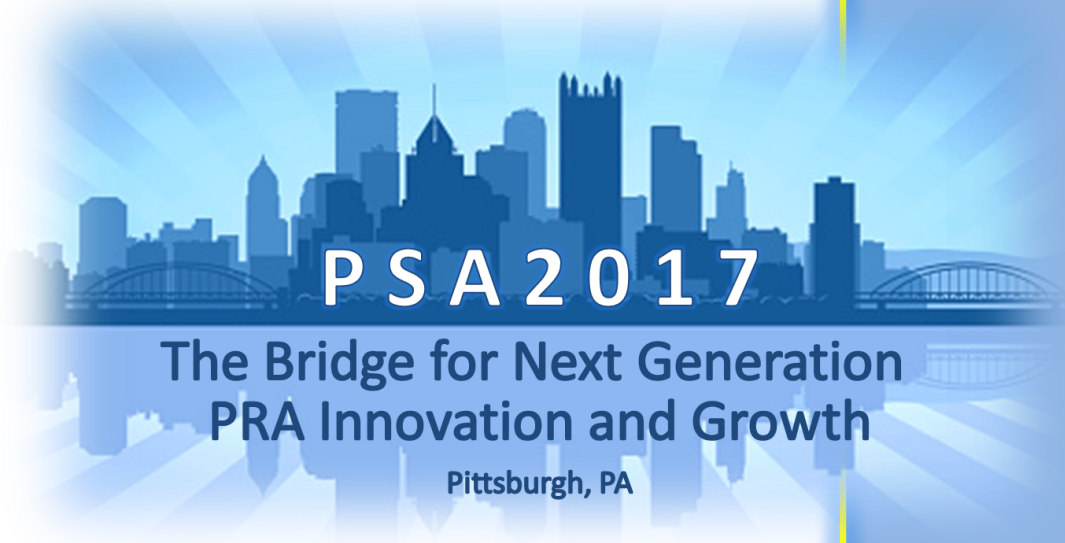


 **ANS** Pittsburgh Section

International Topical Meeting on Probabilistic Safety Assessment and Analysis



PSA 2017

**The Bridge for Next Generation
PRA Innovation and Growth**

Pittsburgh, PA

September 24 – 28, 2017

www.psa2017.org



ANS | PSA 2017

International Topical Meeting on Probabilistic Safety Assessment and Analysis

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Welcome Messages

Welcome message from the General Chair

Welcome to the 2017 International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2017). In my role as key account vice president at Westinghouse Electric Company, I understand the importance of transferring knowledge and skills within our community to continue developing innovative approaches to resolve key challenges facing our industry. This meeting will provide an opportunity to learn from expert presentations and gain knowledge from PSA practitioners with a broad range of experience and expertise.

The theme for this meeting is “The Bridge for Next Generation PRA Innovation and Growth.” I encourage you to consider yourself a pioneer of PSA, no matter how many years you have been working in the field. We hope to see many long-time veterans of PSA as well as early career professionals who are new to this exciting technology.

We need new ideas and perspectives to keep PSA moving forward and continuing to provide value to the nuclear industry. The extensive experience of those who were in the early part of their careers when the Individual Plant Examinations were developed, combined with the energy and vitality of our young professionals, is a key aspect to successfully bring innovative ideas together and to keep growing the value of PSA.

This forum provides opportunity for collaboration, which is critical to assure the safety and long-term sustainability of nuclear power. I care about our industry and its future and hope you are able to participate in this meeting and share your experiences.

Cindy Pezze

Key Account Vice President, Westinghouse Electric Company LLC



Welcome Messages

Welcome message from the International Chair

I am very excited about the 2017 International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2017), whose theme has been selected to be “The Bridge for Next Generation PRA Innovation and Growth.” Indeed, the PSA meeting will provide an extraordinary opportunity to hear old and new problems, learn new methods and exchange experiences and expertise. This is very much needed in the very dynamic world that we live in, where knowledge and technology are advancing at a very fast pace, as we must

appreciate the opportunities and challenges related to ensuring that we continue to search for, and find, innovative solutions to benefit our industry, its safety, and sustainability. To increase and improve the use of PSA for providing value to the nuclear industry, new ideas and developments are needed. Innovation is necessary to continue assuring the required standards of safety and long-term sustainability for nuclear power generation. You can make a difference by participating in the PSA 2017 meeting, sharing your competence and experience with the rest of the participants.



Prof. Enrico Zio

Centrale Supelec and Politecnico Di Milano

Welcome Messages

Welcome message from the Academic Chair

Catastrophic events have made it clear that the integration of physical and social causes of failure into a common systematic modeling framework is essential for the future of risk analysis. There is a need for systematic and theoretical risk analysis frameworks to determine the salient contributing factors (e.g., underlying technological, environmental, and social conditions) that affect risk. Since history has consistently shown that disasters result from dynamic and multi-stranded chains of events, and because accident scenarios must consider the causes of social distress, managerial deficiency, human error and technical system failure, risk analysis requires the development of a common vocabulary within multiple engineering and social science domains in order to address risk emerging from the interface of social and technical systems. The ANS PSA 2017 conference brings together a vibrant and diverse community from industry, governmental agencies and academia to build this common vocabulary and expand upon a shared vision for enhancing risk analysis research and education in order to usher in a new era, void of catastrophic accidents, where scientific discoveries are used to raise social responsibility for the protection of workers, the public, and the environment. Building this community ensures that scientific contributions in risk analysis can continue to benefit complex socio-technical systems by advancing risk informed decision making. As an educator of engineers, social scientists, and policy makers, the PSA conference creates a multidisciplinary environment for uncovering important areas of formal risk analysis training and education, advancing risk analysis knowledge sharing among academia, industry, and governmental agencies by creating an open scientific environment. By promoting risk analysis as its own academic discipline through the ANS PSA conference, we encourage the next generation of risk analysts being introduced into the workforce, thereby contributing to international safety, security, and peace in our global community.



*Prof. Zahra Mohaghegh,
Director of the Socio-Technical Risk Analysis (SoTeRiA) Laboratory
University of Illinois Urbana-Champaign*

Acknowledgement

The PSA2017 Organizing Committee would like to thank the ANS Nuclear Installation Safety Division as the sponsor, the ANS Pittsburgh Local Section as the co-host providing local support, and ANS National for providing support in the review of contracts, advertising and other assistance.

The core of topical meetings is a strong technical program which is the primary responsibility of the Technical Program Chair, Andrea Maioli. I could not thank Andrea more for the exceptional effort and personal time he dedicated to developing this exciting technical program. Prof. Enrico Zio and Prof. Zahra Mohaghegh fostered the Worldwide and academic relevance of the conference, in their role as International Co-Chair and Academic Chair. We also acknowledge the support provided by the Technical Program Committee (TPC) to provide meaningful feedback to our authors. The technical program would not exist if it weren't for all our contributing authors, speakers, panelists and presenters. Thank you for your outstanding technical papers!

It takes a village to organize a large international topical meeting and we couldn't have pulled it off without the help of Melissa Lucci who coordinated the organizing committee activities, Ashlyn Fornear who functioned as finance chair and was responsible for sponsorships, Matt Degonish for website design and AV Support, Tami Stewart / Della DeMaro / Damian Mirizio for logistics, Steven Satter / Richard Roland III for mobile app development, Paige Risley for overall support and Tyler Crummy / Art Wharton / Alex Pingel as our local section liaisons.

Finally, thank you for the significant contributions by the meeting's financial sponsors without whom, none of this would be possible. Please refer to this program and our website for the full list of sponsors.

Organizing Committee

GENERAL CHAIR:

Cindy Pezze (*Westinghouse Electric Company LLC*)

INTERNATIONAL CO-CHAIR:

Enrico Zio (*Centrale Supélec and Politecnico di Milano*)

ACADEMIC CHAIR:

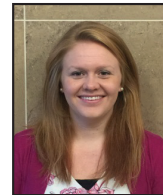
Zahra Mohaghegh (*University of Illinois at Urbana-Champaign*)



ORGANIZING COMMITTEE CHAIR:
Melissa Lucci
Westinghouse Electric Company LLC



TECHNICAL PROGRAM CHAIR:
Andrea Maioli
Westinghouse Electric Company LLC



FINANCIAL CHAIR/SPONSORSHIPS:
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Westinghouse Electric Company LLC



GENERAL SUPPORT:
Damian Mirizio
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GENERAL SUPPORT:
Paige Risley
Westinghouse Electric Company LLC
/ University of Pittsburgh

Technical Program Committee

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TECHNICAL PROGRAM CO-CHAIR: Matthew Degonish (*Westinghouse Electric Company LLC*)

TECHNICAL PROGRAM COMMITTEE MEMBERS:

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Victoria Anderson – Nuclear Energy Institute
Joshua Beckton – Westinghouse Electric Company LLC
Robert Budnitz – Lawrence Berkeley National Laboratory
Daniel Cole – University of Pittsburgh
Ovidiu Coman – IAEA
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Stanley Levinson – AREVA
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Mohammad Pourgol-Mohammad – Sahand University of Technology
Richard Quittmeyer – RIZZO Associates
Paul Rizzo – RIZZO Associates
Eleni Rojtas – JENSEN HUGHES
Ahti Salo – Aalto University
Nathan Siu – U.S. NRC
Barry Sloane – JENSEN HUGHES
Ricky Summitt – RSC
Dave Teolis – Bombardier
Nishikant Vaidya – RIZZO Associates
Dominique Vasseur – Edf
Mark Wishart – JENSEN HUGHES
Clarence Worrell – Westinghouse Electric Company LLC
Yican Wu – Institute of Nuclear Energy Safety Technology
Joon Eon Yang – Korea Atomic Energy Research Institute
Yu Yu – North China Electric Power University
Jing Xing – U.S. NRC
Taotao Zhou – University of Maryland

Daily Schedule

Sunday, September 24

8:00–9:00 am	Registration–Workshop Only	Reflections
2:00–6:30 pm	Registration	Reflections
9:00 am–4:30 pm	RAVEN Workshop	Waterfront
	– Lunch included	Brighton I & II
12:30–4:30 pm	Workshop on Bayesian Inference for PRA	Grand Station III
12:30–4:30 pm	PyCATSHOO Workshop	Grand Station V
6:00–9:00 pm	Welcome Reception	See Welcome Bag for Details

Monday, September 25

7:00 am–4:00 pm	Registration	Reflections
7:00–8:00 am	Continental Breakfast - Chairs and Speakers	Brighton I & II
8:00–9:45 am	Opening Plenary Session	Grand Station II
	Welcome	
	Plenary Lecture #1: Yamaguchi	
9:45–10:00 am	Break	Reflections
10:00 am–Noon	Plenary Lecture #2: Fleming	Grand Station II
	Plenary Lecture #3: Smith	
Noon–1:30 pm	LUNCH	Admiral
1:30 - 3:10 pm	Special session #1 (Bayes)	Grand Station II
	Seismic PSA–I	Waterfront
	External Events Analysis	Grand Station IV
	High Wind PSA–I	Grand Station V
3:10–3:40 pm	Break	Reflections
3:40–5:45 pm	Special session #2 (MUPRA)	Grand Station II
	Seismic PSA–II	Waterfront
	Dynamic PSA–I	Grand Station IV
	Modeling and Simulation–I	Grand Station V

Daily Schedule

Tuesday, September 26

7:30 am–6:30 pm	Registration	Reflections
7:00–8:00 am	Continental Breakfast - Chairs and Speakers	Brighton I & II
8:00–9:00 am	Plenary Lecture #4: Linthicum	Grand Station II
9:00–10:15 am	Special Session #3 (50.69 panel)	Grand Station II
	Seismic PSA–III	Waterfront
	Fire PSA–I	Grand Station III
	High Wind PSA–II	Grand Station IV
	LPSD	Grand Station V
10:15–10:45 am	Break	Reflections
10:45 am–Noon	Risk Aggregation–I	Grand Station II
	Data and Parameter Estimation–I	Waterfront
	Fire PSA–II	Grand Station III
	Risk-Informed Decision Making–I	Grand Station IV
	ATF	Grand Station V
Noon–1:30 pm	LUNCH	Admiral
1:30–3:10 pm	Special Session #4 (ATF Panel)	Grand Station II
	Seismic PSA–IV	Waterfront
	HRA–I	Grand Station III
	Risk Aggregation–II	Grand Station IV
	Dynamic PSA–II	Grand Station V
3:10–3:40 pm	Break	Reflections
3:40–5:45 pm	Mini Workshop: PyCATSHOO	Grand Station V
	Seismic PSA–V	Waterfront
	Data and Parameter Estimation–II	Grand Station III
	Risk-Informed Decision Making–II	Grand Station IV
	Risk-Informed Applications–I (50.69)	Grand Station II
6:00–9:00 pm	Banquet	See Welcome Bag for Details

Daily Schedule

Wednesday, September 27

7:30 am–4:00 pm	Registration	Reflections
7:00–8:00 am	Continental Breakfast - Chairs and Speakers	Brighton I & II
8:00–9:00 am	Plenary Lecture #5: Siu	Grand Station II
9:00–10:15 am	Risk Aggregation–III	Grand Station II
	Seismic PSA–VI	Waterfront
	Human Factors and Behavioral Sciences	Grand Station III
	Level 2 and Level 3–I	Grand Station IV
	High Wind PSA–III	Grand Station V
10:15–10:45 am	Break	Reflections
10:45 am–Noon	Risk Aggregation–IV	Grand Station II
	PSA Standards and Peer Reviews	Waterfront
	Uncertainty Analysis and Modeling	Grand Station III
	Configuration Risk Management	Grand Station IV
	Risk-Informed Applications–II	Grand Station V
Noon–1:30 pm	LUNCH	Admiral
1:30 pm - 3:10 pm	HRA–II	Grand Station II
	Flooding PSA–I	Waterfront
	Fukushima Lessons Learned	Grand Station III
	Level 2 and Level 3–II	Grand Station IV
	Non-Light Water Reactor Safety	Grand Station V
3:10–3:40 pm	Break	Reflections
3:40–5:45 pm	HRA–III	Grand Station II
	Fire PSA–III	Waterfront
	Dynamic PSA–III	Grand Station III
	Level 2 and Level 3–III	Grand Station IV
	Risk Informed Regulation	Grand Station V

Thursday, September 28

7:30–9:00 am	Registration	Reflections
7:00–8:00 am	Continental Breakfast - Chairs and Speakers	Brighton I & II
8:00 am–5:00 pm	TECHNICAL TOUR	
8:00–10:15 am	Passive System Safety and Reliability	Waterfront
	Modeling and Simulation–II	Grand Station III
	Cyber/Security & Cyber/Risk	Grand Station IV
	Fire PSA–IV	Grand Station V
10:15–10:45 am	Break	Reflections
10:45 am–Noon	Advanced Information Technology and PSA	Waterfront
	Safety Culture and Organizational Factors	Grand Station III
	Flooding PSA–II	Grand Station IV
10	Other External Hazards PSA	Grand Station V

General Information

REGISTRATION

The meeting registration desk will be in the Reflections Meeting Room.

REGISTRATION HOURS

Sunday	8:00–9:00 am (Workshop Only)
Sunday	2:00 pm–6:30 pm
Monday	7:00 am–4:00 pm
Tuesday	7:30 am–6:30 pm
Wednesday	7:30 am–4:00 pm
Thursday	7:30–9:00 am

Registration is required for all attendees and presenters.

Badges are required for admission to all events.

Full Conference Registration Fee includes: Sunday Evening Reception, all technical sessions, lunch (Monday – Wednesday), morning and afternoon breaks, Tuesday Evening Banquet, the complete registration package and the proceedings' flash drive.

1 Day Registration Fee includes: All technical sessions, lunch, morning and afternoon breaks, complete registration package, and proceedings' flash drive.

Emeritus Registration Fee includes: Same as full registration.

Student Registration Fee includes: Same as full registration.

Workshops: Workshop registration is separate from PSA 2017 registration and the fees vary. They do not include any of the PSA events Sunday evening through Thursday.

Guests: There is no guest registration. Guests may purchase individual tickets for the following:

- Sunday Evening Opening Reception
- Lunch by the day
- Tuesday Evening Banquet

Westinghouse Technical Tour: The tour will visit the Westinghouse Waltz Mill site and Cranberry Headquarters AP1000® Plant Simulator. The tour will depart from the Reflections Meeting Room at 8 am on Thursday and will return at approximately 5 pm. Lunch is included.

General Information

GUIDELINES FOR SPEAKERS

On the day of your presentation, each speaker should attend the Speakers' breakfast (Brighton I&II, 7:00 am) to meet with the chair of their session. Here the Session Chair will confirm that you will be giving your presentation and will ask for information to be used when introducing your presentation.

Each presentation will be 25 minutes, including time for introduction and questions.

The rooms will be equipped with a laptop, a projector and a laser pointer. Microsoft Windows, Microsoft PowerPoint 2010 and Adobe Acrobat Reader will be installed on the computers.

All presenters should report to the Session Chair in the assigned room 15 minutes before the start of the session. Presentations should be loaded and tested on the computer in the assigned room during the break prior to the session. Presentations may also be tested on similar equipment at the registration desk (Reflections).

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You can also access the PSA2017 guide by scanning the QR symbol below with the built-in QR scanner in the app.



Workshops

SUNDAY, SEPTEMBER 24

Workshop #1 – RAVEN

Time: 9:00 am – 4:30 pm

Room: Waterfront

Workshop on INL RAVEN Software:

RAVEN is a software tool to characterize the probabilistic behavior of complex systems. It might be used for risk analysis, reliability analysis, uncertainty quantification and code validation. In most cases, RAVEN employs a “black box” approach with respect to the external code representing the physical systems (more advanced options like dynamic event trees are also available) and provides sampling strategies to effectively explore the input space. Standard statistical post-processing capabilities are provided to compute mean, variance, etc. of selected figures of merit of the output space. RAVEN relies heavily on artificial intelligence algorithms to construct surrogate models of complex physical systems to perform reliability analysis (limit state surface), uncertainty quantification and parametric studies.

The first objective of the workshop is to acquire a general understanding of the RAVEN package and its main capabilities. Secondly, a series of practical examples are going to be provided, in ascending level of complexity, starting from the simplest statistical analysis to the generation of the complex surrogate models and their utilization in reliability analysis. Users that already have access to the code will be able to run the examples directly on their laptops. Those that do not have access to the software yet, will receive a copy of the example inputs in electronic format. Depending on the Conference room availability “guest” accounts are going to be provided in order to execute workshop examples in a remote server.

The code license may be requested from Cristian Rabiti (cristian.rabiti@inl.gov) or Andrea Alfonsi (andrea.alfonsi@inl.gov). The software (including source) is currently free for non-commercial uses. Commercial usage will be possible in the future under a new license structure.

Workshop #2 – Bayesian Inference for PRA

Time: 12:30 – 4:30 pm

Room: Grand Station III

Workshop on INL RAVEN Software:

This four hour workshop covers the application of Bayesian inference methods in Probabilistic Risk Assessment (PRA). The objective is for participants to be able to describe inference processes as part of PRA applications and to perform hands-on calculations related to the topic. We will describe how to update Bayesian priors and apply tools including Excel and OpenBUGS using the techniques described in the Springer book Bayesian Inference for Probabilistic Risk Assessment (coauthored by the lecturer, Dr. Curtis Smith). In the workshop, we will address a variety of issues related to using probabilistic models for estimating PRA parameters. We will provide background to the analysis framework using a simple fault tree/event tree risk assessment model, then proceed to demonstrate the analysis of varying-complexity problems from traditional conjugate-types of inference through applications including uncertain data and trending. Specific topics of discussion are:

- Introduction to Bayes and Bayesian Networks
- Introduction to OpenBUGS
- Conjugate Calculations using OpenBUGS
- Priors Including Non-Informative Priors
- Non-Conjugate Calculations using OpenBUGS
- Modeling Duration Events such as Off-site Power Recovery
- Bayesian Trending
- Bayesian Regression Models for Fragility Analysis
- Uncertain Data
- Extreme Value Analysis

Attendees are recommended to bring a laptop in order to perform calculations using the software that is provided during the workshop (contact Dr. Curtis Smith, Curtis.Smith@inl.gov, for more information).

Workshops

SUNDAY, SEPTEMBER 24

Workshop #3 – PyCATSHOO

Time: 12:30 – 4:30 pm

Room: Grand Station V

Workshop on a new EDF R&D tool PyCATSHOO for dependability assessment of hybrid systems:

The safety requirements of its nuclear and hydraulic fleet, has allowed EDF to have long-standing experience in using and developing PRA and PSA tools for complex systems.

During PSA 2017 we would like to present to general public our latest development: PyCATSHOO a tool dedicated to dependability analysis of hybrid systems, i.e. systems including deterministic continuous phenomena and discrete stochastic behavior. Currently PyCATSHOO is used at EDF to perform several safety studies. During 2017, EDF will release PyCATSHOO under a freeware license. PyCATSHOO is a C++ written library. It has two APIs (Application Programming Interfaces) in Python and C++. These APIs provide a set of tools, based on distributed hybrid stochastic automata, which help in modelling and assessment of complex hybrid systems.

PyCATSHOO thus combines the power of an object oriented language (Python or C++) and LEGO type modelling approach.

A hybrid stochastic automaton may exhibit random transitions between its states according to a predefined probability law and it may exhibit deterministic transition governed by the evolution of physical parameters.

One can summarize the modelling approach with PyCATSHOO as follows:

- A system is divided in a functional or any other way into elementary subsystems/components.
- Each of elementary subsystem/component is described as a set of hybrid stochastic automata, state variables and message boxes.
- Message boxes ensure message exchange between subsystems/components, that ensures their dependencies
- The system behaviour is simulated using Monte Carlo sampling.
- Sequences that lead to desirable end states are traced and clustered.

PyCATSHOO offers a very flexible modelling framework that allows definition of generic components (classes) that can be reused in different studies. It can greatly reduce model development costs.

During the workshop we will present the tool and we'll work on several practical examples using Python interface.

Workshop is open to general public. Basic knowledge of Python is desired but not required. In order to manipulate the tool we will provide a linux Virtual Machine under Oracle VirtualBox (available for Mac, Windows, Linux) that has to be downloaded on a personal computer.

Plenary Speakers

MONDAY, SEPTEMBER 25

Plenary Lecture #1 – “Confidence in Nuclear Safety under Uncertainties and Unknowns”

Time: 8:45 – 9:45 am **Room:** Grand Station II

Dr. Yamaguchi is a Professor in the University of Tokyo, Nuclear Professional School, Graduate School of Engineering. He has received Ph.D degree in the nuclear engineering from the University of Tokyo in 1984. He joined the Power Reactor and Nuclear Fuel Development Corporation (currently Japan Atomic Energy Agency) and involved in the thermal-hydraulic and safety research of sodium cooled fast breeder reactor. In April of 2005, he moved to Osaka University, Department of Energy an Environment where he performed nuclear thermal-hydraulics, safety and risk assessment studies. Since January of 2015, he is Professor of the University of Tokyo. He has more than 30 years of experience in nuclear engineering. He has been a member of governmental committees on atomic energy policy, nuclear safety, nuclear regulation and nuclear science and technology by the METI, NRA and MEXT. Among them, he chairs the Nuclear Science and Technology Committee of the MEXT and Nuclear Safety, Technology and Human Resource Development Committee of the METI. Currently he is the chair of the Risk Technology Committee of the Atomic Energy Society of Japan and an International Board member of the IAPSAM (International Association of PSA and Management).



The accident in Fukushima Dai-ichi Nuclear Power Plant Station in March 2011 is a remarkable event in the Japanese history of industrial and academic use of nuclear science and technology. The lessons-learned need to be deeply understood and reflected in facilities and activities related to nuclear energy use in Japan and to be shared with the international nuclear community.

It has raised several contradictory paradoxes. It should not be allowed to let an unknown left unresolved; we have recognized there always remain unknowns even after all possible measures have been implemented. If and only if one ensures a nuclear power plant is safe, its restart of commercial operation is accepted; even though a nuclear power plant is in conformity with the regulatory requirements, it is not sufficiently safe. The most interesting notion for us is probably related to the use of the probabilistic risk assessment (PRA). The PRA is very useful for improving safety and ensuring safety; the PRA is only applicable to limited use as the technology and data are incomplete and immature.

Like this, ensuring nuclear safety is a complicated and contradictory notion. Currently five nuclear power plants are in operation in Japan. It seems however, public does not trust the current practice of ensuring nuclear safety and protecting public and environment. Do we use PRA insights in dialog and communication with public? Do we rely on the PRA for ensuring nuclear safety? The PRA is the best available approach to identify and understand uncertainties and unknowns and is a key technology to answer to the complicated and contradictory question. The author believes ensuring nuclear safety implies thorough awareness of uncertainties and unknowns. As a consequence, it is justified and rationalized to continue safety enhancement activities for a nuclear power plant that is already safe enough.

Plenary,
Special
Sessions
and
Technical
Tours

Plenary Speakers

Plenary,
Special
Sessions
and
Technical
Tours

MONDAY, SEPTEMBER 25

Plenary Lecture #2 – “Safety Culture and the One Reactor At a Time Mindset”

Time: 10:00 – 11:00 am **Room:** Grand Station II

Karl Fleming received his B.S. in Physics from Penn State University, Cum Laude in 1969 and his M.S. in Nuclear Science and Engineering at Carnegie-Mellon University in 1974. He has devoted his entire professional career of 45 years to the advancement and application of PRA technology in the nuclear, chemical, process, and aerospace industries. Following professional and executive level assignments at General Atomics, Pickard Lowe and Garrick, and ERIN Engineering and Research he established an independent consulting firm in 2001 and continues to be active in advancing PRA technology.



Mr. Fleming is an internationally recognized expert in the development and application PRA technology. He is a member of the ASME/ANS Joint Committee on Nuclear Risk Management, a co-author of the ASME/ANS PRA Standard, a principal author of the ASME/ANS PRA Standard for Advanced non-LWRs, as well as hundreds of reports, papers, and peer reviewed articles on the development and application of PRA technology to nuclear reactor safety. His 45 years of experience includes more than 30 years in light water reactor (LWR) PRA technology, more than 15 years in high temperature gas-cooled reactor (HTGR) PRA, and extensive experience in applying PRA technology to the aerospace, process, and chemical industries. He was selected by the Advisory Committee on Reactor Safeguards to advise them on technical issues in the advancement of risk-informed decision making. He has given numerous presentations to the ACRS and participated in Commissioner briefings.

His contributions to PRA technology include the development of methods for common cause failure analysis in PRA such as the beta factor, Multiple Greek Letter, and contributing author of the Alpha Factor method, methods for predicting the reliability of piping systems and the influence of inspections, internal fire and internal flooding PRA, PRA of events initiated during low power and shutdown, PRA of multi-unit accidents, PRA database development, probabilistic treatment of severe accident phenomena in Level 2 PRA, and risk-informed applications.

Mr. Fleming's recent contributions include development of an approach to estimate location dependent loss of coolant accident frequencies, co-author of EPRI reports on PRA guidelines and piping system failure rates for internal flooding PRA, and development of methods for evaluating seismically induced multi-unit accidents. He is the principal author of the IAEA Safety Series Report on the Technical Approach to Multi-unit PSA and a contributing author to IAEA TECDOC-1511 and 1804 on technical attributes for PSA Applications.

While there has been some modest progress on this issue in recent years, our safety culture still suffers from can be described as a “one-reactor-at-a-time mindset”. This is true in both the deterministic and probabilistic arenas of safety analysis, although the mindset is arguably more severe in the former. Based on the available evidence even before the Fukushima Daichi accident but now more obvious since that event, the risks associated with multi-unit accidents is certainly significant and may be even dominant at some multi-unit sites. Worldwide more than 80% of the operating reactors sit on multi-unit sites. Although the sharing of systems and structures on multi-unit sites and the issue of seismic fragility correlation may exacerbate this problem somewhat, the fact that all sites share a common electrical grid and ultimate heat sink, and a site-wide susceptibility to most external hazards means that more emphasis needs to be given to multi-unit risk and safety assessment in our safety culture. It is difficult to argue that we have been successful in managing multi-unit risk given we have done little to assess it in our PRAs. There are gaps in the deterministic safety realm that may be more important to address including the application of the defense-in-depth philosophy. In this talk evidence is offered up to support a recommendation that the safety community give much greater priority to addressing this one-reactor-at-a-time mindset.

Plenary Speakers

MONDAY, SEPTEMBER 25

Plenary Lecture #3 – “Computational Risk Assessment”

Time: 11:00 am – 12:00 pm **Room:** Grand Station II

Curtis Smith, Ph.D., is a Directorate Fellow in the Risk Assessment and Management Services Department at Idaho National Laboratory. In this capacity, he is the past project manager for the NRC’s SAPHIRE risk analysis software, serves as a lead instructor and manager for the NRC’s Risk Assessment Training program, is a technical lead for the NASA Safety Mission Success project at INL, and is the Risk Informed Safety Margins Characterization Pathway (RISMC) lead under the DOE Light Water Reactor Sustainability Program



Dr. Smith has been in the risk and reliability assessment field for more than 25 years. He has worked at INL as a risk analysis specialist and has served as a consultant for a diverse set of organizations including the Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), the National Aeronautics and Space Administration (NASA), the International Atomic Energy Agency (IAEA), the Federal Aviation Administration (FAA), and other government and private companies. He also was a visiting scientist to the OECD-sponsored Halden Reactor Project, performing human performance-related research, was the Chair of the ASME Safety Engineering and Risk Analysis (SERAD) Executive Committee, is currently on the Board for the International Association of Probabilistic Safety and Management (IAPSAM) organization, and is a past President of the Idaho State University College of Engineering Advisory Council (EAC). Dr. Smith has published over 200 papers, books, and reports on risk and reliability theory and application. He has taught over 100 technical and university courses on a variety of reliability and safety topics. He holds a Ph.D. in nuclear engineering from Massachusetts Institute of Technology. He is a member of the American Society of Mechanical Engineers, American Nuclear Society, and the Idaho Academy of Sciences.

Assessment is the predictive process that informs decision makers about outcomes, analysis is the process of decomposing an assessment into smaller parts to effectively understand both the analysis parts and the integrated assessment. The Idaho National Laboratory, through the Risk-Informed Safety Margins Characterization (RISMC) project, is demonstrating a next-generation reliability- and risk-assessment method that supports decision-making by combining mechanistic physics-based models with probabilistic quantification approaches. Integrating the two worlds of physics and probability leads us to predictions based upon an approach called “computational risk assessment” or CRA. Several analysis factors are key to the need for a new approach based upon CRA, including: temporal (timing issues), spatial (location issues), mechanistic (physics issues), and topology (complexity issues). We will describe how these factors are being addressed via a new simulation-based risk assessment approach and will provide recent examples highlighting the CRA methods and tools.

Plenary,
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Plenary Speakers

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TUESDAY, SEPTEMBER 26

Plenary Lecture #4 – “PRA Community Support for Delivery of the Nuclear Promise”

Time: 8:00 – 9:00 am **Room:** Grand Station II

Roy Linthicum is the chair of the Risk Management Committee for the PWR Owners’ Group. He has over 30 years Nuclear PRA experience, primarily related to Risk-Informed Applications. Roy has been involved with MSPI and the NEI ROP Task Force since the original MSPI pilot efforts and is a participating member in multiple NEI Risk Related Task Forces. He is also involved in the Risk Informed Operations team for Delivering the Nuclear Promise. Roy has been employed by Exelon Generation since 1999 with previous positions at Sargent & Lundy, Public Service Electric and Gas, Arizona Public Service, Northeast Utilities and Knolls Atomic Power Laboratory. Roy is a graduate of Rensselaer Polytechnic University with a degree in nuclear.



Continued innovation and development of new methods is important to continue to advance the state of the art and improved realism in our risk models and applications. However, we also need to understand that the economic and regulatory environment has changed. Development of new methods and standards needs to be tempered with understanding the cost of implementing these methods and standards and weighed against the benefits to be gained. Given the current economic challenges, new methods and standards should have a specific application in mind before we start down that path. We also need to understand the change in the regulatory mindset towards approving all new methods used in regulatory applications. This adds to the cost and can delay the implementation of new methods. Is this change warranted or are there other more cost effect approaches that can be used? When used in the appropriate way, a smart risk-based culture can help improve performance and reduce operational costs and help deliver the nuclear promise.

WEDNESDAY, SEPTEMBER 27

Plenary Lecture #5 – “PRA R&D–Changing the Way We Do Business?”

Time: 8:00 – 9:00 am **Room:** Grand Station II

Nathan Siu is a Senior Technical Adviser for PRA (Probabilistic Risk Assessment) Analysis in the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission (NRC). He has over 35 years of experience in the development and application of PRA methods, models, and tools and has co-authored over 200 papers and reports. At the NRC, he’s responsible for providing PRA-related advice and support regarding technical programs and issues (including issues requiring research and development) and cooperative activities with U.S. and international organizations. He is also currently involved in broader, strategic discussions regarding the role, performance, and effectiveness of regulatory research at the NRC. He is a past-chair of the OECD/NEA Working Group on Risk Assessment, a Fellow Member of the American Nuclear Society, and a past-member of the International Association for Probabilistic Safety Assessment (IAPSAM) Board of Directors. Prior to joining NRC in 1997, he was on the Idaho National Engineering Laboratory staff (1992-1997), the Nuclear Engineering Department faculty at the Massachusetts Institute of Technology (1986-1992), and the staff of Pickard, Lowe, and Garrick, Inc. (1980-1986). He received his B.S., M.S., and Ph.D. degrees at the University of California at Los Angeles.



Past probabilistic risk assessment (PRA) research and development (R&D) efforts have contributed significantly to the current, widespread use of PRA methods, models, tools, and data in support of safety-related decision making within the nuclear industry. However, times are changing, and PRA R&D practices, as well as topics, may need to change to maintain efficiency and effectiveness. This talk provides a personal perspective on the challenges to and opportunities for the R&D community posed by recent events and trends, considering the varying points of view of the R&D, practitioner, and user communities, and provides a few suggestions for each community.

MONDAY, SEPTEMBER 25
SPECIAL SESSION #1 - 1:30 PM

Bayesian Inference for PRA

Chair: Curtis Smith (*INL*)

Location: Grand Station II **Time:** 1:30 - 3:10 pm

This special 90 minute session covers the application of Bayesian inference methods in Probabilistic Risk Assessment (PRA). The objective is for participants to be able to describe inference processes as part of PRA applications. We will describe how to update Bayesian priors and apply tools such as OpenBUGS using the techniques described in the Springer book Bayesian Inference for Probabilistic Risk Assessment (coauthored by the lecturer, Dr. Curtis Smith). In the session, we will address a variety of issues related to using probabilistic models for estimating PRA parameters. We will provide background to the analysis framework, then proceed to demonstrate the analysis of varying-complexity problems from traditional conjugate-types of inference through applications including uncertain data and trending.



Pictured: Curtis Smith

SPECIAL SESSION #2 - 3:40 PM

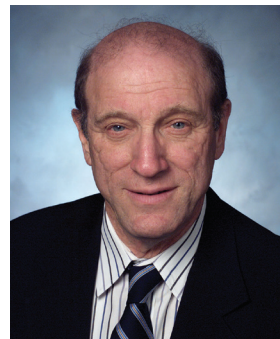
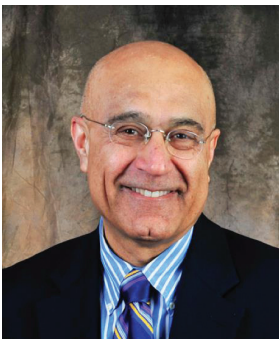
MUPRA Advances, Issues, Impediments and Promise

Panel Moderator: Mohammad Modarres (*Univ of Maryland*)

Location: Grand Station II **Time:** 3:40 - 5:45 pm

Panel: Mohammad Modarres (*University of Maryland*), George Apostolakis (*MIT*), Karl Fleming (*KNF Consulting*), Robert Budnitz (*Lawrence Berkeley National Laboratory*), Nathan Siu (*NRC*), Carlos Lorencez (*OPG*)

The Fukushima Daiichi accident highlighted the importance of risks from multiple nuclear reactor unit accidents at a site. As a result, there has been considerable interest in Multi-Unit Probabilistic Risk Assessment (MUPRA) in the past few years. This panel discusses related issues and developments in MUPRA. Specifically, such topics as risk metrics, risk aggregation, dependency and common cause modeling, safety goals, external event treatments, regulatory and international experiences in the context of multi-unit accidents will be discussed.



**Pictured: Mohammad Modarres, George Apostolakis,
Karl Fleming, Robert Budnitz
Nathan Siu, Carlos Lorencez**

Plenary,
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TUESDAY, SEPTEMBER 26
SPECIAL SESSION #3 - 9:00 AM

Delivering the Nuclear Promise with Risk-Informed Regulations

Chair: Kyle Hope (*Westinghouse*)

Location: Grand Station II **Time:** 9:00 - 10:15 am

Panel: Joseph Giitter (*NRC*), Scot Greenlee (*EXELON*), Brad Adams (*Southern Nuclear*), Roy Linthicum (*PWROG*)

This panel will provide an overview of the potential impact of risk-informed regulations on “Delivering the Nuclear Promise” which is a multi-year strategy to transform the nuclear industry and ensure its viability. A “Business as usual” approach won’t successfully address rising generation costs and increased competition. The DNP team is working to identify ideas that can be implemented quickly and achieve significant reductions in cost while maintaining safety and reliability. Risk-informed applications have a proven track record of reducing organizational impact and cost while maintaining or improving safety. Implementation the 10 CFR 50.69 and Risk-Informed Completion Times applications provide opportunities to go even further. The panel will discuss the current status of risk-informed regulations and applications as well as needs for future regulations that can further expand the application of risk-informed initiatives that can provide new opportunities for cost-effective improvement.



Pictured: Scot Greenlee, Roy Linthicum

SPECIAL SESSION #4 - 1:30 PM

Accident Tolerant Fuel – Panel

Chair: Raymond E. Schneider (*Westinghouse*)

Location: Grand Station II **Time:** 1:30 - 3:10 pm

Panel: Ed Lahoda (*Westinghouse*), Steven Hess (*JENSEN HUGHES*), Robert Rishel (*Duke Energy*)

The nuclear industry has been conducting research and development activities on advanced LWRs for many years. Prior to the events at Fukushima Dai-ichi in 2011, the focus of the fuel improvement efforts was on developing higher burnup fuels for waste minimization, increasing pellet density for power upgrades, plant life extensions and extended cycles and to improve fuel reliability. Following the events at Fukushima Daiichi the Department of Energy (DOE) increased their participation in advanced fuel development, sponsoring an aggressive effort to develop a new generation of fuels with enhanced accident tolerance. In accordance with DOE, ideally, ATF characteristics would include:

- Reduced Hydrogen Generation
- Improved fission product retention
- Improved fuel cladding reaction to high temperature steam
- Improved fuel cladding interactions under extreme conditions

Concurrent with these benefits are overall plant risk reductions and potential increases in typical plant safety margins and operational flexibility. This session will cover various ATF fuel design options, and their expected physical and economic benefits and the role of PRA in identifying these benefits and quantifying cost savings.



Pictured: Ray Schneider, Ed Lahoda, Steven Hess, Robert Rishel

THURSDAY, SEPTEMBER 28

TECHNICAL TOUR - 8:00 AM

Technical Tour of the Westinghouse Waltz Mill and Cranberry Headquarters AP1000® Plant Simulator

Price: \$90 per person

Time: 8:00 am charter departs hotel; 5:00 pm return to hotel.

