, No.117 xisanhuanbeilu, Haidian District, Beijing, 100840, China

Refuelling machine simulator (simulator for short) has been used for operators to learn refuelling operation steps or periodic inspection. Refuelling machine is working in critical path of pressurized water reactor (PWR) nuclear power plants (NPPs), which has complicated control functions and interlock protection for refuelling operation. After finishing the refuelling, the control console of refuelling machine will be removed to a special room outside of reactor building. Then the simulator can be conveniently interconnected with control console for learning operation steps or making sure the control console in good condition before next reactor outage. The paper introduces the design and implementation of simulator control system, including requirements analysis, electrical product selection, control software development, HMI configuration and test validation. For designing and developing the simulator, the paper puts forward an innovative scheme that uses virtual simulation tools instead of physical instruments and electrical components by combining LabVIEW software platform with NI hardware platform based on FPGA. According to control requirements, the paper establishes mathematical simulation models of encoders, motors, load sensors, etc. Finally, the simulator has been validated by lots of tests to ensure the scheme is satisfied with design specification. In fact, the simulator control system has good performances in real time and expansibility. The simulator has been applied in the automatically controlled and digital refuelling machine of 1000MW PWR NPPs in China.

The Analysis and Simulation Study of the Control System for the Floating Nuclear Power Plant ACP100S

Zhi CHEN, Kai YOU, Tao LongTAO

Science and Technology on Reactor System Design Technology Laboratory,Nuclear Power Institute of China,Chengdu, 610213,China

Floating nuclear power plant (FNPP) is built on the offshore platform. As a new way of using nuclear energy in the sea, floating nuclear power plant can be operated in a fixed area, but also can be moved to other areas to provide energy by self propulsion or by means of tugs. The FNPP can provide power supply, fresh water and high temperature steam and other products. So it can be used as special energy sources, such as regional power supply, district heating and offshore oil exploitation. Based on the market demand of floating nuclear power plant, China National Nuclear Corporation has developed different types of floating nuclear power plant. One of them, named ACP100S, was developed on the basis of the technology of land-based small modular reactor ACP100.In this paper, firstly, the general technical parameters and the overall technical scheme of ACP100S are briefly introduced. On the basis of this, the design scheme of the control system is discussed, and the difference of the system design is compared with that of the land based on the characteristics of the application environment. These differences include: tilt, swing and other marine environmental conditions on the impact of the design of the control system, influence of layout space changes brought to the overall design of the control system and the influence of the load following requirements for the control system under the circumstance that the FNPP is running on the sea in the isolate island state, etc. Finally, the simulation results of the control system are analyzed under several transient conditions. The results show that the optimized control scheme can meet the requirements of the floating nuclear power plant.

Multi-unit Small Modular Reactor Control System Experimental Platform Design

Xinyu WEI, Junyan QING, Fuyu ZHAO

Department of Nuclear Science and Technology, Xi'an Jiaotong University, Xi'an 710049, China

Small modular reactor (SMR) is a type of advanced reactor depending on its safety, economy, and multipurpose. SMR always adopts the multi-unit scheme, which uses several nuclear steam supply system (NSSS) units to supply steam to a steam header. We call it multi-unit small modular reactor (M-SMR) here. Because these NSSSs are separate, the header pressure is hard to control. The experimental platform is designed to solve the control problem. The platform contains four NSSS units, which are connected to a header. The work medium in the loop is water. Some experimental plan is also designed based on this platform.

Tuesday, November 28, 2017 10:00-12:00

18. System Reliability and Risk

Room #203

Session Chair: Wu Jie (Institute of Nuclear Energy Safety Technology), Seung Jun LEE (UNIST)

An Approach To Assess The Impact Of Instrumentation With An Embedded Digital Device

Richard T. WOOD[1], Tanner G. JACOBI[2], Dan C. FLOYD[2]

[1]*Nuclear Engineering Dept., University of Tennessee, Knoxville, Tennessee, 37996 USA*

[2]*NE Dept., University of Tennessee, Knoxville, Tennessee, 37996 USA*

The instrumentation and control (I&C) equipment used in currently operating U.S. nuclear power plants (NPPs) is primarily based on mature analog technologies that are progressing towards obsolescence. The continued reliance on this legacy analog technology imposes performance penalties and maintenance burdens in comparison with modern digital I&C equipment.

However, the nuclear power industry has been slow to adopt digital technology in large part as a result of regulatory concerns about common-cause failure (CCF) vulnerabilities. In many instances, currently available I&C equipment contain embedded digital devices (EDDs) such as microprocessors and programmable logic devices. Consequently, there is a clear need to develop cost effective qualification methods to contribute to the assessment of CCF vulnerability posed by EDDs in modern instrumentation that could be used in NPPs.

