ANS Meetings

16th International Topical Meeting on Probabilistic Safety Assessment and Analysis

PSA 2019

Official Program

April 28-May 4, 2019
Charleston Marriott
Charleston, South Carolina, United States

ANS PSA 2019

International Topical Meeting on Probabilistic Safety Assessment and Analysis

Our most sincere thanks to our sponsors for their support of the 2019 PSA International Topical Meeting on Probabilistic Safety Assessment and Analysis

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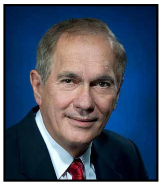
PLENARY, SPECIAL SESSIONS & EVENTS

PSA 2019 TECHNICAL SESSIONS

ADDITIONAL

Organizing Committee

16th International Topical Meeting on Probabilistic Safety Assessment and Analysis



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Robert A. Bari
Senior Physicist Emeritus, Brookhaven National Laboratory (ret.)

GENERAL CHAIR, WORKSHOPS COORDINATOR
Kevin O'Kula
AECOM Technical Services

TECHNICAL PROGRAM CHAIR
Professor Mohammad Modarres
University of Maryland

ASSISTANT TECHNICAL PROGRAM CHAIR
Bonnie Shapiro
Savannah River Nuclear Solutions

STUDENT PROGRAM CHAIR
Professor Travis Knight
University of South Carolina

EARLY CAREER COORDINATOR
Maeley Brown
AECOM Technical Services

FINANCE CHAIR
Damon Bryson
V.C. Summer, SCANA

ARRANGEMENTS
John Dewes
Idaho National Lab

ASSISTANT CHAIR STUDENT PROGRAM
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University of Maryland

ARRANGEMENTS
Beth Vail
AECOM Technical Services

PUBLICATIONS
Tracy Stover
Savannah River Nuclear Solutions

WEBSITE
Liz Henley
AECOM Technical Services

HLW TOUR
Carl Benhardt
AECOM Technical Services

REGISTRATION
Tinh Tran
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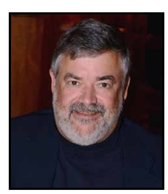

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Duane Twining
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TOURS AND EVENTS: CHARLESTON HARBOR TOUR
Linda Wright
Retirees, AECOM

TOURS AND EVENTS: VOGTLE NEW BUILD
Mike McCracken
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TOURS AND EVENTS: VOGTLE NEW BUILD
TJ Corder
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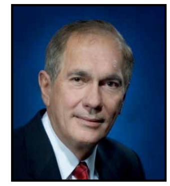
Technical Program Advisory Board

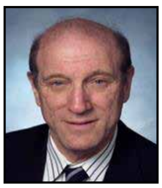
16th International Topical Meeting on Probabilistic Safety Assessment and Analysis

TECHNICAL PROGRAM CHAIR
Professor Mohammad Modarres
University of Maryland

2019 PSA BOARD MEMBER
Dr. George Apostolakis
Nuclear Risk Research Center



2019 PSA BOARD MEMBER
Dr. Robert Bari
Brookhaven National Laboratory



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Lawrence Berkeley National Laboratory

2019 PSA BOARD MEMBER
Ms. Cornelia Spitzer
International Atomic Energy Agency

2019 PSA BOARD MEMBER
Professor Akira Yamaguchi
University of Tokyo

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Vanderbilt University and CRESP

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Shahen Poghosyan
IAEA

Tomoaki Yoshida
Nuclear Risk Research Center

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SCANA & Politecnico di Milano

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EDF Ricky Summit EPM, Inc.

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BNL, retired

C. Ray Lux
AECOM Technical Services

Welcome to Charleston

City of Charleston

JOHN J. TECKLENBURG
MAYOR

April 28, 2019

WELCOME!

It is my pleasure to welcome the 16th International Topical Conference on Probabilistic Safety Assessment and Analysis (PSA 2019) as you gather in Charleston on April 28-May 4, 2019. As Mayor of the City of Charleston, and on behalf of Charleston City Council and all our citizens, I am honored and delighted that you are here.

I am excited that your international community of nuclear safety and risk applications professionals are gathering here. For some of you, this is your first visit to Charleston, while for others it is a welcome return to our historic city. Charleston has been voted twice as the "Top City in the United States" in the Conde Nast Traveler Reader's Choice Awards. Additionally, Travel & Leisure named Charleston as one of the "World's Best Cities" for 2018. These honors are a recognition of the priority we place on ensuring that your visit here is one that is memorable and will inspire you to return again.

As you convene in Charleston, I hope that your group will enjoy the many amazing interchanges and discussions you have planned for the week. I encourage you to relax and enjoy our slower, graceful way of living. Take in all that you can, including the incredible array of attractions, dining, and entertainment the Holy City has to offer. There is so much to enjoy that I am sure you will find that one visit is not enough. We look forward to hosting you again soon.

My best wishes for a great conference!

Most sincerely yours,

John J. Tecklenburg
Mayor, City of Charleston

Welcome to PSA 2019

Dear PSA 2019 participants,

It is my great pleasure to welcome you to the 2019 International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2019). For over 40 years, PSA topical meetings have brought together the brightest minds from the nuclear industry, national laboratories, and regulatory and academic institutions of the United States and other countries of the world. PSA 2019 keeps this tradition with its outstanding collection of participants, papers, and authors.

We have a superb set of accepted publications, fully peer reviewed by our esteemed Technical Program Committee (TPC) members. In addition, we have a number of plenary and panel sessions to bring you the most current and critical topics in PSA, presented by highly recognized authorities in this field. We are offering 38 technical sessions and 11 panels. I am confident that you will find the topics highly stimulating, readily applicable, and thought-provoking. Additionally, we will have some great tutorials toward the beginning and the end of the meeting.

I would like to thank all the members of the TPC for their contributions and their invaluable time spent on peer reviews. I also would like to extend special thanks to the meeting's general chair, Dr. Kevin O'Kula, for his tireless efforts in nearly every aspect of this conference. Without his leadership this conference would have not been possible. Also, special thanks go to Ms. Bonnie Shapiro for her critical assistance to the TPC.

Finally, I would like to welcome you to the historic city of Charleston, South Carolina, and encourage you to enjoy and explore various historic sites as well as the prized Southern cuisine and art galleries the city has to offer, and I especially recommend you to spend time to see and learn more about the earthquake of 1886 and its aftermath.

You will see me throughout the meeting, and I am looking forward to personally greeting and welcoming each and every one of you.

Sincerely,

M. Modarres

Mohammad Modarres
TPC Chair, PSA2019

Foreword

Dear PSA 2019 colleagues, friends, and guests,

Welcome to Charleston, South Carolina, the site of the 2019 American Nuclear Society (ANS) International Topical Meeting on Probabilistic Safety Assessment and Analysis (PSA 2019). This is the 16th edition in the PSA series and is sponsored by the Nuclear Installations Safety Division (NISD) of ANS. The Savannah River and Columbia Sections of ANS are honored to be co-hosts for PSA 2019.

Since 1978, the ANS PSA biennial topical conference has been a forum for communication on major probabilistic risk and nuclear safety topics worldwide, including issues, methods, applications, insights, policy, research, and risk-informed regulation experience. PSA 2019 continues this conversation through presentations, daily plenary and panel sessions, and in general, face-to-face discussions with colleagues in traditional PSA areas as well as emerging work. Included are multiunit and full-site PSA, dynamic PSA, advanced and small modular reactors (SMRs), the human-machine interface and human reliability assessment (HRA), waste processing/cleanup, reliability of systems/structures/components, natural phenomena/seismic and fire risk, the growing area of decommissioning and decontamination of nuclear facilities, testing to reduce uncertainties, as well as advances in core areas of traditional Level 1, 2, and 3 PSA. The overarching theme of PSA 2019 is the role of probabilistic methods in understanding uncertainties and improving the safety and security of nuclear facilities and activities.

In addition to daily plenary sessions on PSA knowledge management, external hazards methodologies, international perspectives at the International Atomic Energy Agency and its member states, and the role of educational institutions on advancing PSA, PSA 2019 will provide 11 panel sessions throughout the five-day conference schedule. Some of the topics covered will be current and emerging work in multiunit PSA, HRA technology, precursor analysis, conservatisms and safety margins, PSA standards, advanced and SMR PSA, research and development prioritization, dynamic PSA, and multiunit and risk aggregation. We are pleased to have a special Thursday panel to recognize the 40th anniversary of the Three Mile Island event and the lessons learned as seen by those directly involved with managing the recovery and addressing major safety aspects of the event. These panel opportunities are complemented by nearly 40 oral presentation sessions throughout the mornings and afternoons of PSA 2019. Together with nuclear facility technical tours and Charleston cultural tour opportunities, PSA 2019 promises to be a full, but rewarding experience for students and PSA practitioners of all ages and experience levels.

As the general chair for PSA 2019, I would like to recognize three individuals in particular and encourage you to greet them during the course of the week. The first is the technical program chair, Professor Mohammad Modarres (University of Maryland). Despite the globe-trotting schedule of an academic, Dr. Modarres agreed to work remotely over the past two-and-a-half years to help organize and promote the PSA 2019 technical program content—no small feat! The second individual is Dr. Robert Youngblood (Idaho National Laboratory), who will be recognized at Thursday's banquet with one of NISD's most prestigious awards, the Theos J. "Tommy" Thompson Award in reactor safety. Finally, take a moment to personally greet Dr. Robert A. Bari (Brookhaven National Laboratory), Senior Physicist Emeritus, as PSA 2019's honorary chair. Many others who helped in extraordinary ways to build this conference are identified elsewhere in this program, including the Senior Advisory Panel, the more than 60 members of the Technical Program Committee, and last but certainly not least, the PSA 2019 Organizing Committee.

Take advantage of the evenings and non-session time to explore the City of Charleston and the surrounding area. Charleston provides one of the most beautiful settings in the world to host an international community of nuclear safety and risk applications professionals. To encourage this exploration, PSA 2019 is paused in midweek to allow off-site travel and casual downtime. Please have fun and network with both longtime and new colleagues. However, all participants are expected back in session on Thursday morning!

On behalf of the members of the Organizing Committee, we invite you to actively engage in PSA 2019 and wish you a great stay in this cultural and historic city. We hope you can experience true Charleston charm and Southern hospitality during your stay. Feel free to call upon the PSA 2019 staff to assist you during your visit.

Sincerely,

Kevin O'Kula

Kevin O'Kula, General Chair, PSA 2019

Daily Schedule

Sunday, April 28

8:00 am-6:00 pm	Meeting Registration	Topaz/Opal Prefunction
8:00 am-6:00 pm	Exhibit Set Up	Topaz/Opal/Emerald Prefunction
9:00 am-12:00 pm	Dynamic PRA Workshop—I	Yellow Topaz & Blue Topaz
12:00-1:00 pm	Lunch on your Own	
1:00-4:30 pm	Dynamic PRA Workshop—II	Yellow Topaz & Blue Topaz
1:00-5:00 pm	Saphire Workshop	Opal 1
3:00-3:30 pm	Mid-Afternoon Break	Topaz/Opal Prefunction
6:00-9:00 pm	PSA 2019 Arrival Meet & Greet	Courtyard *(Weather Permitting)* or Emerald

Monday, April 29

7:00 am-6:00 pm	Meeting Registration	Topaz/Opal Prefunction
7:00-8:00 am	Continental Breakfast–Monday Chairs and Speakers	Opal 1
8:00-9:00 am	Spouse/Guest Breakfast	Opal 1
8:00-10:00 am	Opening Plenary	Emerald
10:00-10:30 am	Mid-Morning Break	Topaz/Opal/Emerald Prefunction
10:30 am-12:15 pm	Opening Plenary Panel	Emerald
12:00-5:00 pm	Exhibits	Topaz/Opal/Emerald Prefunction
12:15-1:45 pm	PSA 2019 Luncheon ***Sponsored by*** Westinghouse	Emerald
1:45-3:25 pm	Multi-Unit PSA and Risk Integration—I	Emerald Salon One
1:45-3:15 pm	Internal Events—I	Emerald Salon Two
1:45-3:10 pm	Working Group & International Program Insights	Emerald Salon Three
1:45-3:15 pm	Risk-Informed Decision-Making—I	Yellow Topaz
1:45-3:15 pm	Passive System Reliability	Blue Topaz
1:30-3:15 pm	Understanding and Managing Conservatisms and Safety Margins to Support Safety Decisions–Panel	Opal 1
3:15-3:45 pm	Mid-Afternoon Break	Topaz/Opal/Emerald Prefunction
3:45-5:25 pm	Multi-Unit PSA and Risk Integration—II	Emerald Salon One
3:45-5:15 pm	Internal Events—II	Emerald Salon Two
3:45-5:00 pm	Risk-Informed Decision-Making—II	Emerald Salon Three
3:45-4:45 pm	Risk Management	Yellow Topaz
3:45-5:15 pm	Extended Sequences	Blue Topaz
3:45-5:15 pm	Criticality Safety Insights	Opal 1
6:00-9:00 pm	PSA 2019 Opening Reception ***Hosted by*** JENSEN HUGHES Advancing the Science of Safety	South Carolina Aquarium

Daily Schedule

Tuesday, April 30

7:00 am-6:00 pm	Registration	Topaz/Opal Prefunction
7:00-8:00 am	Continental Breakfast–Tuesday Chairs and Speakers	Opal One
8:00-9:00 am	Spouse/Guest Breakfast	Opal One
8:00 am-5:00 pm	Exhibits	Topaz/Opal/Emerald Prefunction
8:00-9:30 am	Tuesday Opening Plenary—II	Emerald
9:30-10:00 am	Mid-Morning Break	Topaz/Opal/Emerald Prefunction
10:00 am-12:00 pm	Seismic Multi-Unit PSA: Special Challenges and Opportunities–Panel	Emerald
12:00-1:30 pm	Buffet Luncheon	Topaz/Opal/Emerald Prefunction
1:30-3:00 pm	SMR and Advanced Reactor PSA	Emerald Salon One
1:30-3:10 pm	External Events—I	Emerald Salon Two
1:30-3:10 pm	Level 1/2 PSA—I	Blue Topaz
1:30-3:00 pm	Digital I&C, Software Reliability, and Cyber Risk	Yellow Topaz
1:30-3:00 pm	Internal Events and Common Causes	Opal One
1:30-3:15 pm	Advancing HRA Technology: Short-Term and Long-Term Needs–Panel	Emerald Salon Three
3:00-3:30 pm	Mid-Afternoon Break	Topaz/Opal/Emerald Prefunction
3:30-5:15 pm	Learning from Experienced Nuclear Events: The Role of Precursor Analysis–Panel	Emerald Salon One
3:30-5:10 pm	External Events—II	Emerald Salon Two
3:30-5:10 pm	Level 1/2 PSA—II	Emerald Salon Three
3:30-5:10 pm	Risk-Informed Regulation—I	Yellow Topaz
3:30-5:10 pm	Low Power Risk, Accident Management and Emergency Planning	Opal One
3:15-5:15 pm	PRA Standard Update–Panel	Blue Topaz
5:30-9:30 pm	Charleston Harbor Dinner Cruise	Charleston Harbor

Wednesday, May 1

7:00 am-12:00 pm	Registration	Topaz/Opal Prefunction
7:00 am-6:00 pm	Exhibits	Topaz/Opal/Emerald Prefunction
7:00-8:00 am	Continental Breakfast–Wednesday Chairs Presenters, and Speakers	Opal One
8:00-9:00 am	Spouse/Guest Breakfast	Opal One
8:00-9:30 am	Wednesday Opening Plenary—III	Emerald
9:30-10:00 am	Mid-Morning Break	Topaz/Opal/Emerald Prefunction
10:00 am-12:00 pm	State-of-the-Art Consequence Analysis (SOARCA)/Uncertainty Analysis	Emerald Salon One
10:00 am-12:00 pm	Dynamic PSA—I	Emerald Salon Two
10:00 am-12:00 pm	Level 1/2 PSA—III	Emerald Salon Three
10:00-11:30 am	Risk Informed Regulation—II	Opal One
10:00 am-12:00 pm	Insights from Advanced and Small Modular Reactor PRA Development–Panel	Blue Topaz
10:00 am-12:00 pm	Plant and Site Level PSA Applications—I	Yellow Topaz
12:00-10:00 pm	Tours to Plant Vogtle, HLW at SRS, Hunley, Charleston Walking Tours and Otherwise, Time on your own to explore Charleston	

Daily Schedule

Thursday, May 2

7:00 am-1:00 pm	Registration	Topaz/Opal Prefunction
7:00-8:00 am	Continental Breakfast–Thursday Chairs and Speakers	Opal One
7:00 am-12:00 pm	Exhibits	Topaz/Opal/Emerald Prefunction
8:00-9:00 am	Spouse/Guest Breakfast	Opal One
8:00-10:00 am	Thursday Opening Plenary—IV	Emerald
10:00-10:30 am	Mid-Morning Break	Topaz/Opal/Emerald Prefunction
10:30 am-12:00 pm	Thursday Plenary—V	Emerald
12:30-1:30 pm	Buffet Luncheon	Topaz/Opal/Emerald Prefunction
1:30-3:00 pm	Dynamic PSA—II	Emerald Salon One
1:30-3:10 pm	Human Reliability Analysis and Human Factors—I	Emerald Salon Two
1:30-3:10 pm	Level 3 PSA	Opal One
1:30-3:10 pm	Safety Goals Risk Metrics and Guidance Updates	Yellow Topaz
1:30-3:10 pm	Reliability Estimation and Data Analysis—I	Blue Topaz
1:30-3:10 pm	Plant and Site Level PSA Applications—II	Emerald Salon Three
3:00-3:30 pm	Mid-Afternoon Break	Topaz/Opal/Emerald Prefunction
3:30-5:00 pm	Dynamic PSA—III	Emerald Salon One
3:30-4:30 pm	Education, Training, and Knowledge Management	Opal Two
3:30-5:00 pm	Computer Tools	Emerald Salon Three
3:30-5:10 pm	Reliability Estimation and Data Analysis—II	Yellow Topaz
3:30-5:10 pm	Uncertainty Quantification	Blue Topaz
3:15-5:45 pm	Risk Informed R&D Prioritization: Near-Term and Long-Term Needs–Panel	Emerald Salon Two
6:30-9:00 pm	PSA 2019 Banquet	Emerald

Friday, May 3

7:00 am-12:00 pm	Registration	Topaz/Opal Prefunction
7:00-8:00 am	Continental Breakfast – Wednesday Chairs Presenters, and Speakers	Opal One
8:00-9:00 am	Spouse/Guest Breakfast	Opal One
8:00-9:30 am	Dynamic PSA Standard Development Initiation–Panel	Emerald Salon One
8:00-9:40 am	Human Reliability Analysis and Human Factors—II	Emerald Salon Two
8:00-9:30 am	DOE Accident Analysis: Uncertainty, Conservatism and Testing to Reduce Conservatism	Emerald Salon Three
9:30-10:00 am	Mid-Morning Break	Topaz/Opal/Emerald Prefunction
10:00 am-12:00 pm	Understanding of the Overall Risk Profile: Multiunit Context and Risk Aggregation Topics–Panel	Emerald
1:00-5:00 pm	MACCS Workshop—I	Topaz

Saturday, May 4

8:00 am-12:00 pm	MACCS Workshop—II	Topaz

General Information

MEETING VENUE

The PSA 2019 Conference will be hosted at The Marriott Charleston, April 28-May 4, 2019. The Marriott Charleston is located at 170 Lockwood Boulevard, Charleston, South Carolina 29403. Phone: 843-723-3000

REGISTRATION

Meeting registration is required for all attendees, and speakers. Badges are required for admission to all plenaries, technical sessions and events.

REGISTRATION LOCATION & HOURS

Room: Topaz/Opal Prefunction

Sunday, April 28	8:00 am – 6:00 pm
Monday, April 29	7:00 am – 6:00 pm
Tuesday, April 30	7:00 am – 6:00 pm
Wednesday, May 1	7:00 am – 12:00 pm
Thursday, May 2	7:00 am – 1:00 pm
Friday, May 3	7:00 am – 12:00 pm

NOTICE FOR SPEAKERS:

All Speakers and Session Chairs must check in at the ANS Registration Desk during registration hours. Each day's speakers/panelists should plan to attend a Speakers Breakfast (Opal 1)to meet with their Chairs/Moderators and should bring their Power Point files on a USB drive for transfer/download at the Speaker-Ready desk in Opal 2. The Speaker Breakfast will begin each day (Monday - Friday at 7:00 am). Speakers should provide a short biographical sketch at the time of registration but no later than the Speaker breakfast. Please report to your session room at least 15 minutes before the start of the session. Please sit near the front of the room near the Session Chair to facilitate communications. Unless you need to support another PSA 2019 function, please plan to stay in the session room until the last speaker has presented to actively participate in the session discussion once all speakers have presented.

ATTENDEE MEAL FUNCTIONS

Breaks will be provided to all registered meeting attendees, Monday-Friday.

The Welcome Reception (Sunday), PSA 2019 Opening Reception (Monday), and PSA 2019 Banquet (Thursday) are provided to all registered PSA 2019 attendees. Lunch will be provided to all registered meeting attendees,Monday, Tuesday, and Thursday. Your badge is required for entry to these events. Additional tickets are available for purchase.

Tickets for lunches and events are available for purchase for guests.

General Information

ABOUT ANS

Mission

ANS provides its members with opportunities for professional development. It also serves the nuclear community by creating a forum for sharing information and advancements in technology, and by engaging the public and policymakers through communication outreach.

Statement on Diversity

The American Nuclear Society (ANS) is committed, in principle and in practice, to creating a diverse and welcoming environment for everyone interested in nuclear science and technology. Diversity means creating an environment – both in ANS and in the profession – in which all members are valued equitably for their skills and abilities and respected equally for their unique perspectives and experiences. Diverse backgrounds foster unique contributions and capabilities, and so creation of an inclusive Society ultimately leads to a more creative, effective, and technically respected Society.

ANS believes that everyone deserves opportunities for learning, networking, leadership, training, recognition, volunteering in Society activities, and all the other benefits that involvement in the Society brings, regardless of age, color, creed, disability, ethnicity, gender identity and expression, marital status, military service status, national origin, parental status, physical appearance, race, religion, sex, or sexual orientation. The selection of a member to serve in ANS's volunteer leadership structure shall be based solely on the member's ability, interest and commitment to serve. In particular, ANS encourages members at each level of the Society and in each Professional Division and Technical Group to make special efforts to recruit underrepresented minorities and women to ensure that they are adequately represented in the Society.

Respectful Behavior Policy (Abbreviated)

The open exchange of ideas, freedom of thought and expression, and productive scientific debate are central to the mission of the American Nuclear Society (ANS). These require an open and diverse environment that is built on dignity and mutual respect for all participants and ANS staff members, and is free of bias and intimidation.

ANS is dedicated to providing a safe, welcoming, and productive experience for everyone participating in Society events and other Society activities regardless of age, color, creed, disability, ethnicity, gender identity and expression, marital status, military service status, national origin, parental status, physical appearance, race, religion, sex, or sexual orientation. Creation of a safe and welcoming environment is a shared responsibility held by all participants. Therefore, ANS will not tolerate harassment of or by participants (including ANS volunteer leaders and staff members) in any form. Disciplinary action for participants found to have violated this principle may include reprimand, expulsion from an event or activity with or without a refund, temporary or permanent exclusion from all ANS events and activities, suspension or expulsion from volunteer leadership positions or groups, and/or suspension or expulsion from Society membership, as appropriate.

If you or someone else experiences harassment, regardless of how you otherwise choose to initially handle the situation, you are encouraged to report the situation to ANS. It is possible that the behavior you experienced is part of a larger pattern of repeated harassment. Please alert ANS to behavior you feel to be harassment regardless of the offender's identity or standing in the Society.

The designated contact for reports at PSA 2019 is Dr. Kevin R. O'Kula. He can be reached by phone at 803.640.2572 or by email: kevin.okula@aecom.com, or you can leave a message at the Registration Desk for him to contact you directly.

The complete Respectful Behavior Policy can be found at www.ans.org/about/rbp. If you have questions about the policy, please contact ANS Executive Director Bob Fine at 708-579-8200 or rfine@ans.org.

ANS CODE OF ETHICS

Preamble

Recognizing the profound importance of nuclear science and technology in affecting the quality of life throughout the world, members of the American Nuclear Society (ANS) are committed to the highest ethical and professional conduct.

Fundamental Principle

ANS members as professionals are dedicated to improving the understanding of nuclear science and technology, appropriate applications, and potential consequences of their use.

To that end, ANS members uphold and advance the integrity and honor of their professions by using their knowledge and skill for the enhancement of human welfare and the environment; being honest and impartial; serving with fidelity the public, their employers, and their clients; and striving to continuously improve the competence and prestige of their various professions.

ANS members shall subscribe to the following practices of professional conduct:

Principles of Professional Conduct

1. We hold paramount the safety, health, and welfare of the public and fellow workers, work to protect the environment, and strive to comply with the principles of sustainable development in the performance of our professional duties.
2. We will formally advise our employers, clients, or any appropriate authority and, if warranted, consider further disclosure, if and when we perceive that pursuit of our professional duties might have adverse consequences for the present or future public and fellow worker health and safety or the environment.
3. We act in accordance with all applicable laws and these Practices, lend support to others who strive to do likewise, and report violations to appropriate authorities.
4. We perform only those services that we are qualified by training or experience to perform, and provide full disclosure of our qualifications.
5. We present all data and claims, with their bases, truthfully, and are honest and truthful in all aspects of our professional activities. We issue public statements and make presentations on professional matters in an objective and truthful manner.
6. We continue our professional development and maintain an ethical commitment throughout our careers, encourage similar actions by our colleagues, and provide opportunities for the professional and ethical training of those persons under our supervision.
7. We act in a professional and ethical manner towards each employer or client and act as faithful agents or trustees, disclosing nothing of a proprietary nature concerning the business affairs or technical processes of any present or former client or employer without specific consent, unless necessary to abide by other provisions of this Code or applicable laws.
8. We disclose to affected parties, known or potential conflicts of interest or other circumstances, which might influence, or appear to influence, our judgment or impair the fairness or quality of our performance.
9. We treat all persons fairly.
10. We build our professional reputation on the merit of our services, do not compete unfairly with others, and avoid injuring others, their property, reputation, or employment.
11. We reject bribery and coercion in all their forms.
12. We accept responsibility for our actions; are open to and acknowledge criticism of our work; offer honest criticism of the work of others; properly credit the contributions of others; and do not accept credit for work not our own.

Technical Tours

WEDNESDAY, MAY 1

PSA 2019 Technical Tour: Plant Vogtle Simulator and Units 3 and 4 Tour

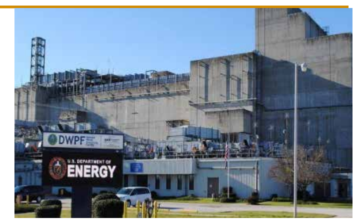
Location: Bus departs from the Marriott Charleston **Time:** 9:30am-9:00pm

A tour is planned of the Vogtle Electric Generating Plant's (VEGP's) Units 3 and 4 new power reactors in Burke County, Georgia on the Wednesday (May 1, 2019) of the PSA 2019 Conference. The tour participants will visit the Units 3 and 4 simulator and the Plant Vogtle Energy Education Center, and will be taken on a driving tour of the construction site. The AP1000 units are the first new nuclear plant construction in the U.S. in 30 years.

The tour will allow participants to view the construction progress being made firsthand and see the modernized control room via the training simulator used to train operators for the two new reactors, which should provide a better understanding of the enhanced safety features that are included in the reactor design.

Driver's license (U.S. participants) and passport (international participants) information is required.
The one-way drive time to the Plant Vogtle is estimated to be two hours and forty minutes (130 miles). Bus departure from the Marriott Charleston is at 9:30am, with the tour scheduled to conclude by 4:00pm. Return to the hotel is estimated to be no later than 9:00pm. The tour fee includes bus transportation, a box lunch, and will stop on the return trip for an on-your-own dinner at Taylor Barbeque restaurant, in Waynesboro, Georgia. The Plant Vogtle Tour is limited to 40 participants.

PSA 2019 Technical Tour: Savannah River Site Liquid Waste Facilities Tour



Location: Bus departs from the Marriott Charleston **Time:** 9:30am-9:00pm

The Savannah River Site (SRS) is a key U.S. Department of Energy industrial complex responsible for environmental stewardship, environmental cleanup, waste management and disposition of nuclear materials. On this tour of the liquid waste portion of SRS, you'll see major liquid waste dispositioning facilities that are used to safely store, process, vitrify, and maintain highlevel waste. Included in the tour is the nation's only operating vitrification facility, the Defense Waste Processing Facility (DWPF) that has been the steady and reliable workhorse of liquid waste operations at SRS for over twenty years.

Bus departure from the Marriott Charleston is at 9:00am, with the tour scheduled to conclude by 4:00pm. Return to the hotel is scheduled to be no later than 9:00pm. The tour fee includes a box lunch, and will stop on the return trip for an on-your-own dinner at Miller's Bread Basket, Blackville, South Carolina. Dress in comfortable clothing, with closed-toe shoes

Charleston Tours

TUESDAY, APRIL 30

Charleston Harbor Dinner Cruise

Location: Charleston Harbor **Time:** 6:45-9:30 pm

Join other PSA 2019 Conference colleagues and guests for a relaxing dinner cruise around the Charleston Harbor on Tuesday evening, April 30th. Sites that we will see during the cruise are Castle Pinckney, Charleston Battery, the Aircraft Carrier USS Yorktown, the iconic Arthur Ravenel Bridge over the Cooper River, and the historical landmark, Fort Sumter (which received the first shots of the U.S. Civil War in April 14, 1861). Enjoy the sites while sipping a cool drink and enjoying a menu of South Carolina BBQ and other fixins', prepared especially for the event.

This unique dinner cruise is on the Spirit Line Cruise Ship "Lowcountry". Bus departure from the Marriott Charleston is scheduled for 6:45 pm. Dress is casual, but attendees are recommended to wear comfortable footwear and bring a sweater or light jacket. Bus departure from the Marriott Charleston is scheduled for 6:45 pm. Cruise is from 7:30 pm to 9:30 pm.

Charleston Area Tours and Recreational Opportunities

With technical sessions not scheduled for Wednesday afternoon, consider the recreational and cultural opportunities available. Individual and group tour possibilities are numerous in the Charleston and surrounding area. See the Marriott Charleston concierge for more information, and the schedule for the free shuttle bus to the Charleston Battery and other city districts. Conference attendees may also select one of the pre-arranged tours.

- H.L. Hunley Civil War Submarine
- Charleston Walking Tour

Workshops

SUNDAY, APRIL 28

Dynamic Probabilistic Risk Assessment Methodologies Workshop

Workshop Coordinator: Professor Tunc Aldemir *(The Ohio State University)*
Location: Yellow Topaz & Blue Topaz **Time:** 9:00 am-4:30 pm

A new generation of methodologies is starting to receive attention for nuclear reactor probabilistic risk assessment (PRA). Often referred to as dynamic PRA (DPRA) methodologies, these methodologies explicitly account for the time element in the probabilistic system evolution and heavily incorporate plant analysis tools (e.g., RELAP, MELCOR, MAAP5) to model possible dependencies among failure events that may arise from hardware/software/firmware/process/ human interactions. DPRA methodologies are also capable of quantifying the effects of phenomenological variability and model uncertainties on the consequences of upset conditions. They can be particularly useful for the PRA modeling of passive safety systems, including representation of aging effects. As shown in the attached table, four plant level applicable DPRA tools will be demonstrated in the workshop:

Time	DPRA Tool	Presenter(s) Institution
9:00 – 10:30 am	**ADS**	Diaconeasa Mihai/ Ali Mosleh (*University of California at Los Angeles)*
10:30 am – 12:00 pm	**ADAPT**	Zachary Jankovsky/ Troy Haskin (*Sandia National Laboratories)*
12:00 – 1:00 pm	**Lunch**	
1:00 – 2:30 pm	**RAVEN**	Diego Mandelli/ Andrea Alfonsi (*Idaho National Laboratory)*
2:30 – 3:00 pm	**Break**	
3:00 – 4:30 pm	**PyCATSHOO**	Valentin Rychkov (*Electricité de France)*

The Dynamic Probabilistic Risk Assessment (DPRA) Methodologies Workshop is scheduled from 9:00pm to 4:30pm on Sunday, April 28, 2019 with a one-hour break for lunch. The DPRA Methodologies Workshop fee includes handouts and Sunday lunch. Registration for PSA 2019 is required to participate in this Workshop.

SAPHIRE Computer Code Tutorial Workshop

Workshop Coordinator: James K. Knudsen *(Idaho National Laboratory)*
Location: Opal One **Time:** 1:00-5:00 pm

The SAPHIRE tutorial will consist of three parts:

1. Introduction to SAPHIRE and its primary capabilities: SAPHIRE's primary capabilities are the development of logic models that will be solved to obtain minimal cut sets. Here we will discuss the creation of an event tree and fault tree logic model. The basic event parameters will be input (failure rates and failure probabilities). These logic models will be solved to obtain their minimal cut sets and then discuss the different risk metrics (i.e., importance measures and parameter uncertainty).
2. Advanced features of SAPHIRE used for more detailed analysis: This part will look at the use of top event substitution (link rules and graphically). Postprocessing rules will also be discussed on how to manipulate the minimal cut sets generated by the logic models. Lastly, some of the advanced basic event options will be presented (i.e., common cause failure calculators, human reliability analysis, and convolution correction factor). Time permitting, end state analysis and different quantification options available will be discussed.
3. Open discussion on use of SAPHIRE for personal applications. The SAPHIRE Tutorial Workshop fee includes tutorial handouts and Sunday lunch. The lunch is available thirty minutes prior to the start of the Workshop. Registration for PSA 2019 is required to participate in this Workshop.

FRIDAY, MAY 3 & SATURDAY MAY 4

MACCS Computer Code Training Workshop

Workshop Facilitator: Dr. Nathan E. Bixler *(Sandia National Laboratories)*
Location: Topaz **Time:** Friday 1:00 pm – 5:00 pm and Saturday 8:00 am – 12:00 pm

The MACCS computer code is an atmospheric transport and dispersion software tool used to support Level 3 PRA and other types of radiological consequence analyses. The MACCS Training Workshop provided here as part of the PSA 2019 Conference is comprised of two half-days. The training is intended to have a hands-on component. Current MACCS users are recommended to bring a laptop with WinMACCS 3.11.2 installed so they can participate in the exercises; prospective MACCS users are also invited to attend the workshop to learn more about the MACCS models and capabilities.

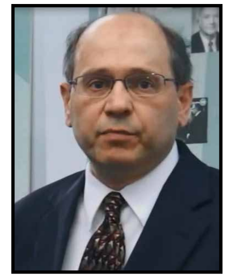
The first portion of the training, on Friday afternoon following the final PSA sessions, is fundamental in nature and is intended to describe many of the MACCS input parameters and provide guidance in choosing values for those input parameters. This material covers all three MACCS modules, ATMOS, EARLY, and CHRONC. The format of this training is lecture with time for questions and answers plus one or two hands-on exercises employing MACCS.