Lunch included in fee.

Attire: Slacks and shirts and steel toed safety shoes required. If you have no safety shoes, close toed/closed back shoes (no sandals or athletic shoes) are allowed and slip-over safety shoes will be provided to you.

Westinghouse's Waltz Mill facility in Madison, Pennsylvania is the global center of excellence for the company's outage services performed throughout the Americas and Asia and is home to the Westinghouse Standardized Nuclear Unit Power Plant Systems (SNUPPS) plant control room simulator. Employees are responsible for delivering predictable, successful results in reactor and steam generator services; shop, service center and training operations for rotating equipment, reactor coolant pumps and motors; and welding, machining and installation services.

The Westinghouse Cranberry Headquarters Site is a 100-plus acre campus located in Cranberry Township, a northern suburb of Pittsburgh, Pennsylvania. Most members of the Westinghouse senior staff are based at headquarters. Center-led support functions, the Americas region staff and various segments of the four product lines also are located here. This site is also home to our AP1000® plant control room simulator.

Waltz Mill Agenda:

- 9:00 am – Check in at Security at Waltz Mill
- 9:30-10:20 am – Tour of "C" and "D" Bay – Steam Generator Channel Heads and typical Steam Generator Services: Manway removal, nozzle dam installation, eddy current inspection, tube repair and plugging. Reactor Vessel Head (stud tensioners, instrumentation conoseals), Reactor Coolant Pump (motor, pump, seals), Fuel Handling (tools, refueling bridge, upenders, fuel transfer canal).
- 10:30-11:20 am – Tour of Plant Control Room Simulator
- 11:30 am – Lunch

Cranberry Headquarters Site Agenda:

- 1:30 pm – Arrival / Check in at Security at Cranberry
- 1:45 – 2:15 pm – Break at Cranberry Headquarters Cafeteria
- 2:30 – 4:00 pm – Tour of AP1000® Plant Control Room Simulator
- 4:15 pm – Depart for Hotel



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Westinghouse Waltz Mill Facility
Madison, Pennsylvania



Westinghouse Waltz Mill Facility
Madison, Pennsylvania

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MONDAY, SEPTEMBER 25
TECHNICAL SESSIONS - 1:30 PM

External Events Analysis

Chair: Martina Kloos (GRS)

Location: Grand Station II **Time:** 1:30 - 3:10 pm

1:30 pm: Effective Use of PRA Walkdown Information for Integrated External Hazard Risk Model Development, Richard Anoba (JENSEN HUGHES)

In the aftermath of the Fukushima incident, there has been increasing interest in addressing the integrated impact of external hazards. The Fukushima incident demonstrated the direct impact of a seismic event was secondary to the consequential flooding of an entire nuclear site. An integrated external hazard risk assessment would explicitly address the direct effects of an external event (i.e., seismic, high winds, external flood, etc.), as well as the indirect impact to the nuclear plant site from consequential events such as flooding and fires. Consequently, Probabilistic Risk Assessments (PRAs) are increasingly being used as a tool for addressing integrated external hazards at nuclear power plants.

The objective of this paper is to provide a summary of selected technical issues related to the use of PRA walkdown information to develop integrated external hazard risk assessment models, including suggested approaches for cost-effective measures to control the scope of the model development effort. The paper will also provide an example method for implementing the integrated external event impacts into an existing PRA model.

1:55 pm: Determination of River Water Level Exceedance Frequency Curves, G. M. Schoen, R. C. Hausherr, A. Ramezani (ENSI)

Flood level exceedance frequency curves are used in the Probabilistic Safety Analysis (PSA) for nuclear power plants (NPPs) or to determine a frequency-dependent design basis for critical infrastructures. Developing such hazard curves involves the consideration of many attributes such as analysis of measured data, assessment of historical flood events, extreme value statistics, evaluation of (interacting) flood phenomena (e.g., flood-induced breaks or blockages of water control structures, landslides, sediment transport) and deterministic calculations of flood levels using numerical simulation tools.

In this paper, an approach is presented which incorporates deterministic and probabilistic calculations as well as the various flood phenomena governing the hazard curves. The proposed approach allows using common tools and to consider phenomena (such as a failure of a large dam or potential new retention areas in extreme events) which may occur but are not covered by the measured data. The calculation flow is illustrated by an example.

2:20 pm: NPP Control Room Habitability Assessment due to an Accidental Toxic Release from an External Event Applying Probabilistic Approach. Part 1: Atmospheric Dispersion, Antonio Gomez, Guillem Cortes, Alfredo de Blas (UPC)

The studied external event consists of an accidental release of toxic gas coming from a rail shipment. For this kind of event, RG 1.78 recommends to apply screening criteria of different parameters that influence on the amount of released hazardous material. In case of relevant values, the atmospheric dispersion must be evaluated by conservative codes considering the worst conditions.

The goal of this study is to obtain additional information about Control Room (CR) vulnerability, by applying a probabilistic methodology.

The selected methodology is Monte Carlo (MC) simulation, which also informs about importance of different combinations of initial and boundary conditions.

Due to the complexity of the analysis, the rail accident and the atmospheric dispersion phenomena have been studied separately. In this paper, part 1, all possible meteorological scenarios that can transport toxic vapors from the different release points to the control room intake of a Spanish Nuclear Power Plant (NPP) are analyzed.

Results provide with a probability density function (PDF) of toxic concentration values in the CR, which combined with train accident frequencies, allow to obtain frequency a year that the Control Room Habitability (CRH) is compromised. The results also indicate that the critical values do not necessarily come from the nearest points to the plant, concluding that the probabilistic approach, in any case, returns useful information about safety.

2:45 pm: A PSA for External Events Endangering Water Intake from the River Danube at NPP Paks, Tamas Siklossy, Attila Bareith (NUBIKI Nucl Safety Research Inst)

The external events probabilistic safety assessment for the Paks NPP, Hungary has recently been extended with an analysis of events that can lead to loss of ultimate heat sink due to the discharge of dangerous substances into the river Danube. Those substances that were considered dangerous in the analysis were those that can directly or indirectly disable water intake from the river. To the extent seen feasible (1) an exhaustive list of external hazards was developed with the associated possible endangering events, (2) dangerous substances were described in terms of chemical, physical and biological characteristics, and (3) those events were selected that needed detailed PSA modeling and risk quantification. Four types of dangerous substances were identified as a result of screening analysis: crude oil or oil by-products floating under water surface, toxic substances causing large scale fish devastation in the Danube, grains and river vegetation. The accident sequence models developed describe the consequences of losing the essential service water supply due the various external events analyzed. This paper gives a concise description of the analysis process focusing primarily on hazard assessment, and it discusses the most important analysis results and findings.

MONDAY, SEPTEMBER 25

TECHNICAL SESSIONS - 1:30 PM

High Winds PSA—I

Chair: Kyle Hope (*Westinghouse*)

Location: Grand Station V **Time:** 1:30 - 3:10 pm

1:30 pm: Integrated Use of Modeling and Simulation in High Winds PRA, Stephen M. Hess (*JENSEN HUGHES*), Steven Prescott, Curtis Smith (*INL*), Linyu Lin, Nam Dinh (*NCSU*), Ramprasad Sampath, Niels Montanari (*Centroid Lab*)

It is generally recognized that high winds pose a significant externally induced hazard to nuclear power plant (NPP) sites. In general, the high wind hazard occurs in three distinct forms: (1) tornados, (2) tropical storms (i.e. hurricanes / typhoons), and (3) straight-line winds (e.g. from thunderstorms and extratropical cyclones). In response to the accident at the Fukushima Dai-ichi NPP, many regulatory authorities and NPP owner / operators are reevaluating NPP vulnerabilities to high wind hazards. Additionally, plant operating experience has shown that high wind events have been responsible for extended losses of offsite AC power (ELAP) events at several NPP sites while simultaneously impacting the plant's ability to cope with these events. An important conclusion obtained from such reassessments is the need to develop methods and tools that can support the assessment of the high wind hazard (including the effects of wind borne missiles) and its effects on NPP risk and safety in a manner that is both efficient and cost effective.

1:55 pm: Plant Walkdown Guidance to Support High Winds Probabilistic Risk Assessments, Nicholas Lovelace, Kelly Wright, Leo Shanley (*JENSEN HUGHES*), Hasan Charkas (*EPRI*)

High winds pose a potential hazard to nuclear power plant (NPP) sites in the form of tornados, hurricanes, and straight winds. Because of the accident at the Fukushima Dai-ichi NPP, analyzing the impact of external hazards on plant risk and safety has become a high priority for both NPP owner operators and regulatory authorities.

The impact of high wind events on NPPs can be assessed in the context of a high winds probabilistic risk assessment (HW PRA). This analysis consists of three specific engineering evaluations.

- *Conduct of a wind hazard analysis to determine the occurrence frequency of winds at various speeds. The results of this analysis are the development of one or more hazard curves.*
- *Evaluation of the strength and response of plant structures, systems, and components (SSCs) to the direct effects of the winds and of wind-borne missile loads to determine the failure probabilities of the SSCs due to these factors. The results of this analysis are SSC fragility curves.*
- *Development of a quantified PRA model to evaluate the risk incurred by the NPP that is associated with the high wind hazard. The results of this analysis are the contribution to plant core damage frequency (CDF) and large early release frequency (LERF) due this hazard*

Similar to the evaluation of other external hazards (e.g. seismic and flooding hazards) a fundamental component of a HW PRA is to identify those plant SSCs which may be vulnerable to the high winds hazard. Additionally, it is important to identify those characteristics of the NPP site that, upon experiencing a high wind event, could generate debris that becomes entrained within the wind field and generates high energy projectiles (wind-driven missiles) that could damage other plant SSCs and impact plant safety. It has been found that the most expeditious and effective approach to obtaining the relevant information to permit development of a HW PRA is to conduct a plant walkdown.

Guidance developed by the Electric Power Research Institute (EPRI) to permit the efficient conduct of a plant high winds walkdown is reported on in this paper. This guidance document was created based on experience obtained from the recent development of HW PRAs at several operating NPPs. The process of creating a plant-specific high winds equipment list (HWEL) is provided in step-by-step format, starting with the internal events PRA model used to generate an initial HWEL which is then validated (with suitable modifications) based on information obtained during the conduct of a plant walkdown. The walkdown guidance also provides detailed instructions with extensive examples for plant personnel to perform a detailed survey of plant SSCs, to identify specific SSC vulnerabilities, and conduct a site missile survey. To support the efficient conduct of the site walkdown additional guidance is given for the following logistic elements:

- *the necessary skill sets for walkdown team members,*
- *use of the HWEL in the conduct of the site walkdown,*
- *the data to be collected, with several examples provided of vulnerable SSCs, potential missiles, and wind and missile barriers.*

To demonstrate the generic applicability of the guidance, three pilot walkdowns were performed at both BWR and PWR sites, and one at a non-US (international) NPP site. Lessons learned and improvements to the guidelines obtained from the pilot plant walkdowns were incorporated in the EPRI guidance document.

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TECHNICAL SESSIONS - 1:30 PM

High Winds PSA—I Continued

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2:20 pm: Excessive Risk Conservatism in High Wind PRA, Mohammad Hadi Hadavi (Duke Energy)

High winds PRA involves a significant level of uncertainty resulted from complexity and unpredictability of high wind events coupled with the plant layout. The existing high wind PRAs show the indirect structural interaction failures heavily influencing the core damage frequency (CDF). There is no industry standard or analysis tool for modeling majority of the wind induced structural interaction failures.

The analysis of high wind induced accident scenarios shows that if these indirect failures could be eliminated or significantly reduced, the CDF would be reduced substantially. Usually conservative, simplified code-based methods are used for estimating fragilities of structures. These calculations assume simple geometries and do not explicitly treat a number of influential factors that provide additional margin.

Duke Energy nuclear fleet high wind PRAs have helped to identify these factors and to uncover existing over-conservatism. This paper discusses a number of factors that are important but are not taken into consideration in current high wind PRAs. A systematic qualitative approach is suggested to complement the quantitative methods leading to a more realistic and less conservative PRA. The qualitative risk evaluation could prevent unnecessary and expensive plant modifications as well as expensive detailed analyses that would eventually confirm the results of the qualitative analysis.

2:45 pm: Improved Tornado Missile Risk Analysis Using Nonlinear Finite Element Analysis of Nuclear Power Plant Structures, Claudia Navarro-Northrup, Robert T. Bocchieri, Virginia Phan, Jeffrey C. Sciaudone, Lawrence A. Twisdale (Applied Research Associates, Inc.)

The TORMIS computer code was developed to estimate the probability of damage to nuclear power plant structures from missile impacts in extreme winds. TORMIS uses a methodology to predict the probabilities of damage to nuclear power plant structures by tornado propelled missiles. The risk probabilities are estimated with Monte Carlo techniques.

An important component of performing this risk analysis is determining the velocity at which critical damage results for each power plant target with a given missile. For some target-missile combinations, test data and simple analytical methods exist to predict damage. However, as the targets and missiles become more complex, the simple analytical methods can produce significant errors in estimating critical velocities. Using nonlinear dynamic finite element analyses to explicitly model the target-missile interaction gives a more accurate representation of the impact damage compared to simplistic loading methods. More accurate calculations lead to higher critical missile velocities, lower risk numbers, and thus cost savings to a power plant in avoiding costly and unnecessary retrofits.

This paper details LS-DYNA finite element analyses that were conducted to determine the critical orientations, impact locations, and velocities for various missile-target combinations. Missiles varied in complexity from wide flange beams (hard missile) to steel grating and metal siding (soft missiles). Assumptions within the target and missile material modeling, missile impact locations and orientations as well as the response of constraining target structures are used to ensure that the results are realistic, but conservatively defensible.

Seismic PSA—I

Chair: Andrea Maioli (Westinghouse)

Location: Waterfront **Time:** 1:30 - 3:10 pm

1:30 pm: Methodology of Treatments of Multiple Failure Initiating Events for Seismic PRA (1) Establishment of Analysis Methodology and Trial Analysis of Core Damage Frequency, Yuki Kameko, Tatsuya Kunishi, Ken Muramatsu, Hitoshi Muta (Tokyo City Univ)

At the time of an earthquake, there is a possibility of multiple failure initiating events. Damage of emergency prevention or mitigation systems due to seismic motion could occur coincidentally or sequentially. In the current seismic PRA, the hierarchical event tree is used to evaluate the occurrence frequency of the initiating event. However, there should be the problem that doesn't consider explicitly the occurrence of the multiple failure initiating events that include the simultaneous occurrence of fracture of the safety injection system piping and failure of the safety injection systems.

The object of this study is to establish the method of multiple failure initiating events analysis which solves this problem.

MONDAY, SEPTEMBER 25

TECHNICAL SESSIONS - 1:30 PM

Seismic PSA—I Continued

1:55 pm: Quantification of Plant-Level HCLPF Capacity in PSA-Based SMA, Zhaoliang Wang, Smain Yalaoui, Yolande Akl (*Canadian Nucl Safety Commission*)

PSA-based Seismic Margin Assessment (SMA), compared to the EPRI SMA and to the NRC SMA, is a more comprehensive and rigorous method for evaluating the seismic margin of nuclear power plants, defined by the plant-level “High Confidence of Low Probability of Failure” (HCLPF) capacity. Two quantification methods have been used for evaluating the plant-level HCLPF capacity in a PSA-based SMA: the min-max method and the convolution method, respectively. These quantification methods, however, differ substantially in concept, rigor, accuracy, and efficiency.

To facilitate the use of quantification methods in a PSA-based SMA and to make substantiated discussions, this paper presents a comparative study to evaluate the performance (i.e., accuracy and efficiency) of the quantification methods. Both simple systems having basic union or intersection logic, and realistic plant systems having complex core damage Boolean logic are evaluated. The results of the system/plant-level HCLPF capacity are compared based on the following computational methods (cases): (1) the min-max method; (2) the convolution method with input of the mean SSC fragilities; and (3) convolution method with input of the entire family of SSC fragilities. Based on the comparative study, the use of different quantification methods in a PSA-based SMA is discussed.

2:20 pm: Insights from the Application of the Hybrid Approach to Seismic Human Reliability Analysis at the Oconee Nuclear Station, Erin P. Collins, Elizabeth Cook (*JENSEN HUGHES*), Scott Hollingsworth, David Garland, Russell Childs (*Duke Oconee Nucl Station*), Paul J. Amico (*JENSEN HUGHES*)

This paper reports on the application of a hybrid approach for the Oconee Nuclear Station (ONS) Seismic PRA (SPRA), which is underway. Some initial insights that have been identified include:

- *the need for a mindset shift by the PRA analysts from internal events, fire or other external events, since seismic introduces a specific set of conditions, influences and uncertainties that need to be addressed as best as possible,*
- *the importance of correctly identifying the SSCs whose failure will affect human performance,*
- *structuring the SSCs into bins of increasing impact on the performance of the operators,*
- *carefully defining the PSF impacts to apply to these bins,*
- *making the PSF impacts sufficiently conservative that they envelope the impacts, since they will be applied to all of the HFEs in the SPRA (otherwise, there is a risk that the HEPs could possibly increase when going from screening to detailed analysis). This is consistent with the need to evaluate the significant uncertainty in the analysis and how to best characterize it in the HRA.*

An important key to this process is obtaining plant-specific information through operator interviews. These interviews were conducted early in the process, which helps to assure that the bins were stable throughout the modeling and quantification effort. Another important aspect is the need to have good fragility estimates so that the bins can be properly linked to the seismic events defined in the model and also that the bin hierarchy is known. These fragility estimates in the case of the ONS SPRA were initially rough estimates and so as the project progressed, it became necessary to alter the definitions of the seismic events so that they match up well with the bins. Finally, the process works well in limiting the need for detailed HRA and associated logic modeling. This tends to be required only for certain bins, and in many cases only for certain actions within a bin; however there are some techniques that are needed in order to make this modeling work.

Finally, the decision on the part of the analysis to delay the consideration of FLEX as a response mechanism available to the operators initially limited the efficacy of interviews and the analysts' ability to postulate a realistic response. In retrospect, it would seem to be more prudent to include the FLEX equipment and procedures up-front in the SPRA/SHRA.

2:45 pm: Methodology of Treatment of Multiple Failure Initiating Events for Seismic PRA (2) Success Criteria Analysis for Multiple Pipe Break Accidents of a PWR, Tatsuya Kunishi, Hitoshi Muta, Ken Muramatsu, Yuki Kameko (*Tokyo City Univ*)

Since the occurrence probability of multiple pipe rupture in pressurized water reactors (PWRs) is low, analytical or experimental investigation for termination of such accidents is not performed explicitly. Therefore, thermal-hydraulic analysis of the plant behavior under the primary system pipe, secondary system pipe, steam generator tube in all loops or in a single loop with loss of more than one safety injection systems were made to clarify the behavior during accidents described above in this study with use of the code RELAP5-3D, and investigations for accident management (AM) of such accidents were made in order to contribute to continuous risk reduction efforts in the future.

In this study provide the thermal hydraulic behavior and success criteria for core cooling of Multiple Failure Initiating Events for Seismic events.

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TECHNICAL SESSIONS - 3:40 PM

Dynamic PSA—I

Chair: Francesco Di Maio (*Politecnico di Milano*)

Location: Grand Station IV **Time:** 3:40 - 5:45 pm

Technical Sessions: Monday September 25

3:40 pm: Dynamic Event Tree Generation With RAVEN-MAAP5 Using Finite State Machine System Models, Claudia Picoco, Tunc Aldemir (*Ohio State Univ*), Valentin Rychkov (*EdF*)

The Dynamic Event Tree (DET) methodology has been developed in order to overcome the limitations of traditional event tree/fault tree approach. One of the advantages of the DET methodology is the capability to model possible complex interactions among hardware/ software/process/human behavior by explicitly accounting for the time element in system evolution.

After Fukushima accident there has been a rising interest for long term analysis of accident evolution (greater than 24 hours). The nuclear industry has developed finite state machine (FSM) models that can probabilistically describe possible changes in the system configuration throughout the accident duration. However, the FSM models do not explicitly account for the physical evolution of the system during this time which may affect FSM transition times.

A framework is presented to enrich the FSM models with information on the physical evolution of the plant by converting them into DETs. The framework is applied to loss of offsite power in a pressurized water reactor. The MAAP5 code is used to simulate the plant behavior. DET generation is accomplished through coupling MAAP5 with the RAVEN code under development at the Idaho National Laboratory.

4:05 pm: Local Fusion of an Ensemble of Semi-Supervised Self Organizing Maps for Post-Processing Accidental Scenarios, Francesco Di Maio, Roberta Rossetti, Enrico Zio (*Politecnico di Milano*)

Integrated Deterministic and Probabilistic Safety Analysis (IDPSA) of dynamic systems is challenged by the need of implementing efficient methods for accidental scenarios generation (that are to be increased with respect to conventional PSA, due to the necessary consideration of failure events timing and sequencing along the scenarios) and for their post-processing for retrieving safety relevant information regarding the system behavior (that, in the context of IDPSA consists in the classification of the generated scenarios as safe, failed, Near Misses (NMs) and Prime Implicants (PIs)). The large amount of generated scenarios makes the computational cost for scenario post-processing enormous and the retrieved information difficult to interpret. To address this issue, in this paper we propose the use of an ensemble of Semi-Supervised Self Organizing Maps (SSSOM) whose outcomes are combined by a locally weighted aggregation: we resort to the Local Fusion (LF) principle for accounting the classification reliability of the different SSSOM classifiers, for the type of scenario to be classified. The strategy is applied for the post-processing of the accidental scenarios of a dynamic U-Tube Steam Generator (UTSG).

4:30 pm: IDPSA Approach to Assess the Potential of a Thermally Induced Steam Generator Tube Rupture, Martina Kloos, Joerg Peschke (*GRS*)

IDPSA (Integrated Deterministic Probabilistic Safety Analysis) is a complementary analysis to the classical deterministic (DSA) and probabilistic (PSA) safety analysis and helps to thoroughly investigate the influence of aleatory and epistemic uncertainties on the behavior of a complex dynamic system. An appropriate tool for IDPSA is MCDET which allows for performing Monte Carlo (MC) simulation, Dynamic Event Tree (DET) simulation or a combination of both. The efficient link between a deterministic computer code for system dynamics simulation and appropriate probabilistic models which can be realized by MCDET essentially facilitates the simulation of the inherent interactions of a complex dynamic system in the presence of uncertainties. An extra Crew Module permits to calculate time-dependent human action sequences interacting with the system dynamics and with any other influencing factor. The MCDET capabilities have already been demonstrated by several applications. The current application case aims to assess the potential of a thermally induced steam generator tube rupture during a high pressure scenario in a pressurized water reactor. The paper gives an overview on this application case.

4:55 pm: Dynamic Approach on Multi-Unit Probabilistic Risk Assessment Using Continuous Markov and Monte Carlo Method, Sunghyon Jang, Akira Yamaguchi (*Univ of Tokyo*)

A risk assessment of multi-unit by a typical event tree (ET) method is insufficient because the state of the plant will vary by a plant status of the adjacent unit. In this study, a new approach of scenario quantification method of multi-unit Nuclear Power Plants (NPPs) using Markov process and Monte Carlo method is proposed to evaluate interactive time-transient accident scenario progression which considered the effect of an adjacent unit. The impact (positive and negative effect) of the adjacent unit (multi-unit) on a single unit is modeled to change failure probability of each heading in the ET. Markov process is adopted to estimate transient plant status using a fault tree (FT) analysis. Markov process decides the plant status at the present time step only based on the status of the previous time step. Also, Monte Carlo method is used to decide the current plant status by comparing a random variable with the transient plant state. Rather than giving branch probabilities in the ET, the current failure probabilities of each heading of the ET are used to decide accident progression. By using this methodology, the influence of multi-unit and dynamic and interactive accident scenario progression can be evaluated. The new methodology applied for a scenario quantification of the primary containment vessel (PCV) failure event. It is showed that the new methodology is an effective approach to evaluate the interaction among multi-units and the particular time sequence of each unit.

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TECHNICAL SESSIONS - 3:40 PM

Dynamic PSA—I Continued

5:20 pm: Surrogate Model Selection in RAVEN for Seismic Dynamic PRA/PSA, Brian Cohn, Richard Denning, Tunc Aldemir, Jieun Hur, Halil Sezen (*Ohio State Univ*)

Surrogate models are used to approximate the response of a large system model based on a number of training runs. Surrogate models allow analysts to more rapidly determine the impact of uncertainties on the system figures of merit of interest than sampling uncertainties on the actual system models, as surrogate models are less complex than the models they are trained on. However, depending on the surrogate model construction scheme used, systematic errors in the results may occur when analyzing seismic effects. It is shown that it is possible to identify if a seismic surrogate model approximates a system well or not by training the surrogate model on a small number of runs.

Modeling and Simulation—I

Chair: Diego Mandelli (*INL*)

Location: Grand Station V **Time:** 3:40 - 5:45 pm

3:40 pm: Implementation of the RCP SHIELD Seal Model in the Comanche Peak PRA, Nathan Larson, Daniel Tirsun, Aaron Moreno (*Westinghouse*)

The reactor coolant pump (RCP) SHIELD®[1] seal provides for many plant safety and Probabilistic Risk Assessment (PRA) benefits due to its potential to prevent an RCP seal loss of coolant accident (LOCA). Typically an RCP seal LOCA represents a significant contributor to the overall risk profile of a Westinghouse designed PWR. The SHIELD seal can have a dramatic impact on the plant PRA results for core damage frequency (CDF) and/or large early release frequency (LERF) by reducing the risk contribution of RCP seal LOCAs.

During the implementation of the SHIELD seal in the Comanche Peak PRA model various lessons learned and insights were noted. These lessons include: the impacts of asymmetric secondary heat removal, loss of all secondary heat removal, impacts to the recovery of offsite power, along with various other in process lessons. This paper will explore each of these lessons learned in more detail and provide an overall picture of the risk impact that was incurred as a result of installing the RCP SHIELD seal.

4:05 pm: Coupling Smoothed-Particle Hydrodynamics and Torricelli's Law-Based Hydraulic Models for Flooding Risk Analysis, Niels Montanari, Ramprasad Sampath (*Centroid Lab*), John E. Weglian (*EPRI*), Robert J. Wolfgang, Donald A. Dube (*JENSEN HUGHES*), Curtis L. Smith, Steven Prescott (*INL*)

Probabilistic risk assessments involving flood require accurate estimations of the time until critical equipment are reached by the water. Conventional approaches for assessing flooding risks employ a variety of simplifications on the water dynamics and geometries involved, limiting their accuracy and reliability. Conversely, 3-D fluid modeling methods make it possible to obtain exploitable data in the highest amount, variety and accuracy, but are associated with a significant computational cost and a limited spatial resolution. We present a hybrid approach, leveraging on the strengths of both previous approaches. An innovative and flexible coupling is realized between: a conventional hydraulic model, based on macroscopic balances and a generalized form of Torricelli's law; and a 3-D fluid model, solving the Navier-Stokes equations with smoothed-particle hydrodynamics. We demonstrate for an internal flooding scenario the benefits of this methodology, making use not only of several kinds of one-way coupling but for modeling flows under doors and through draining systems as well. It is able to provide a significantly more complete, accurate and reliable characterization of the flooding risks than the conventional methods, while keeping the computational trade-off at a moderate level.

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Modeling and Simulation—I Continued

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4:30 pm: Convolution Correction Factor Adjustments on Static PRA Models for Event Assessment, James Knudsen, Ted Wood, Steven Prescott, Curtis Smith, John A. Schroeder (INL)

Current probabilistic risk assessment (PRA) models contain cut sets that have time-constrained basic events such as switched components, recovery of failed components, and failures in time. For these cut sets that have multiple time-constrained events that interact (e.g., offsite power recover at the same time as multiple diesel generators fail to run), there is the possibility that the minimal cut set quantified result is either conservative or non-conservative. Since current PRA models typically assume that the accident scenario starts at “time zero,” the fails-to-run basic events use this starting time as the basis for their respective reliability models. Over time, the standard practice was to recognize this issue and accept it as part of the approximations built into the event tree/fault tree PRA models. However, now that these same PRA models are being used in a regulatory forum, this approximation needs to be addressed. Further, this issue exists not only for the baseline PRA results, but for deficiencies that have been observed or those that have the potential of occurrence, whether the deficiency is complete failure or just a degraded condition. It is found that this approximation becomes very important when evaluating scenarios such as station blackout sequences and other loss of offsite power sequences. This paper will address the theory and implemented process used by the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) PRA software.

Starting with SAPHIRE version 8, the process that was developed is to calculate a “convolution correction factor” based on three distinct calculations. The first calculation is the cut set based frequency calculated via the standard event tree/fault tree PRA model (assuming all run failures start at time equal to zero). The second calculation is to perform the exact time-constrained calculation by integrating the component failure models over the time period of interest (e.g., time to recover, a mission time). For example, this exact calculation for the case of the diesel generators is to convolve their failure models with the offsite power recovery model to obtain the exact probability. The third calculation is to produce the convolution correction factor by dividing the exact probability calculation by the standard cut set probability. This convolution correction factor is then applied to the PRA cut sets in order to remove the time-constrained conservatism.

The process used to obtain the convolution correction factor for the baseline PRA is important; however, in this paper also describe what happens when there is degradation of a component or a potential deficiency. These issues become very important in order to ensure that the convolution correction factor is re-calculated correctly. If the correction factor does not get re-calculated and the same baseline convolution correction factor is used, the final analysis result could be over conservative or under conservative. This paper will discuss how these issues are addressed and provide examples on how SAPHIRE automatically makes the necessary adjustments. MathCad verification results can be computed using the provided equations and input data.

4:55 pm: Accidents, Near Misses, and Probabilistic Analysis: On the Use of CCDPs in Enterprise Risk Monitoring and Management, Nathan Siu, Kevin Coyne, John Nakoski, Christopher Hunter (NRC)

Post-Fukushima estimates of core damage frequency based on global accident statistics have renewed old arguments regarding the use of experience from operational events, including precursors as well as accidents, in estimating risk. This paper discusses the motivation for using precursor information to help the U.S. Nuclear Regulatory Commission assess and communicate its performance in the execution of enterprise risk monitoring (a key element of enterprise risk management); the history, status (including concerns), and potential utility of current approaches; and a number of potential alternatives to these approaches. We find that although precursor-based indices are not good estimators of average, industry-risk, they can be useful for enterprise risk management. We also observe that a formal analysis of the added value of such indices will require a more formal articulation of intended use, and that this articulation will likely have broad-reaching benefits for the agency's operational experience programs.

5:20 pm: Global Importance Measure Methodology for Integrated Probabilistic Risk Assessment, Tatsuya Sakurahara, Zahra Mohaghegh, Seyed Reihani, Ernest Kee (Univ of Illinois)

In this line of research, the authors have developed an advanced PRA methodology, the Integrated PRA (I-PRA) framework, which explicitly incorporates the underlying failure mechanisms into PRA scenarios by integrating the spatio-temporal simulation of underlying physical and social phenomena with classical PRA. The focus of this paper is on developing an Importance Measure (IM) method for I-PRA. The classical IM methods (e.g., Fussell-Vesely IM and Risk Achievement Worth), which are common in the PRA field, are not adequate for I-PRA because they only focus on the risk ranking of components. In I-PRA, the risk importance ranking of input parameters within the simulation models needs to be analyzed and, for that purpose, a moment-independent Global IM, the cdf-based sensitivity indicator $S_i^{(CDF)}$, is selected and tailored for the I-PRA framework. This IM method can capture three key aspects of the I-PRA model: (i) uncertainty associated with the input parameters, (ii) uncertainty of risk outputs, and (iii) non-linearity and interactions among input parameters within the simulation model. This paper shows the progress of the ongoing research, and a case study using a reduced-order I-PRA to demonstrate the feasibility of implementing the Global IM method in a realistic PRA application, is presented.

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TECHNICAL SESSIONS - 3:40 PM

Seismic PSA—II

Chair: Vincent Andersen (*JENSEN HUGHES*)

Location: Waterfront **Time:** 3:40 pm - 5:45 pm

3:40 pm: Probabilistic Fault Displacement Hazard Analysis for Regulatory Decision-Making: A Case-Study for Krško, Slovenia, Richard Quittmeyer (*RIZZO Associates*), Milos Bavec (*Geološki zavod Slovenije*), Igor Riznar (Consultant), Ali Fatehi, Jose E. Blanco, Paul C. Rizzo (*RIZZO Associates*), Mojca Planinc (*GEN energija*)

Probabilistic fault displacement hazard analysis provides information that can be used for rational decision-making in a regulatory environment. A case-study is provided examining displacement hazard for the Krško vicinity of Slovenia. Two sites adjacent to an existing nuclear power plant at Krško are being considered for a second nuclear power plant. Although no active faults have been identified at the immediate sites, the possibility of such faults near the sites is a suitability issue. A probabilistic fault displacement hazard analysis for the sites provides hazard information that can be used to evaluate the potential safety impact of fault displacement relative to other possible hazards. Results show the mean annual exceedance frequency for displacement of engineering significance, taken as 5 cm, is less than about 2×10^{-8} . This low value reflects the fact that faults included in the model generally do not intersect the site areas, have relatively short lengths leading to assessed values of maximum magnitude that are not associated with large on- or off-fault surface displacements, have low slip rates, and are characterized in many cases by uncertainty regarding whether they are active. Sensitivity analyses show that the results are most sensitive to whether a fault passes through the sites, a case that is not supported by available data.

4:05 pm: Correlation of Equipment Failures in Seismic PRAs, Robert W. Drsek, Sean A. McGhee, Lawrence A. Mangan, K. Raymond Fine (*FirstEnergy*), Eddie M. Guerra (*RIZZO Associates*)

In response to the 50.54(f) letter regarding the Near-Term Task Force Recommendation 2.1, many utilities are developing new seismic probabilistic risk assessments (PRAs) to more thoroughly assess seismic risk and identify any potential plant-specific issues to be addressed. A major task of developing a seismic PRA is deciding when to correlate the seismic failures of similar structures, systems, and components (SSCs). This paper will discuss correlation of seismic failures from a PRA analyst's perspective. It includes a discussion of how fragility analysts typically group components for calculations, ultimately leading to the data that is provided to the PRA analyst to be incorporated into the seismic PRA. In the cases of similar types of SSCs, this paper will discuss how the configuration of components in a correlation group will affect risk metrics. Examples will be presented as to when correlation of equipment failures can be conservative, the effects of breaking this conservatism, and how improper correlation can also be non-conservative.

4:30 pm: An Original Approach to Derive Seismic Fragility Curves—Application to a PWR Main Steam Line, Nadia Rahni, Christophe Clement, Georges Nahas, Julien Clement, Lionel Vivan, Yves Guigueno, Emmanuel Raimond (*IRSN*), Thomas Chartier (*ENS*), Marine Marcilhac, Thierry Yalamas (*Phimeca Engineering*)

Seismic fragility curves express the conditional probability of failure of a structure or component for a given seismic input motion parameter. The methodology presented in this paper aims at studying the seismic coupled response of ground, structures and components in a time history analysis. The methodology involves multiple numerical calculations in a short amount of time and the fragility curve is derived by applying linear regression methods traditionally used in the non-destructive testing field.

4:55 pm: Seismic PRA Fragility Sensitivity Studies, Lawrence A. Mangan, Robert Drsek, Sean McGhee, K. Raymond Fine (*FirstEnergy*), Eddie M. Guerra (*RIZZO Associates*)

In response to the 50.54(f) letter regarding the Near Term Task Force Recommendation 2.1, many utilities are developing new seismic Probabilistic Risk Assessment (PRA) models to more thoroughly assess seismic risk and identify any potential plant-specific issues to be addressed. Realistic fragility parameters of Systems, Structures, and Components (SSCs) are necessary to ensure that results and insights from the Seismic PRA reflect site specific conditions. However, doing so for all SSCs within the scope of the Seismic PRA may require significant resources. This paper describes a series of Sensitivity Studies that may be performed with a Seismic PRA to provide justification for the use of generic or conservative fragility parameters for many SSCs without sacrificing model integrity or completeness, thus reducing the overall amount of resources required while still obtaining meaningful results and insights. Examples are provided to illustrate the process of the sensitivity studies.

5:20 pm: Seismic Evaluation of Auxiliary Buildings and Effects of 3D Locational Dynamic Response in SPRA, Jieun Hur, Eric Althoff, Halil Sezen, Tunc Aldemir (*Ohio State Univ*), Richard Denning (*Consultant*)

For the evaluation of operational failure probabilities of safety-related nonstructural components (NSCs) located in an auxiliary building subjected to seismic shakings, the spatially-dependent seismic response of the auxiliary building and in particular the effects caused by the irregularity of building structures on the three directional acceleration responses of the floors of the building are investigated. The findings help to understand the 3D dynamic behavior of the locations where NSCs are mounted. In addition, the approach used is able to identify the correlation of NSC behaviors at different locations of the building, and allows to quantify the marginal and joint failure probabilities of NSCs.

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TECHNICAL SESSIONS - 9:00 AM

Fire PSA—I

Chair: Mark B. Wishart (*JENSEN HUGHES*)

Location: Grand Station III **Time:** 9:00 - 10:15 am

9:00 am: Lessons Learned Applying NRC-Approved Methods for Incipient Detection Credit in Fire PRA, Richard Stremple, Sum T. Leung, F. William Etzel, K. Raymond Fine (*FirstEnergy*), Brian J. Krystek (*EPM*)

In 2013 the two subject plants submitted a License Amendment Request (LAR) to the NRC in support of adopting NFPA 805 as the new fire protection licensing basis under 10CFR 50.48(c). As part of the LAR, each plant committed to add Incipient Detection systems to a number of low-voltage electrical cabinets in order to reduce fire risk in some of the most significant areas. The risk reduction credit taken for these systems was calculated in accordance with FAQ 08-0046 as agreed upon by the industry and NRC. In July 2016 NRC issued a letter to NEI retiring FAQ 08-0046 in favor of NUREG-2180, which was still in draft at that time and for several months after. The beneficial credit allowed by the NUREG for Incipient Detection systems was substantially less than that prescribed in the FAQ, and when applied caused a significant increase in CDF for both subject plants to the point of exceeding RG 1.174 risk criteria in the Fire PRA models in use at the time. Discussions with NRC, followed by an additional RAI, effectively required each plant to incorporate this change in guidance before NRC would complete their review and issue the NFPA 805 Safety Evaluation for each plant. This paper will describe the additional work which was required in order to achieve workable risk values for these plants in light of the reduction in credit for incipient detection; specifically efforts to further refine both the PRA analyses as well as the detailed fire modeling in the affected areas, including taking additional field measurements and incorporating new ignition frequencies and heat release rates from the guidance in NUREG-2169 and NUREG-2178, and committing to additional modifications. Discussion will also briefly cover the attendant schedule delays and cost increases.