This paper describes findings from research regarding qualification methods for equipment with EDDs that is sponsored by the Nuclear Energy Enabling Technologies (NEET) Advanced Sensors and Instrumentation (ASI) program of the U.S. Department of Energy (DOE). Specifically, a classification framework was defined and an extended diversity and defense-in-depth (D3) analysis approach was developed to treat equipment with an EDD based on the functional impact of the device. These outcomes can contribute to a more systematic, predictable assessment of equipment with an EDD that can potentially reduce the burden of having to perform a full D3 analysis for every device.

Analysis on Accident Sequences of SGTR Accident Considering the Status of Safety Valves

Jaehyun HAM[1], Hyun Gook KANG[2]

[1]*Department of Nuclear and Quantum Engineering, Korea Advanced Institute of Science and Technology, 291 Daehak-ro, Daejeon, 34141, Republic of Korea*

[2]*Department of Mechanical, Aerospace, and Nuclear Engineering, Rensselaer Polytechnic Institute, 110 Eighth Street, Troy, NY USA 12180*

A method that classifies accident sequences within single Initiating Event (IE) depends on the status of safety valves in pressurizer and Steam Generators (SGs) to analyze accident sequences that cannot be found in current Event Tree (ET) was suggested in this research. Safety-grade automatic mitigation systems without operator intervention were only considered in accident sequences. Using this method, Steam Generator Tube Rupture (SGTR) accident which is relatively more complicated IE than others to mitigate for operator because condition of each SG is different was analyzed using MARS-KS-003, the thermal hydraulic system code. In the result, accident sequences which can prevent Core Damage (CD) only with safety-grade automatic mitigation systems in SGTR accident sequences was analyzed that cannot be analyzed in current ET.

Uncertainty Characterization for Dynamic Risk Assessment

Robby CHRISTIAN, Hyun Gook KANG

Mechanical, Aerospace, and Nuclear Engineering Department, Rensselaer Polytechnic Institute, Troy, NY 12180, USA

This work proposes a framework to enable dynamic PRA by leveraging information from the well-established static PRA. The framework starts by segmenting accident mitigation process into multiple stages based on static Event Trees and Emergency Operating Procedures. Variations to the initiation of each stage were derived from Fault Trees.

Plant parameters as the exit condition of each mitigation stage were estimated using MARS system code. Because transient simulations through system codes are computationally expensive, the stage's uncertainties were sparsely sampled. A Reduced Order Model (ROM) method was formulated to interpolate and obtain a continuous distribution of the stage's exit condition from these entry condition's samples. It utilizes the Taylor Kriging (TK) method to capture nonlinearities of plant parameters in response to the stage's entry condition variations. Total Variation Regularization (TVR) was used to estimate the Taylor regression order. Numerical error from the time-stepping method, round-offs and ROM construction was formulated and propagated as the next stage's IC error.

A case study on Large Break LOCA (LBLOCA) with uncertainties on Low Pressure Safety Injection (LPSI) actuation timing and capacity is presented. Results show that the proposed ROM method can provide a continuous response and error estimate on plant parameters from finite samples. The methodology enables the plant's safety margin and propagated error to be quantified by using cascading stage ROMs.

A Coordination Review of AVR Limiter and Protective Function in Excitation System for Reliable Power System Operation in NPP

Ji-Kyung PARK, Hong-Seok JANG, Young-Sik CHO

Dept. of I&C and Electrical System, Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon, 34142, Republic of KOREA

Because of inadequate setting values of Automatic Voltage Regulator (AVR) limiter and protective functions, it is possible to happen the unintentional generator trip, which affects to the reliable power system operation. Therefore, it is important to coordinate the setting value of AVR limiter and protective functions in excitation system properly. Even, there could be redundant protective functions in generator protection system. In this paper, we focused on generator under-excitation region to review the performance and coordination of the Under-Excitation Limiter (UEL), Under-Excitation Protection (UEP) and Loss-of-Field (LOF) relay protection, which can affect to generator operation and system reliability. All of the review were conducted by computer simulations with Electro-Magnetic Transient Program (EMTP). After that we concluded the importance of protective coordination of AVR limiter and redundant protective functions. In addition, we discussed the regulatory recommendation for the verification of the protective coordination.

An Estimation of the Effectiveness of an Hybrid-SIT System under SGTR Accident

In Seop JEON, Hyun Gook KANG

Department of Mechanical, Aerospace and Nuclear Engineering, Rensselaer Polytechnic Institute 110 8th Street, Troy, NY, USA

A hybrid safety injection tank (H-SIT) is designed to passively inject coolant into a reactor coolant system under any pressure condition without depressurization, which enables it to be applied in various accident types. In some accident conditions, if long-term cooling is not achievable because required active mitigation components are unavailable, the H-SIT can replace the function of failed components at least for a certain time period. Since the H-SIT has limited inventory, the mitigation strategy needs to be carefully developed to effectively enhance a plant safety. This study focuses on the risk effect of the H-SIT system under steam generator tube rupture (SGTR) accident with safety injection system (SIS) failure, which has been conventionally considered to cause core damage. The use of the H-SIT provides diversity of mitigation options and allows more time to repair the failed components. In order to address plant dynamics in a realistic manner, this study considers the variety of secondary-side cooling performances with corresponding repair probability. Multiple-tube rupture cases are also considered. The analysis results demonstrate that the H-SIT extensively contributes to the plant risk reduction.