The second half of the training, on Saturday morning, is focused on a more advanced topic, how to use the relatively new features in MACCS to perform the consequence analysis portion of a multi-unit, Level 3 PRA. This portion of the training includes a discussion of strategies for keeping the consequence analyses to a practicable number when more than two units (or combinations of units, spent fuel pools, and other sources) are treated in the PRA. It also includes hands-on exercises to make the concepts practical.

The MACCS Workshop fee includes training handouts, Friday lunch and Saturday breakfast. Each meal will be available thirty minutes prior to the start of each session. Registration for PSA 2019 is required to participate in the Workshop.

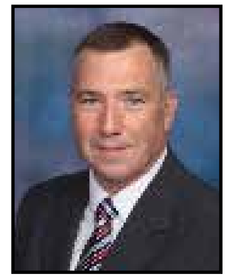
Speaker Biographies

Monday, April 29, 2019 Opening Plenary: Safe Enough? WASH-1400 and Its Legacy
Dr. Thomas Wellock, U.S. Nuclear Regulatory Commission Historian



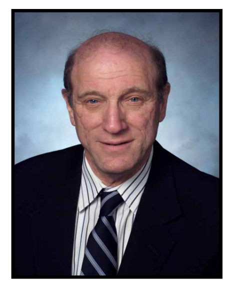
Dr. Thomas Wellock is the historian at the U.S. Nuclear Regulatory Commission (NRC). He is currently at work on a history of the use of probabilistic risk assessment in the regulation of nuclear power, and recently published a article on the topic, "A Figure of Merit: Quantifying the Probability of a Nuclear Reactor Accident." Prior to coming to the NRC, he was a professor of U.S. history at Central Washington University. He has published two books, Critical Masses: Opposition to Nuclear Power in California, 1958-1978 and Conserving the Nation: The Conservation and Environmental Movements, 1870-2000. He has also published numerous articles on the history of nuclear power and environmentalism. In 1995, he earned his Ph.D. in history from the University of California, Berkeley.

Monday, April 29, 2019 Luncheon Presentation (Sponsored by Westinghouse)
Daniel Churchman, Fleet Engineering Director, Southern Nuclear Operating Company



Our Monday Luncheon speaker is Daniel L. (Dan) Churchman. Mr. Churchman is the Fleet Engineering Director for Southern Nuclear Operating Company. Dan joined SNC in 2014 as the Nuclear Fuel & Analysis Director and assumed his current position in 2016. Prior to joining SNC, Dan held various leadership positions with Westinghouse Electric Company, TetraTech, and Black & Veatch. He began his nuclear career serving in the Navy Nuclear Program for 8 years followed by an opportunity to join Entergy as a Shift Engineer at Arkansas Nuclear One. While at Entergy, Dan held various positions in Operations, Engineering, and Project Management. He received an SRO license on ANO Unit 2 and attended the Senior Nuclear Plant Manager Course. Dan has a Bachelors Degree in Mechanical Engineering and an MBA from the University of Oklahoma. He was also certified as a Naval Nuclear Engineer and has a Project Management Certification (PMP).

Tuesday, April 30, 2019 Plenary—II: PSA methodologies for external hazards at nuclear power plants: Current status and future developments
Dr. Robert J. Budnitz, Lawrence Berkeley National Laboratory (retired)



Dr. Robert J. Budnitz has been involved with nuclear-reactor safety and radioactive-waste safety for many years. He is a member of the National Academy of Engineering. He recently retired (spring 2017) from the scientific staff at the University of California's Lawrence Berkeley National Laboratory. From 2002 to 2007 he was at UC's Lawrence Livermore National Laboratory, during which period he worked on a two-year special assignment (late 2002 to late 2004) in Washington to assist the Director of DOE's Yucca Mountain Project to develop a new Science & Technology Program.

Prior to joining LLNL in 2002, he ran a one-person consulting practice in Berkeley CA for over two decades. In 1978-1980, he was a senior officer at the U.S. Nuclear Regulatory Commission, serving as Deputy Director and then Director of the NRC Office of Nuclear Regulatory Research. In this two-year period, Dr. Budnitz was responsible for formulating and guiding the large NRC research program, that constituted over $200 million/year at that time. His responsibilities included assuring that all major areas of reactor-safety research, waste-management research, and fuel-cycle-safety research necessary to serve the mission of NRC were adequately supported. He earned a Ph.D. in experimental physics from Harvard in 1968.

Speaker Biographies

Wednesday, May 1, 2019 International Perspective of Ongoing and Future PSA Activities at the IAEA and its Member States
Ms. Cornelia Spitzer, Section Head, Safety Assessment Section, Division of Nuclear Installation Safety Department of Nuclear Safety and Security, International Atomic Energy Agency

Ms. Cornelia Spitzer is a senior key expert with more than 35 years of leadership, supervision and management experience in nuclear regulatory procedures holding various management positions. Her technical expertise and experience cover the deterministic area as well as the establishment, further development and expansion of the subject areas reliability analysis and risk management, safety assessment and systems analysis, Probabilistic Safety Assessment up to the field of Integrated Risk-Informed Decision-Making. In the context of reviews of comprehensive PSAs, she also performed independent analyses for a multitude of issues particularly in the domain Human-Machine-Interaction as well as guidance for operating procedures design.

Ms. Spitzer joined the IAEA in 2015 to assume the position Head of the Safety Assessment Section in the Division of Nuclear Installation Safety. The Section's programmatic tasks under her responsibility are to support Member States in achieving a high level of safety in nuclear power plant design and excellence in safety assessment through the development and provision of up-to-date safety assessment and design safety standards based on current technology and best practices. Six IAEA safety guides associated with the revised safety requirements under the Section's responsibility were recently approved for publication and complemented by the development of more detailed technical documentation on practical examples to facilitate the understanding of the revised IAEA safety standards, e. g., related to the updated design safety principles.

Before joining the IAEA, Ms. Spitzer had also been working in the international area (e.g. with regulatory bodies and research institutes) and organized e.g. the 2004 edition of the International PSAM-ESREL Conference. She also coordinated and participated in the Safety Reviews conducted after the accident in Fukushima as well as related to the European Stress Tests (in the international area too, e.g. in South Korea).

Under Ms. Spitzer's responsibility, particular attention is given to the review of current practices in topical issues on nuclear installation safety and to the development of documents addressing emerging topics, such as small and medium-sized or modular reactors (SMRs), risk aggregation of various risk contributors and multiunit PSA considerations. One of the main achievements for 2017 was the International Conference on Topical Issues in Nuclear Installation Safety: Safety Demonstration of Advanced Water Cooled Nuclear Power Plants (6 to 9 June 2017) at the IAEA headquarters in Vienna, Austria.

Ms. Spitzer has been active in working groups and committees on national and international level related to the preparation of regulatory guidance. Her experiences include large project´s coordination and team management as well as various professional memberships and international liaisons; she is author of numerous publications as well.

Ms. Cornelia Spitzer holds a University Degree (Diploma) in Mathematics, Minor Subject: Physics from the University of Heidelberg, Germany.

Thursday, May 2, 2019 PSA 2019 Banquet Key Note Speaker:
ANS President Dr. John. E. Kelly

Dr. John E. Kelly recently retired from the U.S. Department of Energy where he was the Chief Technology Officer in the Office of Nuclear Energy. He was responsible for establishing the strategic technical direction for the Office of Nuclear Energy's (NE's) research, development, demonstration, and deployment portfolios.

Prior to assuming the duties of Chief Technology Officer, he served as Deputy Assistant Secretary for Nuclear Reactor Technologies. His office was responsible for the civilian nuclear reactor research and development portfolio, which included programs on Small Modular Reactors, Light Water Reactors, and Generation IV reactors. Additional responsibilities included the design, development, and production of radioisotope power systems, principally for NASA missions.

In the international arena, Dr. Kelly chaired the Generation IV International Forumand former chair of the International Atomic Energy Agency's Standing Advisory Group on Nuclear Energy. Prior to joining the Department of Energy in 2010, Dr. Kelly spent 30 years at Sandia National Laboratories, where he was engaged in a broad spectrum of research programs in nuclear reactor safety, advanced nuclear energy technology, and national security. In the reactor safety field, he led efforts to establish the scientific basis for assessing the risks of nuclear power plant operation and specifically those risks associated with potential severe accident scenarios. His research focused on core melt progression phenomena, which led to an improved understanding of the Three Mile Island accident and, more recently, the Fukushima Dai-Ichi accident.

In the advanced nuclear energy technology field, he led efforts to develop advanced concepts for space nuclear power, Generation IV reactors, and proliferation-resistant and safe fuel cycles. These research activities explored new technologies aimed at improving the safety and affordability of nuclear power. In the national security field, he led national efforts to evaluate the safety and technical viability of tritium production technologies.

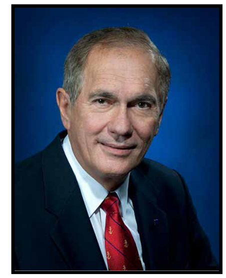
Dr. Kelly received his B.S. in nuclear engineering from the University of Michigan-Ann Arbor and his Ph.D. in nuclear engineering from the Massachusetts Institute of Technology.

Biography and Photo Courtesy of ans.org/about/presidents/jkelly/

Speaker Biographies

Thursday, May 2, 2019 Plenary—V: Perspectives on Nuclear Safety Since the Three Mile Island Event: Learning from the Past 40 Years Panel:

Dr. Robert A. Bari, Chair, Senior Physicist Emeritus, BNL Honorary Chair for the PSA 2019 Meeting



Dr. Robert A. Bari has been recognized for many contributions to nuclear power reactor safety, security, and nonproliferation during more than 40 years of nuclear energy research.

After earning a bachelor's degree from Rutgers University and a Ph.D. from Brandeis University, both in physics, Bari joined the MIT Lincoln Laboratory for a two-year assignment in the solid-state theory group. He then joined Brookhaven Lab as an assistant physicist in the Physics Department in 1971. He was a visiting assistant professor of physics at Stony Brook University during the 1973–74 academic year and he returned to Brookhaven in 1974 as a physicist in the Department of Applied Science. Bari was named associate chair and senior physicist of the Department of Nuclear Energy in 1982 and was awarded tenure in 1984. He became deputy department chair in 1988 and then chair of the Nuclear Energy Department in 1995. Bari served as interim associate laboratory director for applied programs at the Lab for several months in 1999 before returning to his role as department chair. He then returned to his role as a senior physicist in 2000, and in 2004, was presented with a Science & Technology Award—among the highest accolades given by Brookhaven to its employees for distinguished contributions to the Lab's mission. In 2003, the American Nuclear Society presented Bari with Theo J. ("Tommy") Thompson Award for his research contributions in nuclear safety. He retired from the Nuclear Science & Technology Department in 2017.

Bari served for 15 years as co-chairman of the working group on proliferation resistance and physical protection of the Generation IV International Forum, which carries out research and development to establish feasibility and performance capabilities of a new generation of nuclear energy systems. He has served on the board of directors of the American Nuclear Society and as president of the International Association for Probabilistic Safety Assessment and Management. He is a fellow of the American Nuclear Society and American Physical Society, and has been a member of the National Academy of Sciences committee for lessons learned from the Fukushima Nuclear Accident for Improving Safety and Security of the U.S. Nuclear Plants.

Dr. Robert E. Henry, Fauske & Associates, Inc., Emeritus Senior Vice President

Dr. Robert E. Henry Prior to retirement, Dr. Henry was a Senior Vice President and co-founder of FAI. In this position, he was responsible for developing the understanding of PWR and BWR reactors during severe accident conditions. This knowledge base has been integrated into a large system code called MAAP (Modular Accident Analysis Program). MAAP has gained widespread acceptance in the domestic and foreign nuclear industry.

He was a member of the EPRI (industry) team that assessed the TMI-2 accident and also a member of the U.S. delegation to IAEA /Vienna to evaluate the Russian interpretation of the Chernobyl-4 accident. He has served on NRC review panels evaluating ongoing research and was chosen to author the initial EPRI Technical Basis Report for developing Severe Accident Management Guidelines for all U.S. LWR types.

Dr. Henry has published more than 150 articles, authored six U.S. patents and written two books for the ANS (the TMI-2 accident and steam explosions). As is a member of the ANS, he received the Tommy Thompson Award in the field of reactor safety in 1985. He also received an Award for Outstanding Engineering Accomplishments from his alma mater in 1990 and in 2015 he was elected to the National Academy of Engineering.

Dr. Roger Mattson, Consultant

Dr. Roger J. Mattson has more than fifty years of diversified experience in design, regulation and management of nuclear facilities. He began his career at Sandia Laboratory, then transferred to the Atomic Energy Commission, the Nuclear Regulatory Commission and the Environmental Protection Agency. His technical experience is in thermal hydraulics, nuclear facility licensing, risk management, and security of nuclear facilities and materials. He conducted and managed safety reviews for more than 110 nuclear power plants and other radiological facilities. He led the development of NRC's new requirements after the accident at Three Mile Island in 1979 (NUREG-0578, 0585 and 0660). He oversaw early applications of probabilistic risk assessment to regulation of nuclear power plants. After government service, he served as president of International Energy Associates Limited, a nuclear consultancy, and was a founder and chief operating officer of SCIENTECH, Inc., specializing in nuclear safety. Since 2002, he has consulted with a range of clients in the private sector, the DOE weapons complex and the National Labs.

Among his consulting assignments were readiness reviews for startup of Limerick Nuclear Power Plant Unit 2, Rocky Flats Plant, and K-Reactor at Savannah River. He was an expert witness in more than forty legal proceedings involving nuclear facilities. He co-chaired the International Atomic Energy Agency's development of safety principles for nuclear power plants after the 1986 accident at Chernobyl (INSAG-3, updated to INSAG-12). He served on nuclear safety review boards for eight operating nuclear facilities. He aided decommissioning of three nuclear power plants. He led a team to address the risks of hydrogen in process systems at the Hanford Waste Treatment Plant. He served on a team responding to lessons learned from Fukushima Dai-ichi for the American Society of Mechanical Engineers. He is the author of Stealing the Atom Bomb: How Denial and Deception Armed Israel. Dr. Mattson has a PhD in Mechanical Engineering from the University of Michigan.

Plenary, Special Sessions & Events

SUNDAY, APRIL 28

PSA 2019 Arrival Meet & Greet

Location: Courtyard (Weather Permitting) or Emerald **Time:** 6:00-9:00 pm

MONDAY, APRIL 29

Opening Plenary

Location: Emerald **Time:** 8:00-10:00 am

8:00 am Welcome Greeting from Kevin R. O'Kula *(AECOM Technical Services)*, General Chair
8:10 am Welcome from the Mayor Pro Tem of the City of Charleston, Mr. Peter Shahid
8:15 am Welcome from Professor Mohammad Modarres *(Univ of Maryland)*, Technical Program Chair
8:25 am Keynote Presentation, Safe Enough? WASH-1400 and Its Legacy, Mr. Thomas Wellock *(NRC Historian)*
9:15 am Presentation from Professor Mohammad Modarres *(Univ of Maryland)* to WASH-1400 Authors (Dr. Richard S. Denning, Dr. Joseph A. Murphy, and Dr. Ian B. Wall)
9:30 am Remarks from WASH-1400 Authors: Richard Denning *(Consultant)*, Joseph Murphy *(Consultant)*, and Ian Wall *(Consultant)*

Opening Plenary Panel: PRA Knowledge Management: Preserving Data and Information

Chair: Mohammed Modarres *(Univ of Maryland)*
Location: Emerald **Time:** 10:30 am-12:15 pm

There is a vast amount of nuclear power plant related Probabilistic Risk Assessment (PRA) documents that have been developed by regulatory bodies, plant owners, nuclear industry organization and plant owners in the past 50 years. These documents report PRA methods, policy, data, standards, guidelines, risk-informed analyses , etc. To preserve the PRA-related documents and make the information and data contained in them readily accessible, there is an urgent need to gather, digitize and make them searchable. There are many modern computer-based methods for knowledge management and advanced search algorithms that make this objective possible. This panel brings PRA experts to discuss the need for possible initiatives, planned and existing initiatives, scope and difficulties associated with such an effort.

Panelists: George Apostolakis *(Nuclear Risk Research Center, Japan)*
Robert Budnitz *(LBNL - retired)*
Michael Cheok *(NRC)*
Ali Mosleh *(UCLA)*
Marina Röwekamp *(GRS)*

Luncheon Presentation

Location: Emerald **Time:** 12:15-1:45 pm

Mr. Churchman will discuss the status of the Plant Vogtle Units 3 and 4 construction outside of Waynesboro, in eastern Georgia. The AP1000 units are the first new nuclear plant construction in the U.S. in 30 years

Speaker: Daniel Churchman *(Fleet Engineering Director, Southern Nuclear Operating Company)*

Sponsored by Westinghouse

PSA 2019 Opening Reception

Location: South Carolina Aquarium **Time:** 6:00-9:00 pm

Hosted by JENSEN HUGHES
Advancing the Science of Safety

TUESDAY, APRIL 30

Plenary—II: PSA Methodologies for External Hazards at Nuclear Plants: Current and Future Developments

Chair: Mohammad Modarres *(Univ of Maryland)* **Location:** Emerald **Time:** 8:00-9:30 am

In the last few years, the use of PSA methods for studying external-hazard risks at nuclear power plants has finally become widespread worldwide, including PSA studies for hazards other than earthquakes, PSAs during shutdown conditions, Level 2 and Level 3 PSAs, and PSAs during the design phase. Because the bottom-line external-hazard-PSA risk numbers often have significant uncertainties due to uncertain knowledge of the frequencies of the hazards themselves, these PSA studies have sometimes developed an unfair reputation as being less useful than they really are. This talk will explore both the current status and the future prospects (which are bright) for significant further advances in the methodologies. The emphasis will be on insights derived, their application in decision-making, and reductions in uncertainties through use by so many different practitioners.

Speaker: Robert J. Budnitz *(Lawrence Berkeley National Laboratory (retired))*

Plenary, Special Sessions & Events

WEDNESDAY, MAY 1

Plenary—III: International Perspective of Ongoing and Future PSA Activities at the IAEA and its Member States

Chair: Mohammad Modarres *(Univ of Maryland)* **Location:** Emerald **Time:** 8:00-9:30 am

The keynote will provide a brief introduction and overview of the IAEA activities in the area of safety assessment and design safety. The major focus will then be on presenting ongoing and planned PSA projects at the IAEA related to safety standards, Peer Review services, competency building and addressing emerging topics. The presentation will be completed by summarizing insights gained from Member States' PSA activities, their challenges and perspectives.

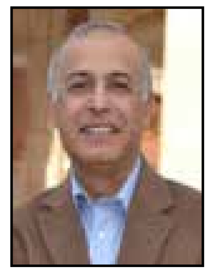
Speaker: Ms. Cornelia Spitzer *(Safety Assessment Section, Division of Nuclear Installation Safety, Department of Nuclear Safety and Security, International Atomic Energy Agency)*

THURSDAY, MAY 2

Plenary—IV Panel: PSA Research and Education at Universities: History, Impact, Challenges, and Future Outlook

Chair: Ali Mosleh *(Prof. UCLA, and B. John Garrick Institute for the Risk Sciences)* **Location:** Emerald **Time:** 8:00 am-10:00 pm

For more than four decades, academic institutions worldwide have played a key role in the development of the PSA discipline, through conducting research and offering formal educational programs and training the workforce. This panel examines the historical roots and impact of the university programs on the evolution of the field, current status, challenges, and outlook for the future. The discussion will start with a brief statement by each panelist to provide initial thoughts. These remarks will then be followed by facilitated discussion involving the panel and the audience.



Chair & Moderator:
Ali Mosleh
Prof. UCLA, and B. John Garrick Institute for the Risk Sciences

Panel Participants:

Moderator: Ali Mosleh *(Prof. UCLA, and B. John Garrick Institute for the Risk Sciences)*

Panelists: George Apostolakis *(MIT Emeritus Prof.)*
Mohammad Modarres *(Prof. UMD)*
Ali Mosleh *(Prof. UCLA)*
Akira Yamaguchi *(Prof. Univ. of Tokyo)*
Katrina Groth *(Assistant Prof. UMD)*
Tunc Aldemir *(Ohio State)*
Wolfgang Kröger *(ETH Zürich)*

Plenary—V Panel: Perspectives on Nuclear Safety Since the Three Mile Island Event: Learning from the Past 40 Years

Chair: Robert A. Bari *(BNL – Retired)* **Location:** Emerald **Time:** 10:30 am-12:30 pm

PSA 2019 marks 40 years and one month since the 1979 Three Mile Island Unit 2 (TMI 2) reactor accident, near Harrisburg, Pennsylvania. Although the partial meltdown from this accident and ensuing radioactive releases had no detectable health effects on plant workers or the public, TMI 2 was the most serious accident in U.S. commercial nuclear power plant operating history. Its aftermath brought about sweeping changes involving emergency response planning, reactor operator training, human factors engineering, radiation protection, and many other areas of nuclear power plant operations. It also caused the NRC to heighten its regulatory oversight. All of these changes significantly impacted how nuclear safety is performed, managed and regulated. The major themes of the session are:

• What occurred? • What have we learned? • What have we done? • Where do we go from here?

This panel features the perspectives from radiological protection, NRC technical coordination, and other technical areas responsible for TMI-2 accident management, recovery, and improving the safety of contemporary and future nuclear facilities. Lessons learned for the industry and the PRA community will be discussed.

Panelists: Dr. Robert A. Bari *(Chair, Senior Physicist Emeritus, BNL)*
Dr. Robert J. Budnitz *(Lawrence Berkeley National Laboratory (retired))*
Dr. Robert E. Henry *(Fauske & Associates, Inc., Emeritus Senior Vice President)*
Dr. Roger Mattson *(Consultant)*

PSA 2019 Banquet

Location: Crystal ABC **Time:** 6:30-9:00 pm

Multi-Unit PSA and Risk Integration—I

Chair: Cornelia Spitzer *(IAEA)* **Location:** Emerald Salon One **Time:** 1:45-3:25 pm

1:45 pm: Development of Multiunit PSA Model for the Case Study of the IAEA Project, Pavol Hlavac, Zoltan Kovacs *(RELKO Ltd.)*, Andrea Maioli *(Westinghouse)*, Denis Hennecke *(Ge Hitachi)*, Paul Amico *(JENSEN HUGHES)*, Paul Boneham *(Jacobsen Analytics)*, Ovidiu Coman, Shahen Poghosyan *(IAEA)*

Based on the interest from the Member States, IAEA launched the project on the Development of a Methodology for Multi-Unit Probabilistic Safety Assessment (MUPSA). The implementation of the project contains the following phases:

- *Phase 1 – develop a document providing a methodology for implementation of MUPSA.*
- *Phase 2 – develop a case study following the methodology developed in Phase 1.*
- *Phase 3 – improve the methodology based on the lessons learned from the case study developed in Phase 2 and integrate the improved methodology and the case study in a single document.*

This paper is focused on MUPSA model development within the case study in Phase 2. A simplified full power PSA model in RiskSpectrum code has been developed for a WWER440 plant under the condition that four units are in operation.

The objective of the case study is to verify the Methodology on Multi-Unit Probabilistic Safety Assessment (MUPSA) proposed with the Phase 1 of the MUPSA project. The verification of the MUPSA methodology is planned to be implemented by applying it to the realistic NPP configuration using the realistic WWER440 PSA model and by providing the feedback on the applicability of the proposed methodology for standard PSA tasks.

It is expected that the case study will provide a base for improvement and increase level of details reflected in the MUPSA methodology. Therefore, the case study should be designed in a way to touch upon and verify various aspects of the MUPSA methodology and should reflect the potential complexity of MUPSA task depending on the number of units available at typical NPP site, their type and configuration. The current state and the results of the case study will be described in the paper.

2:10 pm: Seismic Correlation Modeling in Multi-Module PRAs, Luke McSweeney *(NuScale Power, LLC)*

Correlation between seismic failures is a critical aspect of any multi-module or multi-unit seismic probabilistic risk assessment (SPRA). This paper builds on existing approaches to provide a methodology for systematically addressing dependencies between component failures, using the separation of variables fragility method. Dependent split fractions are assigned to the fragility sub-factor uncertainties of corresponding components. Random sampling is then used to produce the number of failed corresponding components in a set, which is then used to produce a multi-module seismic core damage frequency or multi-module high confidence of low probability of failure (HCLPF). This method is applicable at all ground motion levels for both SPRAs and seismic margin assessments (SMAs).

2:35 pm: Consideration of Multi-Unit Risk Aspects Within an Integrated Risk-Informed Decision-Making Framework, Fernando Ferrante *(EPRI)*, Andrea Maioli, Ken Kiper, Adriana A. Sivori, Carroll Trull *(Westinghouse)*

Multi-unit risk applications have gained significant importance in recent years across various U.S. and international stakeholders. While PRA modeling has mainly focused on single-unit risk since its implementation, the Fukushima Daiichi Accident in 2011 has raised the profile of addressing potential gaps and issues highlighted by the event (which increased regulatory scrutiny and interest in many countries). While approaches and methodologies have been previously proposed to address multi-unit risk, their feasibility and implementation using current tools within a risk-informed decision-making process are still evolving. In addition, unique technical gaps with respect to the explicit modeling of potential drivers for multi-unit scenarios can be found across multiple PRA disciplines, such as the treatment of common cause failure (CCF) for systems and components across units, impacts on human reliability analysis (HRA) during multi-unit accident scenarios, and correlated impacts due to natural hazards (e.g., seismic, flooding). These aspects are compounded when considering more general PRA technical challenges that still remain (e.g., external flooding) and need to be addressed if applicable to multi-unit risk. This work focuses on considering existing approaches, their feasibility, and remaining challenges and technical gaps from a practical perspective (and within an integrated risk-informed decision-making framework, RIDM) that takes into account:

- *A risk-informed approach to characterize the extent and scope that should be considered for multi-unit risk purposes given specific site characteristics and level of technical detail commensurate with the intended decision-making (e.g., a graded approach that accounts for the feasibility of implementation given resources available/required)*
- *Existing PRA models to highlight the basis for such an approach, including the available lessons learned from successful international activities (e.g., COG experience) without replicating efforts*
- *The type of risk outputs can be obtained and considered within a RIDM process in order to provide a robust technical justification for assessing multi-unit risk*

The resulting approach to multi-unit risk aspects presented here includes a risk-informed, graded approach based on extent, scope, and objectives of characterizing multi-unit risk profile that is flexible and implementable, such that the applicability and resources required can be evaluated upfront. In addition, it uses available risk outputs for actual decision-making once the appropriate implementation of a multi-unit risk assessment is developed and results are obtained. Actual PRA models developed for single-unit aspects are studied and modified to support the suggested approach.

3:00 pm: An Approach to Developing an Integrated Site Probabilistic Risk Assessment (PRA) Model, D. W. Hudson *(NRC)*

The U.S. Nuclear Regulatory Commission (USNRC) is performing an integrated site Level 3 probabilistic risk assessment (PRA) for a U.S. commercial nuclear power plant (NPP) site comprised of multiple co-located sources of radiological materials, including: two operating reactor units, two spent fuel pools, and an independent dry cask storage facility. The fundamental objectives of this study are to: (1) develop a contemporary Level 3 PRA generally based on current state-of-practice methods, models, data, and analytical tools that reflects technical advances made in the last three decades and that addresses risk contributors not previously considered, including concurrent accidents involving multiple co-located radiological sources; (2) extract new risk insights to enhance regulatory decision making and to help focus limited resources on issues most directly related to USNRC's mission to protect public health and safety; (3) enhance USNRC staff's internal PRA capability and expertise; (4) improve PRA documentation to make information more accessible, retrievable, and understandable; and (5) obtain insight into the technical feasibility and cost of developing new Level 3 PRAs. Although this Level 3 PRA study is generally being performed consistent with current standards and state-of-practice using existing PRA technology, there are some technical elements that necessitate methodological development due to a lack of sufficient experience to define a current state-of-practice. One such technical element is the Integrated Site PRA technical element. The objectives of this element are to: (1) estimate integrated site risk; and (2) identify and characterize significant contributors to integrated site risk.

Multi-Unit PSA and Risk Integration—I Continued

This paper proposes a definition for the term "integrated site risk" by logically decomposing it into its three constituent elements and defining what is meant by each element that comprises it. Using the traditional quantitative definition of risk and the concept of a risk triplet comprised of an accident scenario, its likelihood in terms of frequency or probability of frequency, and its conditional consequences, integrated site risk is represented as the set of single-source and multi-source risk triplets that encompasses a reasonably complete spectrum of possible single-source and multi-source accident scenarios that can occur at a modelled NPP site. The paper also describes the USNRC's proposed technical approach for developing an integrated site PRA model that concentrates on identifying and prioritizing potential multi-source accident scenarios to make informed approximations that could provide new and useful risk insights. This focused approach utilizes risk insights from single-source PRA models coupled with a systematic search for potential inter-source dependencies and potentially important multi-source accident scenarios that might be missed by relying primarily upon results and insights from single-source PRA models. This three-pronged strategy is intended to provide reasonable assurance that the integrated site PRA model captures important contributors to integrated site risk. The paper summarizes each of the five steps that comprise the proposed technical approach and shares insights and lessons learned from limited-scope pilot applications of the proposed approach. These limited-scope pilot applications were performed to: (1) evaluate the technical feasibility of implementing the proposed approach using existing analytical tools; (2) identify potential barriers to implementation; and (3) identify opportunities for improvement.

Internal Events—I

Chair: Jeff Mitman *(NRC)* **Location:** Emerald Salon Two **Time:** 1:45-3:15 pm

1:45 pm: Simplified Structural Steel Analysis to Support Assumption of Loss of One Column for Building Structural Integrity, Robert J. Wolfgang *(JENSEN HUGHES)*

The analysis of Fire PRAs involves the evaluation of various fire scenarios that can involve whether or not radiant heat flux from a fire scenario impacting exposed structural steel is responsible for building collapse. In analyzing these types of scenarios, a general assumption has been employed that assumes that the loss of one structural column within the interior of a building is not sufficient to result in the loss of building structural integrity and subsequent building collapse. However, because no quantitative analysis has previously been done to substantiate this assumption based on engineering judgment, many peer reviewers have rejected this type of general assumption. This paper provides an example where a typical turbine building is analyzed for a fire scenario in the vicinity of a main feedwater pump, which is a typical fire scenario that arises in most fire PRAs. Because the design loading for structural steel columns within turbine buildings is based on building codes that take into account multiple loads that consider dead loads, seismic loads, wind loads, etc. it is reasonable to postulate that a fire scenario that disables the structural capacity for only one steel column is not sufficient to result in a total loss of building integrity and subsequent building collapse. A simplistic, yet conservative, analysis has shown that the redistribution of building loads given the loss of one column is not sufficient to result in building collapse. It is anticipated that the methodology involved in this type of analysis can also be replicated by analysts in analyzing their own specific structures based on plant specific drawings and design documents. As such, the use of this simplistic analysis will provide the necessary quantitative analysis to support the general assumption that the loss of a single interior structural column within a building or structure will not result in a total loss of structural integrity and subsequent building collapse. Because of the nature of how most plant structures and buildings are constructed, this general assumption is not considered applicable to peripheral columns of a building, such as a turbine building, since peripheral columns are more critical to the structural integrity of a building and would lead to redistribution of loads that may compromise the structural integrity of a large portion of the structure, which could possibly lead to collapse of a major portion of the building.

2:15 pm: An Approach for Apportioning Fire Scenario Frequencies to Induced Initiating Events, Paige Elizabeth Risley *(Westinghouse/Univ of Pittsburgh)*, Clarence Worrell, Kyle Christiansen *(Westinghouse)*

The apportioning of fire scenario frequency to induced initiating event(s) can be a significant source of conservatism in fire probabilistic safety assessments (PSAs). One meaningful approach would be using a structure similar to a seismic initiating event tree, where earthquake occurrence frequency is apportioned based on the fragilities of components whose failure could induce each initiator. This however is difficult in fire PSA due to significant uncertainty in the modeling of fire dynamics and the response of target cables exposed to the fire environment. It is also difficult to rank initiators by severity (in terms of conditional core damage probability for example), given that risk contribution is a function of mitigating equipment failures, which vary greatly by scenario. Given this uncertainty, many fire PSAs map the entire scenario frequency to all potential induced initiators, which results in significantly over-counting the frequency, by a factor of the number of induced initiators. This paper explores an automated approach for initiator selection and frequency apportioning that resolves the identified challenges. The approach is applied to two fire PSAs and the results presented. Initial results from the two pilot studies suggest a 10-50% total fire core damage frequency reduction using the proposed process.

2:45 pm: Characterization of Interruptible and Growth Fires for Nuclear Power Plant Applications, Victor Ontiveros, Margarita Chi-Miranda, Sara Montanez, Francisco Joglar *(JENSEN HUGHES)*, Ashley Lindeman *(EPRI)*

Experience with fire events at NPPs, as captured in the Electric Power Research Institute (EPRI) fire events database (FEDB), indicates that a majority of electrical cabinet fires are extinguished prior to developing into a challenging state. A significant fraction, in excess of 90% of fires that ignite within electrical cabinets are classified as potentially challenging. These are fires that do not reach a challenging state – in other words, the fire was not fully involved, did not impact surrounding equipment, or did not damage cable trays or conduit nearby. Following the current approach described in NUREG/CR-6850 all fires, regardless of fire severity classification (potentially challenging, challenging, and undetermined), are modeled the same way, capable of significant growth (growth to peak in 12-minutes) and causing damage to nearby equipment and cables. The insights from a review of the FEDB data suggests a significant fraction of fires grow in a manner that allows for plant personnel to respond. To capture this experience, events are classified into two growth profile groups, Interruptible Fire and Growth Fire. The Interruptible Fire characterization will be used to classify fire events that grow and progress in a manner that is not at an accelerated rate such that plant personnel are able to discover and suppress prior to the fire becoming a fully involved fire or causing damage to targets outside the ignition source. The Growth Fire characterization will be used to classify fire events that exhibit a rapidly developing and growing fire for which there is a chance responding plant personnel will not be able to discover and suppress the fire prior to becoming a fully involved fire or causing damage to targets other than the ignition source. The Interruptible Fire and Growth characterization is based on the available recorded fire event evidence as included in the FEDB. Subsequent to the review, a procedure and rule set were developed to allow for consistent classification of fire events into two different growth profiles. The current scope is limited to electrical cabinet sources (primarily Bin 15 – electrical cabinets) with fire events occurring between 2000 and 2014. This paper will describe the characterization of the proposed Interruptible Fire and Growth groups, the criteria developed to classify fire events as either an interruptible or growth fire, a split fraction for interruptible and growth fires, and non-suppression probability (NSP) values for interruptible fires, growth fires for use in the NUREG/CR-6850 Appendix P NSP event tree, and revise the HRR profiles using available nuclear power plant electrical cabinet experimental data.