9:25 am: Statistical Characterization of the Time to Reach Peak Heat Release Rate for Nuclear Power Plant Electrical Enclosure Fires, Raymond H. V. Gallucci (*NRC*)

Since publication of NUREG/CR-6850 (EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities) in 2005, modeling of fire growth to peak heat release rate (HRR) for electrical enclosure fires in nuclear power plant probabilistic risk assessment (PRA) has assumed an average 12-minute rise time. NUREG/CR-7197 (Heat Release Rates of Electrical Enclosure Fires [HELEN-FIRE]), published in 2016, has provided substantially more data from which to characterize this growth time. Probabilistic analysis yields distributions that enhance the original NUREG/CR-6850 results for both qualified and unqualified cables. The mean times to peak HRR are 13.3 and 10.1 min, respectively, with a mean of 12.4 min when all data are combined, confirming that the original NUREG/CR-6850 estimate of 12 min was quite reasonable. Statistical-probabilistic analysis, shows that the time to peak HRR for qualified and unqualified cables can again be well represented by gamma distributions. Simulations demonstrate that non-suppression probabilities, on average, are 30% and 10% higher than the use of a 12-min point estimate when the fire is assumed to be detected at its start and halfway between its start and the time it reaches its peak, respectively. This suggests that adopting a probabilistic approach enables more realistic modeling of this particular fire phenomenon (growth time).

9:50 am: Statistical Characterization of Heat Release Rates from Electrical Enclosure Fires for Nuclear Power Plant Applications, Raymond H. V. Gallucci, Brian Metzger (*NRC*)

Since the publication of NUREG/CR-6850 in 2005, the US nuclear industry has sought to re-evaluate the default peak heat release rates (HRRs) for electrical enclosure fires used as fire modeling inputs in fire probabilistic risk assessments (PRAs). An effort by the Electric Power Research Institute and Science Applications International Corporation in 2012 was not endorsed by the US Nuclear Regulatory Commission (NRC) for use in risk-informed, regulatory applications. Subsequently the NRC, with the National Institute of Standards and Technology, conducted tests for representative nuclear power plant electrical enclosure fires to definitively establish more realistic peak HRRs. The results are statistically analyzed to develop two probabilistic distributions for peak HRR per unit mass of fuel that refine the values from NUREG/CR-6850, thereby providing a fairly simple means to estimate peak HRRs from electrical enclosure fires in support of fire PRA. Unlike NUREG/CR-6850, where five different distributions are provided, or NUREG-2178, which now provides 31, the peak HRRs for electrical enclosure fires can be characterized by only two distributions. These distributions depend only on the type of cable, namely qualified vs. unqualified, for which the mean peak HRR per unit mass is essentially a factor of two different. Simulations using variable fuel loadings demonstrate how the results may be used for nuclear power plant applications.

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TECHNICAL SESSIONS - 9:00 AM

High Winds PSA-II

Chair: Nicholas Lovelace (JENSEN HUGHES)

Location: Grand Station IV **Time:** 9:00 - 10:15 am

9:00 am: An Estimation Method for Tornado Missile Strike Probability Under Assumption of Statistically Isotropic Tornado Path Directions, Yuzuru Eguchi, Soichiro Sugimoto, Yasuo Hattori, Takahiro Murakami, Hiromaru Hirakuchi (CRIEPI)

The authors propose an efficient evaluation method for probability of tornado missile strike, without fully employing Monte Carlo method, by assuming statistical isotropy of tornado path directions. In other words, we assume that tornadoes to attack a plant site have uniform probability with respect to the path direction. This assumption allows us to rather easily evaluate strike probability for a horizontal or perpendicular plate per unit area, q , with trajectories of missiles mapped on the rz plane. The trajectories of missiles were evaluated with an in-house code, TONBOS, where wind field of a tornado is modeled by Fujita model (DBT-77), while the motion of a missile in flight is modeled with three degrees-of-freedom translational equations with aerodynamic drag and gravity forces. The near-ground motion of missiles is assumed to be subject to lift force due to ground effect. The strike probability for a plate of unit area, q , can be sorted according to a local wind speed V to obtain the conditional strike probability, $q(V)$. On the other hand, the annual exceedance probability of the local wind speed, $H(V)$, can be used to obtain the probability density function, $P(V)$, via the partial differentiation of $-H(V)$ with respect to V . Then, we can obtain the annual probability of tornado missile strike for a plate with the convolutional integration of product of $q(V)$ and $P(V)$ over V . As an example of this evaluation method, the annual probability of strike of a car for a perpendicular plate with arched cross section was computed, and the validity was quantitatively confirmed by comparing the result with that of another code which is described in our previous paper "An Evaluation Method for Tornado Missile Strike Probability with Stochastic Correlation" published in Nuclear Engineering Technology on Jan. 6, 2017(online).

9:25 am: Comparison of Tornado Missile Fragilities Developed Using TORMIS and Simplified Methods, J. C. Sciaudone, L. A. Twisdale, T. B. Dombrosky, M. B. Hardy (Applied Research Associates, Inc.)

Tornado missile fragilities developed using TORMIS for several plants are compared to results from simplified tornado missile risk analysis tools. The simplified methods investigated include a missile flux density approach, a multivariate model with regression parameters fitted to TORMIS calculations, and a simplified calculation method. This paper provides insights from these comparisons including discussion of the limitations, accuracy, and uncertainties introduced in the use of simplified methods.

9:50 am: High Wind PRA Key Insights and Uncertainties, Nicholas Lovelace, Matt Johnson (JENSEN HUGHES), Lawrence A. Twisdale, Jr. (Applied Research Associates, Inc.)

Recently there has been significant developments in relation to high wind probabilistic risk assessments (PRAs). Several Nuclear Power Plants have built and successfully peer reviewed state of the art high wind PRAs. In each of these high wind PRAs the risks associated with thunderstorms, extra-tropical storms, tornadoes, and hurricanes were evaluated. Based on these recent high wind PRAs and developments in the high wind PRA community, it has been shown that the high wind contribution to the CDF and LERF is higher than previously expected.

This paper is based on several high wind PRAs developed by the authors and focuses on the key insights and uncertainties in the high wind PRA analyses. Examples of these insights and uncertainties are: impact of key assumptions and uncertainties on the PRA model and results; lessons learned that can be used to optimize high wind PRAs; high wind deterministic protection vs high wind PRA vulnerabilities; sensitivity of results to high wind correlations; and important lessons learned from the high wind PRAs.

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TECHNICAL SESSIONS - 9:00 AM

LPSD

Chair: Raymond E. Schneider (*Westinghouse*)

Location: Grand Station V **Time:** 9:00 - 10:15 am

9:00 am: Phenomena Identification Ranking Techniques (PIRT) for Determination of Low Power Shutdown PRA Priorities, Garill Coles, Steve Short (*PNNL*)

In early calendar year 2017, Pacific Northwest National Laboratories (PNNL) coordinated and facilitated for the Nuclear Regulatory Commission (NRC) a Phenomena Identification Ranking Technique (PIRT) process involving expert elicitation to determine priorities associated with performing low power shutdown (LPSD) Probabilistic Risk Assessments (PRAs). The objectives of the work were to develop a process and exercise it to shed light on the important technical challenges associated with performing a LPSD PRA and to identify enhancements to the PIRT expert elicitation process. PNNL notes that the resources required to perform a comprehensive LPSD PRA that encompasses all combinations of LPSD plant operating states, plant outage types, and hazards would be so extensive that its undertaking would be extremely challenging. To address this challenge, PNNL developed and implemented a PIRT expert elicitation process that prioritizes those plant operating states, hazards, and outage types that should be included in a full-scope nuclear power plant Level 3 LPSD PRA.

In writing about the role of the PIRT process, early PIRT developers have recommended use of the Analytical Hierarchy Process (AHP) – developed by Thomas Saaty, professor of statistics and operations research – as a way to formalize subjective decision-making into a product that is defensible, transparent, and complete. PNNL notes that while the AHP does not appear to have been used in PIRTS performed and documented for the NRC up to this point, it has been studied and used extensively since its development in the early 1980s. Moreover, PNNL finds use of pairwise comparisons, a central feature of the AHP along with use of explicit evaluation criteria, to be a helpful way of judging the importance of one factor (or alternative) over another against an overarching goal. This paper describes using the AHP as an integral part of the PIRT process to determine LPSD PRA priorities associated with adequately assessing risk and the kinds of insights that can be achieved with such PRA applications.

9:25 am: Low Power and Shutdown PSA for High Temperature Gas-Cooled Reactor, Tao Liu, Jiejuan Tong, Jun Zhao (*INET of Tsinghua Univ*)

High Temperature Gas Cooled Reactor Pebble-bed Module (HTR-PM) has finished construction design and prepared for the charging license application. And relative technical documents are submitted and under review by National Nuclear Safety Administration (NNSA). Low power and Shutdown operation Probabilistic safety analysis (LSPSA) report is one of charging license conditions according to the requirement of nuclear safety regulatory authority. The result of LSPSA will firstly verify the safety goal compared with full range Probabilistic safety analysis (PSA) and then apply to design direction.

According to these intention, HTR-PM LSPSA report has finished the first version and is upgraded according to the review feedback. The paper will introduce the LSPSA framework and summary several technical issues which have been encountered during the HTR-PM LSPSA development.

9:50 am: Is Green Good Enough? Managing Shutdown Risk in Generation III+ Nuclear Power Plants, Aaron Moreno, Nathan Larson, Ken Kiper (*Westinghouse*)

The new generation of passive safety system nuclear power plants such as the AP1000® presents special challenges of applying generally accepted industry practices on defense-in-depth (DID) to manage shutdown risk. The use of a qualitative shutdown risk monitor has been shown to be effective at planning and executing a risk conscious refueling or shutdown outage. Current US operating plants strive to meet the “N+2” criterion for the five key safety functions of Decay Heat Removal, Power Availability, Reactivity, Inventory Control and Containment Closure. However, with the inclusion of passive safety systems in addition to the normally utilized non-safety systems that function in shutdown modes, is the “N+2” criterion still sufficient for managing the risk for generation III+ plants? There is a potential that some key safety functions may extremely exceed the “N+2” criterion such that scheduled risk may never rise above the Green threshold. This results in the question: does the DID model adequately reflect all of the risks of a passive plant? Additional and/or alternate key safety functions may need to be considered to truly understand the complete shutdown risk picture. Issues like these will be explored further to ensure that defense-in-depth is keeping pace with the evolution of the Generation III+ passive safety feature plants.

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TECHNICAL SESSIONS - 9:00 AM

Seismic PSA—III

Chair: Barry D. Sloane (*JENSEN HUGHES*)

Location: Waterfront **Time:** 9:00 - 10:15 am

9:00 am: Qualitative Approach to Grasp Whole Risk Profile of Multi-Unit Site, Haruhiro Nomura, Yoshiyuki Narumiya (*Kansai Electric Power Co.*), Kensuke Toyoshima, Satoshi Shinzaki, Masayuki Hijya (*Nucl Engineering, LTD*), Akira Yamaguchi (*Univ of Tokyo*)

In this paper, we marshal the items to be considered for multiple unit risk evaluation, referring to episodes of Fukushima Daiichi Nuclear Power Plant accident. And we show trial risk evaluation of the probability that multiple cores damage frequency, caused by earthquake. In this evaluation, we assumed a site with two twin units (four units in total), and we made a simplified fault tree representing core damage of four units. We used seismic hazard data common to four units, and the fragilities of the same installations of the two units which constitute a twin unit was assumed to be equal, and we assumed certain correlation coefficients of response among four units, referring to the correlation coefficient among the components within a unit defined in the NUREG-1150(Ref. 1). The results showed that not only in the correlation considered case but also in completely independent case, the Core Damage Frequency (CDF) of 4 unit simple total was 30% to 40% larger than the total CDF of multi-unit evaluation. This is because the hazard is common for all units in evaluation of multi-unit. The ratio of multi-unit CDF increases in proximity location where hazards can be considered as common, while total CDF becomes smaller.

9:25 am: Probabilistic Seismic Hazard Analyses—An Industry Professional's Perspective, Lawrence Salomone (*Pinnacle Specialty Group, Inc.*)

Recent seismic hazard studies have resulted in "seismic fatigue"! What will be the drivers for performing future Probabilistic Seismic Hazard Assessments? Having viable Senior Seismic Hazard Analysis Committee (SSHAC) options will be key for future use of SSHAC assessment guidance. The paper, in part, discusses these market conditions and presents the lessons learned from recent Probabilistic Seismic Hazard Assessments. The CEUS SSC source model (2012) and the EPRI (2013) ground-motion model developed using SSHAC Level 3 and 2 assessment processes, respectively, were used to evaluate the seismic hazard at existing nuclear power plants in the Central and Eastern United States. NRC is updating NUREG-2117, "Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies," used for these studies. As a professional directly involved in the Senior Seismic Hazard Analysis Committee (SSHAC) process, first-hand "lessons learned" experience from these studies was provided to the NUREG-2117 Working Group updating this NUREG. These lessons learned are shared in the paper.

The paper looks beyond the work of the NUREG – 2117 Work Group and highlights the processes needed to maintain the stability obtained from having a regional source model and a regional attenuation model. Guidance on which SSHAC Level assessment process to use to provide a cost-effective SSHAC assessment methodology which can be performed and approved in 18 months or less is presented. The SSHAC Level 2 assessment process is examined and the enhanced SSHAC Level 2 assessment process outlined to meet the need for a viable assessment process.

A CEUS SSC model maintenance plan and a CEUS website maintenance plan have been proposed to maintain stability in how probabilistic seismic analyses are performed. The role of these plans and their status for implementation are discussed to demonstrate the path forward for Probabilistic Seismic Hazard Assessments.

9:50 am: Seismic Fragility Evaluation of Reinforced Concrete Walls with Arbitrary Shapes in Plan, Enrique Bazan, Yigit Isbilioğlu, Bradley Yagla (*RIZZO Associates*)

Seismic fragility evaluation of walls is required for seismic probabilistic risk assessments (SPRA) of reinforced concrete buildings in nuclear power plants (NPPs). Structures, systems, and components (SSCs) which are significant contributors to seismic risk require realistic fragility evaluation. Unrealistic fragilities for top contributors may lead to inaccurate conclusions about seismic risk and consequently limit the SPRA's value for making risk informed decisions. To this end, procedures and formulas are readily available in the technical literature for walls that are rectangular in plan. However, walls may also exhibit different shapes in plan other than rectangular; for instance, some walls in containment buildings are either totally or partially circular. In such cases, simplifying assumptions can be developed to use existing formulas or programs for rectangular walls, or alternatively, fundamental principles of reinforced concrete behavior can be employed for estimating fragility parameters. Based on the latter option, this paper describes a procedure developed to calculate the median seismic strength factor, FS, and the corresponding logarithmic standard deviation of a reinforced concrete wall as part of the wall fragility evaluation. Two failure modes are considered for concrete walls: i) overall bending about two orthogonal axes, including the effect of axial load, and ii) overall shear along the two axes. The adopted methodology is based on the ACI 318-14 and ACI 349-06 codes, EPRI TR-103959 guidance, and our review of experimental data. The variabilities of relevant parameters are accounted for by industry standard procedures (e.g., EPRI TR-103959). The methodology has been implemented in a special purpose computer code, which after validation and verification has been used to assess the accuracy of some simplifications for non-rectangular walls. On this basis, the paper offers recommendations for estimating the reinforced concrete wall median seismic strength factor as part of the fragility evaluation.

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TECHNICAL SESSIONS - 10:45 AM

Risk Aggregation—I

Chair: Fernando Ferrante (EPRI)

Location: Grand Station II **Time:** 10:45 am - 12:00 pm

10:45 am: Data-Mining Approach for Validation of PSA Models, G. Loskoutov, P. Hellström, C. Karlsson
(Swedish Radiation Safety Authority)

Validation of PSA models is aimed to provide confidence in the PSA results. Common methods for validation of PSA models consider mainly dominating events. While such validation can be sufficient for decision making based on comparison of the total values (for example, total core damage frequency) with some predefined numerical criterion, it does not provide enough confidence for relative comparison of calculated results for different events or event groups. The suggested validation procedure allows to generalize PSA results and calculate the correlation between initiating event frequency and conditional core damage probability. The method has been tested on a real PSA model for a nuclear power plant using relation between initiating event frequency and conditional core damage probability for events leading to core damage. The results show that the method provides important knowledge about the model structure and its assumptions without prior experience of the particular model. Such ability is crucial for parties not directly involved in the development of the PSA model, e.g. during authority review and use of PSA model results in supervision activities.

11:10 am: WGRISK Site-Level PSA Project: Status Update and Preliminary Insights for the Risk Aggregation Focus Area, Smain Yalaoui, Yolande Akl (Canadian Nucl Safety Commission), Marina Roewekamp (GRS), Daniel Hudson (NRC)

Interest in the assessment of the “total risk” that encompasses impacts of both internal and external hazards on all major radiological sources at a nuclear installation has grown since the March 2011 Fukushima Dai-ichi nuclear accidents. Such assessment of total risk goes beyond current practice in probabilistic safety assessment (PSA), which typically focuses on assessing risks from accidents involving individual sources that are caused by individual hazards or hazard groups. In June 2015, the Organisation for Economic Co-operation and Development (OECD), Nuclear Energy Agency (NEA), Committee on the Safety of Nuclear Installations (CSNI) approved a Working Group on Risk Assessment (WGRISK) activity to collect information on whether and how member countries are addressing multi-unit (multi-source) issues in PSA. Three focus areas were identified: (1) risk aggregation, (2) multi-source interactions, and (3) site-based risk metrics and safety goals. This paper addresses the risk aggregation focus area and summarizes: (1) challenges related to defining PSA risk metrics; (2) results from an international survey regarding risk aggregation for comparison with national safety goals; (3) challenges related to risk aggregation; and (4) risk aggregation methods, considering the different degrees of uncertainty and PSA maturity for different hazards.

11:35 am: A Hierarchical Tree-Based Decision Making Approach for Assessing the Trustworthiness of Risk Assessment Models, Tasneem Bani-Mustafa, Nicola Pedroni, Enrico Zio, Dominique Vasseur, Francois Beaudouin (EdF)

Risk assessment models are conceptual constructs (translated into mathematical forms), built on a set of assumptions (hypotheses) made on the available knowledge. In this sense, the risk assessment outcomes are conditional on the available knowledge.

Risk assessment provides informative support to decision making (DM), and assurance must be provided to guarantee that the results are credible and trustworthy for the DM purposes, for which they are employed. The present paper proposes a four-levels, top-down, hierarchical tree to identify the main attributes and criteria that affect the level of trustworthiness of models used in probabilistic risk assessment. Based on this hierarchical decomposition, a bottom up, quantitative approach is employed for the assessment of model trustworthiness, using tangible information and data available at the basic “leaf” sub-attributes level. The analytical hierarchical process (AHP) is adopted for evaluating and aggregating the sub-attributes.

The approach is shown by application to a case study concerning the estimation of failure probability of the Residual Heat Removal (RHR) system of a nuclear power plant (NPP). The trustworthiness of two models of different complexity is evaluated: a Fault Tree (FT) and a Multi-States Physics-based Model (MSPM).

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Fire PSA—II**Chair:** Tatsuya Sakurahara (Univ of Illinois)**Location:** Grand Station III **Time:** 10:45 am - 12:00 pm**10:45 am: Modeling the Interface of Manual Fire Protection Actions With Fire Progression in Fire Probabilistic Risk Assessment of Nuclear Power Plants, Tatsuya Sakurahara, Zahra Mohaghegh, Seyed Reihani, Ernie Kee (Univ of Illinois)**

This paper develops a methodology for “explicit” modeling of the interface between manual fire protection (i.e., manual fire detection and suppression) and a Computational Fluid Dynamics (CFD) fire progression model, utilizing Fire Dynamics Simulator (FDS), in Fire Probabilistic Risk Assessment (PRA) of nuclear power plants (NPPs). A literature review revealed that there had been no research on developing an “explicit” interface between a CFD-based fire model and manual fire protection until very recently, when Kloos et al.^{1,2} integrated FDS with dynamic event trees and Human Reliability Analysis (HRA). The research demonstrated in this paper has been conducted in an Integrated PRA (I-PRA) framework, i.e., an integration of classical PRA of the plant and a simulation-based module, and therefore, using dynamic event trees is not applicable. However, to obtain a more accurate and realistic estimation of fire-induced NPP risk, there is a need to account for (i) the performance of the plant’s crew in manual detection and suppression, and (ii) the interactions of the crew with the fire progression. In the existing Fire PRA methodology (NUREG/CR-6850),³ manual suppression is addressed by a data-driven approach, where the time to manual suppression is estimated by a non-suppression curve - a statistical probability model derived from historical fire event data. Meanwhile, the interactions between manual suppression and fire progression are addressed through an implicit method based on the competition between two separately computed time quantities for “time to target damage” and “time to fire suppression”. In the methodology introduced in this paper, the explicit interface between FDS and manual fire protection is developed using a data-driven model for manual suppression. To build this interface, the Heat Release Rate (HRR) curve, which is an input to FDS, is modified based on data-driven probability models of three timings associated with manual fire protection: time to fire detection, time to fire brigade response, and time duration of fire suppression. A case study, using a typical NPP fire scenario, is conducted to demonstrate the implementation of the explicit interface and to illustrate the impact that the interface can have on the results of Fire PRA. The results show that the fire-induced damage probabilities computed by the I-PRA framework are smaller than those computed by the existing Fire PRA of NPPs (i.e., NUREG/CR-6850 methodology).

11:10 am: High Energy Arc Faults (HEAF) and Their Impact on PRA for a Nuclear Power Plant – Latest International Operating Experience and Research Activities, Marina Roewekamp (GRS), Nicholas Melly, Mark Henry Salley (NRC)

The international operating experience with fires in nuclear power plants (NPP) has provided insights that high energy arc faults (HEAF) may cause significant explosions and create ensuing fires that could damage systems, structures and components (SSC) important to safety. Two so-called Topical Reports on HEAF induced fire events and on event combinations of fires and other events prepared in the frame of the OECD (Organization for Economic Cooperation and Development) Nuclear Energy Agency (NEA) Database Project OECD FIRE have demonstrated that the number of fire resulting from HEAF, even if the HEAF event was an event subsequent to another hazard, is non-negligible. Several of such events did either impair nuclear safety or has the potential for impairing safety under different boundary conditions (precursor events).

For a better understanding of the potential safety significance of HEAF events to safe NPP operation an international in-depth investigation of HEAF was initiated. The objective of this project is to determine damage mechanisms, extent of areas affected, methods of protecting SSC important to safety and possible calculation methods for modeling of HEAF events as applicable to fire protection in NPP environment. As part of this effort, an experimental program was initiated investigating HEAF fire phenomena to inform future deterministic and probabilistic methods. The first phase of the experimental program with seven member countries looked at a variety of different low and medium voltage components (e.g., breakers, switchgears and bus bars).

Some of the general observations into the HEAF phenomena from the data collected in the experiments are:

- Those experiments where electrical components were made of aluminum were much more energetic during the HEAF resulting in more severe physical damage to equipment than those involving only copper and steel at any voltage level.
- The most severe electrical enclosure damage was observed as a result of a low voltage HEAF in an enclosure with aluminum bus bars.
- Increased duration arcing events were more likely to create an ensuing fire.
- HEAF events involving aluminum have the potential to create a new failure mechanism, specifically the aluminum produces a conductive aluminum plasma in the smoke that coats any exposed material causing short circuits and unintended current paths in electrical systems.
- All medium voltage enclosures, approx. 4 kV and above, maintained the arc for more than 2 s.
- Experiments with arc durations less than 2 s typically did not result in ensuing fires.

The paper provides valuable insights for probabilistic risk assessment (PRA) from the operating experience collected in the OECD FIRE Database as well as from the HEAF experimental program.

11:35 am: Development of Fire PRA Guide for Japanese NPPs, Tsuyoshi Uchida, Toshiyuki Zama (CRIEPI), Hiroyuki Takeuchi (Toshiba), Shingo Oda (GE Hitachi), Kakujiro Kadoya (Mitsubishi), Daniel Funk (JENSEN HUGHES), Mardy Kazarians (Kazarians & Associates, Inc.)

Internal fires within a nuclear power plant (NPP) can be a significant risk contributor. Fire PRA is a useful tool to identify vulnerabilities of NPPs for internal fires. In order to assist utilities in Japan in conducting fire PRAs, Nuclear Risk Research Center (NRRRC), in cooperation with Japanese reactor vendors and experts from the U.S. with fire PRA experience, has embarked on the development of a fire PRA guide. The fire PRA guide is intended to provide Japanese nuclear industry a state-of-the-art method and supporting data. The guide is based on NUREG/CR-6850 (Ref.1) and recent developments and research results. Furthermore, the guide incorporates design features and operating experiences specific to Japanese NPPs.

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TECHNICAL SESSIONS - 10:45 AM

Risk-Informed Decision Making—I

Chair: Mark B. Wishart (*JENSEN HUGHES*)

Location: Grand Station IV **Time:** 10:45 am - 12:00 pm

10:45 am: Enhanced Guidance on Integrated Risk-Informed Decision-Making, Donald Dube, Gareth Parry, Stuart Lewis, Doug True (*JENSEN HUGHES*), Fernando Ferrante (*EPRI*), James Chapman (*Jim Chapman Risk LLC*)

Risk-informed decision-making is an approach to regulatory decision-making, in which insights from probabilistic risk assessment (PRA) are considered along with other engineering insights. In the U.S., Regulatory Guide 1.174 provides the framework for using PRA for risk-informed decisions on plant-specific changes to the licensing basis. RG 1.174 and associated guidance documents provide some guidance for addressing individual elements of the key principles including those related to defense in depth, safety margin, risk changes and performance monitoring. There is, however, a tendency in regulatory applications to evaluate each of these key principles individually, almost in a “check-the-box” sort of mindset, rather than in a truly integrated fashion. How should one weigh defense-in-depth impacts against risk changes that are deemed negligible? How do uncertainties in the assessment of risk impacts factor into the final decision-making process? This paper describes guiding principles and the need for a framework for integrated risk-informed decision-making whereby individual elements are all given due consideration in a holistic sense, with no one element carrying so much weight that it effectively overrides all other elements. The development of detailed guidance remains a work in progress.

11:10 am: Probabilistic Risk Assessment Usage in Support of Plant Operations and Management, Mark T. Cursey, Suzanne M. Loyd, Christopher Pupek, Gregory T. Zucal (*JENSEN HUGHES*)

A known strength of both quantitative and qualitative risk assessment techniques is their ability to identify system relationships which are not immediately visible; however, the dominant usage of risk assessment has frequently related to quantitative results in the forms of numerical criteria and importance measures. Utilities have made substantial investments in risk modeling for at-power and outage conditions. These models can be used to enhance safety and efficiency outside of regulatory risk-informed processes.

This paper reviews the different types of information that can be developed from risk assessment results and how they can be used in plant operations and management. It also provides cautions when that information needs to be supplemented to increase its usefulness. Examples that will be discussed include the following:

- *Prioritization of trouble shooting and testing activities,*
- *Prioritization and optimization of preventive maintenance activities,*
- *Identification of Hazard Sensitive Areas,*
- *Development of diverse mitigation strategies, and*
- *Training prioritization and support.*

This paper will finally demonstrate that even with limited risk modeling, insights can be developed that can be applied to various hazard types that can be used to improve safety and efficiency.

11:35 am: Probabilistic Risk Assessment and Knowledge Management at the U.S. Nuclear Regulatory Commission, Suzanne Dennis (*NRC*)

The U.S. Nuclear Regulatory Commission, as a risk-informed agency, is increasingly using multidisciplinary, multifaceted, and technically specialized information to support regulatory decision making. In order to make the best use of this information to support these regulatory decisions, the NRC has undertaken several knowledge management initiatives, including a report that addresses questions surrounding PRA using the format of frequently asked questions (FAQs), a report on the history of defense-in-depth, and a study of knowledge engineering. This paper discusses these knowledge management efforts, including the motivations for performing them, as well as the lessons-learned during development.

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TECHNICAL SESSIONS - 10:45 AM

ATF

Chair: Raymond E. Schneider (*Westinghouse*)

Location: Grand Station V **Time:** 10:45 am - 12:00 pm

10:45 am: Assessing the Business Case for Accident Tolerant Fuel, Stephen M. Hess, Jeff R. Gabor, Jennifer L. Uhle, Tom Elicson, Garrett Geiger (*JENSEN HUGHES*)

Since the accident at the Fukushima Dai-ichi nuclear power plant (NPP), the nuclear power community has engaged in research to develop advanced fuel designs that can provide a significantly more robust response to design basis and severe accident conditions than the current system consisting of ceramic UO_2 fuel pellets encased within a Zirconium alloy cladding. These candidate enhanced fuel designs are commonly referred to as accident tolerant fuel (ATF). Because the current commercially available fuel designs have been developed over many decades of reactor operation, they are highly optimized to provide outstanding performance and reliability during routine reactor operation while meeting all regulatory requirements to ensure an acceptable level of safety during postulated design basis accident (DBA) and reactor transient conditions. As a result, for NPP owner / operators to adopt ATF as a replacement for the current fuel designs, the enhanced safety and performance during accident conditions provided by ATF must be of sufficient magnitude to compensate for any potential economic impacts associated with adoption of the technology when compared to the highly optimized and economical fuel designs currently in use.

A fundamental prerequisite for NPP owner / operators to adopt ATF, is the quantification of the various safety enhancements that can be obtained and comparison of these benefits against the potential economic impacts of the proposed ATF concepts. An appropriate approach to conduct such a comparison is to develop a business case that identifies the safety benefits that would be needed to permit the adoption of any particular ATF design. Such a business case would expand upon and quantify the general high level criteria that have previously been identified as requirements necessary for industry acceptance of new ATF designs.

In this paper we describe an approach to conduct preliminary assessments of the critical safety benefits associated with proposed ATF concepts. We describe results of preliminary studies of the expected safety benefits obtainable from several proposed ATF concepts based on information available from the literature. These benefits then need to be evaluated against potential economic limitations of the particular ATF concept under consideration. From these results an initial business case can be developed. The overall objective of these evaluations would be to develop a set of requirements and a strategic roadmap that would guide further ATF development and validation to achieve the objective of widespread adoption and deployment of ATF technology.

11:10 am: Estimating the Benefits of Accident Tolerant Fuel (ATF), Raymond Schneider, N. Reed Labarge, Hans Van De Berg, Martin Van Haltern, Zeses Karoutas, Edward Lahoda (*Westinghouse*)

Accident tolerant Fuel (ATF) potentially offers many benefits to the nuclear industry. While the benefits will differ among the various ATF design alternatives, the development of this concept offers tangible benefits to all stakeholders in the nuclear industry: utility, regulator and the public. These benefits span the gamut from reducing plant risk, increasing plant operational flexibility and design margins, increasing operator response times to potential enhancements to plant emergency response strategies. From the viewpoint of the utility, ATF simplifies plant operation, potentially extending refueling cycles and reducing the overall cost of plant operation. While design basis enhancements can tangibly simplify and improve plant performance, ATF fuel can also significantly reduce the beyond-design basis plant risk metrics such as core damage frequency (CDF) and large early release frequency (LERF). The improved risk from utilizing ATF can be capitalized on to reduce the cost of plant operation. From a regulator's perspective, ATF increases margin to plant upsets and enhances the reliability of desired operator response. Finally, the public will benefit from increased safety and the knowledge that plant hydrogen challenges may be eliminated.

This paper reviews the ATF design concepts being developed by Westinghouse in terms of their post-accident performance and compares and contrasts the benefits as they impact the various stakeholders. In demonstrating the benefits of ATF this paper includes a spectrum of MAAP5 analyses based on a representative PWR with and without accident tolerant fuels and identifies considerations for establishing target coping times and other key parameters which benefit accident tolerance.

11:35 am: Risk Implication of Using Accident Tolerant Fuels in LWRs, Koroush Shirvan (*MIT*), C. R. (Rick) Grantom (*C. R. Grantom P.E. & Assoc. LLC*)

The severe LWR accidents at the Three Mile Island and Fukushima Daiichi plants all involved reactor fuel damage and generation of hydrogen. These resulted from overheating of the Zircaloy cladding of the fuel, and oxidation reactions of the metal with the water coolant. The goal of ATF cladding is to replace this vulnerable cladding with alternatives that would not react chemically to generate hydrogen at high temperatures, and that could go to higher temperatures than conventional LWR fuel before suffering structural failures. By doing so, ATFs have potential to positively impact the economics of NPPs. The paper will discuss such economic impact in terms of ATF implications to 50.69 safety-related classification, risk-informed 4b/5b programs to increase completion times to meet current tech spec limits and plant security and emergency planning zone boundary. Since the goal of ATF is to substantially improve severe accident performance, the maintenance savings for reduction or removal of already installed FLEX equipment will also be discussed.

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TECHNICAL SESSIONS - 10:45 AM

Data and Parameter Estimation—I

Chair: Zhegang Ma (INL)

Location: Waterfront Time: 10:45 am - 12:00 pm

10:45 am: Nonparametric Empirical Bayes Estimation Through Deconvolution in Probabilistic Risk Analysis, Andrei V. Gribok, Vivek Agarwal, Vaibhav Yadav (INL)

The advent of Markov Chain Monte Carlo (MCMC) simulations and availability of open source software made full hierarchical Bayesian analysis tractable and a popular choice in parameter estimation for Probabilistic Risk Assessment (PRA). However, despite its theoretical attractiveness, the hierarchical Bayesian analysis is not without its problems. Two of the most prominent practical difficulties in applying hierarchical Bayes analysis in practice to analyze source-to-source variability, for example, is its sensitivity to the selection of the first-stage prior and dependence of the rate of convergence on the selection of the first-stage prior. The first-stage prior in hierarchical Bayesian analysis is usually assumed having a parametric form, often conjugate to the likelihood function (aleatory model). To facilitate convergence for hierarchical Bayes models, the first-stage prior is frequently set through empirical Bayes estimate, which can use available historical data to find the parameters of the first-stage prior. This paper discusses a new nonparametric empirical Bayes estimation and compares it to several well-known parametric estimates such as method of moments and maximum likelihood. The new nonparametric technique exploits the prior predictive distribution as an integral equation and proceeds to solve it with respect to prior assuming the sampling data distribution and aleatory model are available. The discussion covers topics such as selection of aleatory model for future data, selection of regularization parameter, and density estimation for historical data. For comparison, the paper is using contrived as well as real-world data from utilities. Utilities data represent failure data with Poisson distribution used for aleatory model.

11:10 am: Methodology for Modeling Manual Valve Plugging Failures in PRA, J. R. Soniker (Dominion Generation), N. M. Passmore (Enercon)

In an initiative to improve the process of analyzing manual valve plugging failures, a systematic approach to evaluate them is put together and comprehensively explained. This is an often overlooked process in PRA modeling, and if not evaluated, manual valve plugging failures cannot be properly placed in the model or screened based on various criteria. This failure mode, although a small contributor compared to other PRA failure modes, is required to be assessed per ASME/ANS Standard SRs SY-A11 and SY-A14. The approach detailed in this paper helps establish technical adequacy of a plant's PRA by improving the process for evaluation, analysis, documentation and screening of manual valve plugging failures. This systematic approach considers the frequency at which the valve is flowed through, the fluid medium, the type of manual valve, and potential factors contributing to stem/disc separation. Screening criteria are discussed; calculation methodologies are given. This analysis develops a basis for the exclusion of certain air/gas valve plugging failures. As such, this paper gives an introduction and overview of manual valves with a concentration on the plugging failure mechanism and the susceptibility of manual valves to plugging failures.

11:35 am: Bayes, Data and NUREG_CR-6928 Caveats, Carroll Trull, Nathan Larson, Craig Navas (Westinghouse)

This paper is a compilation of data-related issues that have occasionally been hot topics throughout various circles within the PRA industry, as well as some seemingly often overlooked but important aspects of data analysis and NUREG/CR-6928 in particular. The focus is largely on some specific details of NUREG/CR-6928 and NUREG/CR-6823, and why these documents should be every PRA analyst's best friends (in spite of known errors in the former). The paper will highlight the issues, discuss misconceptions and provide insights into applying the following data related topics (subsections):

1. State-of-Knowledge Correlation
2. Change-of-State Failure Mode
3. Plant Availability Factor Normalization
4. LOCA Treatment
5. Zero Event Bayesian Updating
6. Statistical Checks – Are They Important?

The discussions in this paper are based on personal experiences and industry interactions (and do not reflect the opinions of any particular organization), so some of them may seem obvious; however, some of them may present a perspective new to the reader.

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TECHNICAL SESSIONS - 1:30 PM

HRA—I

Chair: Amir H. Mohagheghi (SNL)

Location: Grand Station III **Time:** 1:30 - 3:10 pm

1:30 pm: Reassessment of Operator Actions Claimed in the Sizewell B Living PSA, Christopher Eames, John Collins (EDF Energy)

The Probabilistic Safety Analysis for Sizewell B Nuclear Power Station includes a number of claims on the Operator in response to faults and hazards both when the station is operating at power and when shutdown. These Operator Actions were substantiated as part of the production of the PSA model developed to support licencing of the station during construction and commissioning. This model has been developed into a Living PSA which, as well as forming a key leg of the station Safety Case, is used to support risk informed decision making and underpins the station Risk Monitor.

This paper reports a project to reassess systematically the reliability of these Operator Actions to modern Human Factors standards as part of improvements recommended by the second Periodic Safety Review of Sizewell B. The Operator Actions in the LPSA model were ranked by risk significance using Fussell-Vesely and Risk Increase Factor importances, supported by other qualitative measures. Representative scenarios were developed using the LPSA model to enable individual Operator Actions to be reassessed by Human Factor experts using a range of qualitative task analysis methods commensurate with the significance of the action. These included use of the Sizewell B full scope Main Control Room simulator, local to plant walkdowns and desktop procedural reviews. EDF Energy's Nuclear Action Reliability Assessment (NARA) method was then used to quantify new Human Error Probabilities for use in the LPSA model.

The paper also reports a range of safety improvements that have been implemented as a result of this work. These include significant improvements to plant and operational procedures at Sizewell as well as changes to the LPSA model itself.

1:55 pm: Human Reliability Analysis in CAP1400 Nuclear Power Plant, Qiu Yongping, He Jiandong, Hu Juntao, Zhuo Yucheng, He Jie (SNERDI)

It is well recognized that humans play an important role in the safety operation of nuclear power plants (NPPs). Usually three types of human interactions (HIs) are defined in the human reliability analysis (HRA) of probabilistic safety assessment (PSA) for NPPs, i.e., pre-initiating event HIs, initiating event-related HIs, and post-initiating event HIs. In this paper, a brief introduction of the HRA methodology for CAP1400 nuclear power plant is first presented, including internal events and external events (mainly internal fire and flooding) HRA. Next, the content of CAP1400 human failure event quantification is given with a typical example, and some insights and proposals based on CAP1400 PSA/HRA results are discussed. Finally, the application of HRA in human factor engineering design of CAP1400 is described. The human actions (HAs) most important to safety are identified via a combination of probabilistic and deterministic analyses, and then addressed when conducting the human factor engineering program. The CAP1400 HRA is one of the most important PSA elements and provides fundamental support for CAP1400 PSA and the relevant applications.

2:20 pm: Statistical Approaches to Estimation of Nominal HEPs Using Simulation Data, Yochan Kim, Jinkyun Park, Wondea Jung (KAERI)

Human reliability data supporting empirical bases for quantitative information of HRA (human reliability analysis) methods have recently been collected. Because considerable HRA methods calculate the HEPs (human error probabilities) based on nominal HEPs, it is important to accurately estimate the nominal HEPs for each task type or error type from the collected data. This paper discusses considerations for predicting the nominal HEPs from human reliability data with a brief literature survey and a comparison between results in the two empirical studies. On a basis of the understandings of the nominal HEP, two plausible approaches, estimation-by-aggregation and estimation-by-extraction, were compared. Recent HEP estimations were reviewed. In addition, the results of the two statistical analyses that belong to both approaches were discussed. From the comparative study, necessity of careful use of the HEP estimates was emphasized.