Analysis of Operator Available Time for Responding to Accident Situations in a Digitalized Main Control Room

Ji Suk KIM[1], Eun Seo SO[2], Seung Hoon CHAE[2], Jae Seon HA[2], Eun Jin JEONG[2], Man Cheol KIM[2]

[1]*Department of Energy Engineering, Chung-ang university, Seoul, 06974, Korea*

[2]*School of Energy System Engineering, Chung-ang university, Seoul, 06974, Korea*

After the occurrence of an accident, operators' correct understanding of the situation and timely responding actions are important in maintaining the safety of nuclear power plants. Depending on the operator available time, the situation awareness and hence whether the accident situation is properly managed or not can be determined. In this paper, the operator available times for several accident situations involving small break loss of coolant accident (SBLOCA), loss of feedwater (LOFW), excessive steam dump event (ESDE) are evaluated using a simulation code. The results of this analysis can be used as basis for estimating the operators' failure probability in a digitalized main control room of a nuclear power plant.

Tuesday, November 28, 2017 14:20-16:00

19. MFM & Safety Culture I

Room #203
Session Chair: Morten LIND (DTU), Poong Hyun SEONG (KAIST)

Knowledge Acquisition and Strategies for Multilevel Flow Modelling

Morten LIND

Department of Electrical Engineering, Technical University of Denmark, Kgs. Lyngby, DK2860, Denmark

The paper presents three different strategies for building Multilevel Flow Models (MFM). The first two strategies are formulated on the basis of the end-means and whole part abstractions which are foundations for MFM. These strategies are generally applicable but it is concluded that they both are inefficient and error prone. A third strategy is investigated which try to overcome these deficiencies and at the same time meet the needs of industry. The aim of the strategy is to use information from plant engineering documents to facilitate the building of MFM models. The main challenge is to acquire knowledge about design and operational intentions which often are not accessible in explicit form. The solutions proposed is to use domain dependent libraries of MFM models to represent goals and functions of process units. The possibility of extracting the knowledge required for library building from engineering documents is investigated.

Enhanced Reasoning with Multilevel Flow Modeling Based on Time-to-detect and Time-to-effect Concepts

Seung Geun KIM, Poong Hyun SEONG

Department of Nuclear and Quantum Engineering, Korea Advanced Institute of Science and Technology, 34141, KS015, Korea

In order to easily and systematically understand the behaviors of the various industrial systems, various system modeling techniques have been developed. Especially, the importance of system modeling technique is more emphasized in recent years since modern industrial systems become enlarged and more complex. Multilevel flow modeling (MFM) is one of the qualitative modeling techniques for the representation and reasoning about knowledge of physical phenomena and systems which cannot be modelled quantitatively. MFM can be applied to the industrial systems without additional domain-specific assumptions or knowledge, and qualitative reasoning of event causes and consequences can be conducted with high speed and fidelity. However, current MFM has a limitation that it is not able to consider dynamic features of the target system since time-related concepts are not involved. In this paper, the concepts of time-to-detect (TTD) and time-to-effect (TTE) are adopted from system failure model; and the methodology for enhancing MFM-based reasoning with time-series data is suggested. In addition, empirical TTE distribution estimation methods based on Bayesian update and non-Bayesian distribution approximation methods are described.

Identifying Causality from Alarm Observations

Denis KIRCHHÜBEL, Xinxin ZHANG, Morten LIND, Ole RAVN

Department of Electrical Engineering, Technical University of Denmark, Elektrovej, Building 326, Kgs. Lyngby, 2800, Denmark

The complexity of modern industrial plants poses significant challenges for the design of effective alarm systems. Rigorous alarm management is recommended to ensure that the operators get useful information from the alarm system, rather than being overloaded with irrelevant state information. Alarm management practices have been shown to significantly reduce the frequency of alarms in industrial process plants. These practices help focusing the operators' attention on actually critical situations. However, they cannot resolve the cascades of critical situations frequently occurring during emergency situations.

Multilevel flow modelling (MFM) has been proposed as a way of representing knowledge about the industrial process and infer causes and consequences of deviations throughout the system. The method enables the identification of causes and consequences of alarm situations based on an abstracted model

of the mass and energy flows in the system.