Working Group and International Program Insights

Chair: Dennis Henneke *(GE-Hitachi)* **Location:** Emerald Salon Three **Time:** 1:45-3:10 pm

1:45 pm: Summary VVER Regulators' Forum PSA Working Group 4th Mandate, Gurgen Kanetsyan, Armen Amirjanyan *(NRSC)*, Iva Nikolova (BNRA), Petr Adamec *(SÚJB)*, Ari Julin *(STUK)*, Michael Hage *(GRS)*, András Gábor Siklósi *(HAEA)*, Ajai S Pisharady *(AERB)*, Majid Alinejad *(INRA)*, Mikhail Ivochkin *(SEC NRS)*, Jozef Rybar *(UJD)*, Oleg Zhabin *(SSTC NRS)*

The Forum of the State Nuclear Safety Authorities of the Countries Operating VVER Type Reactors (the Forum) was established in 1993. The objective of the Forum is to foster enhancement of the nuclear safety and radiation protection in the interested countries through utilization of the collective experience, information exchange and consolidation of efforts of the national nuclear safety authorities to study safety problems and improve regulatory policies and practices. The Forum acts under the Memorandum signed by the Member Countries in 1998 in Armenia. Under the Memorandum, the Forum establishes working groups at its meetings to discuss issues selected by the Forum. The members of the 4th mandate of PSA Working Group were the nuclear regulatory authorities from Armenia, Bulgaria, Czech Republic, Finland, Hungary, India, Iran, Russia, Slovak Republic and Ukraine. In addition, GRS (Gesellschaft für Anlagen- und Reaktorsicherheit, Germany) participated in the co-operation as an observer.

The aim of the Working Group was to provide useful international experience and references for further improvement in the member countries' legal and regulatory framework as well as to find and compare good regulatory practices in the field of PSA. Based on the objectives of the fourth mandate, the work was divided into the following seven tasks:

1. *External hazards PSA approaches and regulations in order to identify good practices among the member countries;*
2. *Changes and current practices in the legal and regulatory framework based on the lessons learned from the Fukushima accident;*
3. *Risk-informed regulatory decision-making practices in the member countries;*
4. *Risk-informed inspection practices in the member countries;*
5. *Annual evaluation of outage (shut down) risks;*
6. *Reliability-centered maintenance and maintenance effectiveness monitoring practices in the member countries;*
7. *Updates of the Risk-Informed Regulation Indicator System (RIRIS)*

The paper presents the non-confidential information, conclusions and recommendations gathered and concluded during the 4th mandate of VVER Forum PSA Working Group on each determined tasks.

2:15 pm: BWROG Insights Based on PRA Peer Review F&O Closure Workshops, Jonathan Li, Glen Seeman, Steve Alexander *(GE Hitachi)*

GE Hitachi Nuclear Energy (GEH) has led multiple PRA peer review Facts and Observations (F&Os) closure workshops for the Boiling Water Reactor Owner's Group (BWROG) Integrated Risk Informed Regulation (IRIR) committee members. The workshops were conducted following the NEI guidance on peer review F&O closure, which was a consistent industry-wide F&O close-out process. The BWROG F&O closure workshops have been considered a good practice, which has leveraged the available resources from BWROG IRIR members by scheduling the workshops along with the regular IRIR meetings with commitments from all members.

The original BWROG F&O closure workshops also served as pilot applications, which established the BWROG processes to prepare and conduct the workshops, as well as the templates for the F&O closure workshop inputs and outputs.

Best practices for both host utility and independent assessment teams have been summarized. In addition, lessons learned have also been summarized, which have been applied to the F&O closure workshops after the initial pilot applications.

2:45 pm: SAMG Implementation Lessons Learned, N. Reed LaBarge, Kevin Honath *(Westinghouse)*

The purpose of this paper is to present the structure, scope and capability of the Pressurized Water Reactor Owners Group (PWROG) Severe Accident Management Guidelines (SAMGs) for United States (U.S.) Pressurized Water Reactors (PWRs) that were published in February 2016. The PWROG has upgrades the SAMGs for all existing U.S. PWRs in response to the accident that took place at the Fukushima Daiichi Nuclear Power Station and the U.S. PWR industry has been in the process of implementing the new SAMGs since mid-2016. This paper will highlight the features of the PWROG SAMG and will present lessons learned from implementation of the PWROG SAMG at several U.S. Nuclear Facilities.

The first portion of this paper will give an overview of the 2016 PWROG SAMGs with respect to the enhancements that were made in response to post-Fukushima lessons learned. This includes: improvements to human factors, integration of new post-Fukushima equipment and procedure sets, instrumentation guidance, integrated decision maker guidance as well as several other notable improvements. The second portion of this paper will outline the SAMG implementation process for a site and cover lessons learned from writing the plant-specific SAMGs, conducting and coordinating SAMG validations (including integration of severe accident analysis) and implementing SAMG training.

Risk-Informed Decision-Making—I

Chair: Michelle (Shelby) Bensi *(Univ of Maryland)* **Location:** Yellow Topaz **Time:** 1:45-3:15 pm

1:45 pm: Quantitative Risk Analysis Support to Decision-Making for New Systems, R. Youngblood *(INL)*, H. Dezfuli *(NASA)*

Nowadays, it is widely accepted that scenario-based probabilistic risk assessment (PRA) is needed to support key decisions about new flight systems. However, taking a risk estimate at face value and comparing it with a management "threshold" to determine whether a system is safe enough is too simplistic. This paper discusses considerations that need to accompany PRA results in particular decision contexts. The present point is not to criticize PRA itself; others have eloquently stressed the point that whatever PRA's limitations, doing PRA is better than not doing it, for complex, high-stakes systems. But even a high-"quality" PRA does not confer omniscience.

For one thing, PRA results are conditioned on many things that need to be understood by the decision-maker; PRA models reflect not only properties of nature (e.g., materials properties), but also the expected implications of programmatic decisions: what subsystems are credited in the analysis, the level at which they are assumed to perform, and, implicitly, what is done in order to achieve that level of performance, including special treatment of key subsystems, both before and during system deployment.

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Risk-Informed Decision-Making—I Continued

Correspondingly, system acceptance decisions can be usefully informed by applying the PRA model inversely: given a statement of the performance and efficiency characteristics that we want, we can use the PRA model to determine how best to allocate performance over subsystems, and to begin to reason about what actions need to be taken in order to achieve those levels of performance.

Additionally, it is difficult to assure completeness of PRAs of novel systems. Because of completeness issues (potential failure to identify so-called "unknown unknowns," or potential underestimation of the likelihood or consequences of certain failure modes), a performance allocation for a novel system cannot absolutely guarantee attainment of a desirably low level of risk. But it is reasonable to treat the results of a performance allocation exercise as implying a kind of lower bound on subsystem performance requirements. Beyond this, as illustrated in reliability growth modeling, a focus on precursor analysis has very significant potential to accelerate the rate of learning from experience. It is well known that precursor analysis improves the risk situation in long campaigns; it turns out that precursor analysis has the potential to improve the risk situation even for short campaigns. For this purpose, "precursor analysis" needs to be formulated so as to analyze anomalies in general, and not just obvious near misses.

It has been suggested that since total risk is due to known, unknown, and underappreciated failure modes, a PRA done on a novel system should be thought of as portraying the risk of a mature system. This appears to follow from a simple approximation: as failure modes that are unknown to the analysts are experienced, they are eliminated, and when this process is complete, the system is mature, and the PRA has become valid (it now corresponds to the as-improved system). However, even taken at face value, that notion is not useful for short campaigns. For longer campaigns, it is arguably simplistic; for one thing, it relies on essentially complete elimination of UU's discovered in operation, and this cannot be guaranteed a priori. Even if it turns out that the original PRA bottom line was reasonable, the relative dominance of key contributors may have changed quite a bit as the system is modified to respond to lessons learned during operation.

2:15 pm: Use of Risk Insights in the Practical Implementation of Integrated Risk-Informed Decision-Making Framework, Fernando Ferrante *(EPRI)*, Stuart Lewis, Gareth Parry, Donald Dube, Doug True *(JENSEN HUGHES)*, James Chapman *(James Chapman Consulting LLC)*

While general guidance for addressing individual elements of the key principles of risk-informed decision-making (RIDM) are available in literature, the implementation of RIDM can still be challenging, whether a mature RIDM framework exists or not. Issues regarding the consistency in implementation of these key principles can present challenges that could results in obstacles for a truly integrated RIDM approach to be successful. One of the difficulties with addressing the existing principles of RIDM is that the individual principles cannot be easily addressed using a common scale. Traditionally, RIDM approaches have focused strongly on the use of risk information, particularly quantitative results from Probabilistic Risk Assessments (PRAs), with some individual guidance on other key principles such as defense-in-depth (DID) and safety margin (SM). However, while the assessment of risk principle is amenable to calculating risk metrics and comparing them with numerical goals, meeting DID and SM expectations is necessarily more qualitative in nature. While safety margins can be established quantitatively, they result from using deliberately conservative methods for idealized scenarios. Hence, addressing these different principles in an integrated, balanced fashion that utilizes the strengths of each principle while understanding the impact of uncertainties is not as easily implemented. In fact, evaluation of each principle in isolation can lead to inadequate input for decision-making purposes, while heavily relying on any single principle can negate the benefits from using a risk-informed approach. This work focuses on the specific challenges of the implementation of a truly Integrated RIDM (IRIDM) framework and provides specific solutions and recommendations. It discusses important clarifications of the key principles of RIDM and their intended implementation; as well as the interrelationship of the principles. How IRIDM could be implemented for different RIDM applications is also considered, as different decisions may require different considerations in the use of risk and the application of the other RIDM principles (e.g., evaluating changes that could increase/decrease plant risk may require different RIDM considerations than the use of PRA to compare NPP design alternatives). A framework for IRIDM is presented that integrates the information that needs to be considered, documented, and communicated to the decision-makers.

2:45 pm: Identifying Key Factors Affecting a Team Decision-Making Task Based on the Analysis of Investigating Reports Issued from Diverse Industries, Jinkyun Park *(KAERI)*, Dong-Han Ham, Won-Jun Jung, Hyeon-Woo Oh *(Chonnam National Univ)*

After the Fukushima accident, the scope of human reliability analyses (HRAs) enlarged from the support of Level 1 PSA concerning the contribution of human operators in terms of preventing core damage to the supporting of Level 2 PSA, which focuses on the role of human operators mitigating the consequence of core damage. The change of such scope means that an HRA method that allows us to quantify the human error probabilities (HEPs) of human failure events included in severe accident management guidelines is necessary (i.e., Level 2 HRA). However, it is highly questionable that existing HRA methods are directly applicable to Level 2 HRA, especially the quantification of HEPs related to decision-making tasks under severe accident conditions.

For this reason, based on the intensive review of existing literature, the Korea Atomic Energy Research Institute (KAERI) proposed a conceptual model including 34 factors that are closely related to the performance of a group decision-making making task. This conceptual model is composed of five dimensions, which contain several plausible factors influencing the performance of a group decision-making. These five dimensions are: (1) situational factors that refer to contextual and technological factors impacting the group work performance, (2) decision-making factors that come from the nature of a group decision-making itself, (3) individual-level factors that are defined as factors related to the characteristics, capabilities, and limitations of individual group members, (4) group-level factors that coordinate individual-level factors and a group decision-making problem to achieve the goals of a decision-making, and (5) organizational-level factors that influence the way how a group makes a decision from a higher-level in terms of organizational hierarchy.

In this paper, the catalog of key factors belonging to the five dimensions were identified from the analysis of investigation reports issued from diverse industries including the nuclear industry, railway systems, and aviation industry. Based on these key factors, representative decision-making tasks included in SAMGs will be characterized. As a result, it is expected that a couple of factors that are significantly attributable to the performance of the decision-making tasks involved in SAMGs can be soundly picked out. From the viewpoint of a Level 2 HRA method development, these significant factors plays an important role in estimating the HEP of a decision-making task being included in SAMGs.

Passive System Reliability

Chair: Curtis Smith *(INL)* **Location:** Blue Topaz **Time:** 1:45-3:15 pm

1:45 pm: Decision Making for Active and Passive Safety Systems Alternative: Preliminary Assessment, Luciano Burgazzi *(ENEA)*

In this paper a decision-making framework for comparative assessment of active vs passive safety systems, for Decay Heat Removal (DHR), is proposed, based upon a definite set of criteria.

A Multiple Criteria Decision Analysis (MCDA) methodological process, based on Multi Attribute Value Theory (MAVT), is adopted for building a multi-criteria evaluation model.

A multiple-stage model building process is followed and the Simple Multi Attribute Rating Techniques (SMART) is applied, to capture the decision-makers' concerns for assessing the values of the options, i.e. scoring the assumed criteria, assigning relative weights of importance to the criteria and combining scores and weight.

Subjective judgement is used to deal with the measurement of qualitative criteria and the pair wise criteria comparison of alternatives approach is assumed.

Overall the combination of these MCDA modelling techniques provides a new framework enabling the comprehensive measurement of assessment in a structured way, as a decision-support tool for evaluators of innovative systems to be implemented in advanced reactor designs.

2:15 pm: Interfacing Passive System Performance Degradation Initiated by Nuclear Power Plant Operator Action, Douglas A. Fynan *(UNIST)*, Jinhee Park *(KAERI)*

The study investigates the degradation of the heat transfer performance of a closed-circuit intermediate natural circulation heat transport loop used as a passive safety system in a nuclear power plant (NPP). The reference passive safety system and NPP design are based loosely on the passive residual heat removal system (PRHRS) of SMART (System-integrated Modular Advanced ReacTor). The degradation arises from the strong thermal-hydraulic (TH) coupling to the TH boundary conditions imposed on the hot side of the loop by the transitory state of the primary reactor coolant system (RCS) of the NPP. Several operator actions related to a feed and bleed emergency operating procedure (F&B) are postulated, and system TH code simulations are performed to demonstrate how the F&B can induce two-phase flow conditions in the RCS. The resulting large and sustained two-phase flow instabilities in the core and primary side of the steam generators can significantly reduce the heat transfer to the circulating working fluid of the heat transport loop over long periods, sometimes lasting over 24 hours, of passive system mission time. A transient performance indicator for the PRHRS is introduced for use in passive reliability assessment and quantitative comparison of transient simulations.

2:45 pm: Bathtub Shaped Hazard Rate Functions Change Points Determination; Hazard Graph's Properties, Robab Aghazadeh Chakherlou *(Azad Univ)*, Mohammad Pourgol Mohammad *(Sahand Univ of Technology)*

Investigation the shape of hazard function and its alterations are one of the most important issues in life time data analysis. Bathtub- shaped hazard function is a very practical function in reliability analysis and risk assessment. The bathtub- shaped hazard rate function has either one or two change points where the function value changes. In reliability and risk assessment issues, there are several vital decision situations such as burn-in determination, maintenance (repair, replacement policies), warranty determination, and service which are affected by the locations of change points. Therefore, precise change points' identifications are necessary for these strategies to drive higher reliability and quality and, lower risk and cost. As a result, manufacturers always are fascinated by lifetime data analysis. Life period with bath-tub shaped hazard function is involved of three intervals of infant mortality, useful life and wear-out. The change points' determinations are necessary in hazard assessment and to determine and plan appropriate policies and strategies. This paper presents a robust parametric approach to estimate the change points. The approach consists of two steps. The first step deals with fitting a proper model of hazard rate function. The second step uses new method to specify the change points. In this method, the hazard rate function's curve is investigated precisely to find out every change in the shape of hazard rate function. In this article, hazard rate function's curve is divided to small differentiable functions. The variations of bathtub shaped hazard rate functions curve's slope are calculated for every section. Comparing the variation among the different parts and use of a decision criterion for first and second change points, the change points are determined. Two criteria are used for determination of the change point including minimum of hazard rate function and maximum change in slope of hazard rate function, as published by the author in their previous papers. In prior research, these criteria are used only for failure data points in the hazard curve discretely and, the slope variations are remained in intervals between failure data without any investigation. In this research, all parts of hazard curve are evaluated by new method precisely and change points are defined as well. The Kolmogorov- Smirnov test is used as a goodness of fit method for choosing the best distribution. Additionally, Markov Chain Monte Carlo Method (MCMC) and Bayesian inference are utilized for more precise estimating of the model's parameters. A case study illustrates the proposed approach and its features.

Understanding and Managing Conservatisms and Safety Margins to Support Safety Decisions–Panel

Chair: James B. O'Brien *(DOE)* **Location:** Opal One **Time:** 1:30-3:15 pm

The objective of this panel session will be to discuss experience and work in the area of understanding and managing conservatisms and uncertainties in safety analysis. The topics planned to be covered are:

* Sources of uncertainty in safety analysis
* Defining and measuring conservatisms and safety margins in safety analysis
* Experience/practices in targeting and reducing uncertainty
* Use of best estimate safety analysis.

Panelists: Nathan Siu *(NRC/RES)*
Fernando Ferrante *(EPRI)*
Chip Lagdon *(Bechtel National, Inc.)*
Mohammad Modarres *(Univ of Maryland)*
Steven Krahn *(Vanderbilt Univ)*

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Multi-Unit PSA and Risk Integration—II

Chair: Andrea Maioli *(Westinghouse)* **Location:** Emerald Salon One **Time:** 3:45-5:25 pm

3:45 pm: A Review of Selected Multi-Unit PRA Issues, Mohammad Modarres *(Univ of Maryland)*

False interpretations and confusions remain in the Multi-Unit Probabilistic Risk Assessment (MUPRA) methods and applications that should be properly addressed to make them useful. This paper discusses a select set of such issues, underlines the possible reasons for them and their potential impact on the MUPRA risk metrics, safety goals, and risk-informed practices. In particular, technical limitations or misinterpretations about the scope, types and methods used to treat dependencies among multiple radiological sources on a site will be discussed. Improper assessment, aggregation and interpretation of a site risk metric and the resulting effects on the safety goal policies, regulatory and other risk-informed decisions will be highlighted. Finally, concerns with the use of traditional seismic PRAs in the context of modern MUPRA questions will be examined. These include issues in both assessment of spatial variability of the ground acceleration and treatment of fragilities of multiple structures, systems and components (SSCs) when performing a seismic-MUPRA. Examples of promising methods, directions and new research that would be needed to address the identified issues will be presented

4:10 pm: A Method for Considering Numerous Combinations of Plant Operational States in Multi-Unit Probabilistic Safety Assessment, Dong-San Kim, Jin Hee Park, Ho-Gon Lim *(KAERI)*

One of the technical challenges in multi-unit probabilistic safety assessment (MUPSA) is how to consider numerous combinations of plant operational states (POSs) for each reactor unit. So far, most existing studies on MUPSA have assumed that all units being analysed are in operation at full power and therefore considered only one combination with all units in operation at full power. Since single-unit low-power and shutdown (LPSD) PSAs usually consider ten or more POSs, if we consider a large number of units on a site, the total number of possible combinations of POSs is significantly large.

One way to deal with this problem is to select a manageable number of POS combinations and perform MUPSA for each selected combination. There are two difficulties with this approach: one is to select a representative set of POS combinations; the other is to estimate the fraction of time that is spent in each selected POS combination. The selection of POS combinations should consider both the frequency (or fraction of time) and conditional risk (e.g. conditional core damage probability) of each combination, but it is not an easy task particularly when considering a large number of units.

This paper presents a different approach to address this problem. Firstly, an integrated model for each individual unit is developed combining its at-power and LPSD PSA models in the form of a single-top fault tree. This integrated single-unit model includes basic events representing the fraction of time spent in each state (at-power or a specific POS), which can be easily obtained from the at-power and LPSD PSA results for the unit. Then, the single-unit models are integrated into an MUPSA model in the form of a single-top fault tree. When the size of each integrated single-unit model or the number of units is large, the integrated MUPSA model can be too complicated to be quantified using available software. In that case, each integrated single-unit model can be simplified by screening out non-risk-significant accident sequences (e.g. F-V of 0.001 or lower).

To examine the applicability of this method, site core damage frequency (CDF) due to a multi-unit loss of offsite power (LOOP) initiating event was estimated for three cases for the number of units at a site: two, four, and six units. In this case study, all units at the site were assumed to be identical, and the latest versions of the at-power and LPSD Level 1 PSA models for an OPR-1000 unit were used. As a result, in two-unit and four-unit cases, the site CDF due to a multi-unit LOOP was successfully calculated without screening out non-risk-significant accident sequences for each unit. However, in the six-unit case, the quantification using FTREX failed without screening, but succeeded with screening out the accident sequences with F-V importance values of 0.001 or lower in each integrated single-unit model. The resulting minimal cut sets covered a large number of POS combinations and non-risk-significant POS combinations were truncated.

4:35 pm: Simplified Methodology for Multi-Unit Probabilistic Safety Assessment (PSA) Modeling, Dennis Henneke, Jonathan Li *(GEH)*

GE Hitachi Nuclear Energy (GEH) has led the efforts in developing state-of-the-art methodology for the multi-unit PSA (MUPSA) modeling. The MUPSA methodologies include consideration of the scope of the PSA, risk metrics which will be utilized, methods to develop a combined multi-unit model, and methods to calculate the radiological consequence for multi-unit fuel damage events. Such methodologies have been developed for advanced reactor designs such as the UK Advanced Boiling Water Reactor (ABWR) and the PRISM sodium-cooled fast reactor. The GEH MUPSA methodology has been used in preparation for the current (draft) International Atomic Energy Agency (IAEA) MUPSA methodology, being piloted by the IAEA and by GEH for the UK ABWR MUPSA.

While the efforts for development of MUPSA can vary significantly, insights from the GEH MUPSA results and international pilot applications have demonstrated that application of a simplified methodology may be adequate for most of the applications that require a MUPSA. A simplified methodology can be justified by risk insights obtained from the single-unit PSA models, and the results from the international pilot applications, such as for the UK ABWR.

This paper will provide an updated simplified MUPSA approach, starting with the approach published by M. Stutzke [1]. The updated approach will provide for a more accurate MUPSA estimate for CDF, LRF and radioactive release, and will provide guidance on when a more detailed MUPSA may be needed.

1. Stutzke, M., "Scoping Estimates of Multi-unit Accident Risk," PSAM 2012, June 2014.

5:00 pm: Methodological Approach for a Hydrological Hazards PSA for a Multi-Unit, Multi-Source Site, Matthias Utschick, Siegfried Babst, Gerhard Mayer, Marina Röwekamp, Christian Strack *(GRS)*

In the frame of a research and development project GRS is extending PSA methods to address the risk for nuclear sites with more than one nuclear reactor unit and different kind of sources of radioactivity, e.g. interim dry storage facilities or research reactors. Basis for the PSA extensions for the whole site were research activities regarding a systematic extension and completion of methods for internal and external hazards PSA. Insights from international activities and ongoing research have been considered as well. A Level 1 Site-Level PSA aggregates risk from internal events as well as from different internal and external hazards, depending on the various plant operational states. Moreover, the different risks from other sources of radioactivity, such as spent fuel pool, dry interim storage facilities and nuclear waste treatment installations at a site are considered by assigning to these facilities the risk metrics applied within Level 1 PSAs for German nuclear power plants (core damage, fuel damage, etc.). Dependencies between reactor units and other sources are incorporated.

Multi-Unit PSA and Risk Integration—II Continued

The Level 1 Site-Level PSA approach's first step is a comprehensive and systematic screening of hazards and hazard combinations. Correspondingly, the event trees and fault trees in the PSA plant models are extended in a second step. Basis for hazards screening and PSA model extensions are the documentation of the Periodic Safety Reviews and the PSA models for single units. The Site-Level PSA method has been tested by applying it to a NPP site in Germany with two reactor units, one of them in permanent safe shutdown state. After hazards screening according to a hazards screening approach developed by GRS hydrological hazards which affect the whole site have been chosen for detailed analyses. The Level 1 PSA models of the unit in operation and the unit in permanent shutdown have been extended exemplarily for the event of site-level external flooding. This includes both event tree and fault tree extensions. Moreover, hazard occurrence frequencies and reliability data for systems and components shared by both units have been synchronized within the models, including uncertainty distributions alignments. Finally, the Level 1 Site-Level PSA results have been quantified and analysed.

The paper presents an approach for extending Level 1 PSA to the whole site aggregating the risk and extending the PSA plant model for hazards. The approach has been applied to a multi-unit multi-source site for the example of external hydrological hazards.

Internal Events—II

Chair: Sunil D. Weerakkody *(NRC)* **Location:** Emerald Salon Two **Time:** 3:45-5:15 pm

3:45 pm: Modeling of Personnel Suppression in Nuclear Power Plant Applications, Victor Ontiveros, Orelvis Gonzalez, Francisco Joglar *(JENSEN HUGHES)*, Ashley Lindeman *(EPRI)*

Experience with fire events at NPPs, as captured in the Electric Power Research Institute (EPRI) fire events database (FEDB), indicates that a majority of electrical cabinet fires are extinguished by plant personnel, with minimal suppression efforts, prior to developing into a challenging state. A review of the fire event focused on characterizing the suppression response. The event tree in NUREG/CR-6850 Appendix P considers automatic suppression as the first line of suppression capability. If the fire is not suppressed by an automatic system, the next opportunity for suppression is by the plant fire brigade. The event review determined that only 7% of the electrical cabinet fires were suppressed by automatic suppression.

The event review also shows that plant personnel have a strong role in the suppression of electrical cabinet fire events. However, unlike as prescribed in NUREG/CR-6850 Appendix P, only around 30% of these fires are suppressed by the full fire brigade, while some 50% are suppressed by personnel discovering the fire, staff conducting test/maintenance on equipment, or other general plant personnel. This is not currently captured in the Appendix P framework.

An important criteria to the review of fire growth profiles was analyzing the suppression response, specifically characterizing if the suppression response was simple. Examples of simple responses include de-energizing or removing power to the ignition source and the use of a single portable extinguisher. A review of events shows that over 70% of the fire events were suppressed using simple suppression actions.

This paper will provide an approach that more closely models these types of fire progressions observed in operating experience by revising the non-suppression probability tree. The revision of the event tree better reflects insights gained following a detailed fire events review (e.g. numerous reports of operators responding to equipment alarms in the MCR and discovering a fire, as well as numerous events describing plant personnel discovering a fire in the early stages followed by suppression with minimal effort).

4:15 pm: Radiative Heat Flux Zone of Influence for Open Fires and Electrical Enclosure Fires, Jason Floyd, Francisco Joglar *(JENSEN HUGHES)*, Ashley Lindeman *(EPRI)*

For targets exposed to the radiant heat from a fire, the current guidance computing the radiant flux is contained in NUREG-1805. This guidance is based on techniques developed for large, outdoor, hydrocarbon fires (tank farms, pipeline rupture, etc.). There are two shortcomings with this guidance as it is applied in Fire PRA. The first is the guidance uses a correlation for the emissive power of a fire that does not reflect the real-world behavior of small fires. With exception of fires like a catastrophic failure of the turbine lube oil system, fires used in PRA do not have the emissive power seen in large outdoor fires. This results in overly conservative estimates of the zone of influence (ZOI) of a fire, the distance at which a fire can cause damage to target. The second shortcoming is that there is no specific guidance on how to evaluate the ZOI when the fire is inside of an electrical enclosure. In the absence of guidance, the typical approach is to treat the fire as if it were out in the open. This is also overly conservative as the electrical enclosure prevents direct line-of-site to the radiant heat from the fire.

To address these shortcomings in current guidance, the Electric Power Research Institute (EPRI) has sponsored research into developing improved guidance for the radiative heat flux ZOI for open fires and fires in electrical enclosures. New open fire guidance was derived from the basic principles of fire dynamics and the new guidance is validated against test data for small fires. Guidance for electrical enclosures was developed by modeling electrical enclosure fires with the Fire Dynamics Simulator (FDS), a computational fluid dynamics (CFD) model for fire. The FDS modeling approach was validated using full-scale test data of fire in electrical enclosures. Modeling results were used to develop guidance on both ZOI and severity factor, the severity factor is the fraction of expected fires capable of causing damage. This paper will provide a summary of research activities and summarize the guidance.

4:45 pm: The Effect of the Pressurizer Heaters on Spurious Pressurizer Main Spray Initiation, MS036, Scenario in a Reference Plant Fire PRA, Young G. Jo *(Southern Co.)*

NEI 00-01 provides a list of generic Multiple Spurious Operations (MSOs) scenarios which should be considered in fire PRAs. The generic MSO scenarios were based on the collection of MSO scenarios from industry and the evaluation of their applicability to a specific plant is required for more realistic plant specific fire PRA. In this paper, a plant specific analysis was performed using MAAP Code to evaluate the applicability of generic MSO 36 scenario to a reference plant. According to NEI 00-01 Revision 3, MSO 36 scenario is defined as (Spurious opening of pressurizer spray valves) AND (Inability to trip, or spurious operation of RCP) AND (Failure of Pressurizer heaters). The focus of the plant specific MAAP analysis was to evaluate the effectiveness of the pressurizer heaters in preventing spurious safety injection signal due to low pressurizer pressure cased by spurious main pressurizer spray initiation.

Internal Events—II Continued

MAAP analyses were performed for a reference plant for the two cases where the pressurizer main spray was spuriously on with the full flow rate after reactor trip; one case with pressurizer heaters forced off and another case with pressurizer heaters in auto mode. The results showed that low pressurizer pressure decreased to the low pressure safety injection signal set point at t = 116 second and t =148 second with pressurizer heaters off and with pressurizer heaters in auto, respectively. As sensitivity studies, similar cases with lower main spray flow rates (50 % of full flow rate and 25 % of full flow rate) were also performed. With less spray flow rate, the effective of pressurizer heaters increased but in both sensitivity cases, pressurizer heaters could not prevent but only delay the safety injection signal generation time. Based on the results, it was concluded that pressurizer heaters cannot prevent safety injection signal generation if pressurizer main spay is spuriously on. Therefore, in the reference plant fire PRA, fire PRA MSO36 scenario was modified in such a way that initiating event with spurious safety injection signal is generated when the pressurizer main spray is spuriously on and RCPs cannot be tripped regardless of the pressurizer heater status.

Risk-Informed Decision-Making—II

Chair: Fernando Ferrante *(EPRI)* **Location:** Emerald Salon Three **Time:** 3:45-5:00 pm

3:45 pm: Risk-Informed Acceptance Criteria for Evaluating Leak-Before-Break in Piping Susceptible to Primary Water Stress Corrosion Cracking Degradation, Sara Lyons, Mohammad Modarres *(Univ of Maryland)*

A cooperative effort between the U.S. Nuclear Regulatory Commission (NRC) and the Electric Power Research Institute (EPRI) has resulted in the development of the Extremely Low Probability of Rupture (xLPR) Code Version 2.0. xLPR is a probabilistic fracture mechanics code which models nuclear power plant (NPP) piping, including the effects of primary water stress corrosion cracking (PWSCC) and mitigation, to calculate the probability of leakage or rupture. One aspect of the xLPR project was to develop recommended acceptance criteria for use with the xLPR Version 2.0 Code in reviewing potential leak-before-break (LBB) analyses for NPPs containing piping welds susceptible to PWSCC. The recommended acceptance criteria were considered to be overly conservative by some industry representatives. This paper will examine the recommended acceptance criteria, associated regulatory history, and degradation modeling techniques necessary to support potential alternative acceptance criteria. A discussion of the benefits and challenges associated with applying the NRC's risk-informed regulatory framework for this purpose will be included.

4:10 pm: A Condition-Based Probabilistic Safety Assessment Framework for the Estimation of the Frequency of Core Damage Due to an Induced Steam Generator Tube Rupture, Federico Antonello Francesco Di Maio *(Politecnico di Milano)*, Enrico Zio *(Politecnico di Milano/EdF/Kyung Hee Univ)*

Condition-Based Probabilistic Safety Assessment (CB-PSA) makes use of information on the components conditions during operation to dynamically update the PSA and the therein risk measures quantified (i.e., Core Damage Frequency (CDF) and Large Early Release Frequency (LERF)), thus obtaining a more precise, actualized and locally tailored plant risk profile evaluation with reduced uncertainty on the risk measures quantified. These measures can be used for maintenance planning, because enable estimating the risk evolution throughout the components life, when subjected to aging, degradation, but also unexpected shock events.

In this paper, we show the CB-PSA effectiveness for controlling the risk of a Steam Line Break (SLB)-induced Steam Generator Tube Rupture (SGTR) accident scenario in a Pressurized Water Reactor (PWR), by optimizing the maintenance planning on the CB-PSA results.

4:35 pm: Development of 3+ Level Probabilistic Safety Assessment Methodology and Application for Akkuyu Nuclear Power Plant, Veda Duman Kantarcioglu, Sule Ergun *(Hacettepe Univ)*

There are five levels of protection from radiation which are defined for defense in depth principle. These are respectively; control of abnormal operating conditions, control of design-based accidents, accident management, radiation protection and off-site emergency management (OEM). Severe nuclear power plant (NPP) accidents may result in exceeding of four levels of defense in depth principle. Thus, the fifth level, OEM, aims to minimize the effects of radiation on public by effective planning. The timely implementation of urgent and early protective actions and their pursuance over a reasonable period of time have a direct impact on the mitigation of radiation exposures.

The purpose of this study is to develop 3+ level probabilistic safety analysis (PSA) methodology for severe accidents in NPPs and to apply the developed method to Akkuyu NPP (which will have 4 units of VVER-1200 reactors). The proposed methodology is based on the timely realization of evacuation plans and effective implementations. For this reason, mass evacuation practices in USA are investigated in detail to understand the cases leading the evacuation procedures success or fail.

According to the related regulations in Turkey, size of Emergency Planning Zone (EPZ) is a circle with 20 km in diameter. This zone's name is urgent protective action planning zone (UPZ) as in IAEA's documentation. In the light of obtained data from the literature and existing international standards of the OEM planning, four main factors adversely affecting the mass evacuation procedures have been determined. A probabilistic approach to the disruption of evacuation, due to the failures or vulnerabilities in the OEM processes, has been developed. Based on this approach an evacuation model has been generated by using the fault tree methodology and overall probability of disruption in evacuation procedures has been calculated. Calculations have been performed with SAPHIRE 7.0. The uncertainty in the obtained result has been estimated and the confidence intervals have been determined. Furthermore, different combinations of failures that may arise independently from each other have also been studied and their possible consequences have been predicted. Probabilities of each combination have been estimated with their uncertainties. A risk matrix has been constructed to illustrate the probability-consequence diagram. Combinations of probability and consequence presenting high probability and large negative impact and low probability and large negative impact on evacuation have been stated as high risk cases since they may cause serious break downs in the evacuation process.

The proposed methodology was applied for Akkuyu NPP and it has been estimated that the most probable combination of events disrupting the evacuation is traffic congestion and traffic accident with probability of 2,24E-2 / evacuation and its effect on OEM procedures is discontinuance of evacuation. The obtained results show that the probability of disruption during mass evacuation practices is extremely high. The generated risk matrix shows that disruptions can significantly affect the evacuation and the probability of interruption is high when the combination of events are taken into account.

In conclusion, the proposed methodology can be used as a practical tool to inform decision makers who are responsible for planning for and response to emergencies at UPZ. In general, offsite emergency managers do not have deep knowledge about radiation and its effects. Hence, other technical authorities which have the knowledge and experience on nuclear accidents should inform them. Level 3+ PSA methodology may contribute to strengthening the communication between technical and non-technical emergency management authorities and response units by presenting technical information more apprehensible.