2:45 pm: Lessons Learned in Fire PRA Human Reliability Analysis, Pierre Macheret, Nicholas Lovelace (JENSEN HUGHES)

One of the key challenges in the evaluation of fire risk in a probabilistic risk assessment (PRA), especially with regard to its human reliability analysis (HRA) component, is how fire impacts affect performance shaping factors that underpin the reliability of human actions. This paper focuses on two lessons learned from such HRAs. The first discusses the impacts that a complete dependency level between human actions may have on the modeling of main control room abandonment and its integration in the fire PRA. The second focuses on improving the modeling realism of time-sensitive actions by taking credit for hot-short duration probabilities.

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TECHNICAL SESSIONS - 1:30 PM

Risk Aggregation—II

Chair: Stanley H. Levinson (AREVA)

Location: Grand Station IV **Time:** 1:30 - 3:10 pm

1:30 pm: Beyond Basic Events: Measuring the Importance of Hidden PRA Items of Interest, Andrew Miller, Gregory Zucal (JENSEN HUGHES)

Probabilistic Risk Assessments (PRAs) are unique in their ability to provide insights into the operation of complex systems, such as Nuclear Power Plants (NPPs). The desire to obtain insights from PRA models continues to grow, however the complexity of PRA models is also increasing. For example, PRA models based on spatial analyses (e.g., fire PRAs) introduce new challenges to obtaining meaningful insights.

For decades now, PRA practitioners have relied upon the same metrics to analyze, understand and communicate risk insights from PRA models with very few exceptions. The current approach focuses on the contribution of basic events on the analyzed endstate or top gate. Although this has served the industry well thus far for internal events PRAs where the failure of systems, structures and components (SSCs) are easily mapped to basic events, the method provides limited insights for spatial analysis where initiators are inserted to represent failure of many SSCs. Additionally, the presence of an initiator does not guarantee the failed SSCs participate in the path to the analyzed endstate. Using only the data provided in the cutset files, it is not possible to see how important the initiator based failures of components are to the analysis. A technique that determines the participation of SSCs, or other items of interest (IOIs), is needed.

A new generation of risk metrics is poised to address these shortcomings. If a practitioner was able to track the participation of IOIs for the generated cutsets, an amazing amount of additional information would be readily available from a PRA model. Similar to the Fussell-Vesely (F-V) importance measure used to compare basic events, a Participation Index (PI) can be calculated for all IOIs defined in a PRA model. But, tracking just the participation is only half the story; to best understand the results one must know if the participation of an IOI is critical to satisfying the endstate. Unlike basic events, the participation of an IOI does not guarantee that it is part of a critical path from the basic event cutset to the quantified gate. Being able to look beyond basic events and measure the critical participation of several different types of IOIs will greatly enhance the ability to extract insights from the PRA models.

1:55 pm: Multi-Hazard Risk Aggregation in Support of Risk-Informed Decision Making, Robert Boyer (Duke Energy), Eric Thornsbury (JENSEN HUGHES), Stanley Levinson (AREVA), Andrea Maioli (Westinghouse)

Probabilistic risk assessments (PRAs) for nuclear power plants (NPPs) have been evolving and becoming more complex in the last decades. Along with increased realism in the modeling of plant response to internal events, other hazards, most notably internal flood and internal fire, but also external hazards, mainly seismic and high winds, have been explicitly added to the PRA.

Considerations of risk impact from all realistic plant hazards are a requirement when using PRAs to support risk-informed applications at NPPs, as well as the use of risk-informed regulation promulgated by the Nuclear Regulatory Commission. The Electric Power Research Institute (EPRI) recently published a risk aggregation framework to support a comprehensive view to the aggregated risk profile of a plant that goes beyond the mere summation of hazard-specific risk metrics (i.e., core damage frequency and large early release frequency). Such a framework is aimed at ensuring that different degrees of maturity and uncertainties in the analyses of different hazard groups are appropriately taken into consideration in the risk-informed programs pursued by the plant and by the regulator.

Appropriate consideration of maturity and uncertainties is important to ensure that overly-conservative assumptions or modeling approaches are not invalidating the PRA aim of portraying a realistic risk profile for the plant, masking real risk contributors, and ultimately leading to incorrect decisions using risk-informed insights. The framework published by EPRI has been piloted by the Pressurized Water Reactor Owners Group, using real PRA models and real risk-informed applications from two member plants that could identify hidden complexities in the proposed approach for the aggregation.

This paper describes the findings and challenges faced by the pilot plants and outlines some recommendations to improve the EPRI risk aggregation framework with the goal of providing a more streamlined risk profile picture of the plant and support decision-makers using risk-informed insights when pursuing (or assessing) plant decisions.

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TECHNICAL SESSIONS - 1:30 PM

Risk Aggregation—II Continued

2:20 pm: Further Development of a Framework for Addressing Site Integrated Risk, Adriana Sivori, Kenneth Kiper, Andrea Maioli, David Teolis (Westinghouse)

With rare exceptions, all PSAs performed around the world focus on a single accident at a time occurring to a single unit. However, the vast majority of nuclear plant sites are Multi-unit (MU) sites: almost 90% of all nuclear units are on MU sites and most of the new units projected and under construction will be on MU sites. Certain characteristics of the MU sites can be beneficial to safety (shared systems, cross-ties and shared resources) while, at the same time, presenting specific risk vulnerabilities, such as common initiators, impact to shared facilities or equipment, and cascading initiators.

In this context, this paper will present the high-level descriptions and initial insights and challenges from two different approaches that could be used to assess the site integrated risk following the initial assessment: (1) a deductive (top-down) approach where the focus is on hazards that would most likely represent challenges to site risk, and (2) an inductive (bottom-up) approach where models from single-unit risk assessments are combined with some simplification to produce site-risk results. This second approach was exercised with real plant PSA models where items such as multiple initiators or plant-wide CCFs were investigated for their impact to the original single plant risk profile.

A paper presenting an initial framework to address site integrated risk was presented by some of the authors at PSA-2015. Since that time, they have done further work to develop the framework, including trial applications to a number of site PSAs. This paper summarizes the current state of these methods to address site integrated risk.

Moreover, a review of Operating Experience is presented aiming at the classification and identification of MU Initiating Events, including reactor trip events, external events and LOOP events that have occurred world-wide. A discussion of the treatment of Inter and Intra Common Cause Failure for its inclusion to a Site Integrated Risk Assessment is also covered in this paper.

2:45 pm: Multi-Unit Accident Effects on Safety Goal Quantitative Health Objectives: Insights from a Two-Unit Case Study Involving Two Representative U.S. Nuclear Power Plant Sites, Daniel Hudson (NRC), Mohammad Modarres (Univ of Maryland)

The U.S. Nuclear Regulatory Commission (USNRC) safety goal policy broadly defines an acceptable level of radiological risk to the public from nuclear power plant (NPP) operations. It addresses “How safe is safe enough?” by specifying qualitative safety goals and quantitative health objectives (QHOs) used to measure safety goal attainment. Agency screening evaluations compare results from full-scope NPP probabilistic risk analyses (PRAs) with corresponding QHOs to determine whether proposed regulatory actions to further enhance NPP safety should be rejected before performing detailed cost-benefit analyses to estimate their net benefit. Recent operational experience—including the 2011 Fukushima Daiichi Nuclear Accident—indicates concurrent accident scenarios involving multiple units at a shared NPP site can occur with non-negligible frequency. Yet the USNRC applies the safety goal policy on a per-reactor-unit basis and excludes risk contributions from multi-unit accident scenarios for the nearly 75% of U.S. reactors that are located on NPP sites with multiple units. This gap can lead to underestimating the residual public risk and premature rejection of proposed safety enhancements. This paper summarizes research that applied state-of-the-art consequence models to evaluate the effect of including risk contributions from multi-unit accident scenarios to QHO risk metrics for two representative NPP sites.

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Dynamic PSA—II

Chair: Gary W. Hayner (*JENSEN HUGHES*)

Location: Grand Station V **Time:** 1:30 - 2:45 pm

1:30 pm: Timed-Fault Tree Generation from Dynamic Flowgraph Method, Chireuding Zeliang, Lixuan Lu
(*Univ of Ontario*)

Nuclear steam generators (SG) form the normal heat sink for the energy produced by a nuclear reactor, and hence reliable operation of a SG water level control system is important for reactor safety. A control system failure to maintain water level within reasonable bounds typically results in a reactor trip. It was reported that 25% of shutdown for a pressurized water reactor based nuclear power plants (NPPs) are caused by poor SG level control (SGLC). As a result, a reliable level control system is a major factor in overall plant availability. Moreover, many new and old NPPs are intended to upgrade or refurbish the aging control systems from analog to digital technology. This transition in technology is expected to cause impediments due to the currently limited guidance and consensus on reliability modelling of digital system. Furthermore, numerous concern has been raised regarding the capability of traditional static reliability modelling approaches such as fault tree/event tree to account for dynamic interactions in a system. This paper discusses the dynamic (time-dependent) reliability analysis of a SGLC system. The dynamic flowgraph method (DFM) is proposed to address the dynamic characteristic of a digital SGLC system. The focus of this paper is to model a level control system in DFM, and to generate the corresponding prime implicants and timed-fault trees of the model. To illustrate the relationship between prime implicants and timed-fault trees, two DFM models are presented. The generation of timed-fault trees from the DFM model enables the prime implicants to be integrated into the existing probabilistic risk assessment models, and to quantify the failure impact of a digital control system on the plant core damage frequency. The study shows that DFM can adequately account for interactions in a system. This provide an improved risk-informed decision making with regard to a digital level control system's contribution to plant risk.

1:55 pm: The ADS-IDAC Dynamic Platform with Dynamically Linked System Fault Trees, Mihai A. Diaconeasa, Ali Mosleh (*UCLA*)

Discrete dynamic simulation-based methodologies for probabilistic safety assessment (PSA) are used to generate discrete dynamic event trees (DDETs) that contain rich contextual scenarios, and their associated probabilities that could occur given an initiating event. An example of such a simulation platform is ADS-IDAC: the Accident Dynamics Simulator (ADS) coupled with the Information, Decision and Action in a Crew context cognitive model (IDAC), and a realistic nuclear power plant thermal-hydraulic model. In ADS-IDAC the time-dependent changes in the functional state and parameters associated with the system's elements are traced to generate scenarios by branching to new sequences following various branching rules. Smart model-based branching rules have been developed to obtain a more realistic and complete solution space than the traditional static PSA methods, and avoid the sequence explosion phenomenon as the number of system states increases. One of the currently supported branching options is the hardware failure and recovery: failure and success branches are created when any of the thermal-hydraulic system's component (i.e. Frontline Systems) operation is demanded. This paper describes a new version of the ADS-IDAC platform that includes the ability to model and dynamically link Fault Trees (FTs) to explicitly account for the impact of Support System failures on Frontline Systems. This was accomplished by developing and implementing an algorithm for incorporating binary logic of system failures into the dynamic branching rules of the Discrete Dynamic Event Trees DDETs based on the hybrid causal logic (HCL) methodology. Moreover, hardware failure and recovery capabilities are extended to include multiple failures and recoveries during operation. These capabilities will be demonstrated through the analysis of a real accident precursor.

2:20 pm: Development of Integrated Site Risk Using the Multi-Unit Dynamic Probabilistic Risk Assessment (MU-DPRA) Methodology, Matthew Dennis, Mohammad Modarres (*Univ of Maryland*), Ali Mosleh (*UCLA*)

The events at the Fukushima nuclear power station highlighted the need for considering risk from multiple nuclear reactors co-located at a site. Determining site risk will also be important for small modular reactor designs because of the number and proximity of reactor modules on a site. To gain an accurate view of a site's risk profile, the core damage frequency (CDF) for the site, rather than the unit, should be considered. There are many types of events that could create a dependency between multiple reactor units or modules from a risk perspective. In order to effectively account for these risks when looking to create a multi-unit dynamic probabilistic risk assessment (MU-DPRA), six commonality classifications have been established. These commonality classifications and dynamic PRA can be used to establish a system model, as an example, for two adjacent integral pressurized water reactors (iPWRs) with shared safety and support systems. To accomplish this, the previously developed dynamic simulator, ADS-IDAC, is upgraded in three areas: 1) the thermal-hydraulic code responsible for modeling the system performance is improved to a more recent RELAP version, 2) ADS-IDAC is incorporated with a commercial "executive platform" to allow parallel communication between two or more nuclear reactors, and 3) the hardware reliability model within ADS-IDAC is improved to capture accelerated hardware failure. The research described in this paper expands the system state-space to more than one reactor unit and evaluates initiating events and accident sequences composed of hardware failures and human errors on one or more reactor units. The development and demonstration of a novel methodology proposed here provides a framework for more realistic PRA analyses and assessment of the relative contribution of important core damage end states.

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TECHNICAL SESSIONS - 1:30 PM

Seismic PSA—IV

Chair: Eddie Guerra (*RIZZO Associates*)

Location: Waterfront **Time:** 1:30 - 3:10 pm

1:30 pm: Systems-Based Seismic Contact Chatter Analysis, Eric Jorgenson, Robert Miller (*Enercon*), Parthasarathy Chandran (*Southern Nucl Operating Co*)

A systems-based approach to identify contact chatter scenarios focuses on those scenarios that cannot be reliably recovered by the operations crew in sufficient time, and on additional special scenarios not explicitly represented by the internal events models. A systems-based approach can substantially reduce the circuit analysis work scope for a chatter analysis project. This approach begins by reviewing the seismic equipment list to identify This method begins by reviewing the plant's seismic equipment list (SEL) to identify chatter-susceptible equipment – that is, active equipment such as operable valves with the potential to mal-operate given contact chatter. An additional selection process is performed by a seismic expert panel to identify additional special scenarios. The panel identifies seismic scenarios not explicitly represented in the internal events model, and are analogous to multiple spurious operation scenarios modeled by fire PRAs. Such special seismic scenarios include, for example, potential overloading of a diesel generator due to components loading out of sequence, or spurious closure of ac-powered inboard containment isolation valves with potential non-recoverability of turbine-driven pumps if a station blackout develops. A careful review is performed by the expert panel to capture the subtle interactions produced by component mal-operation, with due consideration of the nature / duration of such mal-operation, and the ability of the operations crew to respond. This paper describes the approach and provides lessons learned from several contact chatter analyses.

1:55 pm: Analysis and Examination for Developing Fault Displacement PRA Methodology, Katsumi Ebisawa, Hideaki Tsutsumi (*CRIEPI*), Ryusuke Haraguchi, Kunihiro Sato, Futoshi Tanaka, Daisuke Ochi (*Mitsubishi Heavy Industries, Ltd.*), Yoshinori Mihara (*Kajima Corp*), Shinichi Yoshida (*Obayashi Corp*)

Fault displacement (FD) can cause serious damage to structures and buildings. Even though nuclear at nuclear power plant (NPP) are constructed avoiding potential FDs and the risk from such event is anticipated to be low, the understanding of the risk from such events is important. Recently in Japan, interest on the impact of principal and secondary FDs on nuclear facilities has increased, and it is currently recognized as an urgent issue for investigation. Under this context, the authors are conducting examination of FD PRA framework and identification of technical issues, as part of the development of FD PRA methodology study. FD PRA consists of accident scenario identification, hazard evaluation, fragility evaluation and accident sequence evaluation. Accident scenarios were identified by identification of risk significance scenarios in terms of impact on plant and their frequency. A concept of the event tree and fault tree structure, as well as the fragility evaluating of structures and components for FD events have been investigated, and fragility evaluation has been performed for representative components. Through these analyses and examinations, the authors obtained prospects for FD PRA methodology development.

2:20 pm: In-Cabinet Amplification Factor for Relay Fragility Analysis in Seismic PRAs, Stephen J. Eder (*Facility Risk Consultants*), Kenneth Whitmore (*Enercon*)

Lateral in-cabinet amplification factors for determining seismic demand for conservative deterministic failure margin (CDFM) relay chatter evaluation are provided in the EPRI seismic margins methodology report NP-6041-SL, Appendix Q. As well understood in the industry, the NP-6041-SL amplification values are upper-bound lateral amplification factors, representing the 84%tile response of worst case location in the various types of cabinets. This location is generally understood to be in the center of door panels near the top of a cabinet, and front-to-back cabinet motion generally controls. However, relays are not always located in these worst-case locations. For SPRA fragility analysis, realistic amplification factors are necessary and important to ensure that excessive conservatism is not introduced into the plant risk model.

Dynamic amplification factors must account for the amplification of the in-structure response spectra (ISRS) resulting from flexibility of the cabinet as a whole (cabinet response) and the flexibility of the surface to which the relay is mounted (response of the door). These responses depend on the actual location of each relay and the actual configuration of the cabinet and its doors. In-structure response spectra (ISRS) are multiplied by an in-cabinet amplification factor to determine seismic demand for each relay. The clipped peak amplitude of the ISRS in the frequency range of interest for the relay is multiplied by the amplification factor and compared with the relay capacity data. The relay capacity data are from broad-band testing, so seismic demand spectral peak clipping should be applied, consistent with EPRI NP-6041-SL.

This paper provides an approximate yet realistic methodology to account for actual relay mounting location. This includes consideration of the elevation of the relay within the cabinet and the location of the relay within the door as well as the comparative natural frequency of the cabinet response and the door response. The approach adjusts for the size and configuration of the cabinet door and the characteristics of the cabinet. Plans are in place to further validate the methodology with in-cabinet test data.

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Seismic PSA—IV Continued

2:45 pm: Evaluation of Interinfluence Between Adjacent Units in Seismic PRA, Shuhei Matsunaka, Hiroshi Abe (Tepeco Systems Corp, Inc.), Manabu Watanabe, Yoshihiro Oyama (TEPCO Holdings, Inc.)

Kashiwazaki-Kariwa nuclear power station of TEPCO is the largest nuclear power station in the world, and it has seven nuclear power plants. As the experience at Fukushima Daiichi nuclear power station accident in March 2011 involving concurrent core damage at multiple units, it is considered that the risk derived from hazards of Earthquakes and Tsunamis is relatively significant in Japan, and these events have a high likelihood of damaging multiple units simultaneously. Therefore, it is very important to grasp the multi-unit specific risk.

Although there are some unique accident scenarios of Multi-Unit PRA, this paper focuses on the influence of radioactive materials released outside the containment vessel on the accident management of the adjacent unit. The events including core damage and loss of containment function should be considered as the causes of the release of radioactive substances, and operator's operation or the like should be considered as the objects to be adversely affected. It is necessary to incorporate that into PRA to confirm the effect on risk.

It is very difficult in terms of the maturity of evaluation method and calculation load to accurately incorporate consequences derived from time series of various events and complicated interaction into PRA model. Therefore, as the first step in evaluating the risk of influence of radioactive material release on the accident management, some streamlining efforts are carried out according to the purpose. For example, Kashiwazaki-Kariwa unit 6 and unit 7 were set as the target units for model simplification. We also assume the earthquake as the initiating event due to the strong common factor for multi units. PRA taking into consideration the radiation influence and PRA not taking into consideration it are carried out, and the influence degree of this issue is provided.

TECHNICAL SESSIONS - 3:40 PM

Mini Workshop: PyCATSHOO

Chair: Valentin Rychkov (EdF)

Location: Grand Station V **Time:** 3:40 - 5:45 pm

Overview of a new EDF R&D tool PyCATSHOO for dependability assessment of hybrid systems:

The safety requirements of its nuclear and hydraulic fleet, has allowed EDF to have long-standing experience in using and developing PRA and PSA tools for complex systems.

During PSA 2017 we would like to present to general public our latest development: PyCATSHOO a tool dedicated to dependability analysis of hybrid systems, i.e. systems including deterministic continuous phenomena and discrete stochastic behavior. Currently PyCATSHOO is used at EDF to perform several safety studies. During 2017, EDF will release PyCATSHOO under a freeware license. PyCATSHOO is a C++ written library. It has two APIs (Application Programming Interfaces) in Python and C++. These APIs provide a set of tools, based on distributed hybrid stochastic automata, which help in modelling and assessment of complex hybrid systems.

Data and Parameter Estimation—II

Chair: Zhegang Ma (INL)

Location: Grand Station III **Time:** 3:40 - 5:45 pm

3:40 pm: Estimation of Manually Recoverable Fraction of Common Cause Failures of Motor Operated Valves, Young G. Jo (Southern Nucl Operating Co)

In this paper, motor operator valve (MOV) common cause failure (CCF) experience events were analyzed and the fraction of MOV CCF events which may be recoverable by local manual actions was estimated. MOV CCF experience events compiled in NRC CCF Data Base and Analysis System during 1980 and 1995 were reviewed for this study. After removing duplicate events and removing one event which was considered as not applicable, a total of 147 MOV CCF events were left. MOV CCF events due to the following caused were considered as being recoverable by manual handwheel operation: loss of power/control signal, failure of actuator subparts which can be bypassed by declutching, premature stopping of the motor before valve is properly position due to incorrect limit switch setting or torque switch setting, and failures where the motor experience excessive resistance due to excessive spring pack preload and motor stopped either by thermal overload trip or due to motor burn out before damage to the valve occurred. 102 of 147 events, about 70.0 % of MOV CCFs, were determined as recoverable by local operator manual actions. Crediting local manual recovery of manually recoverable MOV CCFs should also consider the feasibility of operator recovery action.

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Data and Parameter Estimation—II Continued

4:05 pm: Handling Room Cooling in PRA, Carroll Trull, Daniel Tirsun (Westinghouse)

The majority of PRA models deal with room cooling by judging whether or not the equipment being cooled will exceed some limit (e.g., Equipment Qualification limits), then either screening room cooling for that function or modeling loss of room cooling as necessarily failing the function: go/no-go. This can skew the impacts of room cooling either conservatively or non-conservatively, and more importantly, can make emergent issues regarding these systems' availability/reliability difficult to analyze realistically. Key to PRA methods is that it is not realistic to either exclude these failure modes entirely or necessarily fail the cooled equipment given a loss of room cooling, since even with exceeded limits, there is some probability associated with the equipment failing within the PRA mission time (and therefore some probability of success).

The alternative is to assign a probability of failure to equipment, given a loss of room cooling. This paper discusses the details of such an approach using Interference Theory (developing probability density functions for overlapping variables), as applied at one utility. Peer review and NRC questions and resolutions to those questions pertaining to the process are also discussed.

4:30 pm: Expert Elicitation Process for ISLOCA Modeling – Process, Representative Results, and Lessons Learned, Steve Short, William Ivans (PNNL)

The Nuclear Regulatory Commission (NRC) is performing a full-scope, site Level 3 probabilistic risk assessment (PRA) study. During this study, issues were identified pertaining to the modeling (and quantifying) of independent and common cause failure (CCF) of isolation valves (i.e., large internal leakage) resulting in Reactor Coolant System (RCS)/ Emergency Core Cooling System (ECCS) interfacing system loss-of-coolant accident (ISLOCA) sequences. This class of accidents is a significant concern because the flow path can potentially render unavailable the ECCS needed to mitigate the accident and, furthermore, result in a loss of primary coolant outside the containment boundary and consequential core damage with a direct release path of fission products to the environment, referred to as a containment bypass accident. PRA studies have shown that this particular subclass of ISLOCAs can be significant contributors to public health risk resulting from core damage events.

To resolve these issues, the NRC Level 3 PRA (L3PRA) project team decided to employ an expert elicitation process. The focus of the expert elicitation was on ISLOCA sequences in the portions of the ECCS consisting of the Residual Heat Removal (RHR) System and Safety Injection System (SIS) for a Westinghouse four-loop pressurized water reactor (PWR). Information elicited from the expert panel included (1) large internal leakage failure rates for isolation valves, (2) conditional failure probabilities for isolation valves in series, (3) failure-to-close (FTC) probabilities for motor-operated isolation valves (MOVs) when exposed to large differential pressures, and (4) location of external break of pipe or other components due to over-pressurization.

While the NRC has employed expert elicitation for a variety of PRA applications over the years, there is a lack of standardized guidance that defines the necessary elements or minimum requirements of an expert elicitation process. As a result, implementation of expert elicitation by the NRC has not been consistent across applications. One significant exception is the Senior Seismic Hazard Analysis Committee (SSHAC) guidance for performing Level 3/4 probabilistic seismic hazard analyses. However, it is generally recognized that the SSHAC Level 3/4 process is too resource intensive for many applications in addressing PRA technical issues, especially those that are much more narrowly focused. Therefore, another objective of the ISLOCA expert elicitation project was to develop and implement an expert elicitation process to support development of NRC guidance on performing expert elicitation for PRA applications.

This paper describes the process used to elicit from the members of an expert panel the failure rates, failure probabilities, and break locations to be used in the development of the L3PRA ISLOCA model. Description of the process highlights innovative elements used to improve overall cost effectiveness while maintaining consistency with SSHAC Level 3/4 principles. Representative elicitation results are presented and selected lessons learned are discussed.

4:55 pm: Treatment of Expert Opinion Diversity in Bayesian Belief Network Model for Nuclear Digital I&C Safety Software Reliability Assessment, Ming Li (NRC), Hyun Gook Kang, Sang Hun Lee (RPI), Seung Jun Lee (Ulsan Natl Inst Sci Tech), Tsong-Lun Chu, Athi Varuttamaseni, Meng Yue (BNL), Jaehyun Cho (KAERI)

Since digital instrumentation and control systems are expected to play an important role in safety systems in nuclear power plants (NPPs), the need to incorporate software failures into NPP probabilistic risk assessments has arisen. In order to estimate the failure probability of safety software in NPP and incorporate it into a PRA model, a Bayesian belief network (BBN) model was developed which estimates the number of defects in the software considering the software development life cycle (SDLC) characteristics. In the model, due to a lack of sufficient safety software operation experience data, expert opinion was instead used to quantify the distributed node probability tables (NPTs) that are tables of random variables whose probabilistic distributions were aggregated from experts' elicitation. In addition, handbook data on U.S. software developments and V&V as well as the testing results of two example nuclear safety software were used to Bayesian update the BBN distributed NPTs in order to reduce the BBN parameter uncertainty from the diverse expert opinion. Based on the estimated NPTs, the number of defects at each SDLC phase is evaluated for the typical digital protection software (50 function points and Medium development, V&V quality). This study is expected to provide insight on several aspects of BBN model quantification for nuclear safety-related software reliability assessment, including the expert opinion elicitation and aggregation, the representation of the node probabilities using probability distribution, and the Bayesian updating of the NPTs using available software development data.

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Data and Parameter Estimation—II Continued

5:20 pm: Analysis on the Important Indicators in Bayesian Belief Network Model for RPS Software Reliability Assessment, Sang Hun Lee, Hyun Gook Kang (RPI), Tsong-Lun Chu, Athi Varuttamaseni, Meng Yue (BNL), Jaehyun Cho (KAERI), Ming Li (NRC), Seung Jun Lee (Ulsan Natl Inst Sci Tech)

Bayesian belief network model was developed in authors' previous research that quantifies the number of software faults based on software development life cycle (SDLC) characteristics of nuclear power plant (NPP) safety-related software. In a nuclear application, in order to effectively reduce the number of software defects in a target digital safety system, it is important to analyze the SDLC phases or related activities that are the major contributors to the final number of residual faults in the software. First, in order to identify the software development activities (attributes) as strong or weak indicators for the overall development or V&V quality of specific software development lifecycle (SDLC) phase, the indication measure of an attribute for development and V&V quality is proposed and the contribution of attributes' states to the quality nodes were analyzed. Secondly, the contribution analysis of quality nodes on the number of residual software defects is conducted considering the improvement of development and V&V quality from Medium to High quality in each SDLC phase. Furthermore, the cost of fixing detected defects passed from previous phases when the development and V&V quality is improved from Medium to High is assessed. This study is expected to provide an insight on analyzing the important SDLC phases and related software attributes to be considered when one desires to effectively optimize the SDLC for targeting fewer residual software defects in NPP digital safety-related system.

Risk-Informed Decision Making—II

Chair: Amir H. Mohagheghi (SNL)

Location: Grand Station IV **Time:** 3:40 - 5:45 pm

3:40 pm: Exelon Economic Enterprise Risk Modeling of a BWR, Donald Dube, Brian Albinson, Robert Wolfgang, Michael Saunders (JENSEN HUGHES), Gregory Krueger (Exelon)

This paper describes the results of a pilot of the Exelon Economic Enterprise Risk (EER) modeling at a commercial reactor site. EER modeling employs conventional risk assessment techniques but applies those methods to reactor transients and accidents at nuclear power plants where the accidents do not progress to a state of core damage. EER encompasses those efforts to recover from the transient or accident including repair cost, lost power production (or replacement power cost), and Regulatory impact to the extent that the latter can be quantified. A casualty database was developed consisting of historical industry events with significant regulatory and financial impact including reactor trips with complications, loss of offsite power, fire, external hazards, and various equipment performance issues. From these data, a consolidated set of casualty events was tabulated along with their generalized non-core damage plant impact states (PIs). Cost data from the casualty database were used as anchor points to map each non-core damage sequence to a PI and a corresponding generic cost bin. The three principal hazards considered in this study are full-power internal events (non-flood), internal flooding, and fire.

4:05 pm: PRA Insights Relating to the Loss of Electrical Sources – A WGRISK Survey, Jeanne-Marie Lanore, François Corenwinder, Gabriel Georgescu (IRSN), Attila Bareith (NUBIKI Nucl Safety Research Inst), Kevin Coyne (NRC), Per Hellström (Swedish Radiation Safety Authority), Milan Patrik (ÚJV REZ), Marina Roewekamp (GRS)

The Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) Working Group on Risk Assessment (WGRISK) has carried out a member country survey of Probabilistic Risk Assessment (PRA) insights relating to the loss of offsite electrical power sources (LOOP). This survey was performed in order to derive insights relating to PRA issues as well as safety improvements resulting from application of PRA methods. The survey covered both plant safety (e.g., design features that mitigate LOOP events) and PRA methodology aspects (e.g., data sources and LOOP modeling). The survey indicated that LOOP initiating events are included within the scope of PRA and often result in a dominant contribution to the overall plant risk.

Although some results are site or design specific, the survey also identified several important and generic issues resulting in both plant safety and PRA improvements. Examples of plant safety aspects are the use of different diverse electrical power sources, including mobile equipment. An example of improvements in PRA methodology is the need for appropriate data collection and modeling of recovery from failures of electrical sources. Significant modeling insights are related to the importance of the timing of key events (e.g., event sequencing, mission time duration, and station blackout coping time), recovery assumptions, the significance of common cause failures (CCF), the effects of multi-unit interactions (particularly in case of initiating events induced by external hazards), and the dominant role of human actions.

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Risk-Informed Decision Making—II Continued

4:30 pm: Risk Methodology for Assessment of Fuel Debris Retrieval Options at the Fukushima Daiichi Nuclear Power Station, William Ivans, Bruce Napier, Eva Mart, Michael Smith, Sandra Snyder (PNNL), Masahiko Kakami, Mari Marianne Uematsu, (Nucl Damage Compensation and Decommissioning Facilitation Corp)

The mission of the Nuclear Damage Compensation and Decommissioning Facilitation Corporation (NDF) includes the mid-to-long-term decommissioning planning of the reactors at the Fukushima Daiichi Nuclear Power Station. To carry out its mission, NDF needs to be able to assess worker safety and public health risks that currently exist at the Fukushima Daiichi Nuclear Power Station and the implications of various actions that may be taken to reduce such risks. Pacific Northwest National Laboratory (PNNL) entered into a contract with NDF to support NDF's mission by developing a risk assessment methodology that will support NDF's evaluation of options for transitioning melted and solidified fuels within each unit into safer, more stable, configurations given the ever-changing state of knowledge and large uncertainties. The underpinning of this approach, which incorporates both expert elicitation and adapts probabilistic risk assessment and hazard analysis techniques, is its organization of potential release mechanisms and accident scenarios into a more manageable set of safety issues, which are systematically characterized to allow for useful comparison between fuel debris retrieval options as well as the identification of risk drivers and discriminators. The approach also lends itself to be easily updated as the general state of knowledge improves and will serve as a vital input to prioritizing future investigations associated with human health and safety at the Fukushima site. This paper describes the overall approach developed by PNNL to risk-inform the comparison between fuel debris retrieval strategies and discusses lessons learned from implementing the approach.

4:55 pm: Risk Analysis Framework for Severe Accident Mitigation Strategy in Nordic BWR: An Approach to Communication and Decision Making, Sergey Galushin, Dmitry Grishchenko, Pavel Kudinov (KTH)

Severe accident management (SAM) in Nordic boiling water reactors (BWRs) employ ex-vessel debris cooling in a deep water pool. The success of the strategy requires (i) formation of a coolable porous debris bed; (ii) no energetic steam explosion that can threaten containment integrity. Both scenario (aleatory) and modeling (epistemic) uncertainties are important in the assessment of the failure risks. A consistent approach is necessary for the decision making on whether the strategy is sufficiently effective, or a modification of the SAM is necessary.

Risk Oriented Accident Analysis Methodology (ROAAM+) is a tool for assessment of failure probability to enable robust decision making, insensitive to remaining uncertainty. Conditional containment failure probability is considered in this work as an indicator of severe accident management effectiveness for Nordic BWR. The ultimate goal of ROAAM+ application for Nordic BWR is to provide a scrutable background in order to achieve convergence of experts' opinions in decision making. The question is: if containment failure can be demonstrated as physically unreasonable, given severe accident management strategy and state-of-the-art knowledge? If inherent safety margins are large, then the answer to the question is positive and can be demonstrated through risk assessment with consistent conservative treatment of uncertainties and by improving, when necessary, knowledge and data. Otherwise, the risk management should be applied in order to increase margins achieve the safety goal through modifications of the SAM (e.g. safety design, SAMGs, etc.). The challenge for a decision maker is to distinguish when collecting more knowledge and reduction of uncertainty in risk assessment or application of risk management with SAM modifications would be the most effective and efficient approach.

In this work we demonstrate a conceptual approach for communication of ROAAM+ framework analysis results and provide an example of a decision support model. The results of the risk analysis are used in order to provide necessary insights on the conditions when suggested changes in the safety design are justified.

5:20 pm: Enhancing Safety Associated with Post Fukushima Modifications: Role of Risk-Informed Decision Making, Sunil D. Weerakkody, Michael Montecalvo, Matthew Humberstone (NRC)

When nuclear power plant (NPP) systems are used to support emergency functions beyond their original design purposes, without creating unintended adverse effects, safety can be enhanced. In some situations, the potential enhancements to safety can be substantial. Use of Control Rod Drive (CRD) systems to inject water into the reactor coolant systems (RCS) in Boiling Water Reactors (BWRs) and use of the feed-and-bleed strategy to remove decay heat in Pressurized Water Reactors (PWRs) illustrate substantial safety gains that may be accomplished. In light of this, nuclear safety will be best served if the regulators and industry explore additional safety gains that can be gained from the flexible and diverse mitigating strategies (referred as FLEX strategies) whose intent is to mitigate beyond design basis external events (DBBEs).

This paper provides a summary of various activities that some US NPPs have initiated to leverage FLEX capabilities to enhance public safety, and enhance operational flexibility, by using them in areas beyond DBBEE. The paper articulates how a regulators' initiatives to incorporate FLEX strategies in RIDM could manifest into increases in public safety, which in some situations may far exceed the safety gains associated with reducing risks associated with DBBEEs.

**Technical
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TUESDAY, SEPTEMBER 26

TECHNICAL SESSIONS - 3:40 PM

Risk-Informed Applications—I (50.69)

Chair: Tatsuya Sakurahara (*Univ of Illinois*)

Location: Grand Station II **Time:** 3:40 - 5:45 pm

3:40 pm: Implementation of a Risk-Informed Surveillance Frequency Control Program for the NextEra Energy Nuclear Fleet, Anil Julka, Loren Heistand (*NextEra Energy*), Thomas Morgan (*Enercon*)

NextEra Energy obtained NRC approval for implementation of Risk-Informed Surveillance Test Frequency Control Programs (SFCPs) for several of its plants prior to 2015. However, there had been limited progress in actually making changes in surveillance test intervals. NextEra's senior management was aware of the potential benefits of the SFCP and wanted to promote more rapid adoption of surveillance test changes. NextEra management challenged the fleet to implement an extension of the EDG auto-start/loading test prior to each plant's next refueling outage. A strategic initiative was added for the fleet to implement this test extension for all eight units within an eighteen-month period from early 2015 through mid-2016. The effort required significant involvement from both site and fleet engineering, as well as PRA resources. All plants successfully met this challenge.

As a result of this intensive involvement of the staff at each plant in the process of implementing this complex test extension, followed by immediate savings in outage workload in the next outage, interest in the SFCP process was stimulated at most of the plants. Plant groups began to identify additional surveillance requirements as candidates for frequency extension. Many of the plants have since implemented additional test extensions as follow-on projects. The paper describes the process that was implemented, lessons learned from the projects, and insights for other plants that are seeking to promote interest in the SFCP program.

4:05 pm: Integrating Risk and Engineering Skills for 10 CFR 50.69 System Categorization, Brian Nolan, David Passehl (*JENSEN HUGHES*)

Exelon completed a company pilot program for 50.69 categorization in 2016. The overall categorization process is risk-informed, but within the overall process there are both Probabilistic Risk Assessment (PRA) based and deterministic tasks that require the appropriate skills. A key challenge to successful implementation is obtaining the proper balance of PRA and traditional engineering resources and coordinating the tasks appropriately.

The Exelon 50.69 categorization team addressed this issue by effectively integrating these resources into the categorization processes by examining the deterministic and PRA aspects of the system. This was highly dependent on system complexity and the degree to which system functions are modeled in the PRA. In addition, while some processes are deterministic in nature they can be more efficiently addressed by PRA risk engineers.

This paper provides insights into assembly of an integrated team of engineering and PRA personnel and the types of activities that can be effectively implemented in a "production mode". Among the insights gained through this activity was the importance to successful system selection and component categorization of the assembly of a range of engineering skillsets in addition to risk engineers.

4:30 pm: Lessons Learned from Development of the 10 CFR 50.69 LAR for Exelon Plants Using the NEI Efficiency Bulletin 17-09 Template, David Passehl, Barry Sloane (*JENSEN HUGHES*), Shannon Rafferty-Czincila, Eugene Kelly, Philip Tarpinian (*Exelon*)

A key task in obtaining regulatory approval to take advantage of 10 CFR 50.69 is to develop and submit the required License Amendment Requests (LAR). Preparing the LARs requires coordination and early interaction with multiple site resources. In addition, in accordance with the industry Delivering the Nuclear Promise (DNP) Efficiency Bulletin 17-09 on Industrywide Coordinated Licensing of 10 CFR 50.69 [Ref. 1], interactions with the 10 CFR 50.69 LAR Coordinating Committee are planned to support a timely and efficient LAR. To support these interactions, Exelon has worked closely with the Nuclear Energy Institute (NEI) and other utilities to ensure consistency with industry guidance.