The application of MFM for root cause analysis based alarm grouping has been demonstrated and can be extended to reason about the direction of causality considering the entirety of the alarms present in the system for more comprehensive decision support.

This contribution presents the foundation for combining the cause and consequence propagation of multiple observations from the system based on an MFM model. The proposed logical reasoning matches actually observed alarms to the propagation analysis in MFM to distinguish plausible causes and consequences. This extended analysis results in causal paths from likely root causes to tentative consequences, providing the operator with a comprehensive tool to not only identify but also rank the criticality of a large number of concurrent alarms in the system.

Barrier Identification by Functional Modeling of a Nuclear Power System

Jing WU[1], Morten LIND[1], Xinxin ZHANG[1]
Pardhasaradhi KARNATI[2]

[1]*Department of Electrical Engineering, Technical University of Denmark, Kongens Lyngby, 2800, Denmark*

[2]*ELDOR Technology AS, Koppholen 19, 4313 Sandnes*

The paper investigates application of functional modeling for independent protection layer analysis of risk assessment in complex industrial plant with special reference to nuclear power production.

Layer of Protection Analysis (LOPA) is a simplified semi-quantitative risk assessment method that typically builds on the information developed during a qualitative hazard evaluation such as HAZOP. LOPA typically uses order of magnitude categories for initiating event frequency, consequence severity, and the likelihood of failure of independent protection layers (IPLs) to approximate the risk of a scenario. Identifying the IPLs systematically is a fundamental challenge as a basis for estimating the probability of failure on demand of each IPLs and for evaluating the risk to a decision concerning the scenario. Functional safety is the main focus of this study, which shows the modeling and reasoning capability of functional modeling, e.g. Multilevel Flow Modeling (MFM) and its application in IPLs analysis of a design based accident scenario, e.g. Loss of coolant accident (LOCA). Previously, MFM has showed its potential to be used for safety barrier analysis and Defense in Depth.

The main contribution of the study is to explore a procedure using MFM to identify safeguards and then credit some of them as IPLs. Firstly, MFM modeling of the process system including control flow structures is presented. Secondly, the rule-based cause reasoning of MFM is used to identify initiating causes (chain of causes) of a specific consequence. Thirdly, safeguards are derived (safety functions in the system are designed represented by MFM functions) to prevent the consequence to happen. Fourth, judging the initiating causes and safeguards whether they can have common mode failure. If there is no common mode failure, then the safeguard is considered as an IPL. This procedure is demonstrated in a PWR LOCA accident scenario.

Reasoning about Cause-effect through Control Functions in Multilevel Flow Modelling

Xinxin ZHANG, Morten LIND

Department of Electrical Engineering, Technical University of Denmark, Elektrovej Building 326, Kgs. Lyngby 2800, Denmark

Multilevel Flow Modelling has been used for modelling complex system such as nuclear power plants. The causal reasoning capability of the MFM models is explained in various literatures by the authors as well as other researchers. MFM is also used to represent control functions in relation with system objectives. This paper clarify the fulfilment of MFM objectives and extend the MFM causal reasoning rules to the control functions and use reasoning rules to generate explanations for understanding control actions. A case study based on a previous developed PWR model is used to illustrate the new reasoning rules. This work contribute to support human operators to understand system automation under abnormal situations.

Tuesday, November 28, 2017 16:20-18:20

20. MFM & Safety Culture Ⅱ

Room #203
Session Chair: Andreas BYE (OECD HRP), Akio GOFUKU (Okayama University)

Accident Management of the Station Blackout at BWR by Using Multilevel Flow Modeling

Mengchu SONG, Akio GOFUKU

Graduate School of Natural Science and Technology, Okayama University, Okayama 700-0084, Japan

Accident management is a key factor to ensure the defense in depth of the nuclear power plant. One of the requirement is an effective response planning, especially for the unexpected event when pre-defined countermeasures may fail. When operators encounter a planning task, they usually consider the problem within a context of intentions. It is important to apply model-based system to acquire plant's intentionality, which is necessary to manage planning related resources and further to generate plans. In this paper, we first demonstrate how multilevel flow modeling, a method of functional modeling can represent the intentional knowledge of a plant in terms of function and goal. Then we will investigate how the same representation can be used to identify alternative means to realize goals, which may be out of their original purposes considered in design but have positive effects on goal achievement. Based on a previous study, the alternative means generated by MFM can be further expressed as operating procedure, which can include series of human operations. An accident case that is similar with the Fukushima Daiichi accident, i.e. station blackout of a boiling water reactor, shows how MFM can be used to support the activity of response planning. Planning knowledge of the plant that contains functions, objectives, and their relationships is represented by an MFM model, based on which, several response plans are generated to achieve the goal of core cooling.