Risk Management

Chair: Gerald Loignon (SCANA – retired) **Location:** Yellow Topaz **Time:** 3:45-4:45 pm

3:45 pm: Discussion of Risk Aggregation in Three Dimensions for Various Risk Hazards, Robert J. Wolfgang *(JENSEN HUGHES)*

The concept of risk aggregation and the method used to describe it has been fraught with difficulty with many defaulting to the concept that all risk is additive in a linear fashion. This paper explains a method in which different initiators can be categorized as relating to one of three main types of hazards, sort of like a three dimensional orthogonal coordinate system. The components of risk attributed to these three "dimensions" are then added as vectors to arrive at an overall magnitude in this three dimensional risk space. The techniques employ the concept of breaking each of the risk hazards into their fractional components that describe the type of risk, much like the three rectangular coordinates that describe a vector in three dimensional space. These three components, or risk dimensions, are as follows: 1) Operational Factors, 2) Design Deficiencies, and 3) External Phenomenon. To explain how these three components are utilized in describing a risk estimate, e.g., core damage frequency (CDF) in this three dimensional risk space, an example is presented to help illustrate the method for risk aggregation using various risk hazards for an example nuclear plant. For example, the full power internal events (FPIE) model is made up of several plant initiators that vary from turbine trips and other plant transient events to random pipe rupture events that are associated with internal flood events. Although these two types of events may be treated using similar fault tree logic in accident space, their underlying cause of how the different events were initiated can be quite different. The turbine trip may be the result of an I&C technician touching the wrong lead during a test involving reactor protection circuitry, whereas the internal flood event was the result of a pressure boundary failure due to a design flaw involving stress concentration and accelerated corrosion. In any event, although the same risk metric, e.g., CDF, may be quantified for both events, they each came about from a different perspective, or what could be perceived as a different risk dimension. An example of how risk can be aggregated using this particular methodology will help illustrate the conceptual basis behind this technique, and also help show the practicality of how this can be used in a general sense for U.S. nuclear plants when it comes to aggregating the risk from various PRA models. PRA models typically evaluate the traditional categories of risk hazards involving internal events, fire, and external hazards for a multitude of initiating events, and this methodology will help provide a technique for aggregating the risk from each of these models to derive an overall estimate of the magnitude of risk using a three-dimensional perspective.

4:15 pm: Insights from Risk-Related Implementation of Reactor Pressure Vessel Water Inventory Control (RPV WIC), Marie Schmehl, Suzanne Skoras *(JENSEN HUGHES)*, Ian Francis *(Susquehanna Nuclear)*

The Reactor Pressure Vessel Water Inventory Control (RPV WIC) Technical Specification supersedes the previous Technical Specification related to Emergency Core Cooling Systems in Shutdown Modes. This change affects multiple technical specifications and removes any reference to the obsolete term Operations with Potential to Drain the Reactor Vessel (OPDRV). In addition to the removal of the term "OPDRV", a new concept has been added to the technical specifications: Drain Time. Drain Time provides a new means of ensuring that inventory is maintained throughout the outage. The concept of drain time is not dependent on an outage mode and maintains its applicability throughout the outage. The updates required to implement these changes affects many groups, which are not limited to design engineering, operations, work control, and PRA. Regarding PRA-related activities, the scope of this change ranges from removing terminology to complete logic changes for shutdown modeling, which include modeling modifications to two Key Safety Functions: Inventory Control and Secondary Containment Key Safety Functions. Modifications to Safety Function Assessment Trees (SFATs), logic requirements for shutdown modeling, and outage risk monitoring software is necessary to updating shutdown risk. This paper discusses the scope of plant-related documents, which were considered for review and update for risk-related purposes. Additionally, this paper describes the interactions between operations and work scheduling utilized on implementing and effectively communicating risk-related changes due to RPV WIC. It also touches on what types of modeling changes were utilized to achieve the end results. This paper also discusses various interpretations across the industry, and variation of implementation on a plant-specific level. Lessons learned through the RPV WIC implementation process will also be discussed.

Extended Sequences

Chair: Richard H. (Chip) Lagdon *(Bechtel National, Inc.)* **Location:** Blue Topaz **Time:** 3:45-5:15 pm

3:45 pm: PSA Evaluation of the New Independent Feedwater System at Ringhals NPP in Sweden, Cilla Andersson *(Ringhals AB)*

The requirement to install an independent core cooling system has been discussed in Sweden since the 1980ies. After the accident in Fukushima further studies were conducted and in December 2014 a requirement to install an independent cooling system before the end of 2020 was issued by the Swedish Regulator SSM. The new system should be designed to improve the ability to handle extended loss of offsite power (ELAP) and loss of normal ultimate heatsink (LUHS).

The solution that is under construction at Ringhals NPP will provide feedwater on the secondary side and make-up to RCS during power operation and shutdown operation with closed RCS to make it possible to cool down the RCS by residual heat removal by the steam system relief valves and natural circulation. To prevent increased RCP seal leakage new passive RCP seals will also be installed. When the RCS is open residual heat removal by feed and boil will be provided by a new possibility to inject water on the primary side. To achieve independence, protection against extreme hazards and a good physical protection the new system will be located in a new, seismically qualified building.

During the planning and construction of the new system PSA modelling has been used both to evaluate the reliability of the new system during 72 hours and its impact on the PSA results. Special consideration has been given to study the manual start of the system, the choice of test intervals, the possibility to make planned maintenance and the allowed outage time. Insights and results from this evaluation will be described in this paper.

Extended Sequences Continued

4:15 pm: Power Supply and Mitigation System Considerations for Extended Loss of All AC Power Events, James C. Lin *(ABSG Consulting Inc.)*

Since the Fukushima Daiichi accident, an extended loss of all AC power (ELAP) event has become a critical issue considered in the nuclear plant risk and safety analysis. This event can be caused by a single external event (e.g., earthquake, tsunami, external flooding, etc.) alone, compound external events, or external events in combination with random failures.

Following an event involving a loss of all AC power, systems that can be used for accident mitigation reduces to a very limited set of equipment. Whether it is a pressurized water reactor (PWR) or a boiling water reactor (BWR), core cooling can only be provided by DC controlled, steam driven systems. For PWRs, a seal loss of coolant accident (LOCA) may also result which further aggravates the progression of the event. In addition, because of the limited capacity of the DC power system, core cooling supplied by the DC controlled, steam driven systems can only last for a limited period of time prior to the depletion of the DC batteries. To cope with this type of events, additional power supply and coolant delivery capability must be provided to prevent the occurrence of a severe accident.

As such, the greatest risk of an extended loss of all AC power comes from fact that only a small set of mitigation equipment is available to provide core cooling for a limited duration only. To minimize this risk, added mitigation efforts are required in two separate aspects; i.e., restoring power from alternate sources and extending the capability of the core cooling mitigation equipment during the interim period prior to restoration of power from the alternate sources.

In terms of power restoration, the most effective means is to provide AC power from alternate sources directly to the medium voltage Class 1E essential buses, which can then distribute the power to all of the safety-related mitigation equipment including supplying the charging current to the vital batteries.

With respect to extending the capability of the core cooling mitigation equipment during the period of time before power restoration, an AC power independent system or an AC-powered system with a dedicated diesel generator (DG) must be available to serve the core cooling function. This may be accomplished by a turbine-driven system, a diesel-driven system, or an AC-powered system with a dedicated DG injecting cooling water into the steam generators (for PWRs) and/or reactor pressure vessel (RPV). At the same time, the DC power capacity should be expanded to ensure that instrument power is available to the vital instruments used for monitoring the critical plant conditions, such as the steam generator water level and/or the RPV water level. To prevent the occurrence of a seal LOCA in a PWR, an alternate method *of seal cooling can also be provided by an AC power independent system or an AC-powered system with a dedicated DG.*

This paper discusses, from the standpoint of probabilistic risk assessment (PRA), the additional plant options that may be considered in response to an ELAP event.

4:45 pm: Modeling Fire-Induced Main Control Room Abandonment in PRA Fault Trees, Kyle Christiansen *(Westinghouse)*

Nuclear power plants have procedures to mitigate fire events that force operators to abandon the main control room (MCR). Abandonment could be required due to either loss of habitability or loss of functionality caused by a fire inside the MCR, or loss of plant control caused by fire outside the MCR. Fire probabilistic safety assessments (PSAs) must model these scenarios to meet supporting requirement FSS-B2 in the 2009 ASME PRA standard; however, the MCR abandonment scenarios introduce additional complexity compared to a conventional fire scenarios where there is no control room impact. Factors that must be considered include operator actions required to successfully abandon the MCR, independent failure of the equipment used to safely shut down the plant from the remote shutdown panel, and what operators consider to be a sufficient "loss of functionality" to necessitate abandonment. Additionally, it is desirable to model MCR abandonment in a fault tree (rather than by post-processing) to generate MCR abandonment cutsets, quantitatively evaluate the uncertainty, and perform sensitivity studies. This paper describes a methodology used to model MCR abandonment due to loss of habitability and loss of functionality in a CAFTA-based fault tree PSA of a four loop pressurized water reactor

Criticality Safety Insights

Cochairs: Robert Hayes (NCSU), Herbert Carl Benhardt *(AECOM Technical Services)* **Location:** Opal One **Time:** 3:45-5:15 pm

3:45 pm: Defining Realistic Conservatism in Nuclear Criticality Safety Analysis, Robert B. Hayes *(NCSU)*

The default approach in conducting a nuclear criticality safety analysis (NCSA) is that of as assumption of worst case conditions. Lifting the plethora of assumptions that follow from such an array of conservatisms is then allowed with a concomitant assortment of controls. In some cases, the laws of physics are used to lift portions of this conservatism but these laws are typically limited to generalities such as conservation of mass, energy and momentum. These are realized in limits of fissile mass and moderator but even then, worst case conditions are again assumed without controls such as optimum moderator to fuel ratios, infinite reflection, lack of interstitial poisons etc. This approach is easy to defend and demonstrate compliance to the categories of extremely unlikely and/or beyond extremely unlikely for an inadvertent criticality event (ICE). Regulatory drivers tend to focus on an upper limit for keff with appropriate sensitivity and uncertainty analysis along with double contingency. The fundamental technical basis for all these approaches are a negligible probability of an ICE. This work promotes focusing on the foundation of the NCSA drivers of maintaining a negligible probability of an ICE and then building on this foundation rather than the other way around. Effectively using a risk management approach as the driver rather than double contingency and keff limits. In this way, configuration probabilities can be calculated and followed in a probabilistic risk analysis approach. Irrespective of the potential benefits from utilizing such an approach, it is predicted that the largest unavoidable risk comes from operator error.

Criticality Safety Insights Continued

4:15 pm: Criticality Safety Insights for a Nuclear Waste Process Using Hazard Analysis, Herbert C. Benhardt *(AECOM Technical Services)*

The Savannah River Site is processing nuclear waste, and will demonstrate high-level salt waste processing using a modular system. The system is designed to filter salt solution and reduce the amount of Cs-137 using ion exchange (IX) technology. This IX system is designated the Tank Closure Cesium Removal (TCCR) demonstration unit. A criticality safety evaluation was performed to confirm the system remains subcritical under all normal and credible abnormal conditions as directed in DOE-STD-3007-2007. Salt waste processing includes the dissolution of saltcake with water, recirculation of water/dissolved salt solution within the waste tank, filtration and adsorption of Cs-137 onto crystalline silicotitanate (CST) in the TCCR unit, return of filtered solids to the waste tank and receipt of decontaminated salt solution in a different receiving tank.

A hazard analysis was performed and identified 23 abnormal conditions requiring criticality safety evaluation. Scenarios identified included unexpected chemical reactions, leaks, inadvertent transfers, fires, and natural phenomena hazards. These conditions were shown to be subcritical based on mass, concentration, or areal density limits provided in the ANSI/ANS-8.1 standard. There is a sludge layer potentially containing fissile isotopes and other undissolved solids on the bottoms of either tank. This layer is expected to remain relatively undisturbed because vigorous mixing will not be performed. The sludge layer has been shown to remain subcritical due to the presence of neutron absorbers. The analysis concludes that the dissolution, transfer, and processing of the salt waste and subsequent storage of the cesium IX columns will be conservatively subcritical due to limited fissile mass and enrichment.

4:45 pm: Estimating the Probability of Multiple Misloads in Spent Fuel Casks for Light Water Reactor Systems, Ibrahim Jarrah, Rizwan-uddin *(Univ of Illinois, Urbana-Champaign)*

Every core cycle, one-third of the reactor core is replaced with fresh fuel and the discharged spent fuel is stored in the spent fuel pools for about 10 years. The used nuclear fuel inventory increases by 2,000 – 2,300 metric tons per year. The estimated number of spent fuel assemblies is now more than 250,000 assemblies, out of which 57% are BWR assemblies and the remaining are PWR assemblies. Due to the limited pools space, the spent nuclear fuel (SNF) is being transferred from pools to dry spent fuel casks for eventual transportation to permanent future repository or recycling. Currently, more than 83,000 spent fuel assemblies are stored in more than 2,000 dry casks. One of the main considerations of the cask design is to remain subcritical. The subcriticality condition of the cask should be maintained under the normal, abnormal, and accident conditions during loading and transportation processes. This condition can be satisfied by loading the dry cask with fuel assemblies that meet the Certificate of Compliance (CoC) requirements of that cask design. The cask may become susceptible to criticality if it is misloaded with assemblies that do not conform with the CoC, and experiences an accident during transportation. The misloaded cask can be defined as the cask that is loaded with one or more assemblies that are different from what is specified in the CoC. The misloading happens because of errors made during handling, storing, loading and database entry steps. For example, misloading can happen when a spot in the cask is filled with a spent fuel assembly with incorrect burnup value, cooling time, or initial enrichment. This could be the result of errors in the spent fuel pool database, choosing inappropriate assemblies from the pool, or errors during the loading process. Also, misplacing an assembly into an incorrect location inside the cask, or swapping two or more assemblies' positions during the loading process will lead to a misloaded cask. Assessment of risk associated with transportation of dry cask involves assessment of the risk of misloading a cask, the risk of accident during the transportation process, and risk of criticality following the accident. The risk associated with the first of these is the subject of this work.

Criticality analysis of the casks suggests that, due to conservative loading plan, the cask needs to be misloaded with more than one fuel assemblies to become susceptible to criticality. Earlier research focused on quantifying the risk of misloading a spent fuel cask with a single misloaded assembly. In this paper, different misloading scenarios that lead to multiple misloads are being identified. An example is the scenario where two fuel assembly positions in the cask are swapped. Quantitative and qualitative analysis of the scenarios for casks loaded with PWR or BWR fuel is carried out to estimate the risk of multiple misloads. The event tree method is used to model each scenario where the number of the misloads is identified from the tree. The failure probability of the top events in the event tree is calculated using the Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H) method. Due to the large uncertainty of human errors, the uncertainty of the results is quantified. Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE 8) is used to construct and quantify the event tree as well as to perform the uncertainty propagation. The preliminary analysis showed that the probabilities of having at least one misload for casks loaded with BWR and PWR fuel are 2.87E-05 and 5.57E-06, respectively.

TUESDAY, APRIL 30

TECHNICAL SESSIONS - 10:00 AM

Seismic Multi-Unit PSA: Special Challenges and Opportunities–Panel

Chair: Robert J. Budnitz *(LBNL – retired)* **Location:** Emerald **Time:** 10:00 am-12:00 pm

The state of the art for performing seismic PSA for either an operating nuclear power plant (NPP) or an NPP in the design or construction phase is mature. Dozens of seismic PSAs have been performed, going back over 40 years. However, almost without exception these PSAs have been done on a single unit, even in cases where many NPP units (often identical units) are co-located on a site.

In recent years, it has become increasingly clear that these single-unit seismic PSAs are inadequate, in the sense that important safety insights that might be obtained from a multi-unit PSA (MUPSA) have not been developed by these single-unit PSAs. The challenges to performing a MUPSA are important, but not beyond today's state-of-practice except insofar as there has not been enough "practice" in performing enough of these MUPSA studies.

There are particular challenges posed when the initiating event for an NPP accident is a large earthquake that strikes a multi-unit site, because clearly the earthquake can be expected to cause similar damage to structures and components at each of the various co-located units.

The objective of this panel is to explore the PSA methodological issues posed by large earthquakes that might strike an NPP site. There are a number of special technical challenges, none of them beyond our capabilities, even if there have been only a very few multi-unit seismic PSAs performed.

The panelists, between them, represent a wide spectrum of experience and expertise in understanding and addressing the technical issues presented by seismic-MUPSA.

Panelists: Robert J. Budnitz *(LBNL – retired)*
Ovidiu L. Coman *(IAEA)*
Karl N. Fleming *(KNF Consulting Services LLC)*
Sunil D. Weerakkody *(NRC)*
M. K. Ravindra *(M. K. Ravindra Consulting)*

TECHNICAL SESSIONS - 1:30 PM

SMR and Advanced Reactor PSA

Chair: Tom Morgan *(ENERCON Services, Inc.)* **Location:** Emerald Salon One **Time:** 1:30-3:00 pm

1:30 pm: Severe Accident Source Terms for Small Modular SFRs, Richard Denning *(Consultant)*, David Grabaskas, Matthew Bucknor *(ANL)*

Over the past five years, in support of the development of a mechanistic source term methodology for metal-fueled, sodium-cooled fast reactors, Argonne National Laboratory (ANL) has been compiling a database of results of fast reactor safety research that had been generated over the past fifty years. The results of this review have been documented in a series of publicly-available reports and demonstrated within the context of a generic pool-type plant design. Two characteristic severe accident scenarios a protected (accident with scram) loss of flow/loss of heat transfer scenario and an unprotected (accident with failure to scram) transient overpower scenario have provided the focus of the study. As demonstrated in the operation of the EBR-II plant, these designs perform well in unprotected accidents and it is necessary to consider multi-fault events to experience severe fuel damage. In a severe accident, radioactive material released from the fuel must pass through an overlying pool of sodium, leak from the cover gas region in the primary system to the containment building and then leak from the containment building to the environment. The analysis of noble gas release is simplified because of the very low solubility of these gases in sodium. For key non-noble gas radioactive elements, such as cesium and iodine isotopes, there are two principal pathways for release considered: bypass of the sodium pool by bubbles of noble gases containing suspended aerosols and for the material captured in the pool, evaporative release into the cover gas region.

In normal operation, substantial migration of key radionuclides occurs to the gas plenum within each fuel pin. It has been assumed that after pin failure, bubbles of non-condensable gases form, which contain non-gaseous radionuclides in the form of aerosols, and transport as isolated bubbles. A detailed model of aerosol removal during bubble transport has been developed by ANL staff. The current paper describes more in-depth consideration of these assumptions and illustrates the impact on the magnitude of the release to the environment. Mechanisms for the formation of aerosols are examined to determine whether they would be likely to exist in the vapor space of the fuel pin prior to failure or could be formed as the pin depressurizes following failure. If a fuel pin has high internal pressure at the time of failure in an accident, a gas jet is expected to be released to the upper sodium pool, which will subsequently degenerate into a bubble swarm, rather than the formation of isolated individual bubbles. Which model is more appropriate depends on the characteristics of the scenario and the phase of the accident in which release is occurring from the fuel. In general, the sodium pool acts as an effective barrier to the environmental release of radioactive material in severe accidents.

2:00 pm: Development of a Methodology for Early Integration of Safety Analysis into Advanced Reactor Design, Brandon Chisholm, Steve Krahn *(Vanderbilt Univ)*, Andrew Sowder *(EPRI)*, Amir Afzali *(Southern Co.)*

Early integration of safety assessment into the design process via the application of fit-for-purpose tools and methods should support efficient design iteration and improvement as well as productive engagement with regulatory authorities. In addition, advanced nuclear technology development will benefit from a technology-neutral approach that utilizes hazards identification, risk characterization, and systems engineering concepts in a coordinated and efficient process -- from conceptual design through start of operations. In light of challenges and concerns identified via engagement with advanced reactor developers and other stakeholders, EPRI has organized a project to define an approach, and assemble best practices, based on industry-standard process hazard analysis (PHA) methods to initiate and facilitate the design-to-license process. Established qualitative and semi-quantitative PHA methods offer a practical means to begin the development of the building blocks for more quantitative design evaluations, including probabilistic risk assessment (PRA). The intent is to benefit from risk-informed insights early in the design process and to incrementally develop the safety case as the reactor design matures.

SMR and Advanced Reactor PSA Continued

This paper updates the activities of the so-called "PHA-to-PRA" project since the publishing of the preliminary Body of Knowledge and Methodology report in September 2018, which concluded the first phase of the project. The second phase of the project has been a case study to investigate how the proposed methodology could be applied to the Molten Salt Reactor Experiment (MSRE)--a representative, publically accessible, non-LWR design--and revise the methodology accordingly. This paper highlights the results of Process Hazard Analysis (PHA) studies of the MSRE, how these results were used to develop quantifiable event tree and fault tree models, and the risk insights developed during the process. This paper also discusses how the case study has influenced the development of the overall PHA-to-PRA methodology and what the next steps of the project will involve.

2:30 pm: Probabilistic Risk Assessment for a Single-Failure-Proof Crane for Small Modular Reactor Refueling Operations, Nathan Wahlgren *(NuScale Power, LLC)*

NuScale Power, LLC (NuScale) submitted a design certification application for its small modular reactor to the U.S. Nuclear Regulatory Commission (NRC) in January 2017. Each NuScale Power Module (NPM) is a self-contained pressurized water reactor with the pressurizer and steam generators housed within the reactor pressure vessel, which in turn is housed in a high-pressure steel containment vessel. A standard NuScale plant places 12 NPMs in a common reactor building, and all NPMs are immersed in a common reactor pool that becomes the ultimate heat sink for the passive safety systems. The reliance upon passively actuated and operated safety systems means that the NPM can achieve and remain in a safe shutdown state indefinitely without operator action, additional water, or AC or DC electric power.

A unique aspect of the NuScale design is that an NPM is not refueled in place. All NPMs in a plant share a common refueling area, and a custom-designed single-failure-proof (SFP) reactor building crane (RBC) is used to move an NPM between the refueling area and its operating bay. Although cranes are used extensively at current nuclear power plants, transporting a reactor with fuel in place is unique and introduces an entirely new category of potential upset conditions. The design and operation of the RBC has therefore been carefully considered by NuScale, and the probabilistic risk assessment group performed a design-specific analysis of the RBC in order to quantify the risk associated with performance of this key system. This paper describes how the RBC PRA was developed, focusing on the crane design, data sources, and accident sequence modeling.

External Events—I

Chair: Robert J. Budnitz *(LBNL- retired)* **Location:** Emerald Salon Two **Time:** 1:30-3:10 pm

1:30 pm: Once Upon a Time, There was a Total Loss of Ultimate Heat Sink ..., Patricia Dupuy, Gabriel Georgescu *(IRSN)*

The total loss of ultimate heat sink was not foreseen as an accident situation to be addressed in the initial design of generation II French NPP. It was introduced in the safety case afterward, as a "beyond design" multiple failure situation, on the basis of the results of the first PSA developments. This accident situation has been studied in both deterministic and probabilistic safety assessments, however on a single unit basis.

The risk of multi-unit LUHS induced by external hazards was fully identified only in a second step, considering French operating experience – Fukushima accident, occurred in 2011, has confirmed such a correlation between natural hazards and site LUHS.

The paper describes the gradual extension of deterministic and probabilistic safety assessments to include the LUHS. It presents the actions already performed or foreseen to deal with such situation at one or several site units.

1:55 pm: A Study on Probabilistic Risk Assessment Methodology of External Hazard Combinations—Identification of Hazard Combination Impacts on Air Cooling Decay Heat Removal System, Yasushi Okano, Hiroyuki Nishino, Hidemasa Yamano, Kenichi Kurisaka *(JAEA)*

A sodium-cooled fast reactor (SFR) uses the ambient air as an ultimate heat sink to remove decay heat from the reactor core. From this feature, the SFR is robust against terrestrial and hydrological phenomena although meteorological external hazards can be risk factors for loss of heat removal capability. If a rare frequent but large intensity external hazard occurs concurrently with another external hazard that possibly arises, components for transporting heat sink (e.g. air cooler, air filter, or air exhaust stack) can be affected. To evaluate the undesirable effects of such hazards leadings to safety-related event on an SFR, the authors studied to classify simultaneous hazards and associated events, and identify issues that should be quantified in terms of their duration and impacts on the components.

Our study firstly made qualitative and general examination on individual external hazards based on guidelines and determined combinations of external hazards that potentially affect the components. Then, the structure of air cooling systems for decay heat removal and types of failures of each component were related to representative external hazards. In this process, we also focused on durations of hazards and component failures: how long effects of each external hazard on components last, and how long it takes the effects to appear if two hazards occur concurrently. We classified the cases into two in which those effects become evident by overlapped occurrences. One is that the appearance of effects on components depends on the order of hazards, in other words, which hazard comes first greatly affect the consequences because a component's sensitivity to a hazard differs from each other. Another one is that components are affected by both hazards and the effects are amplified due to their overlapped occurrence.

The identified external hazards with rare frequency but large intensity are strong wind, tornado, rainfall, snowfall, and volcanic ash, and associated external hazards are continuous very low temperature, landslide, and forest fire. Potentially affected components for decay heat removal by these hazards are protective roofs on outlets of air exhaust stack, air cooler heat transfer tubes that run between the intermediate sodium and ambient air, air filters, and air dampers located at air inlets. In addition, we identified ex-vessel fuel tank fire as a hazard that potentially affects surrounding air of a reactor building.

This study identified the simultaneous occurrence of external hazards that should be addressed and issues that should be quantified by examining the duration of their effects and the order of occurrence to prevent their impacts on air cooling systems for decay heat removal. Frequency evaluation of two hazards overlapping, event tree development and analyses on the decay heat removal system due to the identified hazard combination, and fragility evaluation on key components on decay heat removal system by selected hazard are recognized as future work.

External Events—I Continued

2:20 pm: **Risk-Reduction Credit for Very Early Warning Fire Detection: From FAQ to Fiction**, Raymond HV Gallucci, *(Retired)*

NFPA-805 Frequently Asked Question (FAQ) 08-0046, "Incipient (Very Early Warning, VEW) Fire Detection Systems (FDS)," has seen a history with rather widely varying results in terms of the amount of credit possible for risk reduction due to the installation and use of these types of fire protection systems. Prior to its issuance by the NRC in November 2009 with a factor of 50 as maximum credit for risk, the proposed reduction credit ranged from a high of 167 to a low of five. As a follow-on to this FAQ, the NRC Office of Nuclear Regulatory Research (RES) performed confirmatory experiments, resulting in an updated and much expanded event tree approach to quantify the risk reduction attainable via VEWFD vs. "conventional spot" detection systems, both in-cabinet and area-wide (NUREG-2180). The "alpha" factor representing the fraction of "challenging" fires detectable by a VEWFD system retained a basis on interpretations from the EPRI fire events database and, in conjunction with maximal credit for human response, constituted much of the reduction deemed attainable. As a result, as much of a reduction by at least a factor of 10 still remains for the maximally creditable case. The dominant contribution to this remains the assumption of the applicability of the non-suppression probability curve for Main Control Room (MCR) fires to non-MCR electrical enclosure fires throughout the plant. However, the maximal reduction factor when the electrical enclosure fire non-suppression curve is applied remains around five, as demonstrated by at least three separate analyses performed since the FAQ was first proposed. The most recent is presented here, using many of the values developed in NUREG-2180 itself, except for the retention of the MCR non-suppression curve. The results indicate that the difference between in-cabinet and area-wide VEWFDS when compared to conventional ceiling-mounted detection is small, with overall reductions in non-suppression probability of approximately five (low voltage electrical enclosures) and three (other). This aligns well with the quantitative results from a maximally creditable bounding approach, also presented here, remaining at least a factor of two less than the current maximal NUREG-2180 risk-reduction factor of at least ten.

2:45 pm: **Incorporation of Spatial Variability of Ground Motions in a Seismic Multi-Unit Probabilistic Risk Assessment**, Jonathan DeJesus Segarra, Michelle Bensi, Mohammad Modarres *(Univ of Maryland)*

During an earthquake, spatial variability of ground motion will be experienced at different locations around a nuclear power plant site (e.g., different reactor units and dry spent fuel storage facilities). However, spatial variability of ground motion is not currently incorporated in the state-of-practice of seismic multi-unit probabilistic risk assessment (MUPRA). That is, in the seismic MUPRA, the ground motion hazard at the different locations around the site is assumed to be perfectly correlated. This paper discusses a method for incorporating different ground motion hazards corresponding to different locations around the site in a seismic MUPRA. The ground motion hazards for the different locations are conditioned on a reference ground motion hazard, which is obtained from an already performed probabilistic seismic hazard analysis (PSHA) at the same site. This paper is based on a case-study using (1) the seismic MUPRA developed by Zhou, et. al. [1], (2) a PSHA for a hypothetical site developed by the authors, and (3) the ground motion hazard at different locations conditioned on the reference ground motion hazard also developed by the authors. Lessons learned regarding the use of different ground motion hazards at the same site for use in a seismic MUPRA are provided.

Reference: *[1] T. Zhou, M. Modarres, and E. L. Droguett, "An improved multi-unit nuclear plant seismic probabilistic risk assessment approach," Reliab. Eng. Syst. Saf., vol. 171, no. Supplement C, pp. 34–47, Mar. 2018*

Level 1 and 2 PSA—I

Chair: Jeff Gabor *(JENSEN HUGHES)* **Location:** Blue Topaz **Time:** 1:30-3:10 pm

1:30 pm: **Accident Sequence Probability in PSA**, Andrija Volkanovski *(Jožef Stefan Inst)*

The safety of the nuclear power plant is assessed with probabilistic safety analysis (PSA). The PSA utilizes fault and event trees analysis for the assessment of the plant safety. The event tree analysis is the technique used to define potential accident sequences associated with a particular initiating event or set of initiating events. The probability of the accident sequence depends on the probability of failure and/or functioning of the systems that constitute given sequence. The frequency of the final state corresponding to the predefined consequence (for example core damage) depends on sequence probability and initiating event frequency. The frequency of the analyzed consequence in overall PSA model, for example core damage frequency, is equal to the sum of the frequencies assessed in all event trees of the model.

One important question that arises from above dependency is if/how we can assess what is overall reliability/safety of the systems that are considered for the given initiating event in the corresponding event tree. The difference of the assessed reliabilities of systems in different event trees is also interesting to analyze and compare.

In order to answer these questions first PSA model created on the basis of Surry Unit 1 PSA model given in NUREG/CR-4550 will be analyzed. The core damage frequency of the PSA model will be assessed including contribution from the specific initiating events. In next phase the frequency of all initiating events in the model will be set on same predefined value. The PSA model will be recalculated with new initiating events frequency. The contribution of the separate initiating events with corresponding event trees in overall core damage frequency will be assessed. This contribution will depend on reliability of the safety systems that are considered in the PSA model for the given initiating event.

Obtained results will be utilized in the analysis of the balance of safety/redundancy implemented in the design of the analyzed plant. Potential application of the presented methodology for improvement of the plant safety will be discussed.

1:55 pm: **Modeling Hydrogen Explosion in Level 1 PSA**, Julien Beaucourt, Gabriel Georgescu *(IRSN)*

Extension of the operational lifetime beyond forty years is currently a noteworthy project in the field of nuclear safety in France, especially for 900 MWe reactors that will be the first ones to go through the fourth periodic safety review. Probabilistic safety analyses (PSA) are to play an important role in this process, especially in the frame of the assessment of the robustness of the plants against internal and external hazards (the PSA are performed by the licensee and reviewed by the institute of radioprotection and nuclear safety, which is the French TSO). Among these hazards, internal explosions induced by hydrogen accumulation following leakage of pipes and/or singularities, or due to the failure of venting systems in the battery locals are now taken into account in the PSA developed by the licensee. IRSN has therefore developed its own Level 1 PSA methodology and a study to analyze the risk related to such an explosion, independently from the analyses performed by the licensee. This analysis is mainly focused on nuclear auxiliary buildings (in which most of the safety systems are located) and electrical rooms (where the battery are stored). The aim of this paper is to present the methodology and some discussions about the key parameters.

Level 1 and 2 PSA—I Continued

The first step is to identify the sources of hydrogen within the different rooms of the auxiliary and electrical nuclear buildings: for that purpose, plant walkdowns have been performed in order to characterize the potential sources of hydrogen (typically, pipes, valves, flanges, batteries…), their locations and environment, and the characteristics of the potential means of prevention and mitigation (detectors, venting systems…). Then, in these rooms, the scenarios that may lead to the accumulation of hydrogen above the flammability limits (as defined by the Shapiro diagram) are defined, and the probabilities of occurrence of such scenarios are quantified. The analysis of operating experience feedback (taking into account human activities) is a key point of the IRSN methodology for this second step. The evaluation of kinetics aspects (namely, the time before the hydrogen concentration overcomes the flammability limits that will be the key parameter to determine the feasibility of human interventions such as manual isolation of valves, doors openings, etc…) is also of major importance at this level. As a result, the probability of hydrogen leakage and explosion within the different rooms of auxiliary nuclear buildings and electrical buildings are calculated.

Finally, based on a simplified functional analysis of the consequences of the explosion within the room where it took place, and possibly within the surrounding area, the core damage frequency induced by internal explosion may be evaluated, using Level 1 PSA for internal events.

2:20 pm: **Practical Application of the Loss of Offsite Power Recovery Analysis Using the Convolution Methodology**, Matthew M. Degonish *(Westinghouse)*

EPRI Report 1009187 details the treatment of time interdependencies in fault tree generated cutset results as applied to accidents initiated by the loss of offsite power. With the growing complexity of PRA models, an efficient way to practically apply recovery factors to loss of offsite power sequences is necessary. Discussion on the convolution integrals, use of modern fault tree software, mathematical simplifications, and other unique insights will be provided. Additionally, this paper will discuss some lessons learned from the application of the convolution methodology to a loss of offsite power recovery analysis.

2:45 pm: **Simplified/Harmonized PSA: A Generic Modeling Framework Applied to Precursor Analysis**, Ali Ayoub, Wolfgang Kröger (ETH Zürich), Olivier Nusbaumer *(Kernkraftwerk Leibstadt)*, Didier Sornette *(ETH Zürich)*

PSAs in commercial nuclear power have been continuously improved over the years, and are of high quality in many respects. However, completeness cannot be proven, and models are large, complex, and difficult to comprehend by "super experts". In this work, in contrast to the existing state-of-art detailed models, we attempt to develop generic PSA models and event sequences. Approaching PSA at a coarser granularity will allow the development of simplified/harmonized models that are neither plant nor site specific and have the potential to represent different plant types and designs.

Following the principle of "learning from experience", we use these developed PSA models for precursor analysis of a number of events from our open comprehensive nuclear events database. Our starting point is to simplify event trees into functional blocks, with the possibility of "zooming-in" -- allowing for more detailed information -- when needed, as well as when more precise probabilities and uncertainties are to be quantified. We focus on internal events but do not separate them from external triggers and take man-machine interactions, including maintenance, into account.