This paper provides an overview of Exelon lessons learned in preparing and submitting its 10 CFR 50.69 LARs. It includes a discussion of (1) Exelon's work with the NEI Risk-Informed Engineering Programs (10 CFR 50.69) Task Force (i.e., RIEP TF) to establish the proper format and content of the LARs; (2) Exelon Risk Management project planning efforts for submitting multiple LARs; (3) Establishing and documenting technical adequacy and scope of the Probabilistic Risk Assessments (PRAs) to conform with the industry standard submittal template, including conduct of PRA Peer Review Finding Closure reviews; (4) Updating external hazard information for the LAR submittal; and (5) Efforts to streamline the Nuclear Regulatory Commission (NRC) review of the technical adequacy of the PRA across other risk-informed applications.

4:55 pm: Insights from Exelon 50.69 Passive Categorization Pilot at Limerick Generation Station, David Bidwell (*JENSEN HUGHES*), Maricarmen Trexler (*Exelon*), Barry Sloane (*JENSEN HUGHES*)

Exelon Corporation undertook a pilot effort beginning in 2016 to investigate the selection and categorization of plant systems at the Limerick Generating Station Units 1 and 2 in accordance with 10CFR50.69, "Risk-Informed categorization and treatment of structures, systems and components for Nuclear Power Reactors." The first two systems selected for categorization were the Core Spray system and the Reactor Enclosure HVAC system. This paper will discuss the challenges and insights gained from performing the passive categorization portion of the process.

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Risk-Informed Applications—I (50.69) Continued

5:20 pm: 10 CFR 50.69 Generic Categorization Process Development, David Teolis, Ryan Griffin, Kyle Hope (Westinghouse), Ogden Sawyer (AREVA), Ralph Chackal (Associated Engineering Resources, Inc.)

The Nuclear Energy Institute (NEI) initiative for Delivering the Nuclear Promise is an industry-wide effort to identify areas of operation at nuclear sites where efficiencies can be implemented to enhance safety, reduce operating costs, and increase the ability of the nuclear power industry to compete in the electrical generation market. Implementation of Nuclear Regulatory Commission (NRC) regulation 10 CFR 50.69 has significant potential to effectively reduce the costs associated with low risk safety-related structures, systems and components (SSCs). 10 CFR 50.69 is designed to permit licensees to categorize SSCs based on risk significance and implement alternatives to special treatments for low risk safety-related SSCs. The alternative treatments may be implemented at the convenience of each licensee with an approved license amendment request (LAR) to the degree that works best for individual utility needs. Recent industry studies and experience have identified substantial cost savings that can be realized by implementing 10 CFR 50.69 and the associated alternative treatments at domestic nuclear power plants. In an effort to reduce implementation costs, the Pressurized Water Reactors Owners Group (PWROG) is evaluating the potential to perform generic categorization of plant systems. The proposed approach is to generically apply the categorization results from one plant to other plants of similar design to the extent possible. This is expected to result in reduced categorization costs for individual sites of similar design. The PWROG is conducting a pilot program using the categorization results for the containment spray system for a Westinghouse Electric Company designed plant. The first step will be to determine the applicability of the results from this plant to other Westinghouse plants, as well as plants designed by Combustion Engineering and Babcock & Wilcox, and apply them to a representative plant of each design to the extent possible. The second step will be to complete the categorization process for each representative plant to account for differences in design. The end result would be a package with generic categorization results that can be used as a starting point for other plants to complete the required steps for site-specific categorization at a significantly reduced cost. This paper provides an overview of the process that will be followed to develop a generic categorization process.

Seismic PSA—V

Chair: Ovidiu Coman (IAEA)

Location: Waterfront **Time:** 3:40 - 5:45 pm

3:40 pm: Integrating Physical Degradation Modeling Within the Seismic Fragility Analysis of Nuclear Power Plant Equipment, Wei Wang (Politecnico di Milano), Sai Zhang (Tsinghua Univ)

In reality, the equipment properties are affected by physical degradation due to equipment usage, so that the actual seismic resistance (capacity) weakens with time. In this study, a physical modeling approach describing the equipment degradation progression with the equipment aging is integrated in a framework of time-dependent estimation of equipment fragility in the Seismic Probabilistic Safety Analysis (SPSA) of Nuclear Power Plants (NPPs), based on the conventional log-normal fragility model. Monte Carlo (MC) simulation is used to propagate the uncertainties affecting both the physical degradation process and the seismic fragility parameters.

A piping system of a secondary cooling circuit of an experimental accelerator-driven system is taken as case study, wherein, a flow of an organic diathermic oil results in a gradual creep deformation process of the piping system.

4:05 pm: Relay Chatter in Seismic PRAs, Andrea Maioli, Clarence Worrell, David Gerlits, Steven Satter (Westinghouse), Andrew Masiunas, Mark Etre (JENSEN HUGHES)

Relay chatter is one of the most characteristic impacts modeled by seismic Probabilistic Risk Assessments (PRA) and has proved a significant risk contributor by a number of recent S-PRAs. The primary effect of relay chatter is spurious actuation of components or the locking and sealing of a control relay into an unwanted position, thus preventing the correct actuation of components. The relay chatter assessment requires significant effort due to the large number of chatter-susceptible devices directly or indirectly modeled by the PRA. In addition many of these devices may not have been completely addressed by previous studies, for example the ASI-46 programs implemented at a number of sites. This paper discusses some of the assumptions, modeling approaches covering initiating events and mitigation strategies, and challenges encountered in modeling relay chatter. The discussion covers topics such as defining the analysis scope (i.e., which relays and which devices will be addressed), challenges associated with relays identification, the necessary steps associated with screening (functional versus seismic screening) to maintain the model solvable, modeling techniques including consideration of correlation between different relays, dealing with generic versus cabinet-specific amplification factors, and capacity. Some lessons learned regarding the plant accident sequences that involve relay chatter are also provided. A critical input is the availability of information from previous studies, which makes a single approach impractical for all plants. This paper is based on multiple seismic PRAs at different utilities, which have been developed by the authors, and therefore provide a somewhat representative landscape of possible variations.

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Seismic PSA—V Continued

4:30 pm: How to Find Synergy Between Different Teams in a Seismic PRA Project, Parthasarathy Chandran, Patrick Fussell (*Southern Nucl Operating Co*), Adam Blood (*Empyrean Services*)

After the Fukushima earthquake, the Nuclear Regulatory Commission had requested through the 50.54(f) letter, that all the utilities assess the effects of ground motion response spectra at their site. This request lead to the development of the Seismic Probabilistic Risk Assessment (SPRA). The SPRA involves several different aspects of science and engineering to create a model of the plant's response to a seismic event. Each field has their own jargon and provides different information that is needed to be effectively communicated and incorporated to build an accurate logic model of the site. The seismic database is a tool that is used to facilitate communication between these groups by showing the relationship between the equipment type, location, relay chatter analysis, fragility analysis, fire and flood source, as well as several other key pieces of information to build a complete picture for the plant on an equipment by equipment basis. By having access to all the information in a single location, the user can quickly find associated documents, notes and results to support and refine the model for the plant. The seismic database tracks and stores decisions and results over multiple hazard versions, which facilitates efficient updates to a new hazard version and maintenance during the updates. Another benefit to storing the information in a single database is the ability to cross reference the data and generate both reports and new values from the database. This paper discusses the communication issues solved by a database, the types of information stored in a database and how the information can be cross referenced to allow efficient updates to the logic model.

4:55 pm: An Effective and Efficient Methodology to Update the Seismic Fragilities of SSCs over Different Seismic Hazard Updates, Benny Jebuna Ratnagaran, Parthasarathy Chandran (*Southern Nucl Operating Co*), Adam Blood (*Empyrean Services*)

US utilities started incorporating the SPRA models in their Risk Informed Applications to attain a more realistic risk insights. Some of the Risk Informed Applications that currently uses SPRA are namely 50.69, Tech Spec Appendix 4B and 5B. As part of these applications, the models have to be maintained and updated on a periodic basis to reflect the current state of knowledge. One of the important parameter that could affect the validity of these models over time is the site-specific probabilistic seismic hazard analysis (PSHA). This is due to the ever evolving understanding of ground motion attenuation and typically updates to the site-specific seismic hazard curves would trigger a change in the seismic response evaluation and in-structure response spectra (ISRS).

The seismic fragilities of equipment are based on the input ISRS and new fragilities must be developed compatible to the updated site-specific seismic hazard curves. But updating the seismic fragilities require a significant amount of effort. To account for higher ground motion levels, scaling of ISRS is considered a technically sound approach and guidance on scaling methods is provided in EPRI documents EPRI NP-6041 and EPRI 103959. But the scaling of these responses will be based on the similarity between the shapes of the previously and newly generated site-specific seismic hazard curves. There is limited guidance on scaling fragilities if the shapes are different.

In this study, we present a framework to minimize the total amount of effort required in acquiring the new fragility data for seismic risk assessment of equipment in a cost effective and efficient manner. The proposed scaling approach would be applicable irrespective of the similarity in shapes of the previously and newly generated site-specific seismic hazard curves. The application of this framework is illustrated using examples of different classes of equipment.

5:20 pm: Recent Advances in Seismic Fragilities for SPRAs, Gregory S. Hardy (*Simpson Gumpertz & Heger Inc.*), Robert P. Kassawara, John Richards (*EPRI*)

Use of seismic fragility assessment within the nuclear power community has increased significantly as a result of the earthquake that affected the Fukushima nuclear power plant. The accident at Fukushima was the initiative in most countries to perform a reassessment of the seismic hazard at all nuclear plant sites using modern probabilistic seismic hazard assessment techniques. Those nuclear plant sites where new seismic hazard assessments indicate a higher seismic response have typically implemented seismic probabilistic risk assessments to assess the impact of the increased seismicity. Seismic fragility is also becoming a more common tool in a broader group of industries, expanding to non-nuclear industries, as risk-informed approaches are becoming the norm for new design and for evaluation. This paper summarizes the key elements of seismic fragility and reports on the most recent changes that impact practitioners.

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WEDNESDAY, SEPTEMBER 27

TECHNICAL SESSIONS - 9:00 AM

Risk Aggregation—III

Chair: Stanley H. Levinson (AREVA)

Location: Grand Station II **Time:** 9:00 - 9:50 am

9:00 am: Spatio-Temporal Probabilistic Methodology and Computational Platform for Common Cause Failure Modeling in Risk Analysis, Tatsuya Sakurahara, Grant Schumock, Tadashi Murase, Zahra Mohaghegh, Seyed Reihani, Ernie Kee (Univ of Illinois)

This research paves the way for a paradigm shift in Common Cause Failure (CCF) analysis. CCF, and in general dependent failure, modeling is one of the most important topics in Probabilistic Risk Assessment (PRA). In classical PRA, CCF is addressed by data-driven parametric approaches and is based on historical CCF data. There are two major problems with the existing parametric methods: (i) their accuracy relies on the availability and quality of historical CCF data, and (ii) they lack an “explicit” connection between the root causes of dependencies, embedded in the failure mechanisms and CCF probabilities in PRA. Therefore, the parametric CCF approaches are limited in providing “cause-specific” quantitative insights that would be helpful in (a) reducing CCF occurrence, and (b) making design decisions for new systems (e.g., new reactors). This research develops a simulation-based CCF analysis that integrates explicit and implicit (stochastic) modeling of root causes and coupling associated with CCFs. This methodology is grounded on numerical outputs from a spatio-temporal probabilistic model of underlying failure mechanisms, which capture chains of causal factors leading to CCF events. The theoretical foundation of this proposed approach is described, focusing on how two key dimensions of CCF, i.e., shared root cause(s) and coupling mechanism(s), are captured in the simulation environment. The computational procedure is developed to operationalize the simulation-based CCF methodology in an Integrated PRA, which is a hybrid PRA framework that connects deterministic simulation models to classical PRA. The implementation of this new CCF methodology is conducted using a case study for a fire scenario at a nuclear power plant. This methodology, not only fills the two major gaps (i.e., relying solely on historical data and lack of explicit connection to root causes of failures) in the existing CCF methods, but also provides two additional features for CCF analysis: (1) modeling of “spatio-temporal” dependencies, and (2) treatment of the “uncertainties” in root causes of dependencies and in the associated failure mechanisms. Ongoing research by the authors is focusing on the incorporation of causal models for social root causes of CCFs (e.g., human and organizational deficiencies in maintenance) into this new methodology to connect them with the physical failure mechanisms associated with dependent failures in PRA.

9:25 am: Issues in Dependency Modeling in Multi-Unit Seismic PRA, Taotao Zhou, Mohammad Modarres (Univ of Maryland), Enrique López Droguett (Univ of Maryland, Univ of Chile)

This paper addresses issues related to dependency modeling in multi-unit seismic probabilistic risk assessment of nuclear power plants. The concept of multi-unit probabilistic risk assessment (MUPRA) is briefly summarized. The current methodologies to seismic-induced dependency modeling are discussed and grouped into four main approaches. Several issues are identified in the present methodologies for consideration of dependencies in seismic MUPRA. It is shown that the β -factor and correlation coefficient approaches to account for dependencies are different. Further, the paper highlights the weakness of the Reed-McCann method in modeling dependencies. These findings underline the need for improved methods for characterizing dependencies in the multi-unit structures, systems and components (SSCs) with shared features and their links in the MUPRAs.

Human Factors and Behavioral Sciences

Chair: Justin Pence (Univ of Illinois)

Location: Grand Station III **Time:** 9:00 - 10:15 am

9:00 am: Optimizing Human Performance Through Systems Engineering: Lessons from the UK Defence Sector, Kelly Davies, Clare Borrás (Corporate Risk Associates)

Since 2003, the UK Ministry of Defence (MoD) has driven forward the development of structured methodologies for improvements to Human Centered Design and Human Factors Integration (HFI). Part of this drive has been to embed HFI within Systems Engineering (SE), the structured engineering method defined by the International Council of Systems Engineering.

This paper presents the key success factors supporting good HFI within UK Defence and how Corporate Risk Associates (CRA) have applied these lessons across other high-risk industries. Through experience, CRA have repeatedly found specific HFI activities to be directly linked to the success factors of a range of projects in terms of meeting requirements, effective identification and prioritisation of human related risks and issues, and stakeholder and end user buy in. These factors address the issues and barriers to effective HFI which include effective base lining, early engagement of Human Factors (HF) expertise and risk analysis.

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TECHNICAL SESSIONS - 9:00 AM

Human Factors and Behavioral Sciences Continued

9:25 am: Functional Allocation to Balance the Requirements of Humans and Automation, Les Ainsworth
(Corporate Risk Associates)

One of the first human factors involvements in designing a new control room or major system is the allocation of functions (AoFs) between humans and the system. Typically this has involved undertaking a high-level task analysis and assigning specific functions to the human or the system, based upon the recommendations from an updated version of Fitts' List.

Developments in computer-based automation present opportunities for effective collaboration between human and automation, so that this is no longer just a binary choice. Automation has also introduced some potential problems, such as reduced situational awareness; automation-induced decision errors; decreased overall vigilance; difficulties when automation only operates effectively only within boundary conditions and 'automation surprises' when an automated system does not function as predicted by an operator.

This paper describes an AoF approach that defines the level of collaboration between humans and automation. The method uses task analysis to define the main functions, followed by an initial AoF process at the subfunction level and then some stress testing of high workload conditions

In order to utilize the potential benefits of dynamic function allocation, the initial allocation process identifies a potential range of collaboration between humans and automation. For normal operations the option with greater human involvement should be selected, but as workload gets higher, more automated support can be provided, whilst still enabling the operator to maintain reliable performance and to sustain other critical functions such as situational awareness and vigilance. However, it is beyond the scope of this study to define the criteria for determining when the system should provide this additional support.

9:50 am: The Fidelity of Human Reliability Data Collected from Training Simulators – Comparing with Human Reliability Data Identified from Operation Experience, Jinkyun Park, Yochan Kim, Wondea Jung (KAERI)

In this study, two sets of HEPs (Human Error Probabilities) are compared. The first set of human reliability data is obtained from the analysis of simulation records that were collected from the full-scope training simulator of domestic nuclear power plants (NPPs). The second set of HEPs is extracted from the analysis of event reports reflecting the operation experience of domestic NPPs. Consequently, it seems that two sets of HEPs are reasonably congruent with each other within a factor of 5.

Level 2 and Level 3 PSA—I

Chair: Kevin R. O'Kula (URS Management Solutions)

Location: Grand Station IV **Time:** 9:00 - 10:15 am

9:00 am: Protective Systems: An Application of Regulatory Guidance for ECCS Sump Performance, Ha Bui, Tatsuya Sakurahara, Wen-Chi Cheng (Univ of Illinois), Timothy Crook, Rodolfo Vaghetto, Martin Wortman (Texas A&M), Zahra Mohaghegh, Seyed Reihani, Ernie Kee (Univ of Illinois), Dominic Muñoz (Alion Sci Technol), David Johnson (UCLA), Wes Schulz, Fatma Yilmaz (STP), Vera Moiseytseva (YK.Risk)

The work described in this paper reports on the additional research and development conducted in support of the Risk-informed Over Deterministic (RoverD) approach to address the concerns raised in Generic Safety Issue 191 (GSI-191). In this work, a method is developed using a RELAP5-3D PWR model coupled with a MELCOR large, dry containment model to obtain conservative containment pressure estimates used together with conservative sump pool temperature estimates in order to comply with the regulatory guidance. The current regulatory guidance on Emergency Core Cooling System (ECCS) performance, "Revised Content Guide for Generic Letter 2004 02 Supplemental Responses, November 2007" (ADAMS Accession No. MLO73110278), and Regulatory Guide 1.82, "Water Sources for Long Term Recirculation Cooling Following a Loss of Coolant Accident," is briefly reviewed as it applies to flashing in the ECCS pumping system. Because the guidance does not specify the method that should be used to comply with the conservative assumptions, an acceptable approach must be developed by the utility engineer.

9:25 am: Full Risk-Informed Resolution of the GSI-191: Simulations of LOCA Scenarios Using RELAP5-3D and MELCOR, Rodolfo Vaghetto, Ernie Kee, Timothy Crook, Yassin Hassan (Texas A&M)

Different approaches have been provided by the US Nuclear Regulatory Commission for addressing the concerns raised by the GSI-191. The approaches proposed by the STPNOC pilot plant adopted deterministic thermal-hydraulic calculations supported by system codes to simulate the plant response under numerous configurations and plant conditions. The results obtained supported the Probabilistic Risk Assessment analysis, and helped the resolution of the GSI-191, proving the importance of selecting and using specialized system codes and modeling technique to perform realistic simulations of the reactor system in support of both deterministic and risk-informed methods.

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TECHNICAL SESSIONS - 9:00 AM

Level 2 and Level 3 PSA—I Continued

9:50 am: Modeling of SiC Cladding Behavior During Severe Accidents, Wison Luangdilok (*Fauske & Assoc*), Peng Xu (*Westinghouse*)

The capability to model the behavior of SiC as a candidate Accident Tolerant Fuel (ATF) cladding material during severe accidents is currently limited by the lack of knowledge in the behavior of SiC at high temperatures. Through the review of current research literature, this paper describes failure criteria of SiC that are needed in a severe accident code modeling of ATF so that a complete evaluation of the benefits/drawbacks of ATF with respect to severe accidents can be performed. In this paper, the time from initiation of an accident to cladding failure is defined as the coping time. The benefit of ATF with respect to severe accidents is realized when the coping time is substantially extended. The new ATF fuel materials including U_3Si_2 that are being developed usually have higher thermal conductivity than the conventional UO_2 fuel which is enclosed inside a fuel rod made of cladding materials such as SiC that have steam oxidation resistance at high temperatures. The cladding failure mechanism and the release of fission products after cladding failure are important considerations in the ATF evaluation. For SiC, in addition to creep, "thinning" of SiC cladding at high steam pressures and temperatures where SiC cladding undergoes oxidation and volatilization is another mechanism that should be taken into account in the modeling of potential failure mechanisms.

High Winds PSA—III

Chair: Nicholas Lovelace (*JENSEN HUGHES*)

Location: Grand Station V **Time:** 9:00 - 10:15 am

9:00 am: Progressive Failure of Building Cladding in High Winds, Sudhan S. Banik, Lawrence A. Twisdale, Jr., Peter J. Vickery, Shahriar Quayyum (*Applied Research Associates, Inc.*)

Turbine buildings at nuclear power plants are typically large metal clad buildings. These buildings may house structures, systems, and components (SSC) used to control and shut down the plant. Failure of the metal cladding during a high wind event poses a threat to SSCs from structural interactions, wind borne missile impact, and wind-driven rain water entering the building. In addition, offsite power supply components are often adjacent to the turbine building. Failed cladding can drop onto or fly into nearby power lines, potentially producing electrical faults. A computer based explicit modeling approach has been developed to simulate the progressive failure of turbine building cladding in high winds. This 3-D computational methodology includes explicit element-by-element modeling, directional wind loads, progressive internal pressure modeling, and load and resistance analysis of cladding failure modes, including supporting girts and purlins. The model uses Monte Carlo simulation method to sample loads and resistances to determine the failure of a building component. Wind loads on a component include the aerodynamic forces produced by the dynamic pressure component of the wind flow, the resultant internal pressure and the atmospheric pressure change (APC) load within the core of a tornado. The resistances of the cladding (wall and roof) and the supported girts and purlins, for all the possible failure modes due to wind loads, are based on strength calculation, experimental results, and product catalog. In this paper, we present the progressive failure modeling methodology and illustrate results for several structures. Cladding fragilities are developed and compared to failures observed in wind storms. Fragilities obtained from the detailed (i.e. explicit) analysis are compared with those obtained from code-based fragility analysis. Comparisons are also made for tornado and straight wind fragilities.

9:25 am: Fragility Analysis of Extreme Wind and Wind-Driven Rainwater, Peter J. Vickery, Lawrence A. Twisdale, Jr., Abby Liu, Sudhan Banik (*Applied Research Associates, Inc.*), Artur Mironenko, Michael S. Kitlan, Jr. (*Duke Energy*), Nicholas Lovelace (*JENSEN HUGHES*)

High winds are often accompanied by wind-driven rain that can enter buildings through failed roof deck and wall cladding elements. In these events, vulnerable interior electrical equipment could fail from rainwater deposition onto the equipment. A simulation methodology to estimate the quantity of rainwater entering a building given failure of the building envelope in a wind storm has been developed for wind-driven rain analysis. A rain model is coupled with a 3-D progressive-failure building analysis methodology that includes explicit locations of potentially vulnerable interior electrical equipment. These two models are then used to develop equipment fragilities from wind-driven rain for high wind Probabilistic Risk Analyses (PRAs). The motivation for this wind-driven rain modeling is to remove conservatisms in the PRA model in which building cladding failures in high winds are assumed to result in vulnerable interior equipment failures. To the authors' knowledge, this is the first such wind-driven rain fragility analysis for plants in the U.S.

9:50 am: Non-Parametric Method for Wind Pressure Fragility Analysis, Lawrence A. Twisdale, Jr. (*Applied Research Associates, Inc.*)

Lognormal probability fragility distributions have been traditionally used to develop fragilities in nuclear plant probabilistic risk assessments. An enhanced non-parametric code-based (NPCB) methodology is presented in this paper for fragility analysis with application to wind pressure fragilities. Arbitrary probability distributions (i.e., non-parametric) can be used in the modeling of the individual factors and the uncertainty parameters. The method incorporates pressure zone and enclosure state (internal pressure) logic that is based on well-established wind code design principles. The NPCB methodology provides for more accurate modeling of wind pressure fragilities and reduced conservatisms in the important lower tail of the fragilities.

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TECHNICAL SESSIONS - 9:00 AM

Seismic PSA—VI

Chair: Paul J. Amico (JENSEN HUGHES)

Location: Waterfront **Time:** 9:00 - 10:15 am

9:00 am: SPRA Alternate Screening Method, Robert Miller (Enercon), Parthasarathy Chandran (Southern Nucl Operating Co)

Determining which equipment and components in the Seismic Equipment List (SEL) will be risk significant while creating a Seismic Probabilistic Risk Assessment (SPRA) model can be challenging. Representative fragilities are typically not available during the early stages in the SPRA development process. Refined SPRA models are also not available until late into the project. S-CDF and S-LERF results can change significantly from those based on initial project estimates and preliminary PRA modeling results. These factors make use of normal PRA importance measures, such as Fussell-Vesely, which are less than optimal for ranking equipment significance while creating a SPRA. Early screening insights are useful for focusing fragility and modeling resources on the systems and equipment where modeling refinement will be most effective. Good being the goal of creating the most accurate and realistic SPRA model possible. An alternate screening method will be described that can be used as a screening tool throughout the project. The alternate method use can start early in the SPRA development process. This can provide useful early modeling and screening insights. This method is based on developing anticipated S-CDF and S-LERF values. SEL components are categorized by their expected failure contribution in the logic model. For early evaluation, it is adequate to classify component failure as either direct or indirect to S-CDF and S-LERF. Using the site-specific hazard, HCLPF values corresponding to threshold values for 1%, 10%, etc. of anticipated S-CDF and S-LERF can be calculated for S-CDF and S-LERF. Critical equipment to the SPRA can be identified early in the SPRA project using the direct and indirect equipment categorization and the threshold levels for S-CDF and S-LERF to evaluate which SSC's in the SEL are going to have significant S-CDF and S-LERF contribution. The calculated HCLPF threshold levels can also be used throughout the project as a screening criteria. All screening decisions, regardless of method, are then verified at the end of the project based on their significance in the final SPRA model.

9:25 am: Seismic PSA of WWER440 Type Reactors in Slovakia, Zoltan Kovacs, Robert Spenlinger (RELKO Ltd)

After the Fukushima accident the seismic risk of the Slovak NPPs equipped with WWER440 type reactors are being re-evaluated. Seismic level 1 and 2 PSA for all operating modes are being prepared. The objective of writing this paper is to compile, in a systematic manner, information on seismic safety technology and knowledge regarding the plants. Information will be provided about the PSHA (Probabilistic Seismic Hazard Analysis) of the sites, fragility analysis of buildings and components and modeling using the RiskSpectrum PSA code. The paper describes the seismic-induced initiating event definition and grouping and the accident sequence modeling. Single systemic event trees were constructed for each group of seismic-induced initiating events to model the plant response in the form of accidents sequences. Generic event trees were constructed to model simultaneous occurrence of seismic-induced initiating events in any combination during a seismic event. Description will be provided also about: overall seismic risk, dominant contributors to the seismic risk, acceleration ranges (earthquake intensities) which dominate the plant risk, comparison of seismic risk to risk originating from other events and proposed safety measures to improve the plant safety given seismic event. The seismically-induced multi-unit interactions are not part of the analysis. It is task for the future.

9:50 am: Updates to the EPRI Seismic-Induced Fire and Flood Methodology Resulting from Pilot Application, Paul Amico, Pierre Macheret (JENSEN HUGHES), Robert Kassawara (EPRI)

EPRI has recently developed a new methodology for assessing the risk from seismic-induced internal fires and floods (SIFF). Before publishing the methodology, it is being subjected to a number of pilot applications. The SIFF methodology was developed over a two year period and, while not yet published, is summarized in a paper presented at ICONE in July 2017 [Amico, Macheret, and Kassawara, "An Advanced Method for Evaluating Risk from Seismically-induced Fires and Floods"]. There are a number of ongoing pilot applications of this methodology, by Duke Energy, Southern Nuclear, DC Cook, and Callaway, some of which are reported in other papers at PSA 2017. This paper reports on the changes that are being made to the methodology as a result of the pilot applications. Although the pilots have not yet been completed, a number of insights have already been coming in and are discussed in this paper. By the time of the conference, the first pilots will have been completed and the conference presentation will present even further enhancements to the methodology right up to the date of the conference.

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TECHNICAL SESSIONS - 10:45 AM

Risk Aggregation—IV

Chair: Ovidiu Coman (IAEA)

Location: Grand Station II **Time:** 10:45 am - 12:00 pm

10:45 am: Treating Common-Cause Failures in Multi-Unit PRAs, Sai Zhang, Jiejuan Tong (Tsinghua Univ), Jing Wu (Huaneng Shandong Shidao Bay Nucl Power Co)

The term “common-cause failures (CCFs)” refers to a failure of two or more components arising from a single shared cause during a short period of time. Methods for incorporating CCFs into Probabilistic Risk Assessments (PRAs) have evolved over the past several decades, including various models (for example Beta Factor, Alpha Factor and Multiple Greek Letter models) and clear guidelines (such as NUREG/CR-5485 and IAEA-TECDOC-1135). However, the PRAs were originally developed for a single reactor, i.e. the analysis is confined to one reactor at a time, rather than the integrated risk from all the reactors and other radioactive sources on the whole site. Regarding the treatment of CCFs in MUPRAs, several questions may arise: (1) should the common-cause component groups (CCCGs) be expanded to include inter-unit CCFs? (2) if the answer is “yes”, how to handle the CCCGs of very high order (e.g. involving more than 5 or 6 components)? This paper conducts base case and sensitivity analyses using different CCF models in an MUPRA, trying to provide insights for the above questions. This paper is based on the MUPRA results of a real two-unit nuclear power station, through which the CCFs are recognized as a primary contributor to the site risk.

11:10 am: Parametric Estimation of Multi-Unit Dependencies, Taotao Zhou, Mohammad Modarres (Univ of Maryland)

This paper presents the parametric estimation of the dependencies and causal events (i.e., the dependencies among diverse events) occurring in nuclear plant sites with multiple reactor units. The approach used is based on the analysis of U.S. Licensee Event Reports (LERs) to identify and parametrically assess the frequency of observed multi-unit incidents and failure events including the uncertainties. The parametric estimates cover the years 2000-2011 of the LERs reported to the U.S. Nuclear Regulatory Commission. A simple example is outlined to illustrate the application of these parametric estimates in multi-unit risk assessments that also confirms the importance of considering these dependencies.

11:35 am: Exploring the Need for Standard Approaches to Addressing Risk Associated with Multi-Module Operation in Plants Using Small Modular Reactors, Mark A. Caruso (NRC)

Due to proposed applications of small module reactors, staff of the U.S. Nuclear Regulatory Commission (NRC) explored the need for standard approaches for addressing risks associated with multi-module reactor plants. Such an exploration is warranted in light of (1) the emergence of new nuclear power plant designs in which multiple small reactors are coupled together in a way that allows variable electric capacity (and associated plant-wide radiological inventories) potentially up to levels approaching those in large nuclear power plants; and (2) assessment of nuclear power plant risk has traditionally focused only on operation of a single nuclear reactor for power production. The NRC has recently received an application for certification of a power plant design that includes 12 small reactor-modules that operate simultaneously for the production of electricity and share structures, systems, and components (SSCs). The application includes a quantitative assessment of multi-module risk. The NRC has also recently engaged with nuclear industry organizations on an approach to license advanced reactor designs that includes a quantitative assessment of risk from multiple module operation. A number of standards for a technically adequate probabilistic risk assessment (PRA) now exist. Some of these standards include requirements to ensure that multi-module or multi-unit operation is accounted for in the identification and quantification of initiating event frequencies and that sharing of SSCs is accounted for in development of success criteria. This paper summarizes research on the development of multi-unit and multi-module PRA that is being done and the extent to which standards for assuring the technical adequacy of such a PRA exist.

Uncertainty Analysis and Modeling

Chair: Martina Kloos (GRS)

Location: Grand Station III **Time:** 10:45 am - 12:00 pm

10:45 am: Interpretation of PSA Results Using Semantic Analysis of Minimal Cutsets, Gennadi Loskoutov, Per Hellström (Swedish Radiation Safety Authority)

Accurate and precise interpretation of PSA results is crucial for identification of weaknesses and planning for safety improvements. The major risk drivers need to be identified, and this is mainly done using different importance measures. Such methods take into account event probabilities and MCS probabilities, which favors events with high probabilities. Identification of risk drivers related to low-probability events may be inaccurate if the identification is done only by using importance measures. To avoid this problem, major analysis clusters can be identified using the technique, which is used to identify topics in corpora of text documents. Such techniques are based purely on occurrence of a certain word in a certain text document. Treating MCS lists as text documents and event identifiers in MCS as words, it is possible to create meaningful clusters of analysis cases and to identify dominant events for each cluster. This information can be used for result interpretation along with identification of risk drivers by calculating importance measures.

One of the strongest features of the suggested method is that it does not require any prior knowledge about particular PSA model to perform clustering. The only inputs required for the method are MCS lists for analyses defined in the model. Therefore, this method can be used by regulatory bodies and other organizations, which are not directly involved in the development of PSA models, but for which accurate interpretation of PSA results is crucial.

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Uncertainty Analysis and Modeling—I Continued

11:10 am: Uncertainty and Sensitivity Analysis of LSTF Small Break LOCA Tests Using RELAP5 and RAVEN, Ikuo Kinoshita (*Inst of Nucl Safety System, Inc.*), Cristian Rabiti, Andrea Alfonsi (*INL*)

RAVEN is a software framework that allows users to perform parametric and stochastic analysis based on the response of complex system codes. In the present study, RAVEN is interfaced with RELAP5/MOD3.2 code which has been successfully used for accident scenario analyses such as small break loss-of-coolant accidents in PWRs. Furthermore, RAVEN is enhanced to treat physical model uncertainties in RELAP5 code by an external input file of uncertainty parameter multipliers. The Best Estimate Plus Uncertainty analysis is conducted by the RAVEN/RELAP5/MOD3.2 code for the “Intentional depressurization of steam generator secondary side” which is an accident management procedure in a small break loss-of-coolant accident (LOCA) with high pressure injection system failure. The input parameter uncertainty propagation analyses are performed for the ROSA/ Large Scale Test Facility (LSTF) secondary-side depressurization tests. The 95%/95% tolerance limit values of the output parameters are obtained for both test cases. It is confirmed that the code predicted well the major event progressions of the accident for both test cases and the 95%/95% uncertainty bands of the peak cladding temperatures include the measured values. It is judged that the RAVEN/RELAP5 code is valid for conducting the uncertainty analysis of the small break LOCA scenarios.

11:35 am: Simple Method to Account for the State of Knowledge Correlation, Michael Lloyd (*Risk Informed Solutions*), Jason Hall (*Entergy*), Ross Anderson (*Enercon*), David Teolis (*Westinghouse*)

This paper describes the State-of-Knowledge Correlation (SOKC), how it impacts risk results, and describes a simple and practical method to incorporate the effect of the SOKC into cutset results without performing an uncertainty analysis. This method involves developing “State-of-Knowledge Correlation Multipliers” and applying them to cutsets containing cutsets affected by the SOKC (i.e., cutsets containing multiple events which use the same component failure mode data). The SOKC Multipliers are always greater than one and, thus, increase the calculated frequency of affected cutsets. Interfacing System LOCA (ISLOCA) cutsets are particularly affected by the SOKC. This paper also describes a practical method to automate the application of the SOKC Multipliers to cutsets quantified using the EPRI CAFTA® platform.

Configuration Risk Management

Chair: Barry D. Sloane (*JENSEN HUGHES*)

Location: Grand Station IV **Time:** 10:45 am - 12:00 pm

10:45 am: Lessons Learned in the Development of an At-Power Risk Monitor for the AP1000® Plant, Nathan Larson, Rachel Christian (*Westinghouse*)

The use of an at-power risk monitor is essential to the day to day operations for many nuclear power plants (U.S. based and overseas) to support at-power testing and maintenance activities. For operating plants, risk monitor models are well developed and exercised on a daily basis such that they can be considered mature and ingrained in the day to day processes. This includes methods in place to ensure seamless use and stable risk metric criteria for determining when plant configurations are such that the risk is increased, requiring appropriate risk management actions.

During the development of the at-power risk monitor model for the AP1000^{®(1)} plant, various challenges and ensuing lessons learned were encountered. Some of these challenges were the result of the nature of the advanced plant design and the absence of operational at-power test and maintenance experience. For example, the application of existing risk metrics (core damage frequency (CDF) and large early release (LERF)) and the industry standard definitions for risk significance present a challenge (1E-06 and 1E-07 for CDF and LERF, respectively) for accurately identifying and presenting all risk associated with the advanced plant design. This paper will present and discuss this and other various lessons learned and challenges that were encountered during the development of the initial at-power AP1000 plant risk monitor model.

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TECHNICAL SESSIONS - 10:45 AM

Configuration Risk Management Continued

11:10 am: The Instantaneous LOOP Frequency, Ross C. Anderson (*Enercon*), Jason Hall (*Entergy*), Michael Lloyd (*Risk Informed Solutions*)

The annual-average Loss of Offsite Power (LOOP) frequency can be estimated as

$$IE - LOOP(all) = \# \text{ events} / \# \text{ unit years} \quad (1)$$

This number consists of several components, including random events and the consequences of specific effects, including switchyard maintenance and severe weather. In recent years, the overall LOOP frequency has been partitioned into several sub-initiators, roughly correlated to some of these effects.

These numbers are valid for annual-average frequencies but are of limited use for instantaneous rates. An improved approach is needed for 10 CR 50.65(a)(4) (the Maintenance Rule) and some Significance Determination Process applications, for example. The baseline for an instantaneous rate will consist only of random events; the risk increase calculation requires consideration of the increased LOOP vulnerability when any additional effects, such as severe weather, are present.

Improved analysis of instantaneous LOOP rates requires an estimate of the time fraction for these correlated effects. For example, the LOOP frequency due to switchyard maintenance (SM)-related events is

$$IE - LOOP(SM) = \#SM \text{ events} / \#SM \text{ unit years} \quad (2)$$

The numerator is readily available in Equation 2 from industry data but the denominator is not. It will be some fraction of the total sample years. Informal data at Arkansas Nuclear One have been collected to develop a ballpark estimate of the Equation 2 denominator for some of these effects. Similar estimates have been developed for other effects.

This paper discusses these estimates, the degree to which they may be representative of the industry database as a whole, and its implications for instantaneous risk analysis.

11:35 am: Lessons Learned from Incorporating Temporary Equipment into the Palo Verde Configuration Risk Management Program Using NEI 16-06, Michael Wittas, Michael Powell (*Arizona Public Service*)

In September of 2016, Palo Verde completed its first integration of temporary equipment into the Palo Verde configuration risk management program following the guidance in NEI 16-06, *Crediting Mitigating Strategies in Risk-Informed Decision Making*. While Palo Verde has incorporated temporary equipment into its qualitative (i.e. defense-in-depth) Shutdown Risk Assessment configuration risk management program before (using NRC RIS 2008-15 as guidance), the incorporation of portable diesel generators in the quantitative PRA model used for online configuration risk management is a first for Palo Verde. Shortly after implementation in September 2016, the NRC conducted an on-site review and did not identify any issues or concerns with the process as implemented via NEI 16-06.

Following this successful visit, Palo Verde incorporated additional equipment into the online configuration risk management program in November 2016, again using NEI 16-06 as guidance. The lessons learned from use of NEI 16-06 to incorporate equipment into the Palo Verde configuration risk management program are documented in this paper.

Risk-Informed Applications—II

Chair: Ernie Kee (*Texas A&M*)

Location: Grand Station V **Time:** 10:45 am - 12:00 pm

10:45 am: Technical Specification Surveillance Requirement Frequency Improvements in Nuclear Power Plants via Risk-Informed Initiative-5b Approach, Luyen D. Nguyen, John T. Kitzmiller (*Westinghouse*)

This paper will provide the technical basis and methods necessary to justify changes to surveillance frequencies consistent with the risk-informed approach defined in Regulatory Guide (RG) 1.174 Revision 2 (Reference 1) and the methodology defined in NEI 04-10, Revision 1 (Reference 2). The impact on plant risk and the impact on deterministic considerations, such as defense-in-depth, safety margins, and component/system performance, while are critical steps of the methodology, will not be addressed in this paper.