Modelling and Validating a Deoiling Hydrocylone for Fault Diagnosis using Multilevel Flow Modeling

Emil Krabbe NIELSEN[1], Mads Valentin BRAM[2], Jerome FRUTIGER[3], Gurkan SIN[3], Morten LIND[1]

[1]Department of Electrical Engineering, Technical University of Denmark, Elektrovej 326, 2800, Denmark

[2]Department of Energy Technology, Aalborg University, Niels Bohrs Vej 8, 6700, Esbjerg

[3]Department of Chemical Engineering, Technical University of Denmark, Søltofts Plads 227, 2800, Denmark

Decision support systems are a key focus in research on developing control rooms to aid operators in making reliable decisions, and reducing incidents caused by human errors. For this purpose, models of complex systems can be developed to diagnose causes or consequences for specific alarms. Models applied in safety systems of complex and safety critical systems, require rigorous and reliable model building and testing. Multilevel Flow Modeling is a qualitative method for diagnosing faults, and has previously only been validated by subjective and qualitative means. This work aims to synthesize a procedure to measure model performance, according to diagnostic requirements, to ensure reliability during operation. A simple procedure is proposed for validating and evaluating Multilevel Flow Modeling models. For this purpose expert statements, a dynamic process simulation in K-spice, and pilot plant experiments are used for validation of two simple Multilevel Flow Modeling models of a deoiling hydrocyclone, used for water and oil separation.

An Overview of the MFM Suite for Diagnostic and Prognostic Reasoning of Industrial Process Plants

Harald P-J THUNEM

Systems and Interface Design Department, Institute for energy technology, P.O.Box 173, N-1751 Halden, Norway

This paper provides the background for and an overview of the development of a software system, the MFM Suite, dedicated to the design and analysis of MFM models related to diagnostic and prognostic analysis of physical processes. The current features of the system are described, as well as some examples of its practical use. The paper also briefly describes how the system facilitates the collaboration between control room and field operators via the Android-based MFM Viewer app.

Development of an Evaluation Method for Nuclear Safety Culture Competency using Social Network Analysis

Sang Min HAN, Poong Hyun SEONG

Department of Nuclear and Quantum Engineering, Korea Advanced Institute of Science and Technology, 291 Daehak-ro, Yuseong-gu, Daejeon, 34141, Republic of Korea

The aim of this study is to propose an evaluation method for Nuclear Safety Culture (NSC) competency using Social Network Analysis (SNA). NSC Competency has been defined and its behavioral characteristics are derived from procedures in nuclear power plant. SNA was applied to evaluate NSC competency of an operation team. The example of an analysis and its validity for application has been addressed in the paper.

What We Have Learned so Far About the Importance of MTO in Control Room Design

Andreas BYE

Institute for energy technology (IFE), OECD Halden Reactor Project, P.O.Box 173, NO-1751 Halden, Norway

The OECD Halden Reactor Project has for many years performed research on the safety of nuclear power plants (NPPs). The focus has been on empirical research in our simulator laboratory HAMMLAB (HAlden Man-Machine LABoratory) as well as on empirical studies in the field and in training simulators in NPPs. The MTO (Man-Technology-Organisation) perspective is a system-oriented perspective in which we seek to understand the dynamic relation between humans, technology and organization. The importance of MTO in control room design is evident by the fact that Human Factors Engineering (HFE) is included as a specific element in regulatory guidance in all countries that operate NPPs. This paper focuses on the need for empirical evidence to support MTO research; experience shows that empirical evidence can often contradict a-priori assumptions. Empirical investigations can help to: 1) identify key questions that control room designers and regulatory reviewers should ask; 2) define ideas for new and innovative designs; 3) evaluate and validate human performance in the control room, with respect to both integrated system validation (ISV) as well as human reliability. This paper outlines examples of results from empirical research carried out by the Halden Reactor Project to address these different needs in the nuclear industry.

Review of Emergency Operating Guidelines from Nuclear Safety Culture Perspectives

Ho Bin YIM, Jae Min PARK, Chang Gyun LEE, Myung Hoon LEE, Jae Young HUH, Gyu Cheon LEE

Department of Safety Analysis, KEPCO E&C Co., Inc., Yuseong-gu, Daejeon, Korea

The term "Safety Culture" first appeared after the Chernobyl accident in the IAEA safety series No. 75-INSAG-1 in 1986. Even though the concept of "safety culture" was born after the severe accident, the safety cultural mind had been diluted with efficiency of NPPs and nearly forgotten until another one, the Fukushima accident, come in 2011. Nuclear Safety Culture (NSC) seems to be more strong connection to plant operation organizations because NSC focuses on prevention of damages and impacts on people and the environment caused by nuclear accidents. However, so evidently, NSC has influence on plan, design, and construction of NPPs, and NPP safety is finally maintained by operating companies. Emergency Operating Guidelines (EOGs) are license documents developed by a design company in Korea. Thus, through examination of EOGs from the NSC perspective is worthwhile since EOGs are the basis documents to handle NPP accidents. Even though EOGs do not mention NSC, EOGs contain key item and philosophy of NSC.