Moving in this direction allows us to gain: 1) more statistics pooled from worldwide experience of different plants and sites (15'000 reactor years of operation to date), 2) a generic order of magnitude CDF comparisons between different reactor designs and operating environments, 3) broader insights and trend analysis, and 4) more transparent and graspable PSA models and results. Naturally, with more generic PSA models one loses plant specific designs and features. Therefore, our proposal is not intended to substitute current PSA; rather it serves as a complimentary framework that could bring new conclusions and insights.

Digital I&C, Software Reliability, and Cyber Risk

Chair: Tunc Aldemir *(Ohio State)* **Location:** Yellow Topaz **Time:** 1:30-3:00 pm

1:30 pm: **Comparative Application of Digital I&C Modeling Approaches for PSA**, Markus Porthin *(PSI)*, Sung Min Shin *(KAERI)*, Tero Tyrväinen *(VTT Technical Research Centre of Finland Ltd)*, Christian Mueller, Ewgenij Piljugin, Jan Stiller *(GRS)*, Richard Quatrain, Léo Granseiggne *(EdF R&D)*, Hans Brinkman *(NRG)*, Paolo Picca, Joshua Gordon *(ONR)*, Jiri Sedlak *(ÚJV Řež a.s.)*

Newly built nuclear power plants are equipped with digital I&C systems. These systems are also introduced in older plants in the course of modernizations. However, in the current situation there is no specific guidance internationally agreed for modelling of digital I&C systems in PSA. Thus, the OECD Nuclear Energy Agency (NEA) CSNI (Committee on the Safety of Nuclear Installations) Working Group on Risk Assessment (WGRISK) initiated a task called "Digital I&C PSA – Comparative application of DIGital I&C Modeling Approaches for PSA" (DIGMAP) in July 2017.

The objective of the task is to compare different modeling approaches and identify possible methods and issues for further development. A simplified boiling water reactor (BWR) design equipped with digital features is used as a common reference plant for the task. Each participating country develops their own PSA model based on the plant description. Through comparison, valuable insights on different modeling approaches regarding digital features (such as software, fault-tolerant techniques, and network communication) for future modeling methods development will be gained. Nine countries are currently participating in the task: Republic of Korea and Switzerland (co-leads), Finland, Germany, France, The Netherlands, UK, Czech Republic (core-group), and Canada (observer). The planned duration for the task is 2017-2020.

The reference plant description was developed for the study based on the DIGREL PSA model of a simplified fictive BWR provided by Finland. The plant design and description were modified in order to focus specifically on modelling issues concerning digital I&C and were otherwise simplified to minimize other modelling efforts. After finalization of the plant description in early 2018, the work for the rest of the year focuses on PSA modelling by each task participant. Two online meetings are planned for 2018 to discuss any matters requiring consensus and coordination in PSA model development. In 2019, two workshops are planned for comparison of PSA models: one to share PSA models and set a comparison framework and another to capture differences among approaches proposed by each participant and identify issues for further development. The final report is aimed to be published in June 2020.

This paper describes the efforts done so far and presents preliminary results and insights.

2:00 pm: **Model Based Reliability Analysis of Digital I&C of the Hoisting Equipment in Nuclear Facilities**, Ewgenij Piljugin, Christian Mueller, Moritz Leberecht *(GRS)*

Digital I&C, Software Reliability, and Cyber Risk Continued

Hoisting technology is used in a wide variety of construction, operation and maintenance processes in nuclear power plants, e.g. carrying heavy loads with the reactor building crane or fuel elements with the refueling machine. Failures of the hoisting equipment may jeopardize nuclear safety, which may result in drop of loads resulting in the release of radioactive material or damage of safety equipment. A hypothetical load drop must therefore be taken into account in the design, operation and decommissioning of a nuclear facility. In the technical supplement on PSA methods of the German PSA Guide the drop of heavy loads is considered as an initiating event; therefore, this event shall be subject to a probabilistic analysis of nuclear power plants. Furthermore, the operating experience from German plants has repeatedly shown malfunctions of the I&C equipment of the refueling machine and other lifting equipment in nuclear facilities. The safe and reliable operation of the hoisting equipment, which are increasingly equipped with digital, programmable I&C technology, is an important prerequisite for the safe operation of this hoisting machinery. For the design and operation of the hoisting equipment in nuclear facilities in Germany nuclear standards (in particular German Nuclear Standards KTA 3902 and 3903) are available, the typical I&C functions of this equipment shall be assigned to the performance level categories according to non-nuclear European Standard DIN EN ISO 13849-1. In the framework of a safety evaluation of the hoisting equipment in a nuclear power plant it is necessary to demonstrate that the given performance levels of the I&C functions are applicable to achieve a sufficiently low probability for a load drop or/and for the violation of the safety goals.

As part of a research project, a methodology for analysis and validation of the proper functioning of the I&C system of hoisting equipment important to safety is being developed. In this context, the I&C system of a hoisting equipment is understood here as a complex structure of sensors, control logic and actuators. The desired assessment approach should provide evidence that the required safety functions (e.g. protective stop of the lifting function, blockage of the drive function) are reliably fulfilled by the I&C system even in case of potential failures. The methodology is being developed by applying a model-based approach, using various deterministic and probabilistic analysis methods, such as Failure Mode Effect Analysis (FMEA), Fault Tree Analysis (FTA), and dynamic simulations to analyze the effects of faults of the control circuits. The paper presents the model description of a generic crane control system in a nuclear facility and the results of the methodological development. For this purpose, advantages and disadvantages of the analysis methods for the application in the developed methodology are discussed.

2:30 pm: Development of Cyber-Attack Complexity Evaluation Model for Cyber Security of Nuclear Power Plants, Jong Woo Park, Seung Jun Lee *(UNIST)*

As adoption of digital technology in instrument and control (I&C) systems of nuclear power plants (NPPs) which is one of the safety critical infrastructures, cyber-attacks have emerged as one of new dangerous threats. Especially "Stuxnet" in 2010, which is one of typical examples of cyber-attacks on nuclear facilities, showed that it is possible to destroy components physically through the cyber-attacks. NPPs must be defended in any situation including cyber-attack scenarios because once cyber-attacks are success to effect digital I&Cs in NPPs that could have severe consequences. Therefore, it is necessary to develop cyber security against cyber-attacks considering specific characteristics of NPPs.

To develop effective strategies for cyber security, the risk of a cyber-attack on an NPP should be evaluated quantitatively. In general, the risk of an NPP is defined as the product of frequency and consequence of accidents. In the same sense, the risk of cyber-attacks on an NPP can be defined as the product of the frequencies of cyber-attacks, the conditional probability of events caused by cyber-attacks, and the consequence of the events. However, the frequencies of intended cyber-attacks are impossible to predict and evaluate its frequencies. For that reason, the concept of cyber-attack complexity which includes both the frequency and the conditional probability could be used instead of general concept of the frequency for assessing the risk.

The aim of this work is to estimate cyber-attack complexity considering specific characteristics of NPPs for developing effective cyber security. To estimate complexity of cyber-attacks, a complexity evaluation model for NPPs was developed based on Bayesian belief network (BBN). The model includes cyber-attack related variables such as vulnerabilities of NPPs, protection and detection systems for NPPs, mitigation systems and back-up actions of operators, and failure impacts of critical digital assets (CDAs) caused by cyber-attacks. By using the developed model, relative complexities of cyber-attacks can be evaluated quantitatively. It is expected that this method can be applied not only to provide quantitative evaluated information of cyber-attack complexities for assessing the risk of cyber-attacks on an NPP, but also to evaluate the effectiveness of developed defense strategies for cyber security quantitatively.

Internal Events and Common Causes

Chair: Rick Summit *(EPM)* **Location:** Opal One **Time:** 1:30-3:00 pm

1:30 pm: Internal Flooding PRA Refinement by Partitioning of Pipe Rupture Frequencies, Matthew M. Degonish, Luyen D. Nguyen *(Westinghouse)*

A common approach for performing an internal flooding PRA (IFPRA) is to group various pipe rupture sizes for a given system by flow rate. The three flood categories commonly considered are: spray events (1 gpm – 100 gpm), flood events (100 gpm – 2,000 gpm), and major flood events (>2,000 gpm).

However, for certain systems, the three flood categories may result in an over conservative application of pipe rupture frequencies. By splitting up the flood categories of risk-significant scenarios into smaller flow rate ranges, more realistic correlation between scenario frequency and scenario impacts can be obtained, which result in more realistic risk metrics. This paper will provide the description of this methodology, comparison of impacts on model results, and lessons learned from application of the methodology.

2:00 pm: Evaluation of Common Cause Failure by an Initiating Event for Multi-Unit Using Bayesian Belief Network, Yun Yeong Heo, Seung Jun Lee *(UNIST)*

Nuclear academia has been focused on probabilistic risk assessment (PRA) for a multi-unit site. Focused attention about multi-unit PRA after the Fukushima Daiichi accident in March 2011 has been maintained and methodology of PRA in the multi-unit site has been developing in many countries. When there are several units in one site, the dependency between units must be taken into account in PRA although there are few shared systems [1]. For that reason, inter-unit common cause failure which is a core element in multi-unit PRA has been developed in Korea [2].

Internal Events and Common Causes Continued

In this study, the probability which represents that if one unit has core damage then the other one unit in the same site also has core damage is estimated using Bayesian Belief Network (BBN). For the simplicity and optimization of the developed model, only common cause failures (CCFs) of the components in units are considered. Since different mitigation systems are considered depending on initiating events, a different correlation factor between units needs to be identified according to the given initiating events. In this work, for a feasibility study, loss of offsite power (LOOP) is selected as the target initiating event because the electric system is a typical example of shared systems between units. By analyzing minimal cutsets (MCSs) and components importance of LOOP event tree (ET)/fault tree (FT) model, systems which affect the initiating event are selected and components which need to be addressed are selected with the systems. Then insignificant components are screened out based on Fussel-Vesely (FV) importance measure for model optimization. In the step of establishing the model, the top node of the model is defined as the correlation factor of inter-unit in the site which represent the probability of core damage on two units by an initiating event. The leaf nodes are component level CCFs related to the initiating event. The relations of component CCFs to system CCFs and system CCFs to the inter-unit correlation factor are represented in the node probability tables (NPTs) of the BBN model.

In this work, a method was proposed to estimate the quantitative correlation representing the probability of core damage in two units at a site for a specific initiating event. Since only CCFs are considered in the proposed model, the size of model is more compact than that of the current PRA model and it is able to illustrate the correlation more intuitively. Furthermore, as taking advantage of the BBN model, components or systems which have a relatively great impact on the correlation can be identified and complemented to improve the safety of the site.

[1] Fleming, K., "On the Issue of Integrated Risk – A PRA Practitioners Perspective", In Proceedings of International topical meeting on Probabilistic Safety Analysis, (PSA2005), San Francisco, CA(2005)

[2] D. S. KIM, "Development of an Inter-Unit Common Cause Failure Analysis Method for Multi-Unit Probabilistic Safety Assessment", KAERI/TR-7061, Korea Atomic Energy Research Institute(2017)

2:30 pm: Development of Inter-Unit Common Cause Failure Methods in Multi-Unit PSA, Seunghyun Jang, Sangyeon Kim, Yein Seo, Moosung Jae *(Hanyang Univ)*

Inter-unit dependency is an element that has a high impact in Multi-Unit Probabilistic Risk Assessment (MUPRA). Among the inter-unit dependency, Common cause failure (CCF) between identical components in multiple units is considered as an important event. The CCF event is an event that multiple components or systems are failed simultaneously due to the same reason. In multiple units on a site, many identical components are installed, and those components can fail due to the inter-unit CCF during operation and maintenance. Therefore, MUPSA with the level 1 PSA models considering inter-unit CCF should be performed.

In this study, we developed methods for inter-unit CCF modeling using a human error dependency model. Because there are considerable similarities in the multiple units in a same site, failure event of identical components between multiple unit can depend on each other. Therefore, inter-unit CCF event was calculated using the dependency between multiple unit and intra-unit CCF data that all same components in single unit are failed. To apply these methods for the case study, inter-unit CCF events between emergency diesel generators (EDGs) in 4 units of different pressurized water reactor (PWR) types (2 units of OPR1000 type, 1 unit of WH600 type. 1 unit of WH900 type) are selected and applied with the multi-unit Loss of Off-site power (LOOP). It is demonstrated that the methodology developed in this study can be applied for modeling the inter unit CCF involved in multi-units as well as evaluating the site risk in the next step.

Advancing HRA Technology: Short-Term and Long-Term Needs–Panel

Chair: Paul J. Amico *(JENSEN HUGHES)* **Location:** Emerald Salon Three **Time:** 1:00-3:00 pm

The panel session will address the short-term and long-term needs to advance HRA technology to better reflect the risk contribution associated with HFEs. It will be a "double panel" over two consecutive sessions in order to provide a broad perspective. The panelists will select their own topics of interest, covering HRA methods, dependency analysis, dynamic HRA, level 2 HRA, HRA in a digital environment, and others.

Panelists: Jeff Julius *(JENSEN HUGHES)*
Susan E. Cooper *(NRC)*
Jinkyun Park *(KAERI)*
Markus Porthin *(PSI)*
Mary R. Presley *(EPRI)*
Cornelia Spitzer *(IAEA)*

TUESDAY, APRIL 30
TECHNICAL SESSIONS - 3:30 PM

Learning from Experienced Nuclear Events: The Role of Precursor Analysis–Panel
Chair: W. Kröger *(ETH Zürich)* **Location:** Emerald Salon One **Time:** 3:30-5:15 pm

Understanding of safety characteristics and further safety improvements is a continuous task within the nuclear community. Growing experience with commercial NPPs (presently more than 15'000 reactor-years) and established databases at national or international level, either restricted like IRS or WANO or open, comprehensive like the ETH database comprising more than 1000 worldwide events over centuries, are a valuable asset. Precursor studies using different approaches, either plant and site specific or generic, have proven really informative and have been carried out or proposed to effectively evidence safety significant insights, either at detailed or order of magnitude level. The Panel will address existing databases and explore on the strengths and weaknesses of precursor-based methods and approaches for different purposes, from different perspectives and backgrounds to gain specific (like CCDF) or big-picture insights.

Panelists: W. Kröger, ETH Zürich Switzerland *(chairman, introductory remarks)*
N. Siu *(USNRC, data evaluation)*
M. Roewenkamp *(TSO, GRS Germany and chairperson of NEA Risk Working Group)*
Fernando Ferrante *(EPRI)*
Yoshikane Hamaguchi *(NRA)*
Cornelia Spitzer *(IAEA)*

Technical Sessions: Tuesday April 30

External Events—II
Chair: Zoltan Kovacs *(RELCO Ltd.)* **Location:** Emerald Salon Two **Time:** 3:30-5:10 pm

3:30 pm: Integrating External and Internal Event Hazard Models at Nuclear Power Plants, Nicholas Lovelace, Matt Johnson *(JENSEN HUGHES)*

Various External Hazards may include more than one hazard contributor that has historically been modeled independently in nuclear power plant Probabilistic Risk Assessments (PRAs). An example is external flooding and high winds, which can occur together during a hurricane or severe thunderstorm that produces tornadoes. External event models typically only include the High Wind hazard as a unique hazard without water impacts and likewise, External Flood models include flood hazards without wind impacts from a severe storm or hurricane. Further, internal event models include a loss of offsite power initiating event with frequency estimated based on data that includes high wind and/or external flooding based events. Power recovery curves are based on the same set of events as included in the internal events model loss of offsite power initiating event frequency. The existing loss of offsite power initiating event data and offsite power recovery curve data may not include wind or flood events of the magnitude that are modeled via hazard curves that are developed using models that predict events over time scales longer than the current operating experience applicable to nuclear power plants.

This paper explores methodology that can be used to integrate the external hazards with combined impacts such as winds and flooding and discusses the validity of internal events loss of offsite power modeling and offsite power recovery in the context of the external events analysis hazard curves. The goal of integrating the impacts is to avoid double counting or underestimating the impacts by treating them independently.

3:55 pm: A Preparatory Study on Systematically Considering Combinations of External Events in the Design Basis and the Probabilistic Safety Assessment of NPP PAKS, Tamas Siklossy, Attila Bareith, Imre Szanto, Barnabas Toth *(NUBIKI Nuclear Safety Research Inst Ltd)*

Mostly single external hazards have been considered in the definition of the design basis of the Paks NPP in Hungary. Accordingly, the analysis and evaluation of plant resistance against design basis loads as well as the probabilistic safety assessment (PSA) of the plant have been limited to these single external hazards. However, the Hungarian Nuclear Safety Codes as high level safety regulations require the inclusion of combined external hazards in the safety demonstration of a nuclear power plant in a site specific manner. This requirement as well as international recommendations and lessons learned from the Fukushima Dai-ichi NPP accident point out the need for a systematic survey and assessment of the combinations of external hazards for the Paks NPP so that their impact on plant safety can be determined and evaluated. As a preparatory step, the available methodologies have been studied and evaluated to underpin the treatment of external event combinations in plant design basis and PSA. In this initial step the Hungarian practices of identifying external hazards to be considered in the plant design basis and the plant PSA were subject to a critical review with respect to hazard combinations in the light of the related Hungarian regulatory requirements and international recommendations. Also, publically available literature and the results of international research and development programs were examined. Eventually, a proposal was prepared for the technical tasks and the schedule thereof to identify external hazard combinations and the assessment of the impact of such hazards on NPP safety. This proposal includes a high level methodology for hazard selection and screening, probabilistic hazard assessment, evaluation of plant protection, plant response and fragility analysis, development of event sequence models for hazard initiated plant transients, and risk quantification and evaluation of results. Identification and screening as well as hazard assessment were in the focus of the study. For severe weather events the multivariate extreme value theory was studied, and a methodology was proposed to establish multivariate hazard curves (surfaces) based on measured data at the Paks meteorological station. The method accounts for the hazard assessment of single hazards used as the marginal distributions for the joint probability distribution of the identified hazard combinations. This paper presents an overview of the preparatory study performed for the Paks NPP. Important methodological aspects are summarized. Key findings and unresolved issues that need further elaboration are highlighted.

4:20 pm: Lessons Learned From Recent Seismic Risk Evaluations Including Probabilistic Risk Assessments to Support Regulatory Actions, Shilp Vasavada, Mehdi Reisi-Fard *(NRC)*

The U.S. Nuclear Regulatory Commission (NRC) licensees may need to quantitatively address the risk associated with seismic events for implementing certain regulatory actions. The NRC staff has recently completed review of several submittals, which include information related to acceptability of seismic probabilistic risk assessments (SPRAs) and other seismic risk evaluations that support regulatory actions. Examples of those actions include adopting the program to risk inform categorization and treatment of structures, systems and components, adopting risk-informed completion time program, and responding to Near-Term Task Force (NTTF) Recommendation 2.1 "Seismic." This paper presents some lessons learned from the NRC staff's review of the technical acceptability of those SPRAs and seismic risk evaluations. These lessons are related to the acceptability of PRA models that are used as the base for the SPRAs, use of acceptable peer-review processes, SPRA acceptability, use of conservative evaluations as an alternative to SPRA models, and disposition of key assumptions and sources of uncertainty in SPRA.

External Events—II Continued

4:45 pm: A New Method to Allocate Combination Probabilities of Correlated Seismic Failures into CCF Probabilities, Woo Sik Jung *(Sejong Univ)*, Kevin Hwang *(Simpson Gumpertz & Heger)*, Seong Kyu Park *(Atomic Creative Technology)*

(CCF method in internal, fire, and flooding PSA) *When component failures in internal, fire, and flooding probabilistic safety assessment (PSA) have positive dependencies (or positively correlated), they are divided into common cause failures (CCFs). After expanding dependent failures (A, B, and C) into independent CCFs (Aind, Bind, Cind, CCFAB, and ,) in a fault tree, the usual fault tree analysis (FTA) is performed. In this way, dependent failures are taken into FTA by splitting dependent failures into independent CCFs.*

(Integration method in seismic PSA) *Instead of the previous CCF application, seismic PSA incorporates the dependencies among seismic failures by means of correlation. This correlation is quantified by using Reed-McCann or Multi-Variate Normal (MVN) integration [NUREG/CR-7237]. If a minimal cut set (MCS) has correlated seismic failure combination ABC, its probability (ABC) is calculated by Reed-McCann or MVN integration. These integration methods require significant effort to extract various combinations of correlated seismic failures from MCSs, and calculate their probabilities. More seriously, when correlated seismic failures exist across many MCSs, it is frequently impossible to extract various combinations of correlated seismic failures from MCSs, and integrate these complex Boolean equations by Reed-McCann or MVN integration.*

(Need of development) *Because of the complexity and difficulty in calculating various combination failure probabilities of correlated seismic failures, there has been a great need for (1) converting all the combination failures of correlated seismic failures into seismic CCFs, and (2) modeling seismic CCFs in a fault tree in advance of MCS generation.*

(Results of development) *This study proposes the first method to convert correlated seismic failures into seismic CCFs that are modeled in a fault tree before generating MCSs. It is accomplished by the procedure to (1) calculate all possible combination failure probabilities of correlated seismic failures (p(A),p(B),p(C),p(AB),p(AC),p(BC),p(ABC)) by Reed-McCann or MVN integration, (2) allocate these combination failure probabilities into seismic CCF probabilities (probabilities of Aind, Bind, Cind, CCFAB, CCFAC, CCFBC, CCFABC), (3) insert these seismic CCFs into a fault tree, and then (4) perform usual PSA as internal, fire, and flooding PSA. The main idea of this study is the first method to allocate or convert combination failure probabilities into seismic CCF probabilities.*

(Strength of development) *The dependent or correlated seismic failures exist as AND, OR, or complex logical combinations in MCSs. Since this new method splits correlated seismic failures into seismic CCFs and inserts these seismic CCFs in a fault tree before generating MCSs, the complexity of logical combinations in MCSs does not affect this new method. Thus, this method will allow systems analysts to quantify seismic risk as what they have done with the CCF method in internal, fire, and flooding PSA*

Level 1 and 2 PSA—II

Chair: Gabriel Georgescu *(IRSN)* **Location:** Emerald Salon Three **Time:** 3:30-5:10 pm

3:30 pm: Assessment for SRV Line Break, Ryotaro Sato, Shohei Yamagishi, Shunsuke Tanno, Teruyoshi Sato, Toshiteru Saito, Masayuki Hiraide, Toshinobu Kita *(TEPCO)*

In BWR plants, SRVs discharge lines are linked to the suppression pool to be condensed of the discharged steam in the suppression pool. If one of the SRVs discharge lines is ruptured, accident progression would be more severe. This scenario was not considered or regarded as negligible in the past internal PRAs because the frequency of this scenario was recognized as extremely low. However, the assumption that discharge lines are intact is not guaranteed in some events like seismic event which hazard could be beyond design basis. Seismic PRA is important in Japan because it is believed that the risk of occurrence of earthquakes is relatively high. Therefore, accident progressions for SRV line break have been assessed not only to recognize the completeness uncertainty of SRV line break in internal PRA but also to apply to seismic PRA in the future.

Kashiwazaki-Kariwa NPP Unit 7 (ABWR) is chosen as the target unit. Then, many events of the SRV line break inside the PCV wetwell airspace were classified into several scenarios systematically according to four points based on emergency operating procedures for internal PRA: i) the status and existence of ADS function of the SRV where discharge line had been ruptured at early phase by reactor pressure control procedures, ii) whether stuck open relief valve (SORV) occurred or not, iii) whether SRV had been operated manually or automatically, iv) status of discharge lines when ADS had been operated at late phase by containment control procedures. Next, several scenarios which are identified through above four points of view were calculated by MAAP, and PCV rupture time was calculated as not so severe in ABWR.

In the results, it has been found that SRV line break scenarios would not need to include the internal PRA for an ABWR. Hereafter, it would be expected to pursue considering in the seismic PRA.

3:55 pm: General Screening Criteria for Loss of Room Cooling in PRA Modeling, Joshua Beckton, Carroll Trull *(Westinghouse)*

Probabilistic Risk Assessment (PRA) modeling practices generally tend to reference equipment operating ambient temperature limits (room temperature for example) as the criteria for determining whether or not room cooling is required to maintain equipment operability. If the room temperature is expected to exceed the equipment qualification temperature following loss of room cooling within 24 hours, the mission time typically used in a PRA model, it is assumed that the equipment will fail. This can result in extensive modeling of room cooling (ventilation and air conditioning) and often introduces additional conservatisms into the model. The equipment qualification temperatures are often intended for continual operation of equipment for the qualified life of the component; not a limited high temperature single use. Limited use of equipment at temperatures above the qualification temperature will not necessarily fail the component, but only shorten its qualified life. Including room cooling in the PRA modeling can also introduce circular logic to the model due to additional dependencies and then additional effort is required to address the circular logic.

Discussions within the industry revealed that utilities were taking different approaches to addressing Heating Ventilation and Air Conditioning (HVAC) dependencies ranging from assuming components fail if Equipment Qualification (EQ) temperatures have been exceeded to the development of screening criteria to show that the equipment will either not fail during the PRA mission time or will have some probability of failure rather than assuming certain failure. This paper will discuss different methods of HVAC screening that are currently being used as well as the development of a consolidated industry approach that will be endorsed by the Risk Management Committee of the Pressurized Water Owners Group under project PA-RMSC-1391. In addition, this paper will look at the potential limitations introduced by equipment protection devices such as molded case circuit breakers and motor overload relays following a loss of HVAC support.

Level 1 and 2 PSA—II Continued

4:20 pm: On Assessing the Risk Related To Consequential Steam Generator Tube Rupture Events in Nuclear Power Plants, Ching Hang Ng, Selim Sancaktar *(NRC)*

Accidents involving steam generator (SG) tube rupture can be contributors to plant risk because of their potential for causing a release outside containment (containment bypass sequences). The U.S. Nuclear Regulatory Commission has developed a method to estimate the large early release frequency (LERF) resulting from C-SGTR events; i.e., events in which SG tubes leak or fail as a consequence of the high differential pressures or elevated temperatures after the onset of core damage during an accident sequence. This paper summarizes the premise of the LERF calculation, which is to assess the contribution to the total LERF by the aforementioned accident sequences that are readily-available in the existing risk assessment process. An advantage of this method is that it avoids (or minimizes) modifying the event tree and fault tree models in a PRA, where C-SGTR is not originally modeled. The conditional probability of C-SGTR given a SG tube challenge, p(csgtr), is developed and utilized to estimate the LERF from the core damage frequency (CDF) sequences. The conditional probabilities of C-SGTR are identified the conditional probability of C-SGTR is considerably different between two categories of U-tube type SGs based on geometry of the lower plenum. Once-through SGs are not susceptible to the challenges discussed. In this study, a simplified method to estimate C-SGTR LERF is developed and applied to 20 sample Westinghouse and Combustion-Engineering plants. The study shows that the overall contribution of C-SGTR scenarios to LERF can be significant based on the type of SGs

4:45 pm: Release Category Characterization: Towards a More Realistic Method, Mohamad Ali Azarm *(Innovative Engineering and Safety Solutions, LLC)*

The key characteristics of a radioactive release to environment would be the timing and duration of the release. The magnitude of release and its consequence is dependent on the starting time, the release rate, and the duration of the release. The release characteristics influence the risk results of both the emergency and long-term phases of an accident. The release characteristics are generally estimated using integral codes (e.g., MELCOR) based on the timing associated with a stylized accident. Some degree of conservatism is included in these best estimates; informed by various sensitivity studies and bounding assumptions to account for uncertainties.

The PRAs for current generation of commercial nuclear power plants has shown that the large early releases are the major contributors to the risk of severe accidents. For PWRs, the risk is generally dominated by ISLOCA (interfacing systems Loss of coolant accidents), SGTRs (Steam Generator Tube Rupture) and some early failure modes of containments. The release characteristics can vary among the classes of accidents and even among different accidents within a class of accident. A more specific characterization of different types of releases for dominant risk contributors (LERF) will help the realism of PRA risk evaluation.

This paper focuses on the methods for better characterization of release categories. The generic method discussed, is piloted to an example of consequential SGTR (C-SGTR) scenario. The emphasis of the pilot application is on the probabilistic aspects and not on the underlying accident progression phenomena. The timing and duration of the releases is presented in the form of probability distributions which can be used to define probability of different release categories resulting from the same accident scenario. The results shows the risk associated with LERF scenarios could be determined much more realistically than the traditional methods.

Risk-Informed Regulation—I

Chair: Susan Cooper *(NRC)* **Location:** Yellow Topaz **Time:** 3:30-5:10 pm

3:30 pm: Office for Nuclear Regulation— Risk Informed Regulatory Decision Making, Joshua Gordon, Shane Turner *(ONR)*

The Office for Nuclear Regulation (ONR) is the independent regulator of nuclear safety and security across Great Britain. ONR recently published Risk Informed Regulatory Decision Making, which provides a re-statement of ONR's risk framework for nuclear installations and aims to further clarify the role of hazard and risk and their relationship with good practice in ONR's decision making. The intent is to give consolidated guidance on our risk and decision making approaches thereby supporting an enabling regulatory approach.

ONR's risk based decision making process builds on the document 'Tolerability of risk from nuclear power stations' (TOR), published in 1992. Risk Informed Regulatory Decision Making reinforces the TOR concept and its relationship to the law; describes application of TOR to the specific challenges presented by the nuclear industry; and clarifies how we take account of wider factors in reaching regulatory decisions. With Risk Informed Regulatory Decision Making ONR has consolidated in one place our approaches and thinking on risk and decision making that are embedded in ONR's lower level guidance.

In setting out our risk based framework in Risk Informed Regulatory Decision Making, ONR aims to: set out its approach to the regulation of risk and the philosophy underpinning it;

- *set out the factors that inform its regulatory decisions;*
- *provide reassurance to the public that risks to people are properly addressed, taking due account of the benefits of the activities giving rise to the risk; and*
- *inform other regulators, whose responsibilities include regulating nuclear sites for matters other than safety and security, about the basis for the management of health and safety risks from work activities, thereby helping to promote consistency of decision making amongst regulators.*

The paper summaries ONR's approach to risk informed decision making, the factors it applies and provides examples of its application.

Risk-Informed Regulation—I Continued

3:55 pm: Assessing the Impact of TSTF 505 Initiative 4b Risk-Informed Completion Times on Baseline Risk, Antonios M. Zoulis *(NRC)*

Many U.S. nuclear power plant licensees are adopting risk-informed initiatives such as 10 CFR 50.69 "Risk-informed Categorization and Treatment of Systems, Structures and Components of Nuclear Power Plants," and Technical Specification Task Force (TSTF) Traveler 505 "Risk Initiative 4b - Risk Informed Completion Times." This paper compares the impact from the potential for increase allowed outage times (AOTs) that may result when a licensee adopts risk-informed completion times or RICTs. The analysis was performed on various plant designs and based on a sample of licensees that have or will adopt TSTF 505. The analysis was performed using the Standardized Plant Analysis Risk (SPAR) Models used by the Nuclear Regulatory Commission (NRC), and developed and maintained by the Idaho National Laboratory (INL). The existing at-power SPAR models were modified to assess the impact of extended AOTs by conducting sensitivity studies on existing test and maintenance terms found in the models. This paper provides an overview of TSTF 505 and presents the results and changes observed on the licensee's baseline core damage frequency.

4:20 pm: Insights From Review of Seismic Probabilistic Risk Assessments in the Context of 10 CFR 50.69, Shilp Vasavada, Mehdi Reisi-Fard *(NRC)*

Nuclear power reactor licensees that are regulated by the U.S. Nuclear Regulatory Commission (NRC) have the option of voluntarily adopting the regulation in Part 50.69 to Title 10 of the Code of Federal Regulations, "Risk-informed categorization and treatment of structures, systems, and components for nuclear power reactors" (hereafter referred to as 10 CFR 50.69). 10 CFR 50.69 allows categorization of structures, systems, and components (SSCs) as either high safety significance (HSS) or low safety significance (LSS). Certain regulatory requirements can be reduced for LSS components that were previously considered to be safety-related. The volume of applications from licensees seeking to adopt 10 CFR 50.69 has increased recently. The NRC staff has been actively involved in detailed reviews of seismic probabilistic risk assessments (SPRAs), which are increasingly being used by licensees in the categorization of SSCs per the requirements in 10 CFR 50.69. This paper will present insights gained from the NRC staff's review of SPRAs in the context of 10 CFR 50.69. Discussion will include insights on fragility evaluation, determination of importance measures across seismic 'bins', mapping of components between SPRAs and PRAs for other hazards, and the relation between reduced regulatory requirements allowed by 10 CFR 50.69 and SPRA maintenance.

4:45 pm: Risk-Deformed Regulation: What Went Wrong With NFPA 805, Raymond HV Gallucci *(Retired)*

The voluntary adoption of National Fire Protection Association (NFPA) Standard 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition), has been celebrated by the regulatory community and most adopters as a major step forward in establishing risk-informed regulation as a framework for this type of licensing basis (a few adopters have been less than enthusiastic, claiming unexpectedly high expenses and inordinate delays and hurdles to finally make it work). What has not been visible are the convolutions that were involved in achieving this "success" – these are the subject of this paper. Nonetheless, this paper does not disparage nuclear power itself since NFPA 805 has enabled plants to be at least as safe as, if not safer than, prior to transition. Even plants that made no change would have at least assessed their fire risks and become more knowledgeable of potential weaknesses that could compromise safety. Having found none, they would not have the need for changes. Plants that made effective changes may be safer than before. What this paper is aimed at are those who believe the means by which the transitions have been "risk-justified" are as important as the transitions themselves. It reveals the "compromises" allowed by the NRC, and the "short-cuts" and "deviations" taken by the nuclear industry, to fulfill the promise of a "sea change" in fire protection at nuclear power plants through risk-informed, performance-based regulation. Even if no diminution of safety occurred, it is possible there were missed opportunities to improve safety if changes might have been made, or different changes substituted for those that were made, if not for these "compromises," "short-cuts" and "deviations."

NFPA 805 was written for use after a risk-informed, performance-based fire protection program had been established. It was not specifically intended to be the mechanism by which this transition took place, although it clearly offered guidance that could be used in this regard. Adopting it as the standard for the actual transition was a choice made by the NRC, with strong encouragement from the nuclear industry, through 10CFR50.48(c) and interpreted via Regulatory Guide (RG) 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (Rev. 1). Via RG 1.205, the NRC created a "transitional plant change evaluation" which it termed a "fire risk evaluation" to represent this same type of post-establishment change analysis to be performed for the transition itself. Never addressed was the question why, when this process was clearly intended to apply after establishing a risk-informed, performance-based fire protection program, was there even a need to perform some sort of "risk comparison" to judge the "propriety" of the final configuration. As has often been the case, the NRC sought to link a risk-informed process to RG 1.174, An Approach for Using Probabilistic Risk Assessment [PRA] in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis. Although only mentioned as an "example approach for acceptance criteria for changes in risk from a 'plant change'" in the appendix section of NFPA 805 (not endorsed in 10CFR50.48[c]), RG-1.174 was deemed as the appropriate guidance by which to determine acceptability during transition. However, since NFPA 805 cited this in connection with a "plant change," a post-established fire protection program activity, it was intended to be applied after transition, not during. Nonetheless, the NRC arbitrarily chose this to apply during transition, not recognizing, or else choosing to overlook, the complications that would ensue.