Objectives of this paper are to: (1) provide a high-level overview of Technical Specification (TS) Surveillance Requirement (SR) frequency improvement program (Risk-Informed 5b) and its benefits and (2) discuss different approaches following the Risk-Informed 5b methodology to justify surveillance test frequency changes.

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Risk-Informed Applications—II Continued

11:10 am: Centralization of Surveillance Frequency Control Program for Enhanced Efficiency, Nicholas C. Sternowski (*JENSEN HUGHES*), Jenna Burr (*Exelon*)

In the current climate of Delivering the Nuclear Promise, Exelon is utilizing the Surveillance Frequency Control Program (SFCP) to pursue Divisionalized Outage schedules for all of the stations in its nuclear fleet. Implementing Divisionalized Outage schedules enhances the scheduling flexibility at the stations while simultaneously reducing burdens to resources, risk, and outage budgets.

As an initial part of overall process, screening criteria were applied to outage surveillances to identify which involved multi-train or divisionalized systems. This resulted in a vast scope of surveillance tests requiring evaluations. This presented resource challenges for both the performance-based engineering analyses as well as the Probabilistic Risk Assessment (PRA) evaluations. In order to achieve this fleet-wide initiative, a centralized team was established to combine efforts and prioritize tasking while targeting specific efficiency gains. This team includes corporate engineering and PRA expertise working in parallel to maximize fleet-wide efficiency.

This paper will provide insights from the centralization of engineering and PRA resources. Specifically, the organizational strategies applied to the scope of PRA evaluations to establish increased productivity and identify efficiencies when pursuing multiple surveillance test interval extensions at a single site.

11:35 am: Global Sensitivity Analysis to Rank Parameters of Stress Corrosion Cracking in the Spatio-Temporal Probabilistic Model of Loss of Coolant Accident Frequencies, Wen-Chi Cheng, Chenghao Ding, Nicholas O'Shea, Tatsuya Sakurahara, Grant Schumock, Zahra Mohaghegh, Seyed Reihani, Ernie Kee (*Univ of Illinois*)

This research conducts Global Sensitivity Analysis (Global SA) on the Spatio-Temporal Probabilistic methodology, developed for steam generator rupture caused by Stress Corrosion Cracking (SCC), to rank the physical causal factors with respect to their influence on the probability of rupture within the lifetime of a nuclear power plant (NPP). The Spatio-Temporal Probabilistic Methodology integrates two types of models: (1) Markov Model to depict the renewal processes associated with the physical degradation and periodic maintenance for repair, and (2) The Probabilistic Physics of Failure (PPoF) model to explicitly incorporate physical failure mechanisms into the estimation of rupture probability. The Spatio-Temporal Probabilistic Methodology enables the possibility for explicitly including the effects of location-specific causal factors such as operating conditions (e.g., temperature, pressure, pH), maintenance quality, and material properties (e.g., yield strength and corrosion resistance), on the probability of a rupture occurrence. The ranking obtained from Global SA could help guide resource allocation to reduce the probability of Loss of Coolant Accidents (LOCAs) in NPPs.

PSA Standards and Peer Reviews

Chair: Vincent Andersen (*JENSEN HUGHES*)

Location: Waterfront **Time:** 10:45 am - 12:00 pm

10:45 am: Insights Into New 2017 Revision of ASME/ANS PRA Standard Part 5—Seismic, Eddie M. Guerra (*RIZZO Associates*), Andrea Maioli (*Westinghouse*), Stephen Eder (*Facility Risk Consultants*), Annie M. Kammerer (*Annie Kammerer Consulting*)

Currently, the practice of seismic probabilistic risk assessment (SPRA) in the United States (US) nuclear industry follows the technical requirements of Part 5 of the American Society of Mechanical Engineers/American Nuclear Society Standard ASME/ANS RA-Sb–2013. Recently, US nuclear plants have followed this Standard to complete SPRAs in response to a 50.54(f) Request for Information letter issued by the US Nuclear Regulatory Commission (NRC) to its licenses in response to the 2011 Fukushima accident in Japan.

As a result of this effort across the US nuclear industry, many technical issues have surfaced that have highlighted the conflicts or disconnects between the current version of the Standard and the current state-of-practice and state-of-knowledge in SPRA. Some areas in which issues have arisen include the degree of realism in seismic fragilities, failure correlation of components, use of new regional hazard models, seismically-induced fire and flooding, among others. To address these issues, a limited revision of Part 5 (seismic) of the Standard was recently undertaken. This revision addressed lessons learned from recent SPRAs, including those performed to respond to the NRC's 50.54(f) letter. The revision also incorporated advances in technology and methodology.

Although this revision did not change the underlying philosophy and main requirements of Part 5, users of the Standard may find value in understanding how the Standard has been revised and the motivation behind these changes. This paper provides users of Part 5 with a high-level overview of revisions for all three main technical elements (i.e., seismic hazard analysis, seismic fragility analysis, and seismic plant response). This paper discusses the motivation and technical basis for the changes, in order to ensure consistent interpretation and application across the technical community.

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PSA Standards and Peer Reviews Continued

11:10 am: The Use of Comprehensive In-Process Peer Reviews in Support of the UK ABWR PSA Generic Design Assessment Process, Dennis Henneke, Jonathan Li, Glen Seeman, Cassandra Ruch (GE Hitachi), Thomas Morgan, Eric Jorgenson (Enercon)

Over the course of an approximately 12-month period, a series of over a dozen peer reviews were conducted on initial PSA models and reports prepared to support the Generic Design Assessment (GDA) application to the United Kingdom's Office of Nuclear Regulation (ONR) for Hitachi General Electric Nuclear Energy's (HGNE's) UK Advanced Boiling Water Reactor (UK ABWR) design. The overall PSA evaluated all internal and external hazards risks for both at-power and low power/shutdown modes conditions, as well as spent fuel pool and fuel route hazards. The scope of the PSA included Level 1 through Level 3 risk assessments,

As many portions of these PSAs were under active development throughout this period, in-process peer reviews were conducted. These reviews initially assessed each draft model/report and assigned Facts and Observations (F&Os) based on the review findings. Follow-on reviews were then conducted on the near-final documents to determine if the initial peer review's findings were fully addressed and to identify any further F&Os that pertained to the final models and reports.

Through this innovative process, the HGNE PSA team was able to resolve the vast majority of technical adequacy issues through the course of this project. The submittal of the enhanced final PSA models and documentation, along with the detailed series of peer review reports, helped to demonstrate to ONR that these PSAs met the industry's technical adequacy requirements.

11:35 am: PRA Peer Review Finding Closure via Independent Assessment, David Passehl, Paul Amico, Barry Sloane (JENSEN HUGHES), Jeffrey Stone, Larry Naron (Exelon)

A process has been developed for performing a focused scope independent assessment to review close out of "Finding" level Facts and Observations (F&Os) of record from prior Probabilistic Risk Assessment (PRA) peer reviews against the ASME/ANS Standard.¹ The results of this independent assessment will be used to support future License Amendment Request (LAR) submittals. F&O dispositions reviewed and determined to have been adequately addressed through this technical review process will be considered as "closed" and no longer relevant to the current PRA model, and thus need not be carried forward nor discussed in future LAR submittals.

The process generally follows the applicable guidance defined in Nuclear Energy Institute (NEI) 05-04, Rev.3 (Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard)²; NEI 07-12, Rev 1 (Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines)³; and NEI 12-13 Rev.0 (External Hazards PRA Peer Review Process Guidelines)⁴. Although those processes were developed for use in performing peer reviews of Technical Elements of the PRA, rather than performing reviews of the resolution of prior peer review finding level F&Os, the NEI documents provide a solid foundation for performing these independent assessments. Performance of independent assessments of F&O resolutions via this process is encouraged by the Nuclear Regulatory Commission (NRC).

The results of this process should obviate the need for NRC review of PRAs allowing the NRC review of LARs to focus on application specific impacts, key assumptions, and areas identified by the peer review as being of concern. The process has been applied to several plant independent assessment finding closure reviews to date with several more planned for the near term. Insights from the technical reviews conducted on Exelon plants as of this writing (Limerick, Peach Bottom, Braidwood, Byron) are summarized in this paper.

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TECHNICAL SESSIONS - 1:30 PM

HRA—II

Chair: Gary W. Hayner (*JENSEN HUGHES*)

Location: Grand Station II **Time:** 1:30 - 3:10 pm

1:30 pm: Approach to Human Reliability Analysis for High Winds Events, Erin P. Collins, Elizabeth Cook, (*JENSEN HUGHES*)

There is currently no guidance specific to high wind events for human reliability analysis (HRA) best practices. Based on operating experience, operator interviews, and HRA analyst judgment, an approach has been developed for applying adjustments to internal events human failure events (HFEs) to represent operator actions during high wind events.

The key factors that have been identified for operator actions during high winds events are timing and location of the action. Timing adjustments depend on anticipated storm durations, which are estimated based on site-specific historical climate data and can vary greatly based on the type of high wind event. Since high wind events include tornadoes, straight winds, and hurricanes, specific sets of rules for each type of high wind initiator typically need to be developed. Actions in outdoor locations or structures that may not withstand high wind events may not be feasible during a high wind event.

Consideration is also given to the cognition, stress, and execution of the actions as some storms may be sudden in nature (e.g., tornadoes), but other storms, hurricanes for example, will provide operators with advanced warning. Thus, operational experience and operator interviews can provide valuable insights into high wind HRA applications.

1:55 pm: Final Conclusions and Lessons Learned from Testing the Integrated Human Event Analysis System for Nuclear Power Plant Internal Events At-Power Application, Huafei Liao (*SNL*), Stephanie Morrow (*NRC*), Gareth Parry (*JENSEN HUGHES*), Dennis Bley (*WreathWood Group*), Lawrence Criscione (*NRC*), Mary Presley (*EPRI*)

The Integrated Human Event Analysis System for nuclear power plant internal events at-power application (hereafter "IDHEAS AT-POWER") is a new human reliability analysis (HRA) method developed by the U.S. Nuclear Regulatory Commission (NRC) in collaboration with the Electric Power Research Institute (EPRI). It was developed to provide a structured approach to the qualitative and quantitative analysis of operator actions during internal, at-power nuclear power plant events. The IDHEAS AT-POWER method was tested to evaluate whether its guidance can be practically applied to produce consistent HRA results. This paper presents study findings and final conclusions on the method performance. Lessons learned on study methodology and recommendations for method improvement are also presented.

2:20 pm: Lessons Learned in HRA Dependency Analysis, Joshua S. Beckton, Mark Wilk (*Westinghouse*)

As PRA models are being updated to meet the 2009 ASME/ANS PRA standard; some analysts have encountered models that have resulted in significant changes from the earlier analyses due to improved and/or expanded methodologies. During a recent model of record update this was experienced when analysts transitioned a previously conducted Human Reliability Analysis (HRA) to the EPRI HRA Calculator tool. Not only did the Core Damage Frequency (CDF) value initially increase by an order of magnitude, but there were new significant events and operator actions.

The associated paper outlines some potential significant changes a plant may experience when they update their HRA Dependency Analysis (DA) methodology, and that these changes are not an outlier in the industry. We will outline many possible drivers for the changes in the application of the DA methods to help analysts interpret the new results. In addition, the paper will attempt to assist the analyst preparing the plant's management expectation of the potential impact of the new methodology.

There have been various discussions within the HRA community on the accepted approach in conducting a DA that have been in the process of being resolved. The updated methodology will usually be less likely to classify operator events in the same accident sequence with zero dependency, and in most cases result in a higher level of dependence than previously determined. This paper would also provide insights for the increased dependency levels and identify key areas that should be reviewed for potential refinement from the initial automation. The impact is expected to be more dramatic for plants (PRA models) that rely heavily on multiple operator actions. The analysts have seen earlier DAs that resulted in a handful of dependent operator actions, which made them independent events. This reduces the significance of these sequences and their associated initiating events. The updated methodology will most likely see these actions having a dependent relationship which introduces higher failure rates for the same combination of actions. If these actions show up in numerous scenarios then they could become very significant.

With these new results there could also be a change in PRA insights that could affect plant operation. With regards to the analysts recent experience changes to operating procedures were made as enhancements with more favorable results. The paper would prepare other analysts for this change (and possibly others) and help prepare them with potential discussions with their management.

In short, the EPRI HRA Calculator implementation of an updated DA methodology has resulted in significant changes in results and insights into overall plant risk.

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TECHNICAL SESSIONS - 1:30 PM

HRA—II Continued

2:45 pm: How Representative Are Our Quantitative Data on Performance Shaping Factors? Les Ainsworth (Corporate Risk Associates)

A comparison of the quantitative values for different performance shaping factors (PSFs) applied in the most widely used Human Reliability Assessment (HRA) methods showed that there is often variability in the PSF multipliers proposed for the same PSFs. A review of the human factors and psychological literature on the error rates was undertaken for two PSFs and for both there was a wide range of values. The constituent data sets were categorised to identify any features of the underlying tasks that would influence the PSF multipliers. These categorisations provided limited evidence that the impact of these particular PSFs was greater for more cognitively demanding tasks. For those studies where the PSF had very little effect, this appeared to be because the experimental paradigm used either a very simple task or a poor criterion for judging errors.

In view of these findings the author believes that there is an urgent need to move the focus away from after-the-event investigations of the literature, towards a targeted experimental investigation of selected PSFs to develop generic values for PSFs that can be used in different HRA methods. To ensure consistency this should be based upon a set of standard tasks that examine the impact of PSFs on the reliability of performing perceptual, motor or cognitive tasks. Revised PSF data are particularly necessary for PSFs that are measured on an interval scale. Otherwise, the quantification of PSF multipliers can become a lottery whereby the PSF value can change dramatically and be dependent upon whether just one particular study is identified or not.

This paper gives some recommendations as to how this could be achieved in a way that also assists analysts to determine whether an identified PSF is actually likely to affect the reliability of a specific task.

Fukushima Lessons Learned

Chair: Barry D. Sloane (JENSEN HUGHES)

Location: Grand Station III **Time:** 1:30 - 3:10 pm

1:30 pm: Mass Relocations After Nuclear Disasters, Vicki M. Bier, Andrew J. Nosek (Univ of Wisconsin)

People sometimes think of displaced persons as a developing-world problem. However, the Fukushima accident made clear that even in the developed world people may need to leave their homes due to disasters. In addition to the social consequences of relocation, mass relocation can have significant economic costs, including not only loss of housing, but also property damage, business interruption, and decline of property values. Economic consequences can be expected to be nonlinear in both magnitude and duration of relocation. With regard to duration, a brief evacuation may be minimally disruptive, if people are able to return to their homes within a few days. A relocation of a few months or a year would be much more disruptive per day, but eventually, costs per day would diminish. By contrast, costs can be expected to increase monotonically but non-linearly in the number of people needing to be relocated, with large relocations being disproportionately more costly. Costs may also vary greatly depending on the nature of the assets that are interdicted. Unfortunately, disasters in populated areas can result in the need to relocate a million people or more. This argues for the need for research on interventions to enhance recovery and resilience after relocation.

1:55 pm: MAAP5 RCIC Model Benchmark Against the Fukushima Daiichi Unit 2 (1F2) RCIC Performance, Christopher E. Henry, Jaehyok Lim, Robert E. Henry (Fauske & Assoc)

One of the most prominent features of the Fukushima-Daiichi Unit 2 (1F2) accident was the operation of the steam turbine-driven reactor core isolation cooling (RCIC) system. The RCIC system operation was the singular injection system that maintained core cooling for nearly three days, until its spontaneous termination ultimately led to core damage and off-site fission product release. After the tsunami inundation event, RCIC operated in an unpowered, unattended mode for the noted extended duration. The unpowered state also entailed loss of reactor pressure vessel (RPV) level control, which resulted in flooding of the main steam line, including the steam extraction line that drives the RCIC steam turbine. The resulting two-phase flow drove the turbine for the nearly the entire extended duration.

This operating state was a surprising outcome. The specific turbine design, termed a Terry turbine, is known to be tolerant of intermittent two-phase transient flow due to transport of individual water slugs. This operating state had never been experienced or even analyzed since it is effectively ruled out by the control system design during normal powered operation.

As part of the MAAP5 code enhancement project, mechanistic modeling of the RCIC steam extraction line and Terry turbine was undertaken as a development task to explain the overall RPV response, which was dominated by the RCIC operation. The modeling encompassed normal single-phase steam operation and the transition to two-phase flow operation after ELAP initiation.

The culmination of the modeling effort showed that the RCIC system, in concert with the connected RPV, yields a self-regulating behavior that gradually adjusts and equilibrates the RCIC pump inflow to the RPV with the RCIC extraction line outflow from the RPV. This integral behavior of RCIC with the RPV was highly successful in predicting the RPV pressure and water level measured data response from the early ELAP initiation through the RCIC system unintended isolation event nearly 3 days later. Furthermore, this fundamental explanation of the RCIC operation under two-phase conditions provides a compelling technical basis that unpowered, unattended, long-term operation is a probable outcome. Thus, the 1F2 experience was not an anomaly but rather a now-known and inherently stable passive operating state that can be utilized in decisionmaking within the industry practice of PSA, SAMG, and FLEX. Finally, the envisioned need for dedicated full-scale RCIC operations experiments to reveal contributing phenomena is now rendered largely unnecessary since the established phenomenology and the 1F2 experience itself already provides the necessary insights.

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TECHNICAL SESSIONS - 1:30 PM

Fukushima Lessons Learned Continued

2:20 pm: Overview of the MAAP5 Benchmark with Fukushima Daiichi Unit 2 (1F2), Christopher E. Henry
(Fauske & Assoc)

One of the most prominent features of the Fukushima-Daiichi Unit 2 (1F2) core damage phase was the presence of three distinct escalations in reactor pressure vessel (RPV) pressure. A later-stage fourth escalation is less evident, but it can also be discerned from the data. In addition to the RPV pressure data, the RPV water level data and primary containment vessel (PCV) pressure data were available (either fully or intermittently). These data sources, combined with the known elements from the accident timeline, provide adequate forensic information to support a computer simulation. While ample, this information is not a complete picture and therefore relies upon key assumptions to fill-in knowledge gaps. However, the assumptions are limited in number and justified by the measured data. Enforced discipline on assumptions is crucial so that the simulation is not adversely prejudiced by the imposed assumptions.

Implementing the forensic information to guide the boundary conditions, the MAAP5 integral computer code is used to simulate the 1F2 core damage phase. The MAAP5 reactor core modeling itself contains embedded technical bases, including the TMI-2 accident investigation and core damage experiments, such as Phebus, Severe Fuel Damage (SFD), and others. The known forensic information, combined with the technical basis-informed insights from the MAAP5 simulation, reveal a plausible scenario for the core damage progression, encompassing both debris in the core boundary and debris relocation to the lower plenum.

The culmination of the simulation is summarized as follows. The initial steps to provide emergency injection to the vessel were not sufficient to avoid severe core damage, which included downward in-core melt progression and collapse of most of the upper portion of the fuel matrix. Subsequent emergency injection was successful in submerging the damaged core, but the collapsed core was in a non-coolable state. Thus, despite the water submergence of the bulk debris pile, significant molten material moved through localized breaches from the core boundary to the lower plenum during the 14-MAR-2011 22:40 event. This event began a longer trend of gradual loss of water level out of the core. The subsequent drying and continued heating of core debris yielded additional material relocation to the lower plenum. However, continued emergency injection appears to have been sufficient to maintain debris cooling in the lower plenum. Furthermore, emergency injection may have maintained water in the RPV downcomer, which provided radial cooling of the core debris periphery (through the shroud wall).

The debris progression evolved into a long-term configuration in which the relocated material in the lower plenum was cooled directly by emergency injection. The remaining debris in the core boundary was not directly cooled by water submergence, but it did benefit from collateral water cooling through the shroud wall to the downcomer. This allowed the in-core debris to achieve an overheated but solidified state. The in-core debris temperatures were substantially elevated, but the elevated temperature near the core plate was not sufficient to fail the underlying structures of the core plate and control rod drive housing, which support the overlying debris weight. This MAAP5 depiction of the debris progression and debris end-state appears to be consistent with the noted forensic information. This depiction means that thermal attack of the RPV lower plenum wall and subsequent large-scale wall failure with majority debris relocation ex-vessel is highly unlikely. This appears to be consistent with forensic information that there is no large-scale reactor vessel failure event. While localized late-stage thermal attack of plenum wall penetrations cannot be ruled out during the timeframe analyzed, if such events occurred at all, it is unlikely that they could have resulted in a significant migration of ex-vessel debris to the pedestal.

This paper is intended as a summary of the analysis. More detailed elaboration on specific technical issues is valuable, and these will be the subject of future papers.

2:45 pm: Quantitative Risk Assessment for Process of Fuel Assembly Retrieval from Spent Fuel Pool in Fukushima Daiichi Nuclear Power Plant Decommissioning, Akira Yamaguchi, Sunghyon Jang
(Univ of Tokyo), Kazuki Hida (Nucl Damage Compensation and Decommissioning Facilitation Corp), Yasunori Yamanaka (TEPCO Holdings, Inc.), Yoshiyuki Narumiya (Kansai Electric Power Co)

A risk assessment is proposed for the management process for fuel assembly retrieval from the Fukushima Daiichi Unit 3 spent fuel pool. Four principles are considered towards the ultimate goal to safely store the retrieved fuel assemblies in dry casks: comprehensiveness, efficiency, measurability and effectiveness. Three risk surrogates are proposed for this activity, concerning personnel safety, public safety, and project schedule. A success path is developed and possible threats are identified along three phases of fuel assembly retrieval work that are established. Possible threats considered are internal/external as well as technical/societal/management. As knowledge is limited on the current situations and of hazard potential and as the decommissioning project is performed with wide-scope and long-term perspective, the risk assessment and management process needs to be updated as the most recent information and knowledge on the plant systems become available.

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TECHNICAL SESSIONS - 1:30 PM

Level 2 and Level 3 PSA—II

Chair: Kyle Hope (*Westinghouse*)

Location: Grand Station IV **Time:** 1:30 - 3:10 pm

1:30 pm: Likelihood that Short Term Station Blackout Scenarios Lead to a Large and Early Release, Donald E. Vanover, Robert J. Wolfgang (*JENSEN HUGHES*)

The Large Early Release Frequency (LERF) risk metric is used in risk-informed regulatory submittals and NRC evaluations. Mark I BWRs generally have assumed a high conditional likelihood of LERF in Short Term Station Blackout (STSBO) scenarios due to the potential contribution from the phenomenon known as liner melt-through of containment following vessel failure. Assignment of LERF in these scenarios has been based on the expected timing of the core melt progression and fission product release magnitude in comparison to the time required to evacuate the nearby population following a declaration of a general emergency.

However, more recent analysis from MAAP and MELCOR and detailed consequence analysis performed by the NRC in support of their State of the Art Reactor Consequence Analysis (SOARCA) program highlight the fact that there is a large amount of uncertainty associated with the base assumption that these scenarios lead to LERF. Additional considerations regarding the potential variability of core melt progression timing and evacuation times with respect to variations in the magnitude of the releases and when they should be classified as a large release all need to be integrated into the LERF likelihood assessment process.

1:55 pm: Insights from MELCOR Independent Confirmatory Analyses for New Reactor Design Certification, Jason Schaperow, Shawn Campbell, Marie Pohida (*NRC*)

The United States (U.S.) Title 10 Code of Federal Regulations (CFR), 10 CFR 52.47(a)(23), requires applicants seeking a design certification (DC) to submit a description and analysis of design features for the prevention and mitigation of SAs (e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass). Applicants perform severe accident analyses for the more likely severe accident scenarios to meet this regulation.

To obtain insights on the results of DC applicant's severe accident analyses, the NRC uses the MELCOR computer code to perform independent confirmatory analyses. MELCOR is a fully integrated, engineering-level computer code developed by Sandia National Laboratories for the NRC to model the progression of severe accidents in nuclear power plants. Such accidents involve long-term loss of core cooling; core uncover, heat-up and degradation; reactor vessel bottom head failure; core relocation to the containment; high temperature and high pressure challenges to the containment; and airborne radionuclide release to the environment. To perform the analyses, the staff selects scenarios to analyze, determines the appropriate modeling approach to perform the simulations, and compares the results to the applicant's simulations. Outcomes from the independent confirmatory analyses for both at-power accidents and shutdown accidents occurring during mid-loop operations are presented.

The United States (U.S.) Title 10 Code of Federal Regulations (CFR), 10 CFR 52.47(a)(27), also requires applicants seeking a DC to submit a description of the design-specific probabilistic risk assessment (PRA) and its results. A DC applicant's final safety analysis report (FSAR) is expected to contain a qualitative description of PRA insights and uses, as well as some quantitative PRA results, such that the U.S. Nuclear Regulatory Commission (NRC) staff can perform its safety review. As referenced in the NRC Standard Review Plan (SRP) (NUREG-0800) Chapter 19, the staff compares the design against the Commission's goals of less than 1×10^{-4} per year for core damage frequency and less than 1×10^{-6} per year for large release frequency. The staff expects that risk is assessed for all modes of operation, including shutdown modes. Although core decay power is lower when the reactor is shut down and the fraction of the time the reactor is shutdown could be small, the risk associated with shutdown accidents could be comparable to at-power accidents. This is especially the case for pressurized water reactors due to factors such as a lower initial water level during mid-loop operations.

2:20 pm: Advanced Containment Design to Enhance Passive Safety Through Phoretic Deposition Phenomena—Preliminary Findings from Empirical Studies, Sola Talabi (*Pittsburgh Technical*)

A nuclear power plant's containment vessel performance is a critical defense in depth barrier for the prevention of radioactive release in the event of a nuclear accident. One of the functions of the containment vessel is to assist in the decontamination of aerosolized radioactive particles through both 'passive' or natural processes and 'active' or mechanical processes. Passive decontamination occurs through natural phenomena, which include gravitational settling, thermophoresis, diffusiophoresis and hygroscopic effects, which are the subject of this study. A series of experiments are currently being performed to quantify the decontamination factors associated with these phenomena for SMR designs. The experiments include a novel approach for in-situ measurement of particle deposition velocities through the use of lasers. Preliminary findings show that a higher combined deposition velocity may be expected due to the effectiveness of the measurement technique, which allows a continuous domain, compared to prior art, which estimated lower velocities with an invasive method in a discretized domain. This work will improve the accuracy of Level 2 Probabilistic Risk Assessments associated with estimation of post-accident in-containment particle deposition.

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Level 2 and Level 3 PSA—II Continued

2:45 pm: Spatio-Temporal Socio-Technical Risk Analysis Methodology: An Application in Emergency Response, Ha Bui, Justin Pence, Zahra Mohaghegh, Seyed Reihani, Ernie Kee (*Univ of Illinois*)

This paper reports on the status of on-going research regarding the development of a Spatio-Temporal Socio-Technical Risk Analysis (ST-SoTeRiA) methodology for Emergency Response (ER) modeling. ST-SoTeRiA is an approach to explicitly incorporate spatial and temporal dimensions, while connected with Probabilistic Risk Assessment (PRA) logic, into the simulation of socio-technical failure mechanisms. The probabilities required for executing PRA are estimated by running a spatio-temporal platform that integrates deterministic simulation methods with probabilistic techniques. In this paper, a case study for fire ER demonstrates one of the building blocks of the ST-SoTeRiA methodology in which Agent-Based Modeling (ABM) technique is combined with physical hazard progression simulation in a shared Geographic Information System (GIS)-based spatial platform, as an input to PRA scenarios. A multi-method coupling between physical progression models and human response models in a spatio-temporal platform is essential to: (i) better characterize dynamic behaviors in ER, given location-specific hazards, (ii) better account for uncertainty, and (iii) better inform decision making in ER contexts.

Non-Light Water Reactor Safety

Chair: Andrea Maioli (*Westinghouse*)

Location: Grand Station V **Time:** 1:30 - 3:10 pm

1:30 pm: Development of a Design-Stage PRA for the Xe-100, Alexander J. Huning (*X-Energy, LLC*), Karl N. Fleming (*Technology Insights*)

Next generation reactor PRAs are in a unique position to influence reactor design prior to construction. Design-stage PRAs typically require more assumptions, are less detailed, and have larger uncertainties due to the conceptual nature of many plant systems and components. However, despite the large number of assumptions on system parameters, human actions, and the internal events only scope, the Xe-100 PRA has provided several risk insights that have supported design decisions. This is one of the two critical benefits for performing a design-stage PRA. The other key benefit is that the PRA will support many highly engaging risk-informed activities such as Licensing Basis Event (LBE) selection, safety classification of structures, systems, and components (SSCs), and Defense-in-Depth evaluation. In addition to these primarily safety-based applications, the PRA will also support an investment risk assessment, ensuring long outages and plant write-off events will be minimized.

This paper presents key features of the rapidly evolving PRA model for the Xe-100, a Pebble Bed-type High Temperature Gas Reactor (PB-HTGR) being developed by X-Energy. Consistent with the ASME/ANS PRA Standard for Advanced non-LWRs, the scope and level of detail of the PRA is consistent with the availability and level of detail of the design, operation, and siting information for the Xe-100. A review of the key event sequences relevant to PB-HTGRs, as well as the respective safety design mitigation strategies for the Xe-100 are presented. A small helium pressure boundary break is presented as an example of the initiating event selection to event sequence analysis approach.

1:55 pm: Evaluating Risk Measures for Non-LWR PSA, Jonathan Li, Jordan Hagaman, Dennis Henneke, Gary Miller, Glen Seeman (*GE Hitachi*)

GE Hitachi Nuclear Energy (GEH) and Argonne National Laboratory engaged in a joint effort to modernize and develop probabilistic risk assessment (PSA) techniques for advanced non-light water reactors (non-LWRs). The primary outcomes of this body of work were the development of PSA methodologies for non-LWR technologies, the demonstration of those methodologies in analysing GEH's PRISM sodium fast reactor design, and the generation of risk insights based on the new PSA.

Typical PSA risk measures of Core Damage Frequency or Large Early Release Frequency are inherently tied to LWR technology. Therefore, the development of the PRISM PSA provided an opportunity to examine various technology-neutral measures of risk. Consequence analysis was performed to study the effects of radionuclide release from postulated reactor accidents. Among the measures of radiological consequences are dose and possibility of developing acute or long-term health effects. The measures of dose included the absolute value of dose, but also the probability of a given release causing an exceedance of a regulatory dose threshold. The measures of health effects include individual effects, or prompt fatalities within the immediate vicinity of the plant site. They also include societal effects, those latent fatalities experienced within ten miles of the plant.

This paper presents the differences between these consequences for various releases postulated in the PRISM PSA. Also evaluated are merits and relative usefulness of each measure to developing risk insights about nuclear reactor design safety.

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Non-Light Water Reactor Safety Continued

2:20 pm: Identification of Research and Development Needs for Non-LWR PSA, Jordan Hagaman, Dennis Henneke, Gary Miller, Jonathan Li, Matt Warner, Jim Young (*GE Hitachi*)

GE Hitachi Nuclear Energy (GEH) and Argonne National Laboratory engaged in a joint effort to modernize and develop probabilistic safety assessment (PSA) techniques for advanced non-light water reactors (non-LWRs). The primary outcomes of this body of work were the development of PSA methodologies for non-LWR technologies, the demonstration of those methodologies in analysing GEH's PRISM sodium fast reactor design, and the generation of risk insights based on the new PSA. Although the focus of the work was on the PRISM technology, many insights were determined to be applicable to PSAs for any advanced reactor design.

This paper describes those conclusions from the project that suggest opportunities for research and development (R&D) to support further development of PSA for non-LWR designs. Two gaps in available methodologies that the PRISM PSA has highlighted are human reliability analysis and software reliability. For advanced reactor designs, especially those employing passive safety features, traditional means of developing human error probabilities or common cause software failures prove to be too conservative. Development in these areas will be critical to supporting risk-informed performance-based reactor design optimization and licensing applications.

2:45 pm: Preliminary Functional Safety Assessment for Molten Salt Fast Reactors in the Framework of the Samofar Project, Anna Chiara Uggenti (*Politecnico di Torino*), Delphine Gérardin (*LPSC-IN2P3-CNRS*), Andrea Carpignano, Sandra Dulla (*Politecnico di Torino*), Elsa Merle, Daniel Heuer, Axel Laureau, Michel Allibert (*LPSC-IN2P3-CNRS*)

In the last decades, the research activity for the development of innovative nuclear systems tries to answer the current needs of safety, reliability and sustainability, including safety assessment and risk analysis.

In this framework, the European project SAMOFAR aims at furnishing the experimental proof of concept of the Molten Salt Fast Reactor (MSFR) and its safety assessment at its present conceptual stage. For this purpose, the Integrated Safety Assessment Methodology (ISAM) is selected and analysed as conceptual methodology and a wide survey on risk analysis tools, international standards and best-practices aims at defining an operational procedure suiting MSFR analysis, including functional safety assessments.

Well-established practices applying "Functional Safety" to conceptual systems do not exist; therefore this work proposes and uses a new method based on functional modelling and on the Functional Failure Mode and Effect Analysis (FFMEA). This approach allows studying systems with a preliminary design, identifying functions deviations able to compromise safety, listing Postulated Initiating Events (PIEs) and recognizing lack of information, criticalities and necessity of supplementary provisions in the current design. Therefore, this methodology aims at influencing the design from its earliest stages. The paper focuses on the application of FFMEA to the MSFR in normal operation conditions.

Flooding PSA—I

Chair: Raymond E. Schneider (*Westinghouse*)

Location: Waterfront **Time:** 1:30 - 3:10 pm

1:30 pm: Nuclear Power Plant Flooding due to a Dam Failure: Teachings, Christelle Weber, Anne Dutfoy, Coraline Gaucher (*EdF*)

The aim of the article is to bring out the teachings brought by different aspects of the study, such as dam failure frequency calculation, global frequency of dependent phenomena assessment, and analysis of flooding impact on the Nuclear Power Plant.

First, the study examined the case of a dam insufficiently drained off during a flood of its river. The assessment was performed with a tool developed at EDF. Initially unexpected main contributors to the dam failure were identified.

The dam failure creates a wave that flows into the valley below. The confluence with another river in the valley below had to be taken into account. The degree of dependence between the flood at the dam and the flood of the river in the valley was characterized by using the theory of statistical extreme values. Eventually, the frequency of dam failure cumulated to the river flood in the valley was assessed.

Then, at the Nuclear Power Plant, looking at the flooding field evolution during the hydraulic simulation, we noticed that the level of water on the NPP platform was not the only parameter to take into account when assessing materials losses.

1:55 pm: Revision and Expansion of ASME/ANS External Flooding PRA Standard, Michelle Bensi (*NRC*), Ray Schneider (*Westinghouse*), Artur Mironenko (*Duke Energy*), Suzanne Loyd (*JENSEN HUGHES*), Zhegang Ma (*INL*)

A significant effort has been undertaken to update the ASME/ANS external flooding probabilistic risk assessment (XFPRA) requirements contained in Part 8 of the Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. This paper provides an overview of the proposed revision of the XFPRA Standard as well as commentary on issues and challenges unique to XFPRA.

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TECHNICAL SESSIONS - 1:30 PM

Flooding PSA—I Continued

2:20 pm: Modeling of Flooding Flow Rates Through Floor Drain Networks Using Mathcad, Robert J. Wolfgang (JENSEN HUGHES)

Internal Flood Probabilistic Risk Assessments (PRAs) involve the accumulation of water in various rooms of plant buildings, with propagation of water to other areas of the plant via stairwells, equipment hatches and floor drains. A companion paper involving the modeling of internal flood scenarios and water propagation using Mathcad software was presented at PSA 2015. However, this paper involving the modeling of floor drains focuses on a more detailed approach that is suggested for the modeling of water flow through a network of multiple floor drains in order to provide a more realistic assessment. This paper makes use of a Reactor Building example with a network of floor drains and develops the equations that can be used in modeling the flow of water through the floor drain network. Although it may be conservative to assume no credit for floor drains in an area where flooding is occurring, the effect of floor drains conveying water from one area of the plant to another should not be ignored. This is especially true for those cases in which the capacity of drain sumps are exceeded and water flows out of sump vents in areas where safety related equipment may be present.

2:45 pm: Development of Internal Flooding PSA for New Build UK Generic Design Assessment, Richard Derrett-Smith (Jacobsen Analytics Ltd), Yuki Ishiwatari (GE Hitachi)

Internal flooding hazards can pose a significant threat to plant safety and can often contribute a significant portion of total plant risk. This level of contribution therefore warrants a probabilistic treatment to identify vulnerabilities and provide insights for design or procedural improvements. Such an analysis was conducted for a new build reactor design of the UK Advanced Boiling Water Reactor (ABWR) as part of the United Kingdom's Generic Design Assessment (GDA) licensing process. This analysis was conducted for both at-power and shutdown operating states to obtain a comprehensive understanding of the potential internal flooding risk for different plant operating states and configurations.

The analysis was conducted for the UK ABWR generic design according to the EPRI method for internal flooding probabilistic assessment¹ making use of the generic piping failure frequencies² as part of that process. The analysis included all buildings within the GDA scope containing equipment with the potential to contribute to overall risk and had to overcome many challenges specific to new-build plant designs such as a lack of detailed design data, evolving design reference points and coordination with other related studies ongoing as part of the GDA process. This analysis was conducted in parallel with an internal fire PSA of the UK ABWR generic design and shared much of the same input data using an innovative data storage and manipulation tool to enable efficient generation of flooding and fire scenarios for use with the quantification software.

The insights for the design and potential solutions for overcoming the lack of necessary design data will be shared as part of this paper as well as insights from performing such studies in the context of the UK new build licensing process.

TECHNICAL SESSIONS - 3:40 PM

HRA—III

Chair: Paul J. Amico (JENSEN HUGHES)

Location: Grand Station II **Time:** 3:40 - 5:45 pm

3:40 pm: Timeline Development for Main Control Room Abandonment Human Reliability Analysis due to Fire Conditions, Kaydee Kohlhepp Gunter, Jeffrey Julius, Erin Collins, Paul Amico (JENSEN HUGHES), Mary Presley, Ashley Lindeman (EPRI), Susan E. Cooper, Tammie Rivera (NRC), Stacey Hendrickson (SNL), John Wreathall (John Wreathall & Co. Inc.)

NUREG-1921 Supplement 1 (to be published in 2017) provides qualitative guidance for how to address main control room abandonment (MCRA) human reliability analysis (HRA) as a subset to HRA in a fire PRA. One of the key components of MCRA is the development of a timeline. In order to be consistent with the rest of the fire PRA, the analysis of MCRA scenarios must adhere to the same fundamental considerations as the fire PRA evaluation of fire areas and different fire scenarios, including timing considerations. The concepts that the Time Available must exceed the Time Required and that the amount of exceedance impacts the reliability of the action remain valid for MCRA. Even so, there are several differences, primarily additional considerations that must also be taken into account in order to ensure that the MCRA actions are both feasible and reliable. The purpose of this paper is to describe the timing considerations associated with MCRA, and to present an approach to overlay the various timing sources (timelines) into a single MCRA timeline.