Wednesday, November 29, 2017 10:00-12:00

21. Testing and Maintenance

Room #203
Session Chair: Richard T. WOOD (University of Tennessee), TBD

Contribution of Electronic Circuit Simulation to Maintain and Exploit Perpetuated I&C Systems in Nuclear Power Plants

Alain OURGHANLIAN

EDF Lab Chatou, 6 quai Watier . BP 49, 78401 Chatou cedex, FRANCE

As part of the extension of the life expectancy of its nuclear power plants, Electricite de France adopts a strategy of refurbishment of Instrument and Control systems, or decides to extend their lifespan them. We describe a practical experimentation showing the contribution of modelling and simulation at the electronic component level of a discrete analog system used on EDF's 900MW Nuclear Power Plant fleet. The interest of a low level model is the capability of sensitivity analysis tools to see the influence of different parameters on the overall behaviour of a function. When replacing faulty modules in operation, this approach can also be used to ensure that the new module is fully operational by comparing voltages and currents measured on the electronic circuit with the results supplied by the model.

An Improved Response Time Test Methodology for the Plant Protection System and Engineered Safety Feature-Component Control System

Chang Jae LEE, Jae Hee YUN

I&C System Engineering Group, KEPCO E&C, 111, Daedeok-daero 989 Beon-Gil, Yuseong-gu, 34057, Daejeon, Korea

The safety instrumentation and control system for the advanced power reactor 1400 (APR1400) nuclear power plant has been improved in terms of hardware configuration and communication system that address critical trip signals. For the optimized power reactor 1000 (OPR1000) nuclear power plants, a lumped test method has been applied in order to conduct the periodic response time test required by the Technical Specifications. This paper proposes a new methodology that covers the response time test for the plant protection system and engineered safety feature . component control system, using a distributed approach. Furthermore, the lumped method used for the OPR1000 is provided in detail to compare with the proposed method for the APR1400. The test results are also presented herein and indicate that the proposed method is appropriate and reasonable to meet the response time design requirement.

Development of an Information Reference System using Reconstruction Models of Nuclear Power Plants

Yuki HARAZONO[1], Taro KIMURA[2], Hirotake ISHII[1], Hiroshi SHIMODA[1], Yuya KOUDA[3]

[1]*Graduate School of Energy Science, Kyoto University, Yoshida Honmachi, Sakyo-ku, Kyoto-shi, Kyoto 606-8501, Japan*

[2]*Present: SoftBank Corp., Higashishinbashi 1-9-1, Minato-ku, Tokyo 105-7317, Japan*

[3]*Fugen Decommissioning Engineering Center, Japan Atomic Energy Agency, Myojin-cho, Tsuruga-shi, Fukui 914-8510, Japan*

Many nuclear power plants(NPPs) in Japan are approaching the end of their operational lifespan. They need to be decommissioned safely in the near future. By using Augmented Reality(AR), workers can understand information related to decommissioning work intuitively. Three dimensional(3D) reconstruction models of dismantling fields are useful for workers to observe circumstances of dismantling fields without visiting the fields. In this study, an information reference system based on AR and 3D reconstruction models has been developed and evaluated. The evaluation consists of questionnaires and interview surveys with 6 workers at NPP, who used this system along with a scenario. The results showed that it will be possible to reduce the time spent and mistakes in dismantling fields. The results also showed that it is easy to refer information in dismantling fields. However, it was also found that it is difficult for workers to build reconstruction models of dismantling fields by themselves.

Model Based Sensor Parameter Estimation and Smart Calibration Scheme

Mujtaba MUJAHID, Ahmed YAR, Talha AZFAR

Directorate of I&C, DNPEP, PAEC, Islamabad, Pakistan, 46300, Pakistani

For online estimation of calibration related sensor parameters, a composite model is proposed by considering a cascade of the plant and sensor as one system. Online parameter estimation is carried out by comparing the system response with the response of an estimated parameter model of the system. The difference between these outputs is minimized by adjusting the estimated parameters according to adaptive laws. If the actual values of the sensor calibration parameters drift with time, it will be tracked by the estimator and notified to the operator if it exceeds some preset bounds.

Updated Electromagnetic Compatibility Guidance For Nuclear Power Plant Instrumentation And Control Systems

Richard T. WOOD[1], David M. DAWOOD[2]

[1]*Nuclear Engineering Department, University of Tennessee, Knoxville, Tennessee, USA*

[2]*U.S. Nuclear Regulatory Commission, Washington, DC, USA*

The typical environment in a nuclear power plant includes many sources of electromagnetic interference (EMI), radio-frequency interference (RFI), and power surges, e.g., hand-held two-way radios, arc welders, switching of large inductive loads, high fault currents, and high-energy fast transients associated with switching at the generator or transmission voltage levels. The increasing use of advanced analog- and microprocessor-based instrumentation and control (I&C) systems in reactor protection and other safety-related plant systems has introduced concerns with respect to the susceptibility of this equipment to EMI/RFI and power surges, as well as the creation of additional noise sources.