From this "fundamental flaw" stemmed the "convolutions" that, in the author's opinion, turned the NFPA-805 transition process into more of a "risk-deformed" rather than "risk-informed" example of regulation, tainted by the series of "compromises," "short-cuts" and "deviations" cited above, including, but not limited to (1) self-serving claims by the nuclear industry that "fire PRA is too conservative," particularly exemplified by two Frequently Asked Questions (FAQs) related to fire ignition frequencies and credit for very early warning fire detection; (2) effectively an NRC philosophy that "rejection is not an option," as exemplified by (i) a compromise that enabled transitions based on promises rather than pre-transition plant changes that, while intended to be an "exception" during the pilot process, became the "rule" for most subsequent non-pilot transitions, and (b) "relaxation" of the criteria for extending enforcement discretion during transition for adopters that failed to show a minimum of progress required to retain this discretion; (3) a debate that continues still over the regulatory acceptability of "unreviewed analysis methods (UAMs)" including the conversion of a UAM "review" panel into a "development" panel, particularly for a UAM for which only a subjective, not phenomenological, basis was pursued; (4) regulatory acceptance of claims of "embedded fire PRA conservatism" that prompted undue "stretching" of the acceptance criteria of RG 1.174 in lieu of firm demonstration or calculational removal of such "conservatism," just to avoid extension of approval deadlines; and (5) finally, NRC's adoption of a philosophy to limit its staff's review scope in an effort to reduce Requests for Additional Information (RAIs) as a result of industry political pressure, as exemplified by (i) a "freeze point' approach to changes in ongoing submittals and (ii) acceptance of analyses for which the required experimental or operational confirmation of the calculational justification remained pending.

Risk-Informed Regulation—I Continued

This paper was prepared by a former employee of the U.S. NRC. The views presented do not nor ever did represent an official NRC position, only those of the author. The author was the first to receive a PhD in nuclear engineering based on a thesis specifically related to fire PRA in power plants and worked for over 35 years in nuclear risk, reliability and safety analysis under both governmental and commercial auspices. He was hired at the NRC in 2003 as the expert in fire PRA for the Office of Nuclear Reactor

Regulation (NRR) and participated in the NFPA-805 program from the start of the pilot process in 2005 until being "phased out" in mid-2014. The perspectives here cover that approximately nine-year time period, with some extended time specific to issues that stemmed from this earlier time period.

Low Power Risk, Accident Management and Emergency Planning

Chair: Bruce Morgen *(EPM Inc.)* **Location:** Opal One **Time:** 3:30-5:10 pm

3:30 pm: Power Restoration Timescales and Probabilities: New Data and a General Theory, Romney B Duffey (DSM Associates, Inc.)

We predict the power restoration timing to facilities including the impacts of catastrophic damage ocurring both onsite and off site. This is relevant to probabilistic assessments of core damage and activity release timing, mitigation action success likelihood, back-up power equipment deployment, and long term passive or active cooling requirements We have already shown the probability of non-restoration was significantly greater, or took much longer for an unexpected extreme storm attributable to the extensive damage, widespread social disruption and overloaded emergency response capability. Using extensive new data, we determine the power recovery timescales, probability and rate for recent massive power outage events due to unexpected major extreme events (storms, hurricanes, fires and floods). These are the types and range of events that are included in current nuclear regulatory and licensing guidelines for deploying back up or alternative power and cooling systems.

Independent of event type, the system repair, replacement and restoration trends share the same fundamental physical, psychological and statistical learning curve behavior inherent in all human decision-making. Despite their totally disparate origins, and urban, rural and island settings, over three orders of magnitude severe wildfires, storms and hurricanes have the same non-restoration probability trends of simple exponential form. The results fall into categories that are well described by new and theoretically based original correlations for use in outage duration risk assessment, dependent on and grouped by the increasing degree of power system and societal damage, and the resulting emergency response disruption. The available nuclear plant data for station blackout (SBO) and Fukushima are shown to follow the same trends. Predictions for the recovery of power in severe events show the non-recovery of power has a probability of 0.8 and 0.5 after 24 and 72 hours, respectively.

Using recent NRC analyses and industry approaches, the influence of multiple back up FLEX or alternate Emergency Power Systems (EPS) has been examined and quantified, and an explicit expression derived for the modified dynamic probability and frequency of extended loss of power. The theory agrees with the trends from the actual Fukushima event data. With EPS, the probability of extended loss of power is predicted to be less than 0.38 at 72 hours, depending on the known restoration and assumed EPS failure rates.

3:55 pm: Low Power Shutdown PRA Modelling Challenges and Recommendations, Garill Coles, Steve Short *(PNNL)*

In 2017, Pacific Northwest National Laboratories (PNNL) coordinated and facilitated for the Nuclear Regulatory Commission (NRC) a study involving expert elicitation to determine priorities associated with performing low power shutdown (LPSD) Probabilistic Risk Assessments (PRAs). PNNL developed and implemented an expert elicitation Phenomena Identification Ranking Technique (PIRT) process to prioritize plant operating states, hazards, and outage types that should be included in a full-scope nuclear power plant Level 3 LPSD PRA. This work, which has been previously reported on, will be documented in a NRC NUREG/CR report. The report will focus on the expert elicitation process that PNNL developed and the resulting plant operating state, outage type, and hazard priorities that were identified by exercising the process with group of LPSD risk experts. However, another benefit of the work, which we highlight in this paper, is the identification of different LPSD PRA modeling challenges that surfaced in discussions with the experts during the expert elicitation sessions.

Though not the focus of the LPSD expert elicitation that was performed, identification of these generic LPSD PRA modeling issues represent significant insights in-and-of themselves. Even if a LPSD PRA is performed to consider the risk-significant plant operating states, outage types and hazards, these modelling challenges are concerning because the lack data, and lack of analyses and studies associated with LPSD configuration and conditions can lead to skewed insights. During the final phase of the LPSD expert elicitation, PNNL facilitated a brainstorming session with the LPSD experts on LPSD PRA to capture modelling issues that surfaced during the expert elicitation process but were not formally documented in the elicitation worksheets. Accordingly, this paper focuses on the side-benefit of holding this brainstorming exercise during the LPSD expert elicitation process performed in 2017 by highlighting the PRA modeling challenges that were identified by the LPSD expert panel as well as the suggestions made by the expert panel for addressing these issues. This paper summarizes those modeling challenges and recommendations, and provides some assessment by PNNL about which suggestions may have the most potential future benefit for LPSD PRA.

4:20 pm: NuScale's Emergency Planning Zone Methodology, Scott J. Weber, Sarah Bristol, Jeremiah Doyle, Bill Galyean, Luke McSweeney, Kent Welter, Cindy Williams *(NuScale Power, LLC)*

NuScale Power, LLC has developed a methodology to determine the size of the plume exposure pathway emergency planning zone (EPZ) for a NuScale plant. The methodology is risk-informed and consequence based. It utilizes risk insights including core damage frequency (CDF) screening limits, as well as a quantitative defense-in-depth assessment. The methodology considers accident sequences from all hazards and all operating modes. It also contains an integrated assessment of multi-module effects and an uncertainty analysis.

An example of EPZ size determination is provided, showing how NuScale's full methodology would be applied using publically available PRA results and hypothetical source terms. This example demonstrates the feasibility of a site boundary plume exposure EPZ, made possible by the increased safety and smaller radionuclide inventory associated with a NuScale Power Module.

Low Power Risk, Accident Management and Emergency Planning Continued

4:45 pm: **Flex Equipment Reliability Data**, Roy Linthicum *(Exelon)*, Michael Powell *(APS)*

Post Fukushima, Utilities have invested significant resources in procuring Flex Equipment and developing guidance for using the equipment for beyond design basis external hazards. NEI 16-08 "Guidance for Optimizing the Use of Portable Equipment" urges utilities to leverage this investment by using Flex or other portable equipment to provide additional safety benefits. These safety benefits can quantified by including Flex equipment in the site specific PRA models. This can provide additional margin for various risk informed applications, such as TSTF-505 "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b", Significance Determination Process evaluations and the Mitigating Systems Performance Index. Modeling Flex equipment in utility PRA models requires the development of reliability data, which is currently unavailable. The PWROG, with support form the BWROG, is currently developing failure data for the most commonly credited Flex equipment. This paper will summarize the results of this evaluation including the approach used in developing the data, component boundaries, failure definitions, sources of uncertainty as well as the failure rates. Finally, lessons learned from the development of the data will be reported.

PRA Standard Update–Panel

Chair: Andrea Maioli *(Westinghouse)* **Location:** Blue Topaz **Time:** 3:15-5:15 pm

The Probabilistic Risk Assessment (PRA) Standard from the American Nuclear Society (ANS) and American Society of Mechanical Engineers (ASME) Joint Committee on Nuclear Risk Management (JCNRM) has been undergoing a significant update following the release of Addendum B (RA-Sb-2013) in 2013 and the seismic specific Code Case in 2017. For the first time since the release of Addendum A in 2009 (RA-Sa-2009), the next edition of the Standard will be endorsed by the United State Nuclear Regulatory Commission (USNRC) via revision 3 of USNRC Regulatory Guidance1.200.

The PRA Standard went through significant modifications in the last few years, including the elimination across the board of Capability Category III requirements (which resulted in some of those requirements being moved into Capability Category II), the elimination of entire Technical Element (TE) such as SF and UNC in Part 4, and significant restructuring of other TEs. The future edition of the Standard is striving to reach consistency on key aspects that have implications that are cross-cutting among the various hazard and parts, such as consistent criteria for screening and a consistent use of the concept of risk-significant.

Individual parts were also updated to reflect technical advancements in the respective areas. Most notably, Part 7 and 8 on High Wind PRAs and External Flooding PRAs are being significantly expanded. An explicit technical element has also been added to address the Configuration Control of the PRA to be used for Risk-Informed Applications.

This panel session provides an overview of the most significant upcoming changes in the next edition of the Standard, along with the inevitable challenges that still lay ahead. The panelists have been leading the respective working groups in this endeavor, now more than five years in the making.

Panelists: Paul J. Amico *(JENSEN HUGHES)*
Vince Anderson *(JENSEN HUGHES)*
Francisco Joglar (*JENSEN HUGHES)*
Larry A. Twisdale *(ARA)*
Shelby Bensi *(Univ of Maryland)*

State-of-the-Art Consequence Analysis (SOARCA) Uncertainty Analysis

Cochairs: Jeff Gabor *(JENSEN HUGHES)*, Carl Mazzola *(PEC)* **Location:** Emerald Salon One **Time:** 10:00 am-12:00 pm

10:00 am: State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analyses for Station Blackout Scenarios, S. Tina Ghosh, Hossein Esmaili, Alfred Hathaway *(NRC)*, Nathan Bixler, Dusty Brooks, Matthew Dennis, Douglas Osborn, Kyle Ross, Kenneth Wagner *(SNL)*

This paper provides an introduction to a special session the State-of-the-Art Reactor Consequence Analyses (SOARCA) Uncertainty Analyses (UAs), and provides an overview of the NRC's project to develop a technical report summarizing the most important insights from the three SOARCA UAs.

The U.S. Nuclear Regulatory Commission (NRC) with Sandia National Laboratories has completed three UAs as part of the SOARCA program. The SOARCA UAs included an integrated evaluation of uncertainty in accident progression, radiological release, and offsite health consequence projections. The UA for Peach Bottom, a boiling-water reactor with a Mark I containment located in the State of Pennsylvania, analyzed the unmitigated long-term station blackout SOARCA scenario. The UA for Sequoyah, a 4-loop Westinghouse pressurized-water reactor (PWR) located in the State of Tennessee, analyzed the unmitigated short-term station blackout SOARCA scenario, with a focus on issues unique to the ice condenser containment and the potential for early containment failure due to hydrogen deflagration. The UA for Surry, a 3-loop Westinghouse PWR with subatmospheric large dry containment located in the State of Virginia, analyzed the unmitigated short-term station blackout SOARCA scenario including the potential for thermally-induced steam-generator tube rupture. These three UAs are currently documented in three reports, encompassing a total of 1800+ pages.

The NRC is currently developing a technical report summarizing the most important insights from the three SOARCA UAs. The purpose of this summary is to provide a useful reference for regulatory applications that require the evaluation of offsite consequence risk from beyond design basis event severe accidents. Examples include regulatory and cost-benefit analyses that rely on offsite consequence projections using the MACCS code, in conjunction with either MAAP or MELCOR.

The SOARCA studies to date have assisted the regulatory evaluation of a variety of issues including the disposition of post-Fukushima Daichi accident regulatory initiatives and the consideration of emergency planning zone size. The SOARCA studies have also informed other NRC research projects, such as the NRC's Spent Fuel Pool Study published in 2013, and the NRC's on-going Site Level 3 Probabilistic Risk Assessment project.

10:30 am: State-of-the-Art Reactor Consequence Analyses Project Uncertainty Analyses: Insights on Accident Progression and Source Term, S. Tina Ghosh, Hossein Esmaili, Alfred Hathaway *(NRC)*, Nathan Bixler, Dusty Brooks, Matthew Dennis, Douglas Osborn, Kyle Ross, Kenneth Wagner *(SNL)*

This paper is the second proposed paper in a special session the State-of-the-Art Reactor Consequence Analyses (SOARCA) Uncertainty Analyses (UAs), and summarizes accident progression and source term insights from the three SOARCA UAs.

The U.S. Nuclear Regulatory Commission (NRC) with Sandia National Laboratories has completed three UAs for particular station blackout scenarios as part of the SOARCA program: for a boiling-water reactor with a Mark I containment (Peach Bottom), for a pressurized-water reactor (PWR) with an ice condenser containment (Sequoyah), and for a PWR with subatmospheric large dry containment (Surry). The Sequoyah and Surry SOARCA UAs focused on an unmitigated short-term station blackout (SBO) scenario involving an immediate loss of offsite and onsite AC power. In the Surry UA, induced steam generator tube rupture was also modeled. The Peach Bottom UA focused on an unmitigated long-term SBO scenario, where battery power is initially available. The MELCOR code was used for accident progression and radiological release modeling. MELCOR models the following:

- *Thermal-hydraulic response in the reactor coolant system, reactor cavity (below the reactor vessel), containment, and confinement buildings (e.g., shield building);*
- *Core heatup, degradation (including fuel cladding oxidation, hydrogen production, and fuel melting), and relocation;*
- *Core-concrete interaction in the cavity after lower reactor vessel head failure;*
- *Hydrogen production, transport, combustion, and mitigation; and*
- *Fission product transport and release to the environment.*

Key input parameters were selected to represent uncertainty in the MELCOR modeling and distributions were assigned. Uncertainty was propagated through Monte Carlo simulation using hundreds of samples of uncertain parameter values.

This paper presents the cesium and iodine release results for the three SOARCA UAs and summarizes some of the important insights and features of the analyses. For example, the performance of passive safety relief valves (SVs) were shown to be important in all three UAs for the unmitigated SBOs modeled, but for different reasons and with different effects on the accident progression. For example, in the Peach Bottom UA, SV behavior strongly influences whether main steamline rupture occurs, which leads to higher releases since the scrubbing benefits of the wetwell are lost in those scenarios. In the Surry UA, SV behavior strongly influences whether steam generator tube rupture is induced, which leads to higher releases since the containment is bypassed in these scenarios. In the Sequoyah UA, SV behavior strongly influences hydrogen generation and migration and whether early containment failure is possible. Considerable uncertainty remains in the distributions of these key safety valve parameters in the current state-of-knowledge.

11:00 am: State-of-the-Art Reactor Consequence Analyses Project Uncertainty Analyses: Insights on Offsite Consequences, S. Tina Ghosh, Hossein Esmaili, Alfred Hathaway *(NRC)*, Nathan Bixler, Dusty Brooks, Matthew Dennis, Douglas Osborn, Kyle Ross, Kenneth Wagner *(SNL)*

This paper is the third proposed paper in a special session the State-of-the-Art Reactor Consequence Analyses (SOARCA) Uncertainty Analyses (UAs), and summarizes offsite consequence insights from the three SOARCA UAs.

The U.S. Nuclear Regulatory Commission (NRC) with Sandia National Laboratories has completed three UAs for particular station blackout scenarios as part of the SOARCA program: for a boiling-water reactor with a Mark I containment (Peach Bottom) located in the State of Pennsylvania, for a pressurized-water reactor (PWR) with an ice condenser containment (Sequoyah) located in the State of Tennessee, and for a PWR with subatmospheric large dry containment (Surry) located in the Commonwealth of Virginia. The Sequoyah and Surry SOARCA UAs focused on an unmitigated short-term station blackout (SBO) scenario involving an immediate loss of offsite and onsite AC power. In the Surry UA, induced steam generator tube rupture was also modeled. The Peach Bottom UA focused on an unmitigated long-term SBO scenario, where battery power is initially available. The MELCOR Accident Consequence Code System (MACCS) suite of codes was used for offsite radiological consequences modeling. MACCS models the following:

State-of-the-Art Consequence Analysis (SOARCA) Uncertainty Analysis Continued

- *Atmospheric transport and deposition of radionuclides released to the environment;*
- *Emergency response and long-term protective actions;*
- *Exposure pathways;*
- *Acute and long-term doses to a set of tissues and organs; and*
- *Early and latent health effects for the affected population resulting from the doses[1].*

Key input parameters were selected to represent uncertainty in the MACCS modeling and distributions were assigned. Uncertainty was propagated through Monte Carlo simulation using hundreds of samples of uncertain parameter values and sampled source terms resulting from the preceding MELCOR analyses.

This paper presents the offsite consequence results, individual latent cancer fatality (LCF) risk and the individual early fatality risk, for the three SOARCA UAs and summarizes some of the important insights and features of the analyses. For example, the early fatality risks are essentially zerio in all three UAs. Long-term exposures are generally more important than short-term (emergency-phase) exposures, indicating that LCF risks are largely controlled by the choice of habitability dose criterion (which can differ by State). Risks within 16 km (10 miles) are slightly lower than those within larger circular areas (except for the 5th percentile results) because the modeled evacuation is effective and most of the emergency planning zone (EPZ) residents receive little or no dose during the emergency phase. Uncertain parameters in both the upstream MELCOR modeling and the MACCS modeling are demonstrated in the integrated UAs to be influential to uncertainty in offsite consequences.

[1]MACCS also models economic and societal consequences such as the population subject to protective actions, however, these were not used in the SOARCA project.

11:30 am: State-of-the-Art Reactor Consequence Analyses Project Uncertainty Analyses: Insights on Methodologies, S. Tina Ghosh, Hossein Esmaili, Alfred Hathaway (NRC), Nathan Bixler, Dusty Brooks, Douglas Osborn, Kyle Ross, Kenneth Wagner *(SNL)*

This paper is the fourth proposed paper in a special session on the State-of-the-Art Reactor Consequence Analyses (SOARCA) Uncertainty Analyses (UAs), and summarizes insights and lessons learned on the application of UA and regression methodologies from the three SOARCA UAs.

The U.S. Nuclear Regulatory Commission (NRC) with Sandia National Laboratories has completed three UAs for particular station blackout scenarios as part of the SOARCA program, as described in the previous three papers. Each of the SOARCA UAs was an integrated analysis of epistemic parameter uncertainty associated with the accident progression and offsite consequence modeling implemented with the MELCOR and MACCS suite of codes respectively.

The UAs used a two-step Monte Carlo simulation process. Simple random sampling was chosen for the MELCOR calculations. From the completed MELCOR realizations, a family of source term results was produced. Either simple random sampling and/or Latin hypercube sampling was used for MACCS, with a sample size to match the number of source terms. The pro's and con's of the two sampling approaches are discussed in this paper, for example, with respect to representation of the output distributions and stability analyses.

Four regression techniques were used for post-processing results, to estimate the importance of the input parameters with respect to the uncertainty in source terms and consequences: linear rank regression, quadratic regression, recursive partitioning, and multivariate adaptive regression splines (MARS). This analysis provides measures of the effects of the selected uncertain parameters both individually and in interaction with other parameters (the Ti index in the three more advanced methods), and helps:

- *Identify which uncertainty in important parameters and phenomena are driving the variability in model results.*
- *Verify and validate the SOARCA model through exploration of unexpected or non-physical phenomena in the distributions of results.*
- *Provide an assessment of the regression techniques and uncertainty analysis approach.*
- *Provide a basis for future work.*

This paper discusses some of the challenges that were encountered in employing the regression techniques, for example, in handling dependent input variables, and the team's approach to overcoming these challenges.

The SOARCA UAs represent a first-of-a-kind analysis in its integrated look at uncertainties in MELCOR accident progression and MACCS offsite consequence analyses. As such, an additional objective of the work was to demonstrate uncertainty analysis methodology that could be used in future combined Level 2/3 probabilistic risk assessment (PRA) and probabilistic consequence analysis studies. This paper also discusses insights and lessons learned with regard to the potential application of the SOARCA UA methodologies and approaches in such future studies.

Dynamic PSA—I

Chair: Tunc Aldemir *(Ohio State)* **Location:** Emerald Salon Two **Time:** 10:00am-12:00 pm

10:00 am: Integration of Recoveries Into Dynamic Event Trees: A Case Study, Claudia Picoco *(Ohio State)*, Valentin Rychkov *(EdF)*, Tunc Aldemir *(Ohio State)*

Dynamic Event Tree (DET) is a methodology used for probabilistic safety assessment (PSA) of nuclear power plants. The DET approach differs from the traditional static methods such as the event-tree (ET) / fault-tree (FT) by explicitly accounting for the time variable in system evolution through a plant simulator. DET allows to model the interactions among stochastic events (such as failure of safety systems) and the dynamic evolution of the plant as a consequence of these events.

The capability for accounting for system recoveries is a consequence of the explicit time modeling within DET approach. Repairs are an important aspect in PSA, however, their modeling has always been an important challenge (and limitation) for the traditional ET/FT approach. When multiple failures and recoveries of plant systems are considered, the thermal-hydraulic modeling o*f plant evolution is not quite straightforward since physical dependencies among system variables have to be taken explicitly into account when needed. That cannot be realistically represented with the ET/FT approach. For example, if a recovery of a safety injection pump is represented within a branch, the pump cannot be switched on if the failure of the diesel generator might have occurred in the meantime. The failure time of a diesel generator is a stochastic variable, thus occurrence of one stochastic event (pump recovery) must be conditioned to another stochastic condition (a diesel generator state). Modeling multiple failure/recoveries can be even more challenging, especially if dependent upon the state of other plant systems.*

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Dynamic PSA—I Continued

This paper presents the case study of a DET, generated by the RAVEN/MAAP5-EDF code pair for a pressurized water reactor with multiple recoveries integrated within the analysis. The initiating event is a loss of offsite power. Possible recoveries are considered as branching conditions for some systems (such as diesel generators and high pressure injection) while for other systems (e.g., seal cooling) only the failure is considered. Result presentation will include branches and histories generated, evolution of the main physical variables along the different sequences, evaluation of the consequences and the conditional probability of core damage. Furthermore, perspectives and lessons learned for DET applications will be pointed out.

10:30 am: India-United States Collaboration on Advanced Dynamic Reliability Modeling, Curtis L. Smith *(INL)*, John Arul *(IGCAR)*, Gopika Vinod *(BARC)*, Darpan Krishnakumar Shukla *(IGCR)*

The countries of India and the United States have an ongoing collaboration under a Civil Nuclear Energy Working Group (CNEWG) arrangement. Within this collaboration, several areas of joint research have been proposed including focusing on the civilian nuclear energy sector in both countries and on the sharing of best practices in safety. A key focal point of the safety collaboration is on advanced modeling and simulation. To promote this area, our three organizations (INL, BARC, and IGCAR) have proceeded on research and development collaboration of advanced PSA methods in order to develop and apply advanced methods to plant design safety optimization and operational decision making. The collaboration between the U.S and India promotes development, validation, and application of advanced and innovative approaches to modeling and assessment. Included in our collaboration are interactions between the experts in both the countries through e-mail, video teleconferencing, and in the working group meetings. During the course of the collaboration, it was noted that the INL has developed an advanced simulation framework and tools for dynamic risk and reliability assessment under the Department of Energy Risk-Informed Safety Margins Characterization (RISMC) Pathway. The tool set consists of finite- element and mesh-free modules for physics, the RAVEN software for dynamic scenario generation and Monte-Carlo calculations, and a state-based dynamic risk analysis software called EMRALD. As part of the collaboration, work was initiated by IGCAR in collaboration with INL to develop a dynamic reliability approach referred to as Smart Component Method (SCM) using an Object Oriented architecture for safety system representation and with compatible Monte-Carlo simulation. Currently, the collaboration team is focusing on demonstrating these approaches using a joint test case problem. The results of this demonstration will be described, including lessons learned, in the paper.

11:00 am: Code Surrogate Development for Dynamic PRA, Robby Christian, Hyun Gook Kang *(RPI)*

The draft rule 10 CFR 50.46c regulates Emergency Core Cooling System (ECCS) criteria for any fuel or cladding types in Light Water Reactors (LWRs). The ECCS criteria for Zr cladding has been established to limit oxidation and loss of ductility. Silicon Carbide however is a ceramic material that lacks ductility. Therefore, the ECCS performance criteria to ensure core safety during Loss of Coolant Accident (LOCA) should be determined differently than the ones used for ductile Zircaloy cladding.

This work reviewed melting and tensile fracture as the clad's deterministic and stochastic failure modes respectively, which limits ECCS performance. The clad structure in this study is a triplex structure of CVD, CVI matrix composite, and an outer CVD barrier layer. The clad failure probability was formulated in a Boolean combination of individual layer's failure probability, and was computed by coupling a PWR 1000 MWe RELAP5 model with an in-house fuel performance assessment code. Because of its stochastic nature of failure, the clad's failure probability was numerically integrated over the range of ECCS performance to estimate the total failure probability due to ECCS uncertainties. The ECCS uncertainties investigated in this study were mass flow rates and actuation timing of active injection systems. To reduce computational cost, a novel method to interpolate clad performance throughout the ECCS uncertainty range was designed. This method clustered sampled data by using a feature-based Dynamic Time Warping and Total Variation Regularization algorithm. Interpolation of clad performance in each cluster was done with a reduced-order Taylor Kriging method. The proposed method was able to generate a full-rank response surface of SiC clad failure probability over the uncertainty range of ECCS performance.

Results showed that SiC clad failure probability spiked less than a minute after a design LBLOCA accident when the current Zr-4 ECCS criteria is maintained. However, it still provides an increased safety margin of 15 to 16 order of magnitude compared to Zr-4. This positive margin was used to relax active ECCS requirement. It was found that active injection flow rate could be reduced up to 10% of its maximum rated flow, or the active ECCS actuation could be delayed up to 420 seconds after accident initiation while still maintaining a positive safety margin. These ECCS criteria relaxation may reduce costs for new reactors, or extend operational life of aged reactors.

11:30 am: Applications of Evidence Theory to Issues with Nuclear Weapons, John L. Darby *(SNL)*

Over the last 13 years, we have applied the belief/plausibility measure from evidence theory to estimate the uncertainty for numerous safety and security issues for nuclear weapons. For such issues we have significant epistemic uncertainty and are unable to assign probability distributions. For example, using a Bayesian approach we cannot assign a reasonable prior and we have insufficient information to update a poor prior to generate an accurate posterior. We have developed and applied custom software to implement the belief/plausibility measure of uncertainty. For safety issues we perform a quantitative evaluation, and for security issues (e.g., terrorist acts) we use linguistic variables (fuzzy sets) combined with approximate reasoning.

In application we perform the following steps:

- *Train the subject matter experts (SMEs) on the assignment of evidence*
- *Work with the SMEs to identify the concern(s): the top-level variable(s)*
- *Work with the SMEs to identify lower-level variables and their functional relationship(s) to the top-level variable(s)*

Then the SMEs gather their State of Knowledge (SOK) and assign evidence to the lower-level variables.

Using this information, we evaluate the variables using custom software and produce an estimate for the top-level variable(s) including uncertainty. For example, if the top-level variable is the probability of failure of a critical component in an abnormal environment we estimate that probability as within a belief to plausibility interval. Belief and plausibility are lower and upper bounds, respectively, on probability.

We will summarize evaluations for both safety and security, and our custom software used to perform the evaluations.

[1]Sandia National Laboratories is a multimission laboratory managed and operated by National Technology & Engineering Solutions of Sandia, LLC, a wholly owned subsidiary of Honeywell International Inc., for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-NA0003525.

Level 1 and 2 PSA—III

Chair: John E. McAllister *(HukariAscendent)* **Location:** Emerald Salon Three **Time:** 10:00 am-12:00 pm

10:00 am: **Extension of a Level 2 PSA Event Tree Based on Results of a Probabilistic Dynamic Safety Analysis (Dynamic PSA) of Induced Steam Generator Tube Rupture (SGTR)**, Sören Johst, Michael Hage, Jörg Peschke *(GRS)*

This paper presents the approach of extending a classical event tree of Level 2 PSA (Probabilistic Safety Analysis) by results of a probabilistic dynamic safety analysis (Dynamic PSA). The example of creep-induced steam generator tube ruptures (SGTR) has been chosen to extend an event tree considering not only operational states and internal events, but also shutdown states, external events and human actions with the corresponding functions and branches. The event tree processor EVNTRE (developed by Sandia National Laboratories (SNL), USA) has been used due to its flexibility in generating different and comprehensive, event trees as the structure of the event tree is directly inputted into its source code.

ATHLET-CD / MCDET results by GRS simulation (Analysis of Thermal hydraulics of LEaks and Transients – Core Degradation / Monte Carlo Dynamic Event Tree) analyzing the failure of reactor coolant system components (hot leg / surge line) by creep in a scenario of a high-pressure core melt accident in a generic pressurized water reactor (PWR) have been implemented into the event tree.

The scenario starts with a station blackout (SBO) under the assumption that the severe accident management (SAM) measure "Primary Side Depressurization" of the reactor circuit is not available. SAM measure "Secondary Side Depressurization" is performed; however, without a subsequent steam generator feed. Amongst others, three aleatory parameters have been varied: the point of time of the secondary side depressurization, the extent of pre-existing steam generator tube damages and the point of time of the failure of the pressurizer valves during normal operation. Particularly, the choice of these parameters determines which components of the reactor coolant system will fail first and/or if a SGTR will be induced.

From the results of these simulations a set of parameters has been extracted and integrated into the event tree (along with their probability distributions). The effect of these parameters on both the progression of the severe accident sequence under consideration and the release categories of an exemplary German PWR plant, respectively, is discussed in this paper.

10:30 am: **Source Term Analysis for PWR ISLOCA Using MAAP5**, Paul McMinn, Chan Y. Paik *(Fauske and Assoc, LLC)*, Kwang-Il Ahn, Soo Yong Park *(KAERI)*, Keo Hyoung Lee *(FNC Technology Co., Ltd.)*, Seok-Won Hwang *(Central Research Inst of Korea Hydro & Nuclear Power Co., Ltd.)*

An Interfacing Systems Loss of Coolant Accident (ISLOCA) is a low probability accident initiator that results in containment bypass. This can lead to a significant release of fission products from containment which are typically assumed to flow directly into the environment. However, there are a substantial number of structures along the flow path to the environment which promote the deposition and retention of fission product aerosols. This paper discusses an improved modeling approach for source term quantification for ISLOCA sequences in a Pressurized Water Reactor (PWR). The approach quantifies the retention of fission products in the piping connecting the primary system to the auxiliary building, the possibility of scrubbing fission products in an accumulated water pool in the auxiliary building, and the deposition of fission products within the auxiliary building. The impact of these retention mechanisms is found to have a significant influence on the magnitude and timing of fission product releases to the environment for a typical 1300 MWe PWR. This paper discusses the modeling methodology used to quantify fission product source terms for an ISLOCA sequence and identifies the key plant features which influence the result. Specific attention is given to thermophoretic deposition, eddy impaction within turbulent pipe flows, aerosol scrubbing in water pools near saturation conditions, deposition within ventilation pathways in the auxiliary building, and the potential for revaporization of deposited fission products. Results are compared to demonstrate the impacts of analyzing the sequences with models of varying detail for piping deposition, pool scrubbing, and auxiliary building modeling. Piping deposition results are shown for a model that does not consider piping deposition, a simple pipe model with only turbulent deposition considered, and a detailed pipe model that considers thermophoretic and turbulent deposition. Pool scrubbing results are shown for releases that are not scrubbed by an overlying pool and releases that consider scrubbing in an overlying water pool. Auxiliary building deposition results are shown for an auxiliary building model with a simplified consideration of ventilation pathways and for an auxiliary building model with detailed consideration of auxiliary building ventilation pathways. Best practices are identified for the representation of plant structures as they pertain to fission product aerosol dynamics.

11:00 am: **Recent Developments on a Level 1 PSA for a Research Reactor**, Matthias Utschick, Gerhard Mayer, Siegfried Babst *(GRS)*

Despite the phase-out for Nuclear Power Plants (NPPs) in 2022, the operational licenses for Research Reactors (RRs) in Germany are not limited to a certain date. Consequently, the RRs will be operated for several more years. Regulatory requirements for conducting PSA are available for NPPs in Germany, but not explicitly for RRs. However, a PSA for a German RR does already exist and is currently under review.

Furthermore, GRS is developing a Level 1 PSA for a German reference RR within the frame of a research and development project funded by the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) to gain insights on PSA issues specific for RRs. Starting point was a literature study showing the main contributors (initiating events) to damage states in PSAs for RRs outside Germany. The study also revealed the well-known fact, that most RRs are very different and diverse leading to PSA results which are hardly comparable.

This paper presents the status of a Level 1 PSA development process for a German reference RR. The RR being analyzed is an open-pool reactor with 20 MW thermal power and only one fuel element. One hot and one cold neutron source produce neutron fluxes which are guided by beam tubes to the experiments halls. Additionally, radioactive isotopes for medical applications can be produced with the reactor. The relevant initiating events and accident sequences during power operation are, e. g., inadvertent control rod withdrawal, loss of offsite power, loss of converter plate cooling or loss of coolant outside the pool. The initiating events pool leakage, seismic hazard or aircraft crash are relevant for all operational states. The initiating event frequencies have been mostly estimated based on the operating experience of German RRs.

Damage states can be fuel element damage and converter plate damage. Main safety functions required to reach the safety goals after an initiating event are reactor shutdown functions and cooling system functions. Event sequence analysis and derivation of success criteria in the PSA are based on the Safety Report and the Operating Manual. Selected event sequences are evaluated by means of thermal hydraulic calculations. The unavailability of safety functions is computed by means of fault trees, which are the result of systems analyses performed for safety systems as well as for operating systems credited for cooling functions after initiating events. The PSA model also includes simplified modeling of the power supply.

Level 1 and 2 PSA—III Continued

The used reliability data for single failures are mainly based on an IAEA data source (TECDOC-930). The data have been applied to the RR using the superpopulation method (a two-stage Bayesian approach). Common cause failure (CCF) data are mainly taken from a database comprising the operating experience in German NPP. The paper concludes with some lessons learned and open issues.