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TECHNICAL SESSIONS - 3:40 PM

4:05 pm: Dynamicizing the SPAR-H Method: A Simplified Approach to Computation-Based Human Reliability Analysis, Ronald Boring (INL), Martin Rasmussen (NTNU Social Research), Curtis Smith, Diego Mandelli, Sarah Ewing (INL)

In this paper, we discuss the adaptation of the Standardized Plant Analysis Risk-Human (SPAR-H) human reliability analysis (HRA) method to dynamic risk modeling. SPAR-H was developed as a worksheet-based method in which human reliability analysts assign the appropriate level of influence for performance shaping factors (PSFs). These PSFs then serve as multipliers to calculate the human error probability. In the adaptation presented here, PSFs are auto-calculated based on plant parameters and scenario context. Auto-calculation enables the dynamicized version of SPAR-H to be coupled to thermo-hydraulic code to estimate event outcomes. The approach demonstrates the value of adapting existing static HRA methods for dynamic modeling.

4:30 pm: Implementation of FLEX Operator Actions in PRA Models, Parthasarathy Chandran (Southern Nuclear Operating Co), David L. Moore (DL Moore Consulting), John Reddington (Empyrean Services)

One of the primary lessons learned from Fukushima Dai-ichi after the seismic and tsunami event was the importance of understanding realistic risk drivers for the plant. An effective and efficient way of understanding and managing the risk drivers is by building a PRA model that reflects the as-built, as-operated plant. To achieve this, the seismic PRA model developed for the Hatch Nuclear Plant included the newly implemented FLEX strategies, both in terms of FLEX-related equipment and the associated procedures. These strategies are implemented as per the order from the NRC after the Fukushima event, with a goal of lowering the risk of core damage and improving containment integrity. The procedures and processes were updated to ensure the plant operators and personnel can implement FLEX in the event of, but not restricted to, Extended Loss of AC Power (ELAP) and/or Loss of Ultimate Heat Sink (LUHS). The FLEX strategies incorporated into the PRA model included Phase 1 and Phase 2, i.e. utilizing plant equipment and on-site FLEX equipment. A key part of FLEX is the series of interrelated operator actions that would be required to be performed over a period of 24 hours for the strategy to be successful. This paper provides a discussion of steps used to develop a realistic model of the FLEX-related Human Failure Events (HFEs) and their implementation using current state of practice methods such as CBDM/THERP. The process of defining the HFEs, developing the time constraints, and incorporating the dependencies is discussed.

4:55 pm: Human Reliability Analysis for Implementation of Incipient Fire Detectors in Fire PRAs, Susan E. Cooper, Amy D'Agostino, Gabriel Taylor, Nicholas Melly (NRC)

NUREG-2180, Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detector Systems in Nuclear Facilities - Final Report, provides information on the performance and use of "very early warning fire detector systems" (VEWFDS) and the associated fire human reliability analysis (HRA) and probabilistic risk assessment (PRA) quantification, for electrical cabinet installations. This report documents the results of confirmatory research for an earlier, interim NRC staff position on the use and PRA quantification of these systems.

The fire risk reduction approach used in NUREG-2180 credits any additional time provided by VEWFDS toward an earlier time for fire suppression initiation. The HRA approach for this research addressed a number of novel aspects for HRA/PRAs, including:

- all operator actions taken without a reactor trip
- actions of licensed operators in the main control room, as well as that of field operators and instrument and control technicians
- no standard requirements for job aids (e.g., procedures) supporting operator actions
- time available for operator actions represented by a probability distribution, rather than a single point estimate

This paper summarizes key aspects of the HRA provided in NUREG-2180, especially focusing on how current HRA approaches and quantification methods can be used for this non-traditional HRA/PRA application.

5:20 pm: Developing Human Reliability Analysis Guidance for Main Control Room Abandonment Scenarios in Fire PRA: What's Different About "Command and Control"? Susan E. Cooper (NRC), Stacey Hendrickson (SNL), John Wreathall (John Wreathall & Co. Inc.), Jeff Julius (JENSEN HUGHES), Mary Presley (EPRI), Erin Collins (JENSEN HUGHES), Kaydee Kohlhepp Gunter (Curtiss-Wright), Paul Amico (JENSEN HUGHES), Ashley Lindeman (EPRI), Tammie Rivera (NRC)

Main control room abandonment (MCRA) due to fire is complex to model in probabilistic risk assessment (PRA) because there are a wide range of fire scenarios and, typically, operator actions are taken at multiple locations throughout the plant. While the US Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (NRC-RES) and the Electric Power Research Institute (EPRI) collaboratively published fire human reliability analysis (HRA) guidance in 2012 (see EPRI 1023001/NUREG-1921), it was recognized that MCRA scenarios would require additional HRA research. In 2015, a second joint EPRI/NRC-RES fire HRA project was initiated to develop HRA methods and guidance for MCRA scenarios due to either loss of habitability or loss of control.

During the course of this development, the joint EPRI/NRC-RES team recognized that "command and control" (C&C) in MCRA scenarios is a unique contextual influence. Initially, the team has developed qualitative explanations for the difference between C&C in the MCR versus that following MCRA, building upon explanatory models from psychology and military perspectives. In addition, reviews by the team of actual events (including those that did not involve MCRA) provide insights on C&C breakdowns.

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TECHNICAL SESSIONS - 3:40 PM

Dynamic PSA—III

Chair: Francesco Di Maio (*Politecnico di Milano*)

Location: Grand Station III **Time:** 3:40 - 5:45 pm

3:40 pm: A Dynamic Assessment of an Interfacing System Loss of Coolant Accident, Zachary K. Jankovsky (*Ohio State Univ*), Matthew R. Denman (*SNL*), Tunc Aldemir (*Ohio State Univ*)

Accident scenarios in nuclear power plants that bypass containment have the potential for large and early releases of radionuclides. They are typically guarded against using means such as redundant valves arranged in series and interlocks for systems that interface with the high pressure reactor coolant system. Some of these preventative arrangements rely on active systems that may fail in unique ways with the introduction of digital instrumentation and control. A hypothetical scenario in a pressurized water reactor plant is examined in which the digital controllers for the residual heat removal system intake valves are subjected to a common cause failure. This failure may cause simultaneous unintended valve opening while the reactor is at power, which has the potential to overpressurize and damage piping in the residual heat removal system and cause a leak of primary system water past containment into the auxiliary building (interfacing system loss of coolant accident). If the controllers are in a persistent fault condition, plant personnel will have to traverse the potentially contaminated auxiliary building to override at least one controller and close its associated valve. A dynamic case is assembled and run using the ADAPT dynamic event tree driver and the MELCOR severe accident analysis code in which uncertainties in the progression of the accident as well as mitigating operator actions are explored for an interfacing systems loss of coolant accident initiator. The results are assessed using recently-developed tools to gain insight into the likely outcomes and key events.

4:05 pm: Dynamic PRA of a Multi-Unit Plant, D. Mandelli, C. Parisi, A. Alfonsi, D. Maljovec, S. St. Germain, R. Boring, S. Ewing, C. Smith, C. Rabiti (*INL*)

Dynamic Probabilistic Risk Analysis (PRA) methods couple stochastic methods (e.g., RAVEN, ADAPT, ADS, MCDET) with safety analysis codes (RELAP5-3D, MELCOR, MAAP) to determine risk associated to complex systems such as nuclear plants. Compared to classical PRA methods, which are based on static logic structures (e.g., Event-Trees, Fault-Trees), they can evaluate with higher resolution the safety impact of timing and sequencing of events on the accident progression. Recently, special attention has been given to nuclear plants which consist of multiple units and, in particular, on the safety impact of system dependencies, shared systems and common resources on core damage frequencies. In the literature, while classical PRA methods have been employed to model multi-unit plants, Dynamic PRA methods have never been applied to analyze a full multi-unit model. This paper presents a PRA analysis of a multi-unit plant using Dynamic PRA methods. We employ RAVEN as stochastic method coupled with RELAP5-3D. The plant under consideration consists of the three units and their associated spent fuel pools (SFPs). The studied initiating event is a seismic induced station blackout event. We will describe in detail how the multi-unit plant has been constructed and, in particular, how unit dependencies and shared resources are modeled.

4:30 pm: Measuring Risk Importance in a Dynamic PRA Framework, D. Mandelli, Z. Ma, C. Parisi, A. Alfonsi, C. Smith (*INL*)

Risk importance measures are indexes that are used to rank systems, structures and components (SSCs) using risk-informed methods. The most used/known measures are: Risk Reduction Worth (RRW), Risk Achievement Worth (RAW), Birnbaum (B) and Fussell-Vesely (FV). Once obtained from classical Probabilistic Risk Analysis (PRA) analyses, these risk measures can be effectively employed to optimize component testing and maintenance. In contrast to classical PRA methods, Dynamic PRA methods couple stochastic methods with safety analysis codes to determine risk associate to complex systems such as nuclear plants. Compared to classical PRA methods, they can evaluate with higher resolution the safety impact of timing and sequencing of events on the accident progression. The objective of this paper is to present a series of algorithms that can be used to determine classical risk importance measures (RRW, RAW, B and FV) along with newly developed ones from a Dynamic PRA analysis.

4:55 pm: Dynamic PRA with Component Aging and Degradation Modeled Utilizing Plant Risk Monitoring Data, Vaibhav Yadav, Vivek Agarwal, Andrei V. Gribok, Curtis L. Smith (*INL*)

In the nuclear industry, risk monitors are intended to provide a point-in-time estimate of system risk given current plant configurations. Current risk monitors are limited in that they do not properly take into account the deteriorating states of plant systems, structures, and components (SSCs), which are unit-specific. Current approaches to computing risk monitors use probabilistic risk assessment (PRA) techniques, but the assessment is typically a snapshot in time. Living PRA models attempt to address limitations of traditional PRA models in a limited sense by including temporary changes in plant and system configurations. However, information on plant component health is not considered. This often leaves risk monitors using living PRA models incapable of conducting evaluations with dynamic degradation scenarios evolving over time. There is a need to develop enabling approaches to solidify risk monitors to provide time and condition-dependent risk by integrating traditional PRA models with condition monitoring and prognostic techniques. This paper presents a novel hazard-rate based model of incorporating degradation of SSCs into an existing PRA model. Time-varying instantaneous hazard-rate of component is used to model degradation based on time-varying performance measure. The instantaneous hazard-rate is used to determine and update time-varying failure probability of the components in their PRA models. The method is applied to estimate the evolution of failure probability of an in-service pump using vibration measurements taken while the pump undergoes degradation.

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TECHNICAL SESSIONS - 3:40 PM

Dynamic PSA—III Continued

5:20 pm: Emulation-Based Uncertainty Quantification of a Fire Dynamics Simulation, Clarence Worrell, Raymond Schneider (*Westinghouse*), Brian Leyde, Max Xu (*SmartUQ*)

Emulation, also known as meta or surrogate modeling, provides a feasible approach for conducting quantitative uncertainty analyses where computationally intensive modeling codes have been applied. This paper documents initial efforts to generate and validate accurate emulators for simulation of fire scenarios. Representative simulations were conducted using the publicly available Fire Dynamics Simulator, the emulator development was performed with the SmartUQ software, and the uncertainty quantification was performed in MATLAB (note that this paper is not an endorsement of any particular software or product). The initial design of experiments is discussed as well as some of the theory and tradeoffs of the different emulation techniques that were tested. The resulting emulators were evaluated for accuracy and their utility in probabilistic risk assessment demonstrated.

Level 2 and Level 3 PSA—III

Chair: Stanley H. Levinson (*AREVA*)

Location: Grand Station IV **Time:** 3:40 - 5:20 pm

3:40 pm: Determining Source Terms for Releases from PWR Spent Fuel Pools in Case of Severe Accidents, Michael Hage, Horst Löffler, Michael Kowalik (*GRS*)

In the event of a severe accident in a nuclear power plant, when airborne radioactive substances may be released to the environment (so-called source term), emergency disaster control authorities have to take measures early enough in order to protect the general public. Computerized analytical tools to guide and assist the plant crisis team or an external emergency team in order to estimate the radioactive release are helpful and time saving in such events. For core melt accidents, most nuclear power plants in Germany apply a fast running tool for predicting source terms. Consecutive to the Fukushima-Dai-ichi reactor accidents, GRS has been contracted by the Federal Office for Radiation Protection to develop a similar tool especially for accidents in a spent fuel pool. It is well accepted in the expert community, that such accidents are extremely unlikely and can be considered hypothetical.

4:05 pm: Crediting the Use of a Rapidly Deployable Mobile Pump to Recover from and Core Damage Events Caused by a Failure of the Turbine Driven Auxiliary Feedwater Pump, Michael Powell (*Arizona Public Service*), Roy Linthicum (*Exelon*), Richard Haessler, Jeffrey Taylor (*Westinghouse*)

The turbine-driven auxiliary feedwater (TDAFW) pump provides a critical safety function for most pressurized water reactors (PWRs) at the onset of an extended loss of all AC power (ELAP) event. If all other auxiliary feedwater (AFW) pumps are unavailable due to the loss of electrical power, the steam-driven TDAFW pump injects water into the steam generators and maintains a heat sink to remove decay heat from the reactor core; however the TDAFW pump cannot run indefinitely. Therefore, most if not all PWRs have implemented post-Fukushima safety enhancements that include the use of portable pumps that can be deployed to feed the steam generators to maintain a long term heat sink in an ELAP event when normal plant pumps are not available. These portable pumps are a backup to the TDAFW pump, and would typically be deployed within six to eight hours of the initiation of the event to take over as an injection source. The key assumption used in an ELAP or Station Blackout (SBO) analysis is that the TDAFW pump will start and operate during the event. If the TDAFW pump fails to start at the beginning of an ELAP event and there is a loss of feedwater (LOFW), thermal-hydraulic analyses show that the steam generators will boil dry and the core could uncover in two hours or less (this timing is highly dependent on reactor design). Since the normal deployment time of the portable pumps is six to eight hours, a TDAFW pump failure early in the event would pose significant risk to recovering from an ELAP event. Contingency plans using a highly mobile makeup pump have been analyzed to determine if a success path for preventing core damage exists. This approach relies on a focused effort to rapidly depressurize the steam generators and quickly deploy a makeup source to supply water to them. Thermal-hydraulic analysis shows that a fire truck or large air driven pump can successfully remove decay heat and keep the fuel covered. An additional scenario was analyzed to determine the system response if the air driven makeup pump is undersized and unable to initially remove the decay heat that is generated by the core. This scenario was also successful but a new set of complications arise. This paper provides insight on the importance of steam generator makeup early in the event, how the plant would respond if makeup flow is not initially sufficient and where plant operators should prioritize resources.

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Level 2 and Level 3 PSA—III Continued

4:30 pm: Protective Systems: Definitions and Terms in the Regulated Risk Assessment Setting, Ha Bui, Tatsuya Sakurahara, Justin Pence (*Univ of Illinois*), ZiHui Gu (*Univ of Illinois, STP*), Zahra Mohaghegh, Seyed Reihani (*Univ of Illinois*), Martin Wortman (*Texas A&M*), Ernie Kee (*Univ of Illinois*), Vera Moiseytseva (*YK.Risk*), David Johnson (*UCLA*)

This paper reports on the initial results of ongoing multidisciplinary research to generate a common language for protective system risk analysis. Although established for risk analysis in the commercial nuclear power industry, these definitions can be generalized and used in other regulated profit-making enterprises. As collaborators in large-scale industry-academia-regulatory Probabilistic Risk Assessment (PRA) projects, the authors have found that misunderstanding the terms used in communication between the industry and regulator can reduce efficiency. For example, if the industry practitioner believes “deterministic analysis” is unrelated to PRA, product rework may be indicated. A review of academic literature and the Nuclear Regulatory Commission’s (NRC’s) website shows that some “definitions” have been assigned, through, for example, a glossary developed for general use. When agreed upon among industry and regulatory investigators, more precise definitions for several common terms such as; Risk, Risk-Informed, Probabilistic Risk Assessment, Prescriptive, Deterministic, Conservative, Risk-Based, Performance Based, and Uncertainty would be helpful in a setting of commercial nuclear power protective systems under regulation. The suggested definitions are based on the authors’ information extraction and interpretations from diverse references, or based on industry expert experience. The authors will continue this research to expand the scope of these terms and to seek feedback from industry, academia, and regulatory agencies, hoping to create a consensus among them. Defining these terminologies is important for efficient implementation of risk assessment information relevant to protective systems used in regulated for-profit enterprise.

4:55 pm: A Bayesian Approach to Estimate Failure Probability of Nuclear Turbine Blades due to Several Degradation Mechanisms, David Quintanar-Gago, Pamela F. Nelson (*UNAM*)

A quantitative methodology to estimate the probability of turbine blade failure-modes caused by typical degradation mechanisms in nuclear turbine units is being developed. The mechanisms and their failure modes, which affect nuclear turbine blade integrity, include pitting, droplet erosion, fatigue, corrosion fatigue, stress corrosion cracking, and fretting. It has been found that from a probabilistic perspective these mechanisms have a conditional behavior that can be described by a Bayesian Network. There are causal relationships between them (e.g., the phenomenology dictates that when pitting is found in a blade, the probability of corrosion fatigue increases) that can be estimated from turbine reliability databases. This way, a prototype network has been constructed as a first qualitative approximation. It is expected that introducing specific plant data from studies, inspections, and/or nondestructive reports, failure modes can be computed as ending nodes of the network and vice versa; that is, if a failure mode occurs, then the most likely set of causes is revealed. The model described here will help optimize maintenance strategies to reduce costs.

Risk-Informed Regulation

Chair: Raymond E. Schneider (*Westinghouse*)

Location: Grand Station V **Time:** 3:40 - 4:55 pm

3:40 pm: Main Applications of Probabilistic Safety Analysis Technology in Nuclear Power Plant Design in China, Gong Yu, Li Chun, Ni Man, Qian Hong Tao (*NSC*)

Probabilistic safety analysis (PSA), as a systematic analysis tool, plays an increasingly important role in the design of nuclear power plants. Synchronous PSA work in the plant design process and throughout the design process can identify potential weaknesses in plant design, provide a PSA-based view of risk, and evaluate compliance with relevant nuclear safety regulations. In the Nuclear Power Plant Design Safety Regulations (HAF102-2016), the National Nuclear Safety Administration (NNSA) specified that a comprehensive probabilistic safety assessment must be carried out throughout the design process of a nuclear power plant. Consideration must be given to all operating modes and all states of the nuclear power plant (Including the shutdown condition) under the probability of safety analysis. This paper summarizes the main PSA work carried out during the design stage and design change of nuclear power plants in China. It is demonstrated that the PSA work in the design stage is of great significance to enhance the safety of nuclear power plants.

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TECHNICAL SESSIONS - 3:40 PM

Risk-Informed Regulation Continued

4:05 pm: A Tool for Planning the Safety Review of Structures, Systems and Components: Development and Application, Tony Nakanishi, Mark A. Caruso, Lynn A. Mrowca (NRC)

The U.S. Nuclear Regulatory Commission (NRC) is accountable to its stakeholders to complete the safety review of an application for design certification of a nuclear power plant in an effective and efficient manner. To that end, the NRC established an initiative, entitled, "NuScale Enhanced Safety Focused Review", in the NRC's Office of New Reactors to prepare for the review of the NuScale design certification application and formed a working group to lead the preparatory activities. NRC is committed to the NuScale enhanced safety focused review approach as an important enabler for the successful completion of the NuScale design certification review. The working group has developed tools and approaches that complement existing NRC review guidance. These tools are helping staff focus review efforts on the most safety-significant aspects of the design while expending less effort on design aspects that are not safety-significant. This holistic and graded review approach aims to establish the scope and depth of each review activity. In this regard, the working group has developed and applied a new tool to help plan review activities for structures, systems and components (SSC) in the NuScale design. This tool is known simply as the "SSC Review Tool". The SSC review tool provides a framework for identifying review activities that may require more or less emphasis during the review of an SSC. Implementation of the tool requires reviewers to consider plant design features in terms of eleven key considerations (such as, novel design, regulatory compliance issues, risk insights, relationship to defense-in-depth and relationship to safety margins) to help formulate the scope and depth of review activities. The tool, in conjunction with available design information and existing review guidance, is used to facilitate structured discussions among groups of staff members about the review of a particular SSC or specific functions of that SSC. The information gathered from these discussions is documented and provides the basis for prioritizing review activities. This paper will identify the factors in the SSC review tool framework and discuss their genesis. It will also provide some examples to illustrate application of the tool.

4:30 pm: Evaluation of the Cumulative Impact of Pending Model Changes to Address MSPI FAQ 14-01 Requirements, Joseph Lavelline (Enercon)

FAQ 14-01 to NEI 99-02 Rev. 7 (Regulatory Assessment Performance Indicator Guideline) was approved in 2014. The new guidelines require that pending model changes that cannot be incorporated into a revision to the site PRA model of record prior to the next reporting quarter should be assessed consistent with the PRA Configuration Control program. The guideline further states that the configuration control program supporting the MSPI should contain a process that ensures that the cumulative impact of pending changes is considered. Pending model changes to be considered for MSPI are those related to implemented plant design and operational changes, identified errors in the PRA model, and F&Os characterized as findings for designed ASME standard supporting requirements. The overall goal of a licensee in addressing this guideline is to ensure that all pending model changes are assessed for their impact on MSPI in an integrated fashion while at the same time limiting the analysis burden associated with this requirement.

This paper describes the systematic approach developed for a single unit BWR for addressing these requirements. The approach utilized relies upon a robust screening evaluation to filter modeling issues that are not applicable to the requirement; the results of this screening evaluation are then incorporated into the PRA model configuration control database. Model changes that are found to be applicable are addressed in either a best-estimate or bounding fashion and are incorporated into a "working MSPI model". A quantification, which incorporates the integrated impact of these model changes, is then performed and the effect on the MSPI values is subsequently calculated.

Fire PSA—III

Chair: Mark B. Wishart (JENSEN HUGHES)

Location: Waterfront **Time:** 3:40 - 5:45 pm

3:40 pm: Development of Computer Program to Model Cylindrical Incident Discrete Emissive Radiation (CINDER) from Pool Fires, Robert J. Wolfgang (JENSEN HUGHES)

Fire Probabilistic Risk Assessments (PRAs) require the analysis of fire hazards in the presence of exposed structural steel to determine whether the structural integrity of buildings is compromised. Specifically, the combined ASME/ANS PRA Standard requires the quantitative assessment of the risk associated with such selected fire scenarios in a manner consistent with the Fire Quantification (FQ) requirements, including collapse of the exposed structural steel. The current available simplistic methods, such as the point-source and solid flame models, may not account for near-field radiation effects. The Fire Dynamics Simulator (FDS) software can certainly be utilized to model such scenarios in detail, but involve expertise in creating such simulations that can be labor intensive. The purpose of this paper is to provide the analyst a tool to model such scenarios using existing correlations and fundamental concepts to provide a more realistic analysis without requiring modeling details that would be required for such general purpose software codes. Because of this, the Cylindrical Incident Discrete Emissive Radiation (CINDER) computer code was developed to address this particular need, in that the ability to analyze structural capability in the presence of oil pool fires can be more easily analyzed without requiring the modeling details and expertise that would be required given the absence of such a method. The CINDER software program was designed to calculate the temperatures of exposed structural steel that would be experienced given a wide range of oil pool fires and target geometry in an interactive format with limited knowledge of fire modeling details.

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TECHNICAL SESSIONS - 3:40 PM

Fire PSA—III Continued

4:05 pm: Insights Gained from a Review of Fire PRA Risk Contribution by Ignition Frequency Bin, Jazmin NeVille, Usama Farradj, Kiang Zee (*JENSEN HUGHES*), Ashley Lindeman (*EPRI*)

Fire Probabilistic Risk Assessments (FPRAs) continue to generate results that dominate the overall nuclear power plant risk profile. Historically there have been many discussions, presentations, and reasons that have all generally made claims that the available consensus methods are generating unrealistic results. In 2013, an effort to illustrate this was presented using a plot of the percentage of plant fire risk contribution for each ignition frequency bin. A comparison of fire event frequency predicted by the FPRAs models and actual fire events also indicated that the analyses over-predicted the frequency of fire events with significant potential consequences. That insight led to industry efforts and NRC tests that ultimately resulted in NUREG 2178 (and EPRI 3002005578) [1]. This NUREG modified the consensus values for electrical cabinet heat release rates and introduced the obstructed plume. Various other methodology and data changes have also occurred since 2013. These changes in FPRAs data and treatments were predicted to have a notable impact on the calculated fire risk.

An effort was undertaken to collect updated (current) fire risk metrics from the U.S. nuclear power industry to re-assess the risk contribution, per ignition source, in light of developments in FPRAs methodology and the larger number of available FPRAs for review. The predicted frequency of high risk consequence fires is compared to NRC data regarding the actual number of high risk fire events experienced to determine if the FPRAs results are consistent with actual operational occurrences.

4:30 pm: Modeling of NUREG/CR-7150 Spurious Duration Probabilities, Nathan Larson, Daniel Tirsun, Kyle Christiansen (*Westinghouse*)

Industry experience has shown that modeling of fire induced hot shorts in fire Probabilistic Risk Assessment (PRA) models has a significant impact on the fire induced risk profile. NUREG/CR-7150 (Ref. 1) provides guidance on how to apply a spurious duration probability to fire induced hot shorts for solenoid and air-operated valves, i.e. a probability that a hot short will maintain the valve's position in the undesired state after a specified point in time. The application of the spurious duration probability can greatly reduce the impact of the fire induced hot short.

NUREG/CR-7150 provides clear guidance for conditions in which the component is in the undesired position for the specified time and does not change the accident progression. However, the analyst must take into account all potential plant responses/conditions that could occur prior to, during, and after the fire induced hot short. If a component returns to its original position and plant response/conditions are not affected, then the application as defined in the NUREG/CR-7150 (Ref. 1) applies. If plant conditions change during the time the fire induced hot short occurs, additional modeling considerations may be necessary. This paper will describe and discuss a case study that was found during a recent fire PRA model development with the application of NUREG/CR-7150 (Ref. 1).

4:55 pm: Impact of New NUREGs on Fire PRAs, Mark Humphrey (*EPM*), Harold Stiles (*Duke Energy*), Bruce Morgen (*EPM*)

Recently, three new NUREG's have been issued which impact the fire modeling and results of Fire PRA. NUREG 2169 (Reference 3) updated ignition frequencies, bins, and non-suppression probabilities. NUREG 2178 (Reference 4) updated heat release rates for electrical cabinets. NUREG 2180 (Reference 5) addresses updates to area wide incipient detection. These NUREGs were implemented in the recent update to Harris and Robinson Nuclear Plant Fire PRA models with clear trends and deltas from the values from NUREG 6850 (Reference 1). The purpose of this paper is to discuss larger trends noticed when each NUREG was implemented. Finally, some of the fire modeling changes that were required to implement the NUREG's will be briefly covered with conclusions about their impact to the model.

5:20 pm: Modeling Incipient Fire Detection with Updated Guidance, Brian Krystek (*EPM*)

The updated guidance for modeling Very Early Warning Fire Detection Systems (VEWFDS), also referred to as incipient detection, in NUREG-2180 replaces guidance found in FAQ 08-0046. To better understand the full impact of differences in guidance, the individual changes to the event tree approach from the FAQ are evaluated by solving the Fire PRA models for several nuclear power plants. Because incipient detection is often applied to fire scenarios with high CCDPs, changes in guidance are magnified. In some cases, detailed fire modeling of specific electrical cabinet and compartment configurations is required to isolate the exact impact of the NUREG-2180 event trees. With the additional detail available from this approach, more refined results are obtained. From these general cases, conclusions are presented for areas of future research and refinement.

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THURSDAY, SEPTEMBER 28

TECHNICAL SESSIONS - 8:00 AM

Modeling and Simulation—II

Chair: Bruce Morgen (EPM)

Location: Grand Station III **Time:** 8:00 - 10:05 am

8:00 am: Implementation of FLEX Strategies in Surry PRA, Aram Hakobyan, Craig Nierode (Dominion Resources Inc.)

In response to Fukushima accident in 2011, US nuclear industry developed an initiative to implement the mitigation strategies (FLEX) that are designed to mitigate Extended Loss of All Power (ELAP) events. As part of that initiative, Dominion purchased the necessary portable equipment, developed procedures, and implemented the FLEX strategies at its Surry, North Anna and Millstone nuclear power plants. A decision was also made to take credit for the FLEX strategies in Dominion fleet Probabilistic Risk Assessment (PRA) models. FLEX was first implemented in the Surry PRA model. Since Surry Core Damage Frequency (CDF) was dominated with internal flooding scenarios in Turbine Building that propagate to Emergency Switchgear Room resulting an ELAP event, implementation of FLEX strategies in Surry PRA was expected to have a significant impact on Surry PRA results. In addition to flooding, FLEX recovery was also applied to a Station Blackout sequence with successful turbine-driven auxiliary feedwater (AFW) pump and failure of offsite power restoration. The main function recovered by FLEX in either scenario was power supply for Steam Generator (SG) level control instrumentation. At Surry, this is provided by a Uninterrupted Power Supply (UPS) system connected to the Remote Monitoring Panel (RMT) for the first 12 hours. This design change was added as part of FLEX implementation. Plant procedures direct personnel to bring in and use a portable generator from the storage facility to power the RMT within 12 hours.

Implementation of FLEX strategies in Surry PRA resulted in about 75% overall CDF reduction, which was reduced from 1.0E-05/yr to 2.53E-06/yr (similar reduction was seen for Unit 2). Flooding contribution to CDF was reduced from 47% to 17%, and Station Blackout (SBO) contribution to CDF reduced from 39% to 26%. The results for Unit 2 were similar to those of Unit 1. The conclusion is that implementation of FLEX strategies in Surry PRA does result in significant reduction in CDF and significant change in overall risk profile in terms of CDF contributors.

8:25 am: Probabilistic Analysis of Electrical Faults Affecting Multiple Redundant Trains of the Electrical Power Supply System of Nuclear Power Plants, B. Brück, G. Gänßmantel, A. Kreuser, C. Müller, E. Piljugin, J. C. Stiller (GRS)

Faults simultaneously impairing multiple trains of the electrical power supply system may pose serious threats to the safety of nuclear power plants. Such faults may have different causes like open phase conditions. Events with such faults have recently occurred in different plants. Such faults are generally not modelled in probabilistic safety analyses. Therefore, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) has initiated a research project to analyze such faults and develop methods to model and quantify them in PSAs.

8:50 am: Fitting of Failure Rate Data to Gamma-Poisson Distribution Utilizing Method of Moments, Sarah M. Ewing, M. R. Kunz (INL)

Because of the responsibility and gravity involved in estimating nuclear power plant failure rates, a consistent and accurate methodology is required for predicting the likelihood that an event will occur. However, the methodology currently employed can vary depending on the data used, as well as subjective conjecture from the expert performing the analysis. The current implementation of the empirical Bayes method to a gamma-Poisson (GaP) distribution utilizes algorithms to solve for the parameters that do not provide consistent answers.

Additionally, to achieve a distribution of the likelihood, the variance of each parameter of the GaP distribution must be determined. There is no exact solution to one of the parameter's variance, and it is typically estimated using techniques like the Kass-Steffey adjustment. Thus, a new approach to the problem is proposed, built upon the method of moments for a negative binomial. Using the method of moments approach, we are able to achieve a closed-form estimation of the mean and variance for each parameter in the negative binomial distribution. Due to the relationship between the negative binomial and GaP distribution, comparisons can be made between the distributions. The hyper-priors defined assume a beta prime distribution that is appropriately informed; the results of this application translate back to the gamma distribution for easy utilization in SAPHIRE. Additionally, two cases are explored using publically available data from the Nuclear Regulatory Commission that consider zero-inflated, over-dispersed Poisson data.

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TECHNICAL SESSIONS - 8:00 AM

Modeling and Simulation—II Continued

9:15 am: Exploring Network Metrics for Accident Scenarios: A Case of Study of the Uncontrolled Level Drop, Mouna Rifi (EdF, LIPN), Mohamed Hibti (EdF), Rushed Kanawati (LIPN)

In this paper, we consider exploring network metrics to highlight “sensitive” components in accident scenarios as implemented in Probabilistic safety models, using low complexity algorithms. We model event sequence diagrams in the form of networks over which we apply networks analysis. The most common approach to “classify or characterize” these components is to use fault tree and event tree analysis in PSA and then importance factors. However, it turns out that, in some models, many component’s failures do not appear in the cut set list and some pretend only 10% or less of the modeled components occur in the cut set list due to truncation procedures. Here we show how graph theory and network analysis can be used in PSA, and what kind of additional knowledge can we get using network metrics. This method consists of modeling nuclear systems into networks and mining those networks using graph theory approach and metrics. In this study, we revisit the uncontrolled level drop accident ULD, of a PWR and try to analyze the components with important metrics from network analysis’ perspective and check the importance of those components in PSA. If those components don’t appear important in the PSA, we will try to figure out and identify the functional or structural reasons of this (these) difference(s). This approach enables us to validate the PSA’s classical metrics (conventional importance factors) and possibly fill the gaps, in other words to find components which can be important and missing in classical PSA’s results.

9:40 am: A Multi-State Physics Modeling for Estimating the Size- and Location-Dependent Loss of Coolant Accident Initiating Event Probability, Francesco Di Maio, Davide Colli (Politecnico di Milano), Enrico Zio (Politecnico di Milano, EdF), Liu Tao, Jiejuan Tong (Tsinghua Univ)

Multi-State Physics Modeling (MSPM) integrates multi-state modeling to describe a component degradation process by transitions among discrete states (e.g., no damage, micro-crack, flaw, rupture, etc.), with physics modeling by (physic) equations to describe the continuous degradation process within the states. In this work, we propose MSPM to describe the degradation dynamics of a piping system, accounting for the dependence on the size and location of the Loss of Coolant Accident (LOCA) initiating event of the Reactor Coolant System (RCS) of a Pressurized Water Reactor (PWR). Estimated frequencies of LOCA as a function of break size are used in a variety of regulatory applications and for the Probabilistic Risk Assessment (PRA) of Nuclear Power Plants (NPPs). Traditionally, two approaches have been used to assess LOCA frequencies as a function of pipe break size: estimates based on statistical analysis of field data collected from piping systems service experience and Probabilistic Fracture Mechanics (PFM) analysis of specific, postulated, physical damage mechanisms. However, due to the high reliability of NPP piping systems, it is difficult to construct a comprehensive service database based on which perform statistical analysis. On the other hand, it is difficult to utilize PFM models for calculating LOCA frequencies because many of the input variables and model assumptions are over-simplified and may not adequately represent the true plant conditions. We overcome these challenges and propose a size- and location-dependent LOCA initiating event frequencies estimation by resorting to the novel MSPM modeling scheme. Benchmarking is done with respect to the results obtained with the Generic Safety Issue (GSI) 191 framework that makes use of field data for LOCA initiating event probability calculation.

Cyber Security/Cyber Risk

Chair: Clarence Worrell (Westinghouse)

Location: Grand Station IV **Time:** 8:00 - 10:05 am

8:00 am: A Risk-Informed Cyber-Security Program Approach, F. Gregory Hudson, Meredith M. Allen (Metcalf PLLC)

It is generally accepted that the effort required to comply with 10CFR73.54, “Protection of Digital Computer and Communication Systems and Networks” is much greater than initially anticipated. For example, instead of cyber security programs addressing handfuls of plant digital assets (CDAs), hundreds if not thousands of CDAs must be addressed. This situation is aggravated by the regulatory framework of 10CFR73.54 excluding risk management attributes of other cyber-security risk frameworks. Although this current situation has been improved by using a consequence-based methodology for CDA assessments, a more effective resolution will require fundamental changes to current regulatory requirements via rulemaking.

Recognizing the Federal rulemaking process is lengthy, this paper outlines a near-term, risk-informed approach that improves cyber security program efficiency and increases program focus on public safety. This approach complies with 10CFR73.54 and builds upon Industry’s current CDA assessment methodology, facilitating incorporation into existing plant programs. With a December 2017 deadline for 10CFR73.54 compliance, incorporating a risk-informed approach into cyber security programs is realistically a post Milestone 8 program optimization initiative. Although this risk-informed approach is not a substitute for rulemaking, it does improve program performance in the near-term and is a meaningful step towards the long-term solution of rulemaking.

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TECHNICAL SESSIONS - 8:00 AM

Cyber Security/Cyber Risk Continued

8:25 am: An Approach for Evaluating the Consequence of Cyber Attacks on Nuclear Power Plants,
Athi Varuttamaseni, Robert Bari (BNL), Robert Youngblood (INL)

Evaluating the effectiveness of cybersecurity measures implemented at a nuclear power plant requires an understanding of the types of attack that are possible and the consequences of those attacks. In this paper, a systematic approach to determining the system response to various cyber attack scenarios is described. Starting from the initial attack vector, the chain of events taking the plant from the initial state to a compromised state is constructed based on the type of components in the system and the method of attack. System response is evaluated using the standard thermal-hydraulics model of the plant. The paper demonstrates the approach by analyzing the low pressurizer pressure trip function of the reactor protection system and shows how an attack against plant network infrastructure can lead to a noticeable impact on the peak clad temperature following a small loss of coolant accident.

8:50 am: Estimation of Failure On-Demand Probability and Malfunction Rate Values in Cyber-Physical Systems of Nuclear Power Plants, Wei Wang, Francesco Di Maio (Politecnico di Milano), Enrico Zio (Politecnico di Milano, EdF)

Nuclear Power Plants (NPPs) are making increasing use of digital Instrumentation and Control (I&C) systems, which makes them Cyber-Physical Systems (CPSs). In CPSs, cyber and physical processes are dependent and interact with each other: sensors, actuators, communication and computational units are all interconnected to realize functionalities of real-time monitoring, dynamic control and decision support, for normal operation as well as in case of accidents. However, an emerging concern is that the use of computer-based technologies might increase the exposure to failures and accidents, providing new channels for their initiation and propagation. System integrity can be, indeed, affected by hardware component failures, human errors, communication malfunctions and software errors, but also compromised by security breaches and cyber attacks. In practice, these latter could be confused with random components failures on-demand and malfunctions, misjudging their actual nature of malicious cyber attacks and, thus, leading to wrong counteractions.

In this study, we analyze and model stochastic failures in components of CPSs, with the purpose of estimating reference values of failure on-demand probabilities and malfunction rates. Considering these as true values, then, significant difference with statistical estimates from field data collected on the real CPS can be used to detect malicious attempts at altering the safety of a NPP. A digital I&C system of a NPP is taken as illustrative case study, in which components stochastic failures resulting in different system responses are analyzed, and Fault Tree Analysis (FTA) and Markov Chain Modeling (MCM) are taken as approaches to estimate the reference failure on-demand probabilities and malfunction rates.

9:15 am: Research on Vital Digital Assets for Nuclear Cyber Security, Kookheui Kwon, Hyundoo Kim, Siwon Kim (Korea Inst of Nucl Nonproliferation and Control)

Digital systems are used in nuclear facilities to monitor and control various types of field devices, as well as to obtain and store vital information. Therefore it is getting important for nuclear facilities to protect digital systems from cyber-attack in terms of safety operation and public health since cyber compromise of these systems could lead to unacceptable radiological consequences. In this paper, the research on development of methodology for identifying vital digital assets based on nuclear risk assessment is introduced. Initiating Event and Event Tree/Fault Tree model for implementing Probabilistic Risk Assessment was considered to identify vital digital assets which could provoke design extension condition in this paper. And upcoming plans to apply this methodology are touched.