Regulatory Guide (RG) 1.180, *Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems*, was developed to provide guidance to licensees and applicants on methods acceptable to the U.S. Nuclear Regulatory Commission (NRC) staff for complying with the NRC's regulations on design, installation, and testing practices for addressing the effects of electromagnetic and radio-frequency interference (EMI/RFI) and power surges on safety-related instrumentation and control (I&C) systems. The regulatory guide is applicable to both safety-related I&C systems and non-safety-related I&C systems whose failures can affect safety functions The regulatory guide endorses grounding and shielding practices from the Institute of Electrical and Electronic Engineers (IEEE), EMI/RFI emissions and susceptibility test methods from the U.S. Department of Defense and the International Electrotechnical Commission (IEC), and surge withstand capability (SWC) test methods from IEEE and IEC.

RG 1.180 was last revised in 2003. At the time, the regulatory guide represented the most accurate and reasonable collection of practices available to ensure the capability of I&C systems to accommodate the effects of and be compatible with the expected (nominal and abnormal) electromagnetic environment in nuclear power plants. Since, its revision, new or revised guidance and standards have emerged, the use of wireless transmitters has increased and additional technical information has become available. This information has been evaluated and the findings provide the technical basis for revision of this guide. This paper presents the technical basis for an update of RG 1.180.

Equipment Testing for Severe Accident Conditions

Vít PLAČEK[1], Zuzana KONEČNÁ[2]

[1]*Radiation Chemistry and Envirenmental Qualification Dept., ÚJV Řež, a. s., Hlavní 130, Řež, 25068 Husinec, Czech Republic*

[2]*Dept. of Electrotechnology, FEE – Czech Technical University in Prague, Technická 2, 166 27 Praha, Czech Republic*

The operational states of a nuclear power plant can be generally categorized as normal operation and accident conditions. The latter one can be further divided into so-called design basis accidents and design extension conditions. Design extension conditions include in the worst case conditions with core melting, called severe accidents. Instrumentation and equipment important for mitigating severe accident need to be tested to ensure the functionality during the severe accident environmental parameters and the mission time, which may be in the order of weeks or even years. The traditional environmental qualification method is not suitable in all cases for testing the severe accident instrumentation and dedicated mitigation equipment. Moreover, design extensions conditions and especially severe accidents are mostly not addressed in equipment qualification standards. This paper summarizes some aspects of the severe accident conditions testing.

Technical Tour

- **Destination** Korea Radioactive Waste Agency (KORAD)
- **Date and time** 11. 30. 2017 (Thu) 08:30 ~ 13:00, local time
- **Schedule**

Time	Detail
08:30 ~ 09:20	Movement (HICO → KORAD)
09:20 ~ 09:35	Entrance and exit application -Original passport necessary
09:35 ~ 09:40	Greeting and photo opportunity
09:40 ~ 11:10	Field trip to Cheongjeong Nuri Park, Surface Facilities, LILRW Disposal Facility
11:10 ~ 11:50	Movement (From KORAD to Restaurant)
11:50 ~ 12:50	Luncheon meeting
12:50 ~ 13:00	Movement (Restaurant → HICO)

- **LILRW Disposal Facility & Cheongjeong Nuri Park**

NUCLEAR R&D GLOBAL LEADER

KAERI
Korea Atomic Energy Research Institute

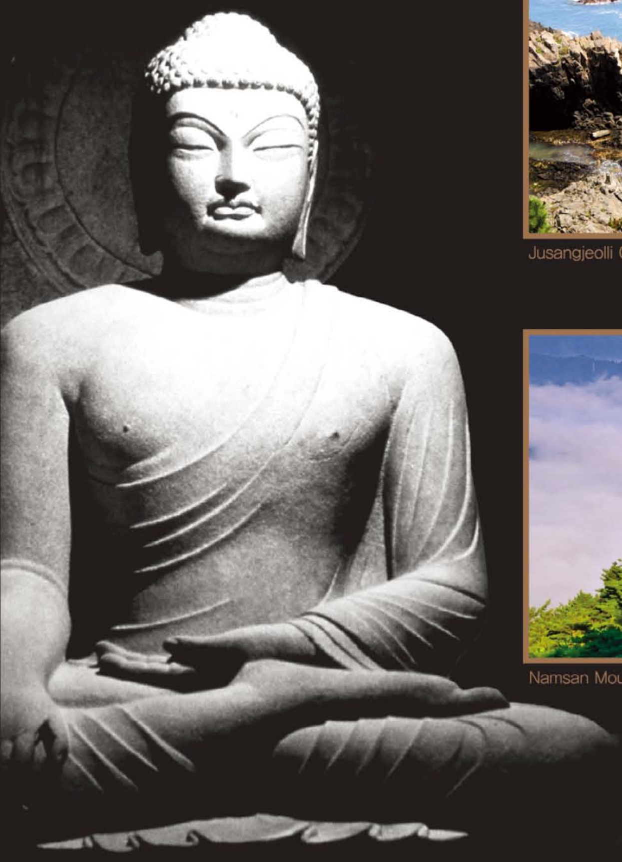
Wherever light is lacking,
our technology is delivering
Building your tomorrow today
DOOSAN
Doosan Heavy Industries & Construction