11:30 am: A Source Term Evaluation in a SGTR Accident, Hoyoung Shin, Youngho Jin, Dong-Ha Kim, Moosung Jae *(Hanyang Univ)*

The nuclear safety act revised in 2015 provides the safety goals for the nuclear power plants in Korea. Prior to the enactment of this act, only the level 1 and 2 PSAs had been practiced conventionally. However, due to the revision of the act, risks due to accidents at nuclear power plants should be evaluated comprehensively through the implementation of the level 3 PSA.

The source term released to the environment due to the steam generator tube rupture (SGTR) accident was evaluated very conservatively because the level 2 PSA has a lot of uncertainty as wells because the level 3 PSA had not been performed in detail. That is, all the SGTR core damage sequences have been classified as a single source term category (STC) because there is no need to confirm the quantitative safety objectives through the level 3 PSA. However, classifying the SGTR core damage sequences as a single STC can have unreasonable effects on the risk assessment results. Because even the cumulative released radionuclides amounts are same, the different available emergency response time can make a different expected early fatality and latent cancer fatality.

Therefore the source terms have been evaluated using MELCOR code in the SGTR core damage sequences for a reference plant. As a result, the single STC was re-classified into three STCs, SGTR-Early, SGTR-Intermediate, and SGTR-Late, based on the start time of the core damage, fission product release and general emergency declaration as well as the amount of fission product released. This re-classification of the STC can be a key variable in risk assessment results of the level 3 PSA. This study can be used as a basis for both a more realistic comprehensive risk assessment and multi-unit consequence analysis.

Risk-Informed Regulation—II

Chair: Antonios Zoulis *(NRC)* **Location:** Opal One **Time:** 10:00-11:30 am

10:00 am: Demonstration of NEI 18-04 RIPB Guidance for Non-LWR Licensing Basis Development Using the PRISM PRA, Matthew Warner, Jonathan Li, Gary Miller, Dennis Henneke *(GEH)*

GE Hitachi Nuclear Energy (GEH) in coordination with the Licensing Modernization Project (LMP) Team performed a demonstration of the NEI 18-04 (Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development) process for the PRISM Generation IV sodium-cooled fast reactor. This work leveraged the recently developed PRISM Internal Events At-Power (IEAP) PRA. The demonstration included Licensing Basis Events (LBE) Selection, structure, system and component (SSC) Classification, and an example Plant Capability Defense-in-Depth (DID) adequacy review across the five DID layers established in NEI 18-04, Table 5-2.

Out of the 591 event sequence families identified, 26 met the criteria to be declared LBEs. Each LBE was further categorized as either an Anticipated Operational Occurrence (AOO), a Design Basis Event (DBE), or a Beyond Design Basis Event (BDBE) in accordance with the NEI 18-04 guidance. PRA sensitivity studies were then utilized to determine the required Safety Functions and to select the Safety-Related (SR) SSCs that mitigate the consequences of DBEs to within the LBE Frequency-Consequence (F-C) Target that is established in NEI 18-04.

A DID adequacy evaluation, performed against the Heat Removal Required Safety Function, identified several non-safety-related SSCs that were required for DID adequacy. These SSCs were classified as Non-Safety Related with Special Treatment (NSRST). SSCs not classified as neither Safety-Related or NSRST in this demonstration were classified as Non-Safety Related with No Special Treatment (NST).

10:30 am: Decommissioning Rulemaking at the Nuclear Regulatory Commission, Alysia Bone *(NRC)*

The U.S. Nuclear Regulatory Commission (NRC) staff is proposing rulemaking in 8 parts of Title 10 of the Code of Federal Regulations (10 CFR), involving 14 technical areas. The NRC's goals in amending these regulations are to provide for a safe, effective, and efficient decommissioning process; reduce the need for exemptions from existing regulations and license amendment requests; address other decommissioning issues that the NRC staff considers relevant; and support the principles of good regulation, including openness, clarity, and reliability. For several technical areas, the NRC staff is proposing to adopt a graded approach that is commensurate with the reductions in radiological risk at four levels of decommissioning. Further, to allow maximum flexibility while maintaining adequate protection of public health and safety and the common defense and security, the NRC staff is proposing to make several of the new requirements alternatives to the current requirements in these areas. The NRC staff is also proposing conforming changes to the regulations for power reactors beyond those related to the decommissioning of nuclear reactors. This presentation will provide background and an overview of this rulemaking effort.

11:00 am: PRA Maintenance and PRA Upgrade, N. Reed Labarge, Andrea Maioli, Rachel Christian *(Westinghouse)*, Roy Linthicum *(Exelon)*

Part 1 of the Probabilistic Risk Assessment (PRA) Standard from the Joint American Nuclear Society (ANS) and American Society of Mechanical Engineers (ASME) defines two types of PRA updates. PRA maintenance activities represent routine updates that retain the methods used in the original development of the PRA; an update of reliability data or initiating events frequency is a classic example of a PRA maintenance activity. A PRA upgrade is an expansion of the scope and/or the capability of the PRA, or implies the use of a method that was not used before for the PRA in consideration and modifies the significant accident sequences and the overall PRA insights. In this context, the definition of what a new method is becomes critical.

The technical adequacy of the PRA is not challenged by maintenance activities, for which the peer review of record remains applicable. In case of a PRA upgrade, on the other hand, a focused scope or even full scope peer review is required to confirm the technical adequacy of the updated PRA model.

Risk-Informed Regulation—II Continued

A characterization in terms of maintenance versus upgrade of the activities performed in support of the closure of Facts & Observations (F&Os) is also required as part of the F&O closure process (i.e., the Appendix X process) that was recently agreed upon with the United States Nuclear Regulatory Commission (USNRC). Such characterization is to be provided by the Utility and agreed upon by the Independent Assessment Team (IAT) and is needed to understand the scope and level of review needed by the IAT. If an F&O is closed via a maintenance activity, then the IAT can limit the review to the specifics of the F&O, without the need for a re-assessment of the associated Supporting Requirement (SR). If, on the other hand, the F&O is closed with an upgrade of the PRA, the associated SR becomes the scope of a focus scope peer review.

The Pressurized Water Reactors Owners Group (PWROG) has been leading a number of F&O closing activities in the industry and has been observing a trend of disagreement and potential confusion between the F&O characterization as maintenance versus upgrade among different stakeholders (e.g., teams defending PRAs in peer reviews and F&O closures, IATs team members, USNRC).

While these disagreements are not hindering the F&O closure process, they have raised the interest on the differentiation between upgrade and maintenance, with the recognition that more clarity is needed in the definition of these PRA update types and on the implication of new PRA methods are adding to the industry proposed license condition for TSTF-505 for model upgrades following recipe of the safety evaluations. This paper summarized the observations of two workshops held by the PWROG on the topic of upgrade vs. maintenance.

Insights from Advanced and Small Modular Reactor PRA Development–Panel

Chair: Sarah Bristol *(NuScale Power)* **Location:** Blue Topaz **Time:** 10:00 am-12:00 pm

The objective of this panel session will be to discuss experience and work in the area of PRA development for small modular reactors and advanced designs. Topics for discussion include:
* unique details discovered developing a design certification PRA for a small modular reactor
* quantifying risks involving multiple modules
* developing regulatory guidance for licensing advanced reactors using a modular design approach
* small modular reactor source terms
* severe accidents in small modular reactors

Panel Members include representatives from the NRC, NEI, and various industry consultants.

Panelists: Tom Morgan *(ENERCON Services, Inc.)*
Dennis Henneke *(GE-Hitachi)*
Karl Fleming *(KNF Consulting Services LLC)*
Martin Stutzke *(NRC)*
Greg Krueger *(NEI)*

Plant and Site Level PSA Applications—I

Chair: Zhegang Ma *(INL)* **Location:** Yellow Topaz **Time:** 10:00 am-12:00 pm

10:00 am: **Insights From a WGRISK Activity on the Status of Site-Level PSA Developments**, Yolande Akl, Smain Yalaoui, Michael Xu *(CNSC)* , Marina Roewekamp *(GRS)*, Daniel Hudson, Nathan Siu *(NRC)*, Joshua Gordon *(ONR)*, Gabriel Georgescu *(IRSN)*

Two important lessons learned from the March 2011 Fukushima Dai-ichi reactor accidents were: (1) There can be significant interactions between multiple co-located radiological sources on a shared nuclear power plant (NPP) site in response to concurrent or consequential initiators; and (2) the timing of concurrent accident sequences involving multiple site radiological sources can challenge shared systems and resources available for severe accident management and emergency response. Since this event, there has been increasing concern among the international nuclear community that (1) the traditional single-unit probabilistic safety assessment (PSA) approach might not be adequate for assessing the total radiological risk to the public from NPP sites comprised of multiple co-located radiological sources; and (2) there is a need for an integrated multi-unit PSA (MUPSA) or Site-Level PSA approach that includes consideration of the potential for concurrent accidents involving multiple co-located radiological sources. 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 (1) how member countries are addressing challenges and developments of site-level PSA and (2) actual or intended uses and applications of Site-Level PSA. This WGRISK activity was completed in two phases. Phase 1 included a preliminary survey and follow-up questionnaires to obtain information about ongoing and future Site-Level PSA activities in WGRISK member countries. This preliminary survey was then used to identify and prioritize three focus areas related to challenges in Site-Level PSA that were of common interest to member countries: (1) site-based risk metrics and safety goals; (2) risk aggregation; and (3) modeling of multi-source (including multi-unit) interactions or dependencies. Follow-up questionnaires were then developed and administered to member countries to identify and obtain more detailed information about specific technical challenges within each focus area. Findings from these focus area questionnaires then provided the technical basis for Phase 2. Phase 2 included an international workshop that was held in July 2018 to (1) support the assessment of the current state of site-level PSA methods, models, and tools; (2) support the evaluation of Site-Level PSA studies; (3) share methods, good practices and experiences among member states on Site-Level PSA; and (4) identify new potential topics for further WGRISK activities related to Site-Level PSA. This paper summarizes the overall WGRISK activity on the status of site-level PSA developments, including key findings, conclusions, and recommendations for the path forward.

10:30 am: **TMRE Implementation Experience at Duke Energy and Southern Nuclear Company Pilot Plants**, Scott A. Brinkman *(Duke Energy)*, J. Alex Gilbreath *(Southern Nuclear Co.)*, Leo Shanley *(JENSEN HUGHES)*, Artur Mironenko *(Duke Energy)*

Plant and Site Level PSA Applications—I Continued

Since the issuance of RIS 2015-06, the industry has been working on a reasonable, cost-effective risk-informed alternative to address legacy tornado missile licensing non-conformances at nuclear power plants in the United States. This Tornado Missile Risk Evaluator (TMRE) methodology was developed with input from the utilities, industry groups, nuclear vendors, and the U.S. NRC.

The purpose of the TMRE is to provide the industry with a RG 1.174 risk-informed option to assess the risk posed by tornado missiles at any site and determine whether additional physical protection is warranted for existing non-conformances. The risk-informed nature of the TMRE allows it to be applied regardless of the vintage of the plant or the content of the plant's licensing basis. The purpose of this paper is to discuss the experience of implementing the guidance, documented in NEI-17-02, at the Duke Energy and Southern Nuclear Company pilot plants. A separate paper will be submitted that discusses the development of the TMRE methodology.

The insights gained from applying different aspects of the guidance, as well as some plant-specific challenges, will both be discussed. Topics ranging from the development of the high winds equipment list (HWEL), the experience from the different types of walkdowns, and specific vulnerabilities will be discussed. Important aspects of the TMRE model calculations and application-specific sensitivities will be reviewed. Also, the interaction with the regulators during the pilot process, changes to the guidance resulting from the regulator's input, and their impact on the site analysis will be mentioned. High level results and conclusion will be presented to give the audience an understanding and perspective of the TMRE application end products.

11:00 am: **Incorporation of Surveillance Frequency Control Program Risk Evaluations in PRA Models**, Brian Burgio *(JENSEN HUGHES)*, Shannon Rafferty-Czincila *(Exelon)*, Gordon Salisbury, Nicholas Sternowski *(JENSEN HUGHES)*

In the current climate of the Nuclear Promise, Exelon Generation has capitalized on the Surveillance Frequency Control Program (SFCP) to drive cost savings and enhance resource flexibility across their nuclear fleet. Enhanced utilization of the SFCP has resulted in an expansive list of extended surveillance intervals and PRA risk evaluations which support those changes.

In accordance with the industry guidance documented in NEI 04-10, PRA model failure probabilities adjusted as part of a surveillance extension are retained in the PRA model. Depending on the total quantitative results of each individual risk assessment, cumulative contributions following retention of adjusted probabilities can vary.

This paper will provide insights from Exelon's developed strategy for inclusion of adjusted independent and common cause failure probabilities from implemented surveillance extensions. Specifically, this paper will highlight the process used to update events during routine model updates, methods to accommodate data updates unrelated to the SFCP while retaining the impacts from an extended test interval, and the capability to include data collected and analyzed over the long-term to more accurately depict the reliability of the components affected by surveillance extensions. Also included will be strategies to document PRA model changes and cumulative SFCP calculational changes post-model update.

11:30 am: **A Simplified Probabilistic Model for Flywheel Integrity Using "R"**, Raymond Schneider, Gordon Hall *(Westinghouse)*

"R" is an open source statistical analysis platform that has been applied to a wide range of issues. This platform allows for rapid simulation and analysis of many stochastic problems that in the past have been quite challenging. This application focuses on the use of the "R" platform to perform a probabilistic fracture mechanics analysis of an RCP flywheel. The use of the "R" platform allows rapid simulation of crack initiation and growth models with various material and failure properties to establish a flywheel failure probability during representative reactor transients. One unique feature of this platform is the ease of use of alternate statistical assumptions and the ease of visualization of the role of randomly selected operational parameters on flywheel failure. Example flywheel failure calculations are provided.

Analytical pitfalls associated with use of traditional distribution selection methods as applied to extremely low probability events are illustrated and discussed along with methods to explore treatment of uncertainty in key parameters. Bounding analyses regarding the impact of failed flywheels on core damage are also presented.

Dynamic PSA—II

Chair: Zachary Jankovsky *(SNL)* **Location:** Emerald Salon One **Time:** 1:30-3:00 pm

1:30 pm: Simulation Based Dynamic Event Tree Analysis, Mahendra Prasad, Mithilesh Kumar, Gopika Vinod, J. Chattopadhyay *(BARC)*, Curtis Smith *(INL)*

The current Probabilistic Safety Assessment (PSA) methodology considers conservative success criteria for the safety systems such as the primary shutdown system, secondary shutdown system etc. This aspect may not be realistic when accident sequences are analyzed for PSA due to the conservatism in the scenario phenomena. The proposed methodology for dynamic evaluations is the Dynamic Event Tree (DET) framework to assess the impact of the variability and scenario dynamics on success criteria and its impact on the PSA model for the initiating event. The DET framework couples the stochastic model (number of component/trains that start on demand, operator action timing, etc.) with a Thermal-Hydraulic (TH) model of the plant. The initiating event selected for the study was class IV failure event. The systems which are modelled in the static portion of the event tree are reactor protection system, passive poison injection system, isolation condensers, emergency power supply system, class IV power recovery, and the shutdown cooling system. The TH analysis was performed with variations in the number of isolation condenser available for the safety function. The analysis showed that the availability of more than one isolation condenser is sufficient to keep the clad temperature within limits under class IV and class III failure. Hence, further uncertainty analysis was carried out with one isolation condenser available and three isolation condensers not available. The uncertain parameters taken in analysis were Heat Transfer Coefficient, Non Condensable gas flow rate, Power, and Loss coefficient. This analysis was performed using RELAP software Mod 3.4. The conclusion of the study broadly is that the clad temperatures are within limits in all the code runs implying a high-degree of safety margin. The time to reach the peak clad temperature is varying but the variation is not very high. This analysis has helped decipher that the operator might have a relatively-stress free state during such accident scenarios due the design of isolation condensers, even if uncertainties in some of the TH parameters are considered. The work is a part of Indo-US ongoing collaboration under a Civil Nuclear Energy Working Group (CNEWG) arrangement.

2:00 pm: Mutual Integration of Classical and Dynamic PRA, D. Mandelli, C. Wang, A. Alfonsi, C. Smith, R. Youngblood *(INL)*, T. Aldemir *(Ohio State)*

Dynamic Probabilistic Risk Assessment (PRA) methods couple stochastic methods (i.e., sampling methods) with system simulators (e.g., RELAP5-3D and MELCOR) to determine the risk associated to complex systems such as nuclear power plants. Compared to Classical PRA methods they can evaluate with higher resolution the safety impacts of timing and sequencing of events on the accident progression without the need to introduce conservative modeling assumptions and success criteria.

When comparing Classical and Dynamic PRA approaches, two considerations might arise from the analysis. First, as part of a Dynamic PRA, it is not uncommon that some components of the system under consideration might not require a complex and computationally expensive simulation model due to its intrinsic characteristics (e.g., no time or physics dependency). From a modeling point of view, such components could be actually included in the analysis by employing simpler Classical PRA models such as Event Trees (ETs) or Fault Trees (FTs).

Second, recent studies have indicated that timing/sequencing of events might impact accident progression modeled in an ET. Typically, only a limited number of aspects of Classical PRAs might require simulation enhancements due to the intrinsic dynamic characteristics of the considered system.

This paper addresses both of these issues by presenting several methods and algorithm that can be employed to link Classical PRA models into a Dynamic PRA (i.e., first issue) and to integrate Dynamic PRA results into Classical PRA (i.e., second issue). These algorithms have been developed within the RAVEN statistical framework and applied for few test cases that will be described in more detail in the full paper.

2:30 pm: Dynamic Probabilistic Risk Assessment with PyCATSHOO: The Case of the Emergency Power Supply of a Nuclear Power Plant, Keoni Sanny, Claudia Picoco, Tunc Aldemir *(Ohio State)*

Component recoveries and system reconfigurations cannot be fully treated by static methods such as the traditional event-tree (ET)/fault-tree (FT) approach as times of occurrence of events are not explicitly considered. Dynamic approaches (and tools), also known as of Dynamic Probabilistic Risk Assessment (DPRA) methodologies, are capable to overcome and solve this major limitation.

In this context, PyCATSHOO software has been developed by the Électricité de France (EDF) research and development team as a DPRA tool that aims at overcoming the shortcomings of static ET/FT approach.

PyCATSHOO explicitly accounts for the time of occurrence of events, allows components recoveries and system reconfigurations. By implementing the concept of Piecewise Deterministic Markov Process (PDMP), it models both deterministic continuous phenomena (described by a set of differential equations) and stochastic discrete events.

In this paper, the application of the PyCATSHOO software is introduced for a simplified AC power system as well as for a nuclear power plant's emergency power supply system (EPSS). The EPSS was previously established as a benchmark for dependability assessment techniques applied to dynamic stochastic systems. The results previously obtained in the benchmark have been used to validate the PyCATSHOO model. Results from the analysis presented in this paper for the two cases study include:

- *Most common sequences (in terms of events and probabilities),*
- *Mean Time To Failure (MTTF),*
- *Unreliability (i.e., number of failed missions/total number of missions),*
- *Unavailability (i.e., time unavailable/total simulation time), and,*
- *System failure and recovery probability distributions.*

Human Reliability Analysis and Human Factors—I

Chair: Mary Presley *(EPRI)* **Location:** Emerald Salon Two **Time:** 1:30-3:10 pm

1:30 pm: Models for Human Performance Improvement, Pamela F. Nelson *(Universidad Nacional Autónoma de México)*, C. R. Grantom *(C. R. Grantom LLC)*

Human Reliability Analysis and Human Factors—I Continued

The "debrief" portion of the SACADA process requires the Control Room crew to critique their performance for a given scenario exercise. In the "debrief" phase crew performance degradations are discussed for crew consensus on the factors associated with SATΔs and UNSATS. The "debrief" part of SACADA was not included in the BNs for calculating the HEPs; however, this aspect of data collection can be used to understand ways to improve human behavior modeling, training effectiveness, and thus, human performance. The relationship between context and the error causes was also investigated for this paper.

These performance indicating factors (PIFs) are referred to as error causes in Tables B2 to B10 in the SACADA paper [1]. A Bayesian Network model intended to evaluate the probabilities of error causes, given a specific scenario context was developed. This model goes beyond the purpose of the data collection use in HRA for PRA purposes. That is, this model provides information that can be extremely valuable to the training organization at a nuclear power plant to improve training and human performance.

The error causes are defined by the PIFs included in the SACADA system. The PIFs include scenario specific causes and person specific causes, such as knowledge gap, slow, lack of questioning attitude, failing to stop, think, act, and review, rushing, and distracted. Additionally, there are PIFs specific to the macrocognitive function of detection / monitoring. These include identification of the error resulting from multiple or unexpected alarms, label, mimic or display issues, among others. Due to the debriefing process, the operators are encouraged to openly express their views relative to scenario issues and UNSAT/ΔSAT causes. The simulator instructors also participate in the debrief and support the debrief evaluation. The simulator instructor also reviews the final inputs and makes changes where considered necessary.

The benefit of using BNs is that the model allows for the determination of the probability of human errors given a cause or group of causes (PIF) as well as the probability of the PIF given a human error, which will be demonstrated in the paper.

1:55 pm: Human Reliability Analysis Quantification Guidance for Main Control Room Abandonment Scenarios in Fire PRAs: What's New and When Can Existing Methods Be Used? Susan E. Cooper *(NRC)*, Ashley Lin`deman, Mary Presley *(EPRI)*, Erin Collins, Jeffrey A. Julius, Kaydee Kohlhepp Gunter *(JENSEN HUGHES)*, John Wreathall *(John Wreathal & Co.)*, Stacey Hendrickson *(SNL)*, Paul Amico *(JENSEN HUGHES)*, 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 U.S. 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. Joint EPRI/NRC-RES guidance for qualitative MCRA HRA was published as Supplement 1 to NUREG-1921 by EPRI in August 2017. (NRC's publication is pending.) Subsequently, EPRI and NRC RES have developed HRA quantification guidance for MCRA scenarios in fire events that is expected to be published at the end of 2018 as Supplement 2 to NUREG-1921.

MCRA HRA quantification guidance addresses three time phases: before the decision to abandon, the decision to abandon, and after the decision to abandon. The approach for HRA quantification is different for each time phase. Also, in some time phases and contexts, the guidance identifies existing HRA quantification tools as being appropriate for MCRA scenarios. However, there are some special cases that required the development of new HRA quantification tools. One such example is the human failure event (HFE) that represents the decision to abandonment for "loss of control" MCRA scenarios (as opposed to "loss of habitability" scenarios).

2:20 pm: The Use of Expert Judgment to Support Human Reliability Analysis of Implementing Flex Equipment, Jing Xing, Michelle Kichline, John Hughey, Matthew Humberstone *(NRC)*

Your Implementation of Flexible Coping Strategies (FLEX) following the accident at Fukushima Dai-Ichi resulted in the purchase of equipment and addition of coping strategies specifically intended to support plant response after extreme external events. Yet, much of the equipment may also be used as added defense-in-depth to mitigate the consequences of non-FLEX-designed scenarios where installed plant equipment fails. Many nuclear power plants have considered using FLEX equipment during non-FLEX-designed scenarios and are taking credit for the additional equipment and mitigation strategies in their Probabilistic Risk Analyses (PRAs). The U.S. Nuclear Regulatory Commission (NRC) needs to Quantify Human Error Probabilities (HEPs) of FLEX types of actions (such as transportation, placement, connection, or local control of portable equipment) in order to support risk-informed licensing activities. The NRC staff performed a formal expert elicitation with the purpose of supporting quantification of HEPs associated with the use of portable FLEX equipment. The Expert Elicitation employed a structured process following established NRC guidance. The expert panel consisted of six experts with expertise in HRA, implementation of FLEX strategies, and typical maintenance practices in nuclear power plants. This paper will describe the Expert Elicitation Process and results as well as the implications on HRA tool development.

2:45 pm: Human Action Dependency Development in the Age of Automation, Ricky Summitt *(Engineering Planning and Management, Inc.)*

The ability of applications to automate the identification and assessment of human action dependency has provided an opportunity to apply so called "recovery factors" for more and more restrictive and specific cut set combinations. This has resulted in tens of thousands of dependencies being defined and applied to the cut sets in a post processed manner that is stretching the current recovery tools and requiring more and more quantification time to arrive at a solution.

In many applications most of the combinations impact few cut sets of meaning and their omission would not alter the overall results. A further complication is added when the combinations are applied to specific assessments in support of SDPs. In these assessments it is possible for some previously assessed actions to be truncated or the event timeline shifted such that a new dependency assessment is required.

This paper will examine an alternative approach for selected critical dependencies for assessment. The paper then considers a mixed modeling approach between model inclusion and post processing to improve quantification time. The conclusions from application of the approach to a typical PRA model are then provided.

Level 3 PSA

Chair: Nathan Bixler *(SNL)* **Location:** Opal One **Time:** 1:30-3:10 pm

1:30 pm: Level 3 PSA Application for Akkuyu Nuclear Power Plant, Veda Duman Kantarcioglu, Sule Ergun *(Hacettepe Univ)*

During severe accidents in nuclear power plants, containment may fail and radioactive isotopes may release into atmosphere. As a result of meteorological conditions and atmospheric dispersion of radioactive isotopes, radiation may reach to public. Urgent and early protective actions are implemented to protect public from the effects of radiation. Level 3 PSA produces data to implement appropriate protective actions and to plan on-site and off-site emergency response actions.

In this study, Level 3 PSA application was performed for Akkuyu Nuclear Power Plant (NPP) in Turkey. The selected accident scenario, which is the basis for the analysis, is the loss of coolant accident that is initiated by the 850 mm diameter guillotine break at the main coolant loop. In this scenario it is also assumed that the entire AC power supply has been lost for more than 24 hours. This accident is the beyond design basis accident (BDBA) and leads to severe off-site radiological consequences.

Amount of radioactive isotopes released as the result of this BDBA and meteorological conditions on site have been reported in the Environmental Impact Assessment (EIA) report of Akkuyu NPP. In this study, by using the data from the EIA report, radiation doses for the public were estimated for locations at different distances from NPP. Calculations were performed to estimate dose values for releases originated from bypass and ventilation channels. For calculations, releases from all possible exposure pathways and a period of one year for the residence at the associated distances were taken into account. RASCAL 4.2 was used for the analysis. Results were used to validate the dose values presented in the EIA report.

The results of the study show that at the distances greater than 3 kilometers, estimated dose values for public exposure were within acceptable limits (5mSv/year) and in good agreement with the values presented in the EIA report.

1:55 pm: Ingestion Dose Evaluation Reflecting Korean Environments, Hyunae Park, Yein Seo, Dahye Kwon, Moosung Jae *(Hanyang Univ)*

In the case that radioactive materials are released to the off-site environment due to the nuclear plant accident, the area in the vicinity of the site and the food from them will be contaminated for relatively long time. Thus, It is important for level 3 probabilistic safety assessment (L3 PSA) to evaluate the ingestion exposure from the intake of contaminated food with the consideration of the agricultural and dairy characteristics and conditions in the interested area, especially in the intermediate and long-term phase. COMIDA2 is a preprocessor for the ingestion dose evaluation using the semi-dynamic food chain model of MELCOR accident consequence code system (MACCS) which is one of the L3 PSA computational code widely used in the U.S. Korea, and the other countries.

In this study, input variables of COMIDA2 such as food category, consumption rate for each category, time parameters and etc. were studied reflecting Korean environmental characteristics from the valid statistic data. For example, instead of other livestock category, pork, the foodstuff whose consumption of Korean is relatively large, was included in the food category. Then, the individual effective ingestion dose for the several accident date cases was evaluated according to the food category. This analysis was conducted on the single radionuclide, Cs-137 because it is one of key contributors to the long-term food contamination and ingestion exposure. It is figured out that the time relation between the accident date and the sowing data of agriculture products would have a great impact on the results. The results of this study would be expected to enhance the reliability and the completeness of a L3 PSA as well as that of multi-unit PSA to be analyzed for developing a site risk assess model for regulation in Korea.

2:20 pm: Development and Status of the ASME/ANS RA-S-1.3-2017 Level 3 PRA Standard, Grant Teagarden *(JENSEN HUGHES)*, Keith Woodard *(ABS Consulting, Inc. - retired)*, Stanley H. Levinson (*Framatome, Inc. - retired*)

This paper discusses the development and current status of the Level 3 Probabilistic Risk Assessment (PRA) Standard (ASME/ANS RA-S-1.3), whose full title is "Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications." The standard development effort began in 2005 with the formation of a standard writing group involving professional practitioners from industry, the national laboratories, and the NRC. The writing group worked for over a decade to develop the technical requirements, interact with standards committees, resolve review comments, and successfully pass the standards balloting process. The Level 3 PRA standard is currently publically available for trial use and pilot application.

This paper also summarizes the technical elements included in the Level 3 PRA standard, challenges encountered by the writing group and lessons learned in the standard development process, including incorporation of insights from two trial applications using a draft of the standard. The completion and availability of the standard for trial use now provides a basis for assessing the technical adequacy of a conditional consequence analysis for atmospheric dispersion or a full Level 3 PRA to support risk decision making, such as potentially justifying a smaller emergency planning zone around a nuclear facility.

2:45 pm: A Focused Sensitivity Study on the Key Input Parameters Important to Long-Term Level 3 PSA 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 general, these contemporary analyses used in quantifying the Level 3 Probabilistic Safety Analysis (PSA) metrics important for performing the subsequent cost/benefit analysis process apply the MELCOR Accident Consequence Code System Version 2 (MACCS2) in calculating consequences as a resulted of postulated severe accidents. The two offsite metrics calculated include population dose and offsite economic cost incurred within a fifty-mile radius of the plant. While many tens of inputs shape the outcome of these analyses, the 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 limited 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 72 and FGR 15 information. The Level 3 PSA SAMA metrics of population dose and offsite economic cost are assessed relative to 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 and FGR 15 changes lead to small decreases in the two offsite PSA metrics; (2) deposition velocity has a significant impact decreasing both PSA metrics; and (3) introducing all three changes leads to an appreciable decrease, again in both metrics. The major sources, or drivers, of these outcomes are discussed, as well as the limitations to and insights obtained from the study.

Safety Goals, Risk Metrics, and Guidance Updates

Chair: S. Tina Ghosh *(NRC)* **Location:** Yellow Topaz **Time:** 1:30-3:10 pm

1:30 pm: Re-Evaluating the Current Safety Goals, Vinod Mubayi *(Consultant)*, Robert Youngblood *(INL)*

In 1986, the United States (U.S.) Nuclear Regulatory Commission (NRC) adopted Safety Goals (SGs) for the operations of nuclear power plants [1], comprising qualitative goals backed up by Quantitative Health Objectives (QHOs), with these QHOs for severe accidents being expressed as the average prompt fatality and the average latent cancer fatality to members of the public exposed to radiation following an accidental release from a nuclear power plant.

The original SGs adopted in 1986 also included a quantitative goal limiting the risk of a "large release." However, after some effort spent to develop a useful definition of "large release," this goal was implicitly abandoned in the 1990s, on grounds that its defined value of a frequency of 1.0E-06/year conflicted with the early fatality QHO. It was replaced, in effect, by the risk guideline that Large, Early Release Frequency (LERF) should be less than 1.0E-05/year (see, for example NRC's Regulatory Guide 1.174). (For a review and discussion of this background, see SECY 13-0029 [2]).

This paper is devoted to a re-examination of the usefulness of the current SGs for informing decision-making considering the risk posed to the public due to nuclear power plant accidents. The most recent Level 3 Probabilistic Risk Assessments (PRAs) show that early and latent fatalities and/or an increase in anticipated cancer rates are quite low, and this is consistent with what happened at Fukushima in 2011. However, the events at Fukushima resulted in very significant societal costs [3, 4], which are not currently reflected in the SGs.

Two recent studies (Denning and Mubayi [3] and Bier et al. [4]) have concluded that for current plants, the major risk is societal risk: extensive land contamination, long-term relocation of large numbers of people, loss of productive farm area, loss of industrial production, etc., and the large costs of remediating contaminated land to make it habitable. Correspondingly, this paper argues that in order for the safety goals to inform decision-making most appropriately, societal risk should be addressed in the safety goal policy. This would have the practical effect of changing the emphasis placed in evaluation of risks posed by plant operation. The paper also discusses some risk measures that could be offered as surrogates for societal risk.

1. *"Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Republication," 51 FR 30028; August 21, 1986.*
2. *History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission, SECY 13-0029 (USNRC, 2013).*
3. *Richard Denning and Vinod Mubayi, "Insights into the Societal Risk of Nuclear Power Plant Accidents," Risk Analysis 37, No. 1 (Society for Risk Analysis, 2017).*
4. *Development of an Updated Societal-Risk Goal for Nuclear Power Safety, Vicki Bier, Michael Corradini, Robert Youngblood, Caleb Roh, Shuji Liu, Proceedings of PSAM-12 (Probabilistic Safety Assessment and Management), 22-27 June 2014.*

1:55 pm: Technical Evaluation of the Margins Between Established Risk Goals and Health Objectives for Nuclear Power Plants, Fernando Ferrante *(EPRI)*, Stuart Lewis *(JENSEN HUGHES)*, Doug True *(NEI)*

This work explores the available sources of information regarding the quantitative risk criteria or goals typically employed in countries that have implemented risk-informed applications with respect to quantitative health objectives. Given that these metrics play a significant role in how risk applications can be implemented and used, especially when results are to be compared with thresholds, it is important to recognize the evolution and current understanding of associated embedded margins. Given that such issues are associated with the level of safety expected from the operation of nuclear power plants, many of these concepts and derived thresholds are often established via policy making. In the U.S., for example, these thresholds were derived and implemented by the U.S. Nuclear Regulatory Commission during the late 1980s/early 1990s. Higher-level safety goals were formulated such that the operation of nuclear power plants would pose no significant additional risk to an individual and the risks to society would be comparable to or less than those associated with other forms of generating electricity. Recognizing the existing challenges at the time the higher-level safety goals were developed in terms of the practical calculation and comparison of numerical results to health objectives (i.e., indirect and direct cancer fatalities), surrogates in terms of core damage frequency (CDF) and large early release frequency (LERF) were established in such a way as to provide margin to the actual safety goals. This margin reflects insights from risk assessments and severe-accident analyses available at the time. Given the additional 30 years of insights, the expansion of risk application in the commercial nuclear reactor industry along with improvements in methodologies and computing capabilities, significant additional information has been gained. Hence, it is now possible to reevaluate the margins between the subsidiary objectives and the actual safety goals, and between estimates of plant risks and the safety goals. This work explores recent developments in severe-accident analysis and risk assessment to inform and expand on these perspectives. Variations in nuclear reactor safety policy, reactor designs, extent of use of risk information in decision-making, and other aspects can impact the conclusions; hence, this work is currently focused on the U.S. perspective. However, the concept of considering margins with regards to high level requirements and subsidiary quantitative goals could be more widely applicable to any framework that has implemented or is considering the development of a risk-informed framework that considers the aspect of safety margins under a similar approach.