9:40 am: Improvement of Vital Area Identification Method, Woo Sik Jung (Sejong Univ)

A new efficient vital area identification (VAI) method was developed in this study. This new VAI method improves Probabilistic Safety Assessment (PSA)-based VAI that takes advantage of PSA results. Since this new method drastically reduces VAI problem size by (1) performing PSA event tree simplification, (2) calculating preliminary prevention sets with event tree headings, (3) converting the shortest preliminary prevention set into a sabotage fault tree, and (4) performing usual VAI procedure.

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TECHNICAL SESSIONS - 8:00 AM

Fire PSA—IV

Chair: Tatsuya Sakurahara (*Univ of Illinois*)

Location: Grand Station V **Time:** 8:00 - 10:05 am

8:00 am: Advanced Techniques for Modeling Fire-Induced Circuit Failures of Digital Control Systems in Support of Fire Probabilistic Risk Assessments, Nicholas Lovelace, Dane Lovelace, Brett Long, Daniel Funk (*JENSEN HUGHES*)

In this paper, blended approaches are presented for model development and circuit analysis of digital control systems for Fire PRA. As with analog control circuits, a “one-size fits all” approach for digital control circuits is not appropriate. Digital circuits can be functionally categorized, which allows a customized and more representative strategy to be applied to the circuit analysis of each category. This approach focuses on establishing different circuit analysis boundaries best suited for each of the different functional categories, which in turn results in more optimal and realistic model development in support of Fire PRA. These analysis methods can be used as tools to reveal potential vulnerabilities in nuclear power plant digital circuit designs for internal and external hazards, reduce biases and undesired conservatism, provide more realistic failures and effects, improve consistency, minimize unproductive analysis time, and control project cost.

8:25 am: Moving Forward with Developed Fire PRA Models – A Model Owner’s Perspective, Young G. Jo (*Southern Nucl Operating Co*)

After transition to the risk informed performance based fire protection or NFPA 805, US utilities encounter new challenges in using, updating, and upgrading their fire PRA models for continuously supporting NFPA 805 implementation and other risk informed applications. To overcome such challenges, it was proposed to set up three tier goals, identify technical skills/expertise to achieve each tier goal, and develop training programs to acquire the required technical skills/expertise. The proposed approach will help U.S. utilities in acquiring technical capabilities for dealing with the most urgent needs, or plant change evaluations for NFPA 805, as soon as practically possible and for gradually developing in-house technical capabilities for updating and upgrading their fire PRA models.

8:50 am: Is Fire Modeling Getting Better? – A Comparison of Recent Fire Dynamics Simulator Versions for Result Trends, Jeffrey D. Miller (*JDM Consulting, LLC*)

As state-of-the-art fire modeling tools such as Fire Dynamics Simulator (FDS) continue to evolve and hopefully progress with each version release, curious minds ask “how much” though exact metrics are often difficult to quantify. It is generally accepted that software developers are continuously refining their product both for improved functionality and accuracy, but also to take advantage of advances in computing hardware. A properly benchmarked fire modeling workflow may, and should in the author’s opinion, produce a comparison between the output from FDS and known experimental data. This can be done using publicly available industry verification and validation (V&V) fire models. However, historical trends are often not a typical interest of probabilistic safety assessment (PSA) fire modeling studies; often being left as research or educational topics. This paper will present a comparison of a common V&V fire model’s benchmarking outputs between recent major version releases of FDS and to the known experimental data which the fire model is attempting to replicate. In addition, fire model execution time differences, if any, between FDS versions on this same workstation, will be presented.

9:15 am: Predictive Model of the Degradation of Cable Insulation Subject to Radiation and Temperature, Yuan-Shang Chang, Ali Mosleh (*UCLA*)

A physics-based model is developed to represent the elongation at break (EAB) of cable insulation subject to temperature and radiation in Nuclear Power Plants (NPPs). It addresses a safety concern about the degradation of critical cables in long term exposure to radiation, particularly in the context of plant life extension. The proposed model provides a theoretical basis for the EAB curve as a function of time. The curve is divided into two phases: the incubation phase where the virtual degradation rate is indiscernible, and the degradation phase in which the virtual degradation rate is considered constant. When the parameters of the model are modified according to a specific material, the EAB curves at different ageing temperatures and dose rates can be predicted. Data of two widely-used insulating materials is incorporated in this research to validate the proposed model. They are cross-linked polyethylene (XLPE) and ethylene propylene rubber (EPR). Besides fitting the experimental data well, the developed model can also render the value of the activation energy of a degradation progress, which is a necessary parameter in accelerated ageing experiments predicting the remaining life of a material.

9:40 am: The “Incredible” Difficulty of Proving “Incredibility” – Example of Fire-Induced Multiple Spurious Operations, Raymond H. V. Gallucci (*NRC*)

“Risk-informed” regulation is often an alternative to “deterministically-based” regulation that offers relaxation in criteria for acceptability while possibly requiring greater analytical effort. “Risk-informed determinism” is an attempt to meld the best of both worlds by using risk information to set deterministic acceptance criteria a priori. A recent joint effort by the U.S. Nuclear Regulatory Commission’s Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI) originally endeavored to do this for several examples involving fire-induced multiple spurious operations (MSOs) in electrical circuits at nuclear power plants. While a noble effort, this did not consider the actual distributions involved in the events, originally limiting the analysis to mean values and, in some cases, qualitative considerations. A much more comprehensive and defensible approach is performed here where the probabilistic distributions for all the factors are considered via simulation to meet quantitative acceptance criteria related to the concept of “incredibility” that is often the figure of merit that must be met in a deterministic world. The effort demonstrates that it can be “incredibly” difficult to prove “incredibility” in this context.

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TECHNICAL SESSIONS - 8:00 AM

Passive System Safety and Reliability

Chair: Luciano Burgazzi (ENEA)

Location: Waterfront **Time:** 8:00 - 10:05 am

8:00 am: Comparative Assessment of Passive and Active Systems for the Development of Advanced Reactors, Luciano Burgazzi (ENEA)

The introduction of passive systems is regarded as one of the most important factors for safety increase of Gen III/III+ reactors, as well as for the development of small reactor size or the SMR (Small Modular Reactors). However, more detailed studies reveal how such an advantage deriving from the use of passive safety systems than the active ones is not so obvious: thus the assessment of the benefits and the challenges that the adoption of the two types of systems in the various reactors pose. The study is aimed at developing the methodical approach in order to identify the criteria suitable to drive the assessment process at the system level, regardless, in any case, of economic factors.

For this reason both active and passive systems designed to accomplish the required safety functions, as the decay heat removal, have been deeply investigated mainly in terms of their safety performance and reliability shaping factors. Through a qualitatively conducted analysis, factors of both active and passive design are identified and compared on a reliability plane. The analysis points out the relevance of the reliability figure of merit as the most important criterion in the process of opting out of one system in favour of the other alternative. In particular any aspects like the functional failure probability and the large amount of uncertainties, are identified as most relevant factors affecting the system performance assessment in passive design.

8:25 am: A Mechanistic Reliability Assessment of RVACS and Metal Fuel Inherent Reactivity Feedback, David Grabaskas, Acacia J. Brunett, Stefano Passerini, Austin Grelle (ANL)

GE Hitachi Nuclear Energy (GEH) and Argonne National Laboratory (Argonne) participated in a two year collaboration to modernize and update the probabilistic risk assessment (PRA) for the PRISM sodium fast reactor. At a high level, the primary outcome of the project was the development of a next-generation PRA that is intended to enable risk-informed prioritization of safety- and reliability-focused research and development. A central Argonne task during this project was a reliability assessment of passive safety systems, which included the Reactor Vessel Auxiliary Cooling System (RVACS) and the inherent reactivity feedbacks of the metal fuel core. Both systems were examined utilizing a methodology derived from the Reliability Method for Passive Safety Functions (RMPS), with an emphasis on developing success criteria based on mechanistic system modeling while also maintaining consistency with the Fuel Damage Categories (FDCs) of the mechanistic source term assessment. This paper provides an overview of the reliability analyses of both systems, including highlights of the FMEAs, the construction of best-estimate models, uncertain parameter screening and propagation, and the quantification of system failure probability. In particular, special focus is given to the methodologies to perform the analysis of uncertainty propagation and the determination of the likelihood of violating FDC limits. Additionally, important lessons learned are also reviewed, such as optimal sampling methodologies for the discovery of low likelihood failure events and strategies for the combined treatment of aleatory and epistemic uncertainties.

8:50 am: Mission Definition for a Passive Reliability Assessment of SMART Passive Residual Heat Removal System, Douglas A. Fynan, Yong Hun Jung, Jin Hee Park (KAERI)

SMART (System-integrated Modular Advanced Reactor) is an integral pressurized water reactor concept that adopts a passive residual heat removal system (PRHRS) in lieu of conventional auxiliary feedwater and pump/heat exchanger RHR systems. This paper presents mission definitions for the PRHRS performance required to achieve the safe shutdown condition and the supporting engineering analysis as the first step of a passive reliability assessment and optimization study of the PRHRS. Integral-time heat removal capacity and continuous-time heat removal rate missions are derived for the PRHRS and a minimum average heat removal rate of 1.7 MW is required for a 36 hour mission time to achieve safe shutdown. Operating characteristics of single-phase fluid and two-phase water natural circulation modes of the PRHRS are investigated and compared. Two-phase natural circulation is the optimal natural circulation mode of the PRHRS and the loop heat transfer characteristics should be optimized for condensing of saturated steam. Inlet subcooling is a second-order importance design objective and boiler superheat is not important.

9:15 am: Passive System Reliability Analysis Using APSRA+ Methodology and Its Application to Passive Isolation Condenser System of an Advanced Reactor, A. Chandrakar (Homi Bhabha Natl Inst), A. K. Nayak, Vinod Gopika (BARC)

Passive systems are being utilized extensively in current and future generations of reactors for their normal operations as well as safety critical operations during any accidental conditions. In this paper, we present a methodology called Analysis of passive system reliability plus (APSRA+) for evaluating reliability of passive systems. This methodology is an improved version of existing methodology APSRA. The methodology has been applied to the passive isolation condenser system of advanced heavy water reactor (AHWR). With the help of APSRA+ methodology, probability of passive isolation condenser system fail to maintain the clad temperature under 400°C is estimated to be of the order 1×10^{-10} . Important features of APSRA+ are: a) it provides an integrated dynamic reliability method for the consistent treatment of dynamic failure characteristics such as multi-state failure, fault increment and time dependent failure rate of components of passive systems; b) this methodology overcomes the issue of process parameter treatment by just probability density function or by root cause analysis, by segregating them into dependent and independent process parameters and then giving a proper treatment to each of them separately; c) treating the model uncertainties and independent process parameter variations in a consistent manner.

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THURSDAY, SEPTEMBER 28 TECHNICAL SESSIONS - 10:45 AM

Safety Culture and Organizational Factors

Chair: Justin Pence (*Univ of Illinois*)

Location: Grand Station III Time: 10:45 - 11:35 am

10:45 am: Data-Theoretic Methodology and Computational Platform for the Quantification of Organizational Mechanisms in Probabilistic Risk Assessment, Justin Pence, Yicheng Sun, Xuefeng Zhu, Zahra Mohaghegh (*Univ of Illinois*), Cheri Ostroff (*Univ of South Australia*), Ernie Kee (*Univ of Illinois*)

The quantification of organizational mechanisms remains a vital challenge for Human Reliability Assessment (HRA) and Probabilistic Risk Assessment (PRA). This paper reports on the progress of ongoing research by the authors to (a) identify the social and organizational factors that affect technological system risk, (b) examine how and why these factors influence risk, and (c) specify, in conjunction with PRA, how much they contribute to risk. The authors advance the Socio-Technical Risk Analysis (SoTeRiA) framework that explicitly integrates both the social aspects (e.g., safety culture) and the structural features (e.g., safety practices) of organizations with technical system risk models. While SoTeRiA represents a significant improvement in the integration of organizational factors in PRA, the theoretical details of underlying contributing factors and causal interrelationships for each generic node should be expanded. In addition to the theoretical advancements, a major challenge of quantifying organizational factors for PRA relates to lack of reliable data. This research uses a new approach, highlighting that data is available for organizational factors; however, with a different nature from tabular numerical equipment reliability data. The nature of data for organizational factors is evolving to an 'organizational-level' of analysis, i.e., organizational communications. The Data-Theoretic methodology is a hybrid combination of theoretical conceptualization and data analytics to quantify organizational mechanisms. With this approach, the strength of human interpretation (i.e., systematic interpretation and expert opinion) is utilized along with theoretical constructs to develop adequate causality for guiding new big data analytic algorithms aimed at large unstructured and heterogeneous data of textual organizational communications. This paper highlights the two primary parts of the Data-Theoretic methodology: (1) DT-BASE: Develop factors, sub-factors, and causal relationships in SoTeRiA, equipped with a base quantification grounded on theories and interpretation of information in the literature and reports and (2) DT-SITE: Develop site-specific measurement techniques to quantify the causal models based on site-specific data, utilizing automated data extraction and inference methods.

11:10 am: Employee Well-Being Framework to Facilitate a Total Safety Culture Within a Nuclear Power Plant, N. J. F. van Loggelenberg, Hester Nienaber (*Univ of South Africa*)

Employee well-being, i.e. a total state of physical, mental and social health, is a prerequisite for organizational performance. Given its importance, employee well-being is receiving increased attention in the literature from a variety of perspectives. Studies focusing on occupational disease show that occupational stress is on the increase. In this regard the United Nation's International Labor Organization recently described occupational stress as a worldwide epidemic. Occupational stress cannot therefore exclude complex nuclear facilities. Probabilistic Safety Assessment techniques were subsequently used to develop scenarios for hypothetical accidents that might result in severe core damage and to estimate the frequency of such accidents. The US Nuclear Regulatory Commission places emphasis on a Safety Conscious Work Environment as an attribute of a safety culture within a nuclear plant and the SECY-04-0111 regulator stipulates that the necessary full attention should be given to safety matters and that personal dedication and accountability of all should engage in activities which have bearing on the safety of nuclear power plants. Findings of occupational stress studies indicate that the workplace is the main source of occupational stress, which spills over to the environment, family and society. Well-being studies focusing on employee engagement show that few employees are engaged, while the vast majority is not engaged, or even disengaged. This finding suggests that employees' well-being may be at risk, affecting the organization's risk profile. Additionally, these studies reiterate the role of leadership and management in ensuring employee well-being. If organizations do not attend to employee well-being, it may have detrimental consequences for both the employees and the organization. Leadership is ultimately charged with the responsibility of creating an environment nurturing employee well-being, in shaping a total safety culture. The purpose of this paper is to present a theoretical framework of nurturing employee well-being, which aims to facilitate a total safety culture within a nuclear power plant. This framework integrates some of the most often-used tools to improve employee well-being. These include the (i) job-demands-control-support model of stress of Karasek and Theorell, which proposes that work should be reconstructed to minimize, if not avoid bad stress; (ii) the job-diagnostic survey of Hackman and Oldham proposing the redesign of work as organizational change strategy directed at increasing employee motivation and productivity and thus improving organizational performance; and (iii) Kahn's concept of psychological presence, which forms part of employee engagement, which allows employees to be fully present in performing their work roles. This theoretical framework will be empirically tested in subsequent research.

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TECHNICAL SESSIONS - 10:45 AM

Flooding PSA—II

Chair: Zhegang Ma (INL)

Location: Grand Station IV **Time:** 10:45 am - 12:00 pm

10:45 am: A Case Study of Simulation-Based Dynamic Analysis Approach for Modeling Plant Response to Flooding Events, Zhegang Ma, Curtis Smith, Steven Prescott (INL)

External flooding events such as local intense precipitation (LIP), flooding due to upstream dam failure, and coastal flooding due to storm surge or tsunami have the potential to interrupt nuclear power plant operations by challenging offsite power, threatening plant structures, systems and components (SSCs), and limiting plant access. Detailed risk assessments of external flood hazard are often needed while many unique challenges exist in modeling the complete plant response to the flooding event. A framework of simulation-based dynamic flood analysis (SBDFA) has been previously proposed to model the performance of SSCs and operator actions during an external flooding event. This paper presents a case study to apply the SBDFA framework in a LIP event. A state-based PRA modeling tool, EMERALD, is used in the study to incorporate time-related interactions from both 3-D physical simulations and stochastic failures into traditional PRA logic models. An example EMERALD model and the associated 3-D flood simulation models are developed for the LIP event. The quantification results from the EMERALD model and 3-D simulations are compared with those from a traditional PRA model using SAPHIRE. The study shows that the dynamic flood analysis approach could be very useful in modeling plant response to external flooding events with their appealing features. Additional thoughts from the study such as the potential roles the dynamic flood analysis approach might play now and in the future are discussed.

11:10 am: Monte Carlo Simulations for Probabilistic Flood Hazard Assessment, Ahmed "Jemie" A. Dababneh, Mark A. Schwartz (RIZZO Associates)

Development of Flood Hazard Curves (FHC) is a necessary step for the development of Fragility Curves as part of Probabilistic Safety Assessment (PSA). The FHC, developed using a Probabilistic Flood Hazard Assessment (PFHA), illustrate the probability of water levels, flow rates, or velocities. The Fragility Curves used in the PSA process, illustrate the probability of system or component failure for a given load. Typically, these loads are (or are derived from) water levels, flow rates, or flow velocities associated with an external flood event. An example site is used to illustrate the series of calculations involved in a potential uncertainty analysis for flood protection, involving Monte Carlo simulations to evaluate potential levee failure.

11:35 am: Lessons Learned in the Development and Characterization of an External Flood Hazard for a Plant Site with Multiple Upstream Dams, Raymond Schneider, Gary Douglas, Jay Fluehr (Westinghouse), H. A. Hackerott (Omaha Public Power District)

An important aspect of a flood hazard Probabilistic Risk Assessment (PRA) is the establishment of flood hazard curves. In practice the flood hazard development and the characterization of the hazard as they affect human actions and fragility of site systems, structures and components (SSCs) can be complex. One of the more complex situations involves the treatment of sites downstream of one or more dams. This paper focuses on the challenges and considerations involved in developing site hazard curves for a plant downstream of multiple dams and the interaction of the hazard curve with other elements of the flood hazard PRA.

Other External Hazards PSA

Chair: Kyle Hope (Westinghouse)

Location: Grand Station V **Time:** 10:45 am - 12:00 pm

10:45 am: Hazard Curve Evaluation for Forest Fire Smoke Effects on Air-Cooling Decay Heat Removal Systems, Yasushi Okano, Hidemasa Yamano (JAEA)

This study evaluates a hazard curve of smoke effects generated by a forest fire by applying a new method using a logic tree which consists of variable parameters on a forest fire (e.g. fire breakout time and its location), weather conditions (e.g. prevailing wind velocity), types of vegetation and topography (e.g. yield of particle matters and a land elevation map), and simulation conditions (e.g. a model of smoke captured on air filters). A response surface of the smoke spatial density at a nuclear power plant is evaluated using two simulation codes: FARSITE for forest fire propagation and ALOFT-FT for smoke transportation. It is followed by a Monte Carlo simulation on a certain set of parameters for the logic tree followed by obtaining a corresponding result of the amount of the smoke by the response surface, and finally the histogram of all the Monte Carlo sample results gives the hazard curve representing the annual exceedance frequency of the total amount of the smoke captured on air filters of the decay heat removal system. The evaluated hazard curve, normalized per air filter area (1 m^2) and per intake air velocity (1 m/s), is about 1×10^{-1} per year for $1 \text{ kg/m}^2/(\text{m/s})$ and about 1×10^{-2} per year for $3.5 \text{ kg/m}^2/(\text{m/s})$.

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TECHNICAL SESSIONS - 10:45 AM

Other External Hazards PSA Continued

11:10 am: Screening Approach for Systematically Considering Hazards and Hazard Combinations in PRA for a Nuclear Power Plant Site, Marina Roewekamp, Silvio Sperbeck, Gerhard Gaenssmantel (GRS)

As a lesson learnt from the investigations after the reactor accidents of Fukushima, probabilistic risk assessment (PRA) has to systematically address external and internal hazards including potential combinations of hazards with other events and hazards. Several international activities carried out in the recent past, in particular in the frame of the ASAMPESA_E project, have revealed lists with several groups of different hazards, for which it has been checked site and plant specifically, if these need to be considered in the PRA. Moreover, a huge variety of potential event combinations of hazards and other events should be screened as well.

In principle, three types of event combinations involving hazards can be distinguished:

- *Consequential or subsequent events: A (external or internal) hazard induces one or more additional hazards;*
- *Correlated events: A common initiating event (including external hazard) results in one or more hazards, which even may occur simultaneously with a certain probability;*
- *Unrelated events: An initiating event (including hazard) occurs independently from, but simultaneously to a hazard.*

A screening approach has been recently developed by GRS for systematically screening those hazards and hazard combinations, which have to be addressed in PRA on a site and plant specific basis. The approach starts with a generic list of the entire hazards by screening out those ones that can be qualitatively excluded for the nuclear site under investigation. For the remaining hazards, probabilistic screening criteria are applied.

Screening of the hazard combinations starts from those hazards not screened out site specifically as physically impossible. For these hazards not screened out, all physically possible event combinations are identified. Thereafter, detailed hazard combination screening is performed applying again quantitative criteria.

The paper presents the systematic screening approach in general as well as an application example for a reference nuclear power plant (NPP) site in Germany.

11:35 am: Analysis of the Risk of Aircraft Crash Hazard, James C. Lin (ABS Consulting, Inc.)

The aircraft hazard is considered in the nuclear plant design. The Standard Review Plan (SRP) describes the acceptance criteria for regulatory review of the aircraft hazards. When the proximity criteria for nearby airports, military training routes, federal airways, and holding/approach pattern cannot be met, a probabilistic analysis of the crash frequency is performed to demonstrate that the risk of aircraft hazard is insignificant. The risk analysis of aircraft crash resulting from takeoff and landing operations at nearby airports are more straightforward. The evaluation of the frequency of an aircraft crash during the in-flight phase along the airways has become more challenging in recent years due to changes in the flight paths and difficulty in collecting flight frequency data. In present-day aviation, airplanes can fly using the Global Positioning System (GPS) and do not always have to follow the airways. Flight paths are primarily based on the shortest routes between the origin and destination navigated by the GPS. Therefore, air traffic in the airspace nearby the plant cannot be just estimated by the air traffic along the nearby airways. Since the Federal Aviation Administration (FAA) does not maintain records of air traffic along specific airways, the air traffic in the airspace nearby the plant should be estimated using the FAA records on the flights crossing specific latitude/longitude boundaries, which includes not only aircraft operations into and out of a specific nearby airport, but also overflights through the same airspace without landing at that airport. Another interesting factor in estimating the aircraft crash frequency is the effective target area. DOE-STD-3014-2006 provides the most comprehensive model to evaluate this parameter. The DOE-STD-3014-2006 formula can be applied to standalone buildings. For NPPs, however, the shielding and slope effects due to the surrounding buildings and nearby slopes should be addressed. Besides the general insights from the analysis of the aircraft crash risk, this paper also discusses how the aircraft crash frequency from overflights can be analyzed using the available FAA flight data, and how the DOE-STD-3014-2006 formula for effective target area should be modified to account for the shielding and slope effects. Finally, this paper recommends a change to the SRP in the evaluation of aircraft crash due to overflights along the airways.

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TECHNICAL SESSIONS - 10:45 AM

Advanced Information Technology and PSA

Chair: Clarence Worrell (*Westinghouse*)

Location: Waterfront **Time:** 10:45 - 12:00 am

10:45 am: Protective Systems: Possible Extensions to the RoverD Method, Tatsuya Sakurahara, Ha Bui Hong, Justin Pence, Wen-Chi Cheng, Zahra Mohaghegh, Ernie Kee, Seyed Reihani (*Univ of Illinois*), Martin Wortman (*Texas A&M*), Vera Moiseytseva (*YK.Risk*), Fatma Yilmaz (*STP*), David Johnson (*UCLA*)

*The Risk-informed Over Deterministic (RoverD) method has been developed to address the U.S. Nuclear Regulatory Commission's Generic Safety Issue 191 (GSI-191). As part of a broader line of research to support the Nuclear Energy Institute's "Nuclear Promise" initiative that seeks to find ways to reduce Nuclear Power Plant (NPP) costs, the RoverD method is being expanded by the authors (a multidisciplinary group from academia and industry) for applications beyond GSI-191. The central research question in this work is: "to what extent can the RoverD method be useful in evaluating cost-saving alternatives in those applications where a tested design level is being questioned with respect to its adequacy for providing protection against internal and external events (e.g., fires, floods, earthquakes, reactor trip transients) that may lead to a release of radioactivity from the containment of NPPs?" Occasionally, changes in the regulatory setting lead to new design requirements which may question the protective system's capability to meet safety margin and defense-in-depth requirements. These new requirements could result in a significant cost for the operation and maintenance of NPPs. High costs are typically associated with the redesign and modification of Systems, Structures, and Components (SSC). In addition, well-intended plant modifications may introduce new failure modes due to unforeseen organizational factors (e.g., training and maintenance) or inadequacies of new designs. The RoverD method is a framework that provides the possibility to quantify, using a conservative approach, the extent of safety margin and defense-in-depth in the current design of protective systems and to identify the SSC modifications that can be avoided or addressed at a lower cost. Comparing conservative plant risk estimates against regulatory risk acceptance criteria (e.g., Regulatory Guide 1.174), the RoverD method indicates if an emerging safety concern is risk significant on a plant-specific basis. This research also introduces a two-step risk-informed strategy (i.e., **Step 1: RoverD**; and if the plant risk does not fall within the regulatory risk acceptance criteria, **Step 2: Integrated Probabilistic Risk Assessment**).*

11:10 am: Protective Systems: Margins of Safety, Regulatory Authority, and the Calculus of Negligence, Martin Wortman (*Texas A&M*), Ernie Kee (*Univ of Illinois*), Vera Moiseytseva (*YK.Risk*), Fatma Yilmaz (*STP*), David Johnson (*UCLA*)

The design and operation of protective systems is an essential engineering responsibility. While ensuring public safety is essential, it must be accomplished at a reasonable cost and within the regulatory guidelines set by the government; hence, protective system design and operational decisions must be evaluated with respect to benefit (both enterprise profit and social benefit) and cost (both enterprise and social costs).

Analytical arguments are made that establish the economic relationship between protective system margins of safety, regulatory authority, and the calculus of negligence. Within this risk-based analytical framework, protection efficacy is explored. In particular, the risk-economics of margins of safety are examined by identifying the referenced efficacy with respect to which margins of safety are measured. Engineering design and operations decisions intended to improve protection efficacy can, thus, be gauged as the degree to which they advance a risk-based margin of safety.

The proposed analytical framework is then exercised to show how risk-based margins of safety reveal the relationship between uncertain costs and regulatory activity that is focused on ensuring a public welfare that is backstopped by liability in the event of catastrophe. How both prescriptive and performance based regulations influence margins of safety with respect to protective system innovation can be identified here.

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THURSDAY, SEPTEMBER 28

TECHNICAL SESSIONS - 10:45 AM

Advanced Information Technology and PSA Continued

11:35 am: Sensitivity Analysis of Dose Conversion Factor and Deposition Velocity Input Parameter Impacts on Level 3 PSA Sama Metrics, Kevin R. O’Kula, David C. Thoman, Maeley K. Brown
(AECOM Technical Services)

Most U.S. Severe Accident Mitigation Alternatives (SAMA) analyses supporting the relicensing of nuclear power plants (NPPs) have followed U.S. Nuclear Regulatory Commission-endorsed industry guidance and earlier relicensing precedent in quantifying the Level 3 Probabilistic Safety Analysis (PSA) metrics important for performing the subsequent cost/benefit analysis process. For this purpose, U.S. licensees typically apply the MELCOR Accident Consequence Code System Version 2 (MACCS2) in calculating consequences as a result of postulated severe accidents, including population dose and offsite economic cost incurred within a fifty-mile radius of the plant. While many tens of inputs shape the outcome of the analysis, key input parameters are: (1) the deposition velocity during atmospheric transport over the region of interest; and (2) the factors to calculate the dose from internal and external exposure pathways from radionuclides released in the severe accident. Precedent from the first group of plants applying for life extension has been to input a standard single-value deposition velocity to radionuclides that are subject to dry depletion mechanisms. For the second parameter, numerous SAMA analyses have applied inhalation dose conversion factors and external dose coefficients from Federal Guidance Reports (FGRs) 11 and 12, respectively. In this paper, a layered Level 3 PSA sensitivity analysis is performed for a nominal, inland U.S. plant site, applying the earlier described inputs and then recalculated using the updated input parameter values for the two sets of key inputs. The updated deposition velocity information is taken from the State of the Art Consequence Analyses (SOARCA) study, and the alternative inhalation dose conversion and external dose factors are based on ICRP 70 and FGR 13 information. The Level 3 PSA SAMA metrics of population dose and offsite economic cost are assessed against changes in deposition velocity and the dose conversion and external dose factor files. The outcome of the MACCS2 sensitivity study suggests that: (1) internal dose coefficient file change leads to an incrementally small decrease in population dose; (2) external dose factor changes to the FGR 13 recommendations have about the same small impact to population dose; (3) deposition velocity has a significant impact decreasing both population dose and the offsite economic cost; and (4) introducing all three changes leads to an appreciable decrease in population dose and economic metrics. The chief causes of these impacts are discussed, as well as the limitations and insights obtained from the study.

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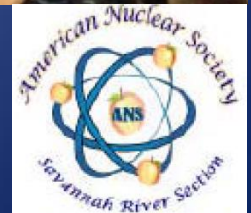
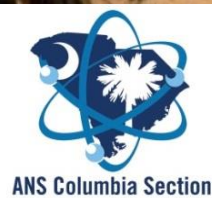
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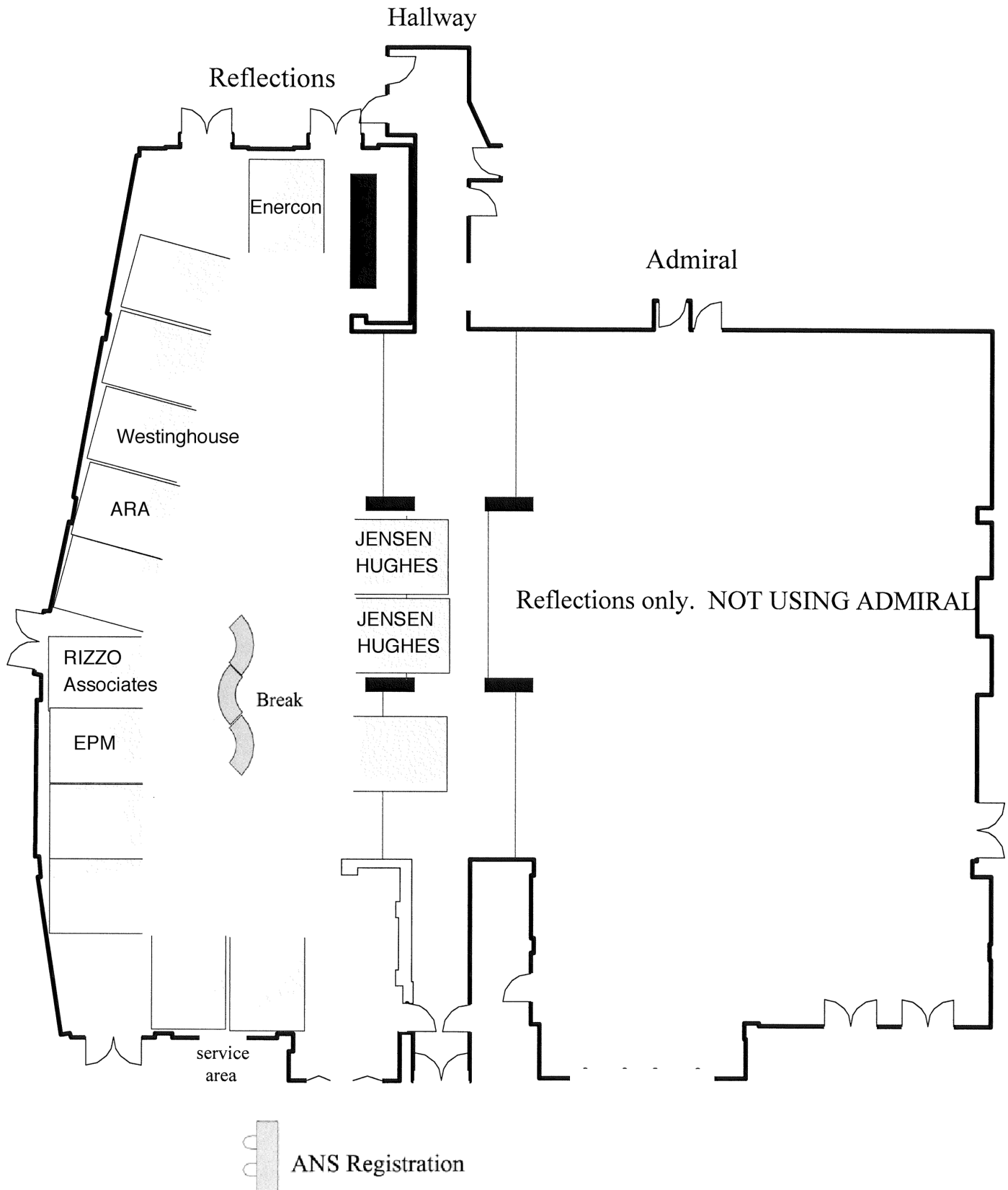
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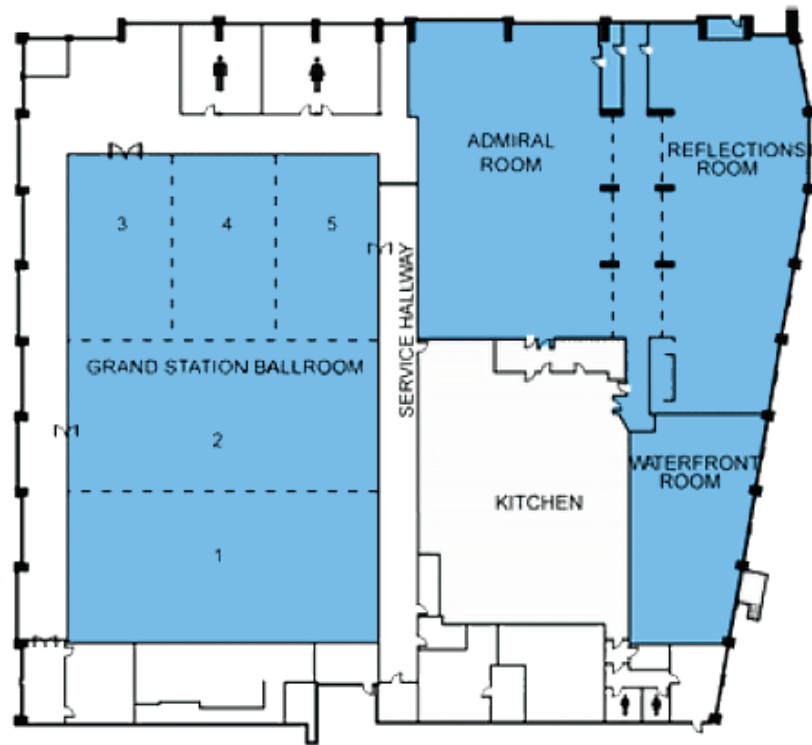
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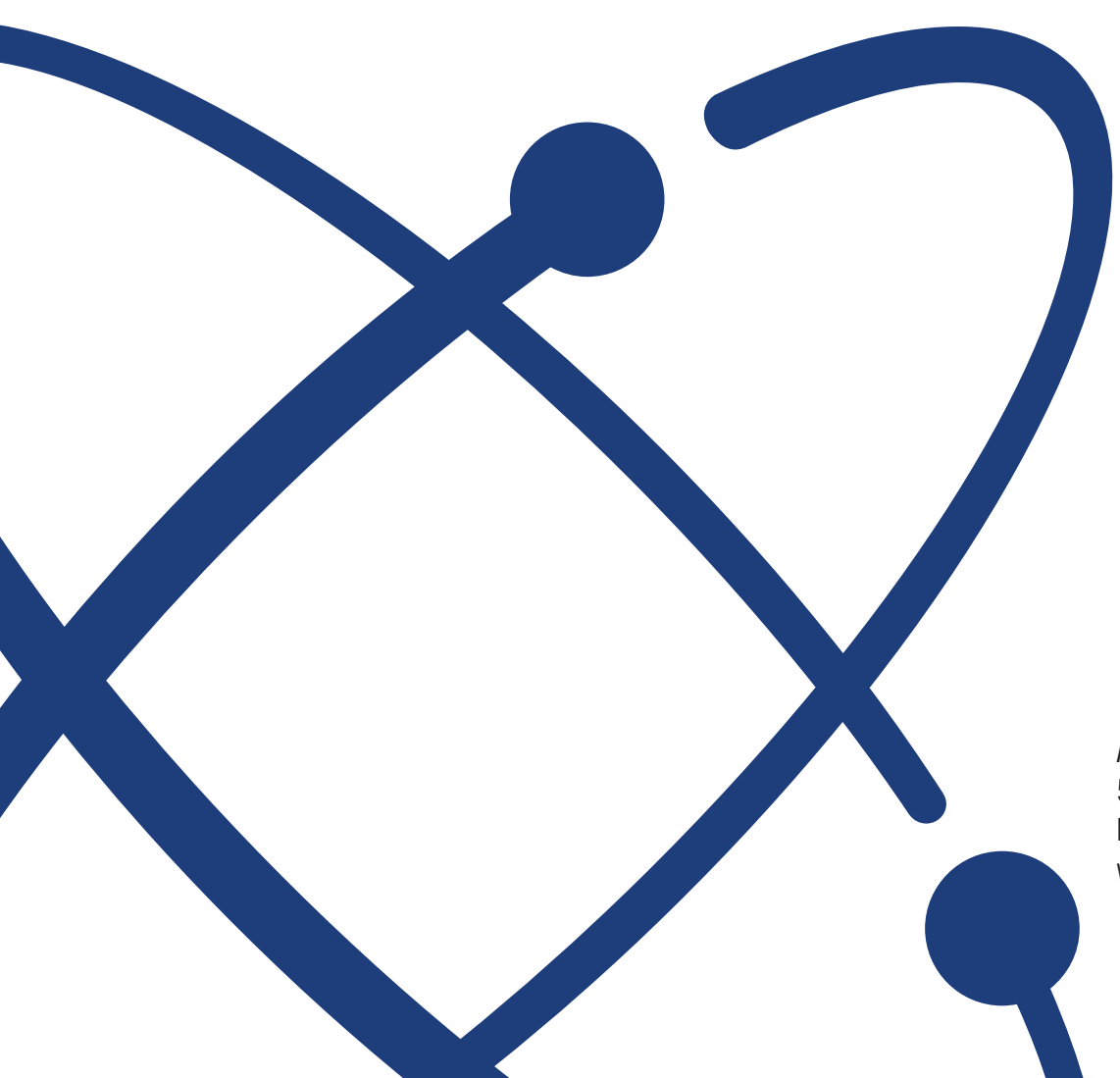
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