The treasure of a brilliant cultural heritage

Welcome to Gyeongju

Cheomseongdae Observatory

Donggung Palace and Wolji Pond



Jusangjeolli Cliff

Bomun Lake

Namsan Mountain

Hwabaek International Convention Center

Pride GyeongBuk

Gyeongju CVB
경주화백컨벤션뷰로
Gyeongju Hwabaek Convention and Visitors Bureau

MEMO

MEMO

Program at a Glance

	Room #201	Room #202	Room #203
November 26, 2017 (Sunday)			
17:00~19:30	Registration		
18:00~19:30	Welcoming Reception : Hilton Hotel Pine Room (1st Floor)		
November 27, 2017 (Monday)			
08:00~18:00	Registration		
08:30-09:00	Opening Ceremony (HICO 1st Grand Hall) Opening Remarks : Man Gyun Na (General Chair) Welcoming Address : Hark Rho KIM (President, Korean Nuclear Society) (Honorary Chair) Congratulatory Address : Soon Heung CHANG (President, Handong Global University)		
09:00~09:30	Plenary Talk 1 : Prof. J. Wesley Hines (University of Tennessee, Knoxville) Advanced Prognostics Using Both Process and Maintenance Data		
09:30~10:00	Plenary Talk 2 : Prof. Takeshi Matsuoka (Utsunomiya University) Fukushima Nuclear Power Plant Accidents in the Viewpoint of PSA		
10:00-10:30	Plenary Talk 3 : Mr. Ian Jung (US Nuclear Regulatory Commission) Introduction of Embedded Digital Devices – What Are They Telling Us?		
	Photo time		
10:30~10:40	Coffee Break		
10:40~12:20	Methods of Sensing, Processing, and Communication	Modernization of I&C and Control Room	Cognitive Systems Engineering for Process Control
12:20~14:00	Lunch Break		
14:00~15:40	Surveillance, Diagnostics, and Prognostics Ⅰ	Future I&C Technologies for Nuclear Applications Ⅰ	Human Factors/Human Reliability Assessment
15:40~16:00	Coffee Break		
16:00~17:40	Surveillance, Diagnostics, and Prognostics Ⅱ	Future I&C Technologies for Nuclear Applications Ⅱ	System Simulation Technologies
18:00~20:00	Banquet (HICO 101~103)		
November 28, 2017 (Tuesday)			
08:00~18:00	Registration		
09:00~09:40	Keynote Speech : Dr. Kook-Hun Kim (Doosan Heavy Industries & Construction) Nuclear I&C: Issues and Way to Go		
09:40~10:00	Coffee Break		
10:00~12:00	Robotics & Automatic Remote Technologies Ⅰ	Special Session:Cyber Security Ⅰ	System Reliability and Risk
12:00~13:30	Lunch Break		
13:30~14:10	Keynote Speech : Prof. Akio Gofuku (Okayama University) Co-operator as an Intelligent Operator Support System for Resilient Operation of NPPs		
14:10~14:20	Coffee Break		
14:20~16:00	Robotics & Automatic Remote Technologies Ⅱ	Special Session:Cyber Security Ⅱ	Special Session:MFM & Safety Culture Ⅰ
16:00~16:20	Coffee Break		
16:20~18:20	Robotics & Automatic Remote Technologies Ⅲ	Special Session:Cyber Security Ⅲ	Special Session:MFM & Safety Culture Ⅱ
November 29, 2017 (Wednesday)			
08:00~18:00	Registration		
09:00~09:40	Keynote Speech : Dr. Chang Hwoi Kim (Korea Atomic Energy Research Institute) Current Status of an Information and Communication Technologies for Nuclear Power Plants		
09:40~10:00	Coffee Break		
10:00~12:00	Wireless Technologies in Nuclear Applications	Safety Critical Software Development and Qualification	Testing and Maintenance
12:00~14:00	Lunch & Best Paper Award (HICO 1st Grand Hall)		
November 30, 2017 (Thursday)			
08:30~13:00	Technical Tour (Korea Radioactive Waste Agency (KORAD))		

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