2:20 pm: Application of Qualitative Importance Measures, Andrija Volkanovski *(Jožef Stefan Inst)*

The importance measures are utilized in the probabilistic safety analysis (PSA) to assess the impact of risk contributors on the selected risk measure. The importance measures are divided into quantitative importance measures, obtained from qualitative results of PSA, and qualitative importance measures.

The qualitative importance measures are derived from the qualitative, logic structure of the PSA. The logic structure of the PSA includes the fault tree and event tree models, the failure combinations causing undesired events and the success paths preventing undesired events. The exact logic expression of the selected risk measure is required input for assessment of the qualitative importance measures. The assessment of the exact logic expression is complex even for a small system as one of the main limitations for their application.

The qualitative importance measures are not depending on the probabilities of the events modelled in the PSA. Therefore they can be assessed for systems under development or when uncertainties considering components reliability are large.

In this paper several approaches for the assessment of the qualitative importance of the events in PSA will be presented. Their applicability for PSA model of real NPP will be discussed. The potential applications of the qualitative importance measures for improvement of the plant safety will be given.

Safety Goals, Risk Metrics, and Guidance Updates Continued

2:45 pm: Overview of the Society of Fire Protection Engineers (SFPE) Latest Engineering Guide to Fire Risk Assessments, 2nd Edition, Rob Plonski *(SRNL)*, Francisco Joglar *(JENSEN HUGHES)*

The Society of Fire Protection Engineers (SFPE) has a Task Group revising the Engineering Guide to Fire Risk Assessments dated November 2006. This Task Group is developing updated guidance on the application of risk assessment in fire protection design/analyses and the use of various risk assessment methodologies in the design and assessment of building and/or process fire safety. This will involve qualitative, quantitative, and semi-quantitative methodologies, presented in a logical order, and based on the latest fire risk-analysis techniques.

The SFPE Fire Risk Assessment Guide is also being written to interface with SFPE's new Standard on Design Fire Scenarios; this new standard is under development and will assist fire risk professionals in the determination of design fire scenarios to be used in performance-based designs and risk analyses.

This presentation will briefly cover the challenges in developing an adequate fire risk analysis, the changes that will be seen in the 2nd edition of SFPE's Guide to Fire Risk Assessments to address these challenges, and demonstrate the application of the updated guide by walking through several real-world examples. These examples will give an attendee the understating for the diversity that this new guide has in the application of fire risk assessments in today's challenging technical environment

Reliability Estimation and Data Analysis—I

Chair: Felix Gonzales *(NRC)* **Location:** Blue Topaz **Time:** 1:30-3:10 pm

1:30 pm: Analysis of Loss-of-Offsite-Power Events 1987-2017, Zhegang Ma, Nancy Johnson, John Schroeder *(INL)*

Loss-of-offsite-power (LOOP) can have a major impact on a nuclear power plant's ability to achieve and maintain safe shutdown conditions. LOOP event frequencies and times required for subsequent restoration of offsite power are important inputs to plant probabilistic risk assessments. This paper provides an overview of the analysis of LOOP events at U.S. commercial nuclear power plants conducted by Idaho National Laboratory (INL) for the U.S. Nuclear Regulatory Commission (NRC). The paper then presents the updated statistical and engineering analysis performed in 2018 based on the operating experience during calendar years 1987 through 2017. Finally, the paper provides the thoughts on some existing concerns in the current LOOP analysis such as the treatment in dependency for multi-unit LOOP events.

1:55 pm: Pilot Applications of SACADA Database for Feed and Bleed Operator Action, Mohamad Ali Azarm, Clifford Marks *(Innovative Engineering and Safety Solutions, LLC)*

The U.S. Nuclear Regulatory Commission (NRC) is collecting the licensed operator performance information from simulator exercise of nuclear power plants (NPPs) to develop empirical basis to better understand the elements affecting operator performance and to help with estimating the associated human error probabilities (HEPs). Scenario Authoring, Characterization, and Debriefing Application (SACADA) system is designed to collect the licensed operator data in a consistent manner. This study uses a selected SACADA Data Set provided by the NRC.

In an earlier study, the authors developed methods based on the concept of context similarity [PSAM 14, USA, LA, Sept 17-21, 2018]. The underlying premise of context similarity was based on the assumption that the HEP values associated with two actions are close as long as all or the majority of their Situational Factors (SFs) are the same. For cases of partial matches, the validity of the data point were examined using statistical significance tests and utilizing the MCF (Micro Cognitive Function) tree for Bayesian updating.

The focus of this paper however is on the pilot application of these methodologies for estimating the failure probability of feed and bleed operation for a four loop Westinghouse plant. The feed and bleed operator action is first divided to several subtasks or procedural steps (also referred to as TOEs in SACADA). Each TOE was then characterized by the associated SFs similar to other SACADA entries.

The entries from SACADA database are then examined to identify those which closely match the SFs associated with the input TOEs. The operator performance in SACADA database then is used to estimate the HEP associated with the input action using evidential data obtained from SACADA database. The HEP values for all subtasks were estimated using Bayes method and were aggregated to get the feed and bleed initiation error probability. The result shows the feasibility and the reasonableness of the SACADA data and the methods developed in this study.

2:20 pm: A Complex Network Analysis for Balanced Design Verification, M. Rifi *(EdF/LIPN)*, M. Hibti, S. Vermuse *(EdF)*, Y. Bennani, R. Kanawati *(LIPN)*

Probabilistic Risk Analysis (PRA) is an analytical technique for integrating design features and operation aspects to assess the safety of nuclear power plants even during the initial plant phase. In the early stages of the design process of some complex systems, it would be nevertheless sometimes difficult to get insights, such as safety significance of design features, through a quantitative probabilistic assessment.

This is mainly due to the lack or the limited level of detail of design information and data. Indeed, the functional requirements of frontal or support systems are generally not yet well defined, established or detailed, such as the level of independence, redundancy or diversity, the sizing of the trains in normal operation or in standby, etc. System architecture modifications are part of a normal plant design process which can prevent from building a consistent and stable PRA model for quantitative assessment. Thus, a global view is missing particularly when different systems are designed by different teams.

Moreover, reliability data are not always available for new components and therefore, by waiting for appropriate FMEA of manufacturers, one could use data of supposed "similar" components or generic ones from public database. Unfortunately that would be probably not appropriate and would increase the uncertainty of the quantified risk.

This paper explores a new approach to study the global balanced character of system design. The accident sequences are modeled as complex networks for which different specific metrics are calculated. The main idea is to assess the prediction of importance factors such as RIF using complex networks centrality metrics (e.g. Betweenness and Page-Rank) without reliability data and to check whether the design is well balanced with regards to the sensibility distribution between component failures, but also between the different accident sequences.

Reliability Estimation and Data Analysis—I Continued

The approach is based on the modelling of structures and dependencies of the mitigation systems rather than on the quantitative assessment of different accident sequences to overcome PSA limitations regarding the uncertainties of the design and the lack of data. It can be used as a complementary verification tool by a large panel of users without prior PRA skills.

2:45 pm: **A Guidance for the Scoping and the Frequency of a PRA Data Update**, Young G. Jo *(Southern Co.)*

The data used in a Probabilistic Risk Assessment (PRA) model should be periodically updated to make the PRA model represent as-built as-operated plant. In general, a PRA data update task is a very resource intensive task because large amount of raw data should be collected, reviewed and classified. For a utility which has to maintain multiple PRA models for its nuclear power plants, periodically updating PRA data for all the PRA models is a significant burden. In this paper, a guidance for determining the scope and the frequency of a PRA Data Update was developed. The developed guidance was intended to reduce the burden for PRA data updates while maintaining PRA data to represent as-built as-operations plants as much as practically possible. A basic rule is that the PRA data should be updated each time a PRA model goes through a periodic model update only if PRA basic events are risk significant and are not rare events. PRA data for risk insignificant events or rare events may be updated less frequently, like every two PRA model periodic update. Risk significance of an event is based on the ASME/ANS PRA standard and other references. Rare events here are events whose PRA data are based on expert elicitation process because there is none or little operating experience data is collected for data evaluation. Events for which only industry pooled data can provide statistically meaningful data set may also be considered as rare events. Some exceptions to the rules were made, for example, if a component experienced actual failure at the plant or there is a significant change in generic data for the component since the last PRA data update, the PRA for the component should be updated in the next periodic model update. Separate recommendations were developed for updating initiating event frequencies, components failure data, common cause failure parameters (alpha factors), and common cause failure probabilities.

Plant and Site Level PSA Applications—II

Chair: Marina Röwekamp *(GRS)* **Location:** Emerald Salon Three **Time:** 1:30-3:10 pm

1:30 pm: **Complex Modeling for Surveillance Test Interval Extensions**, Justin Sattler, Matthew Johnson *(JENSEN HUGHES)*

The Surveillance Frequency Control Program (SFCP) is an industry-wide effort; it involves relocation of time-based surveillance frequencies to a licensee-controlled program. Application of PRA models is used to evaluate each surveillance test interval (STI) extension. When the proposed STI extension involves structures, systems and components (SSCs) explicitly modeled in the PRA and/or involve low-risk-significant SSCs, the STI extension analysis can be relatively straight-forward. On the other hand, analyses involving high-risk-significant SSCs where the tested functions are not explicitly modeled require more refined analyses. Examples of such analyses are presented here.

One example of such an analysis was for Core Spray and Low Pressure Coolant Injection (LPCI) Response Time Functional Test. Adjustment factors are applied to failure probabilities based on the ratio of the proposed STI to the current STI. Applying the entire adjustment factor to the failure probabilities of Core Spray and LPCI would have been too conservative, so a more detailed method was used to apply the adjustment factor to a small portion of the failure probabilities throughout the model. Additionally, to focus specifically on the response time aspect of the surveillance requirement, the adjustment factor was applied to the entire failure probabilities in the portions of the PRA model that reflect modeled accidents similar to those used to develop the Core Spray and LPCI timing criteria.

Another example of a refined analysis was for Control Rod Scram Time Surveillance. The Fire PRA assumed fire-induced failure to scram does not occur, and fire-induced failure of ATWS mitigation systems will not affect fire risk because fire-induced failure to scram will not occur. Therefore, ATWS mitigation systems were not analyzed to determine the required cables and cable routing for the equipment to function. Using the Fire PRA as-is would produce a lower bound estimate because the Fire PRA assumes the ATWS mitigation systems are available given random failure of the RPS to scram the reactor. Therefore, an analysis method was developed to modify the Fire PRA model and produce upper-bound fire risk results to facilitate the STI extension.

1:55 pm: **Application of Electrical Power Recovery in the South Texas Project Electrical Power Generating Station (STPEGS) PRA Model**, Russell Jones, Levi Holden *(STPNOC)*, Shawn Rodgers *(Retired)*

Probabilistic Risk Analysis (PRA) models are charged to be a realistic representation of the as-built as-operated nuclear power plant. A complete PRA model does not only represent failure of equipment, it also represent actions taken by operators (recovery actions) to restore the plant to a safe condition after an initiating event. Recovery actions play a significant role in reducing the values of the Core Damage Frequency (CDF) and the Large Early Release Frequency (LERF). Loss of Offsite Power (LOOP) events and Losses of Offsite Power following an initiating event (known as consequential LOOPs) are significant contributors to CDF and LERF in the STP PRA model.

A methodology that considers the recovery of offsite power, the time dependent nature of the failure of Standby Diesel Generators (SDGs) following a demand, and the potential to repair at least one SDG is developed and being used at STP. This methodology was developed partially in response to updated LOOP initiating event frequency data categorized as Plant-Centered, Switchyard-Centered, Grid-Related, or Weather-Related.

This method utilizes probability distributions for LOOP duration and for SDG repair time provided by the Nuclear Regulatory Commission (NRC). Recovery curves are developed which account for both recoverable and non-recoverable SDG failure types. The PRA software that is used develops cutsets (possible failure combinations) from a fault tree representing failure of all SDGs, which are evaluated in order to determine which type of recovery curve applies. Each cutset is then combined with the appropriate SDG and offsite power recovery curve. The results are used to produce recovery probabilities for times from 0 to 24 hours following an event.

This method has been proven to reduce the value of CDF and LERF by more than 30% in the current STP PRA model.

Plant and Site Level PSA Applications—II Continued

2:20 pm: **Whole-Site Risk Characterization Approaches in Canada: Regulatory and Technical Challenges**, Smain Yalaoui, Yolande Akl *(CNSC)*

There are two types of Canada Deuterium Uranium (CANDU) nuclear installations in Canada. The multi-reactor installations which are mainly located in Ontario province (Pickering A and B, Bruce A and B, and Darlington), and the single unit CANDU 6 reactor located in Quebec (Gentilly-2, now in decommissioning state), and New Brunswick (Point-Lepreau).

The early designs of the CANDU reactors were based solely on deterministic rules and criteria, single/dual failure, and reliability design such as redundancy, and diversity. However, the Probabilistic Safety Assessment (PSA) has long been recognized as an important tool for assessing and managing nuclear power plant risk, and to support the adequacy of the plant safety provisions.

As part of their effort to comply with the Canadian Nuclear Safety Commission (CNSC) regulatory document on PSA, the licensees have conducted Level 1 and Level 2 PSA for internal and external events, during both at-power and shutdown states, and to include the uncertainty, sensitivity, and importance analyses. In the aftermath of the Fukushima Dai-ichi reactor accidents, during the Pickering Licensing Hearing in May 2013, the Commission directed Ontario Power Generation (OPG) to conduct a whole-site PSA for the Pickering Nuclear Generating Station. In addition, the regulatory document S-294 was updated and reissued as REGDOC 2.4.2 "Probabilistic Safety Assessment for Nuclear Power Plants" in 2014, specifically requiring the inclusion of multi-unit impacts, the consideration of other radioactive sources such as the spent fuel bay, and the consideration of potential combinations of internal and external hazards.

Risk aggregation to assess the "total risk" for a site introduces additional regulatory and technical challenges that need to be addressed. These include the methods of aggregating risk contributions from all reactor units and other onsite sources of radioactivity; all hazard groups; as well as all operating states with due regard to differences in level of realism/conservatism, level of detail in PSA modelling, and uncertainty treatment.

This paper presents the approaches used to characterize the whole-site risk for the single-unit and the multi-unit CANDU installations. These approaches for risk characterization were performed either through a careful risk aggregation, where the per-unit based PSA results are extrapolated through the cutset interrogation to quantify site-based risk metrics for a given hazard type, or through the development of an integrated master PSA model. The latter approach was used only for the single unit CANDU installation.

This paper also discusses the regulatory and technical challenges, as well as the overall results and insights from the two approaches.

2:45 pm: **Reliability Analysis of a Safety System Using Petri net and Comparison with Smart Component Methodology**, Darpan K. Shukla, A. John Arul *(IGCAR)*, Mark James Wootton, John Andrews *(Univ of Nottingham)*

For the reliability analysis of advanced nuclear reactor safety systems, though event tree-fault tree (FT) approaches have been used over the years, they are inadequate from a modeling perspective. First, it involves making various levels of approximations depending on the complexity of the system being modeled and second, the responsibility of deriving the correct reliability model rests with the analyst. To overcome the problems mentioned above various methods for the reliability analysis of dynamic systems are being developed. Though many of the methods can more closely reflect the dynamic reliability aspects of the reliability model, they lack the features required for a user-friendly approach. Recently, a Smart Component Methodology (SCM) based on the object-oriented representation of system structure and behavior, to perform dynamic reliability analysis has been proposed.

The dynamic reliability methods could be divided into two categories based on how close the initial formal representation is to the actual system description. For example, in the case of Petri nets, which is often used to perform dynamic reliability analysis, a dynamic system's structure and behavior have to be manually translated (as of now) to a Petri net to perform reliability analysis. Petri net would fall into one category. In SCM since it uses object-oriented representation, which is closer to the system's design description/representation, this method would require the least reliability expertise to perform dynamic reliability analysis (would be the other category). Future methods which would automatically translate a system description/representation into a reliability model or automatically generate reliability metrics would fall into the latter category.

In this paper, we perform a comparative study of the dynamic reliability modeling of an emergency power supply system with Petri net model as well as the newly proposed SCM to bring out the differences and advantages of these two methods. The results are also compared with that from the traditional FT method. The running time complexity and ease of modeling and correctness verification aspects would also be brought out.

TECHNICAL SESSIONS - 3:30 PM

Dynamic PSA—III

Chair: Valentin Rychkov *(EdF)* **Location:** Emerald Salon One **Time:** 3:30-5:00 pm

3:30 pm: **Advanced Tolerant Fuels: A PRA Comparison**, D. Mandelli, C. Parisi, N. Anderson, Z. Ma, H. Zhang *(INL)*

Accident Tolerant Fuels (ATFs) are new nuclear fuels that have been developed in light of the accident at the Fukushima power station in March 2011. The goal of ATFs is to be able to withstand accident sequence with better performances than the currently employed fuels (e.g., smaller hydrogen generation). This paper targets a method to evaluate and compare ATF performances from a Probabilistic Risk Assessment (PRA) perspective.

Given the nature of the problem, Classical PRA methods (based on Event-Trees and Fault-Trees) show their limitation to analyze ATF. Instead, we will employ a newly developed combination of Dynamic PRA methods.

Dynamic PRA methods couple stochastic methods (i.e., sampling methods) with system simulators (e.g., RELAP5-3D) to determine the risk associated to complex systems such as nuclear power plants. Compared to Classical PRA methods they can evaluate with higher resolution the safety impacts of timing/sequencing of events on the accident progression without the need to introduce conservative modeling assumptions and success criteria.

Dynamic PSA—III Continued

In this paper we analyze the impact of three different fuels configurations from a PRA perspective for a Large Break Loss of Coolant Accident (LB-LOCA) scenario. The goal is to present the major differences that ATF types can create when compared to classical Zr-based fuels.

4:00 pm: Exploring Integrated Safety/Security Dynamic Probabilistic Risk Assessments (DPRA) for Nuclear Power Plants, Brian Cohn *(Ohio State)*, Adam Williams *(SNL)*, Tunc Aldemir *(Ohio State)*

Security at nuclear power plants (NPPs) in the United States is currently based on vital area identification (VAI)—a procedure to determine locations within a nuclear facility that need to be defended from adversaries in order to avoid damage to the facility and/or release of radionuclides to the environment. This procedure heavily leverages a Level 1 probabilistic risk assessment (PRA) which identifies combinations of events that can lead to core damage. Current approaches to VAI for NPPs, however, are determined on a "snapshot-in-time," and therefore unable to include the time-dependent effects of safety systems within a NPP

A novel "leading simulator (LS)/trailing simulator (TS)" methodology is proposed to integrate the thermal hydraulic-based safety analysis of a NPP with a physical security analytical tool to model vital area boundaries and related potential consequences. The methodology will use dynamic event trees to systematically explore the uncertainties in an adversary attack scenario at a hypothetical NPP while incorporating the timing and repair effects that are not captured using the available modeling approaches to physical security practices. Ultimately, the LS/TS methodology will enable NPPs to incorporate the full complement of safety systems and procedures when performing security analyses.

4:30 pm: A Dynamic Safety Margins Estimation with a Limited Number of PWR Large Break LOCA Simulations, Francesco Di Maioa, Ajit Raia *(Politecnico di Milano)*, Enrico Zioa *(Politecnico di Milano/ MINES/Kyung Hee Univ)*

The development of methodologies for Risk-Informed Safety Margin Characterization (RISMC) in presence of stochastic and epistemic uncertainties affecting the system dynamic behavior is one of the objectives of the Light Water Reactor Sustainability (LWRS) Program.

In the present work, the characterization of safety margin uncertainties is handled by Order Statistics (OS) (with both Bracketing and Coverage approaches) to jointly estimate percentiles of the distributions of the safety parameter and of the time required for it to reach these percentiles values during its dynamic evolution. The novelty of the proposed approach consists in the integration of dynamic aspects (i.e., timing of events) into the definition of a dynamic safety margin for a probabilistic quantification of margins.

The methodology is applied to a pilot case study of a TRACE model of a Large Break Loss of Coolant Accident (LBLOCA) occurring in the Zion Nuclear Power Plant (NPP).

Education, Training, and Knowledge Management

Chair: Maeley K. Brown *(AECOM Technical Services)* **Location:** Opal Two **Time:** 3:30-4:30 pm

3:30 pm: Communicating PRA Concepts to Non-Practioners, Bruce Morgen *(EPM)*

PRA is widely understood to be a tool providing the risk of a nuclear power plant (NPP) using metrics such as core damage frequency (CDF). NPP technical organizations such as Engineering, Operations, Maintenance and Work Control well understand CDF and other basic PRA risk ranking metrics. With the move to adopt risk-informed categorization of SSCs (50.69) and risk-informed completion times (RICT or TSTF-505/4b) these organizations as well as Management and Licensing staff will be exposed to additional PRA concepts and metrics.

While PRA is based on relatively simple concepts, additional skill and understanding is necessary to effectively apply insights and results. Because of this, NPP technical staff knowledge can be limited due to a perceived complexity associated with PRA. Based on anecdotal evidence and a heuristic approach for developing training for targeted NPP staff, this paper relates selected populations of technical nuclear workers to current risk-informed applications and then to key PRA concepts that should be easily understood. These relationships can be used to focus PRA knowledge to specific targeted populations so that PRA insights and results can be more effectively leveraged into job tasks.

4:00 pm: RISKAUDIT and CNEN Cooperation on Probabilistic Safety Analysis, Gabriel Georgescu *(IRSN)*, Marcos E. Nunes *(CNEN)*, Patricia de Silva Pagetti de Oliveira *(IPEN)*, Andreas Wielenberg *(GRS)*, Ilkka Niemela *(STUK)*, Patrical Dupuy *(IRSN)*

The Brazilian project BR3.01/12 financed by the European Union in the framework of its Instrument for Nuclear Safety Cooperation INSC, and accomplished by a RISKAUDIT consortium (IRSN, GRS, TECNATOM and STUK) consisted of a support for enhancing and strengthening CNEN expertise in regulatory and licensing activities. CNEN is the Brazilian regulatory authority in nuclear safety, and the activities carried out during the project included among others probabilistic safety analysis (PSA), deterministic analysis and ageing management. The objectives of the project PSA task were to provide support to CNEN in the enhancement of its regulatory capability related to PSA development, review and applications and assist CNEN staff in the review of the documents related to parts of Level 1 and Level 2 PSAs submitted to CNEN by the utility, Eletronuclear (ETN). These PSAs refer to the Brazilian NPPs of the CNAAA nuclear station, located in Angra dos Reis, Rio de Janeiro. The activities focused on: updating and enhancing the Brazilian regulatory requirements regarding the development and use of PSA; elaborating the Brazilian guide on regulatory review of PSA; supporting CNEN on the preliminary review of the PSAs for Angra 2 NPP; supporting CNEN on the management of Level

1 PSA computer codes, as well as training CNEN staff on PSA methods, PSA review and PSA applications. The project started in June 2015 and successfully ended in May 2018. This paper will present the main results and lessons learned from the BR3.01/12 project PSA task development.

Computer Tools

Chair: John E. McAllister *(HukariAscendent)* **Location:** Emerald Salon Three **Time:** 3:30-5:00 pm

3:30 pm: The GRS Source Term Prognosis Software FaSTPro for PWR and BWR Spent Fuel Pools, Michael Hage, Sören Johst *(GRS)*

In the event of a severe accident in a nuclear power plant, when airborne radioactive particles may be released to the environment (so-called source term), emergency disaster control authorities must take measures early enough to protect the general public. Computerized analytical tools to guide and assist the plant crisis team or an external emergency team for estimating the source term are helpful and time saving in case of such events. Information about expected source terms should quickly be provided to the authorities via data transmission or ready-made forms to make forecasts of the radiological situation, e.g., with the decision support system RODOS (Realtime Online Decision Support System) developed by the Institute for Nuclear and Energy Technologies (IKET). Consecutive to the Fukushima-Dai-ichi reactor accidents, GRS has been contracted by the German Federal Office for Radiation Protection (BfS) to develop a software tool especially for accidents in a spent fuel pool, although it is well accepted by the expert community that such accidents are extremely unlikely and can be considered hypothetical.

For core melt accidents, most nuclear power plants in Germany apply the GRS software FaSTPro (Fast Source Term Prognosis) for predicting and transmitting source terms originating from the nuclear inventory of the reactor pressure vessel. Two newly developed versions of FaSTPro, focusing only on the nuclear inventory of a pressurized water reactor (PWR) and a boiling water reactor (BWR) spent fuel pool, are presented in this paper. The tool uses a probabilistic approach based on a best estimate plant-specific PSA Level 2 combined with actual data of the plant condition using Bayesian belief networks (BBN). Basically, the BBN contains information from the PSA Level 2 and further information about the current plant status given by the plant personnel during the accident. The plant personnel insert this information by answering given questions. In the updated FaSTPro versions, the sets of these questions were extended by questions about the BWR and PWR spent fuel pool. In order to predict source terms, selected severe accident sequences have been analyzed by MELCOR calculations. On this basis, different source terms have been derived and implemented into the updated software. The GRS software for source term prognosis has been extended and further advanced. As a result, spent fuel pool source terms of PWR and BWR have been implemented in standalone software versions outlined in this paper. The updated versions of FaSTPro are used at the GRS crisis center and aim on improving plant external protection measures.

4:00 pm: New Functions and Features Associated with EPRI HRA Calculator Version 5.2, Kaydee Kohlhepp Gunter, Jeff Julius, Michael Hirt *(JENSEN HUGHES)*, John Weglian, Mary Presley *(EPRI)*

The purpose of the EPRI HRA Calculator®1 is to analyze and document the human reliability analysis (HRA) of pre-initiator and post-initiator human failure events (HFEs) using a consistent, standardized HRA framework designed to ensure that the HRA elements performed using the HRA Calculator meet the requirements of the ASME/ANS PRA Standard as endorsed by R.G.1.200.

The EPRI HRA Calculator® is designed to interactively apply current HRA methodologies that are used by members of the EPRI HRA Users Group and it is intended for use by probabilistic risk assessment (PRA) practitioners, who may or may not be HRA experts. The software provides functionality to interface with traditional PRA software such as to exporting HEP data to PRA models and identification of HFEs which appear in combinations in PRA models.

The initial software requirements were developed by HRA user group members in 2001 and version 1 of the EPRI HRA Calculator® was released in 2003. Since then 10 versions of the software have been released with the intention of updating the code to ensure it continues to supports current HRA research and industry initiatives. In 2017, EPRI released EPRI HRA Calculator® version 5.2. This new version builds upon previous versions with the following new functions and features:

- *New timeline module - Timeline pictures for individual HFEs are now drawn to scale, and the timeline display was added for dependent combinations of HFEs.*
- *New instrumentation library – This library is used to link the cues for an HFE to associated plant instrumentation components.*
- *New feasibility criteria – The feasibility criteria is automatically applied to HFEs under specified conditions, and sets the HEP to 1.0 if it is determined to not be feasible.*
- *New uncertainty parameters that allow the HRA Calculator interact with EPRI Uncertainty code, including uncertainty parameters for JHEPs.*
- *New batch process features – The new tool allows the user to create multiple copies of HFEs using a CSV file.*
- *Improved HRA Helper tool – This tool supports implementation of the HRA dependency analysis into PRA models.*
- *New guidance for CBDTM recovery factors.*
- *New reporting features.*
- *Improved Dependency Analysis module design*

This paper provides a review of the new functions and features of the HRA Calculator version 5.2.

4:30 pm: SAPHIRE's Current "State of Practice" to Meet PRA Demands, James Knudsena and S. Ted Wooda, Curtis Smith *(INL)*

The Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) is a software tool developed for use on a personal computer (PC) to perform complete probabilistic risk assessments (PRA). SAPHIRE Version 8 is funded by the U.S. Nuclear Regulatory Commission (NRC) and developed by the Idaho National Laboratory (INL).

SAPHIRE is used to model complex systems using fault tree analysis or system responses to transient events at facilities event tree accident sequence analysis. The quantification of these different analysis types results in either probabilities or frequencies. For nuclear power plant applications, SAPHIRE 8 solves the Level 1 PRA (core damage frequency) to provide different risk metrics. Some of these risk metrics are core damage frequency and important contributors, i.e., dominant contributors, importance measures and uncertainty. SAPHIRE 8 has been enhanced to handle the expansion of event sequences and different questioning necessary to evaluate Level 2 PRA. Features have been added to SAPHIRE in order to manage data to handle external initiating events, e.g., fire and seismic.

Computer Tools Continued

SAPHIRE 8 contains essentially four different workspaces that is used for analyses of the developed model. The four workspaces are: (1) "default" where model development is performed along with base model evaluation; (2) Events and Condition (ECA) where event analyses are performed; (3) Significance Determination Process (SDP) where straight forward condition analyses are performed; and (4) General Analyses (GA) where sensitivity analyses can be performed. These workspaces are designed to make full copies of the base model and allow users to make changes to the model and perform different analyses without affecting the original model. Each of these workspaces are geared to provide specific reports.

Along with the overview of the workspaces and key features of Level 2 PRA analysis, part of the "State of Practice" features added to SAPHIRE will be discussed in this paper. SAPHIRE 8 contains an automated report generator. This report generator will output multiple quantification result tables along with tables that identify all of the inputs into the model, e.g., list of all basic events, operator actions and fault trees. Another report output developed into SAPHIRE is the Plant Information e-Book (PRIB). Other important features added to SAPHIRE to keep it current with "State of Practice" are: Category fields, common cause failure expansion (for initiating events and enabling events), Integrate and Check project, large early release factors (LERF) via global rule editor and others.

As PRA models grow in complexity, software tools have to grow to handle these complexities. This paper will cover the "State of Practice" that has been developed into SAPHIRE in order to meet these new challenges. This software like all PRA software programs have to keep evolving to meet these demands, since PRA is in all aspects of plant and/or process development. With faster computers and the need for refined evaluations, software is being challenged to meet these needs.

Reliability Estimation and Data Analysis—II

Chair: Rachel E. Vail *(AECOM Technical Services)* **Location:** Yellow Topaz **Time:** 3:30-5:10 pm

3:30 pm: Human Reliability Dependency Analysis and Configuration Risk Management, John E. Weglian, Mary Presley *(EPRI)*, Kaydee Gunter, Michael Hirt, Jeffrey Julius *(JENSEN HUGHES)*

A key element of a Probabilistic Risk Assessment (PRA) of a nuclear power plant is the Human Reliability Analysis (HRA). One requirement for the HRA is to assess the level of dependency between human actions that occur in the same cutset or in the same accident sequence. The goal of the HRA dependency analysis is to identify important combinations of human failure events and calculate a joint human error probability for that combination. There are various approaches to account for the human failure event dependencies, and each approach may have a different effect on the model results depending on how previously-identified human failure event combinations were addressed.

The HRA dependency analysis is typically performed by analyzing cutsets from the PRA model with human error probabilities set to a high value, such as 1.0, to generate a set of cutsets containing all of the potential combinations of human failure events that are expected to occur. However, this process neglects changes in the PRA model that take place when it is used for Configuration Risk Management (CRM). In a PRA model used for CRM, the average maintenance terms are zeroed out (e.g., set to FALSE) and, if a component is in maintenance or otherwise unavailable, basic events in the model for that component are set to TRUE, then the model is re-quantified. The re-quantification thus accounts for the out of service equipment and the actual plant alignments, to generate the risk value for a particular plant configuration. It is possible for a CRM model to reveal unexpected behavior in the PRA as a result of the HRA dependency analysis, such as a reduction in the calculated risk when a mitigating system is removed from service. This paper will look at the issues that can occur in a PRA model used for CRM and the impact of various approaches for incorporating the results of the dependency analysis in the PRA model.

3:55 pm: Sodium Pump Performance in the NaSCoRD Database, Zachary K. Jankovsky, Zach Stuart *(SNL)*, Matthew R. Denman *(Kairos Power)*

Sodium Fast Reactors (SFRs) have an extended operational history that can be leveraged to accelerate the licensing process for modern designs. Sandia National Laboratories (SNL) has recently reconstituted the United States SFR data from the Centralized Reliability Database Organization (SFR) into a new modern database called the Sodium (Na) System Component Reliability Database (NaSCoRD). This new database is currently undergoing validation and usability testing to better understand the strengths and limitations of this historical data. Development of this database helps to address key knowledge management and preservation issues as identified in the multi-year study entitled the SFR Safety and Licensing Research Plan.

Pumps are a major class of equipment found in the NaSCoRD database. NaSCoRD contains a record of 117 pumps, 60 with a sodium working fluid, that have operated in EBR-II, FFTF, and test loops including those operated by both Westinghouse and the Energy Technology Engineering Center. Pump failure events in NaSCoRD can categorized by working fluid (e.g., sodium, water), pump type (i.e., electromagnetic and mechanical), failure mode (i.e., failure to start, failure to run, leakage, and rupture), operating facility, operating temperature, or other user defined categories. This data was collected during the early periods of probabilistic risk assessment and thus the reliability database collection effort had inconsistent component failure definitions which impedes direct incorporation of this data into modern risk analyses. Historical efforts were conducted by EG&G to recategorize failure modes for pump and other components into modern definitions. The impacts of these sodium component reliability improvement efforts were unfortunately minimized when the CREDO database was temporarily lost in the 1990s.

This paper will present sodium pump reliability failure probabilities for various conditions allowable with the U.S. facility CREDO data that has been recovered under NaSCoRD. The current sodium pump reliability estimates will be presented in comparison to estimates provided in historical studies. The impacts of EG&G's suggested corrections and various prior distributions on these reliability estimates will also be presented.

4:20 pm: Development of a Reliability Data Toolkit for Component Analysis in Liquid Waste Nuclear Facilities, J. Patrick Folk, C. Ray Lux, and Kevin R. O'Kula *(AECOM Technical Services)*

Within the U.S. Department of Energy (DOE) Complex are numerous nuclear facilities which rely on active and passive components to safely control, store, transport, test and process high-level waste (HLW) materials. A major category consists of high-level liquid waste storage, waste processing, vitrification, laboraty and related facilities. These facilities must be safely designed, constructed, and operated in accordance with applicable federal regulations, DOE orders, and DOE standards. To support robust design and safe operation processes, and the related facility Safety Basis, sound reliability data are required. Currently, defensible failure rate data for active and passive components are often not readily available from similar process facilities, difficult to apply because of differences in the operating environment, or are of earlier vintage equipment. A Reliability Data Toolkit has been developed to improve the reliability and related data availability for all types of DOE liquid waste nuclear facilities, particularly those categorized as Hazard Category 2 facilities under DOE-STD-1027.

Reliability Estimation and Data Analysis—II Continued

The Reliability Data Toolkit focus is initially on failure rate estimates for active and passive components for various types of postulated liquid waste accident conditions (e.g., pump and valve failure, spray leaks, piping breach). The customized quantitative data informs engineering evaluations, failure modes, effects, and criticality analyses (FMECAs), Safety Integrity Level (SIL) analyses and quantitative risk assessments (QRAs). The implementation of this methodology also allows contractor maintenance and system health organizations to share equipment performance trends, and proactively address potentially affected facilities. While initial baseline data sources include Savannah River Site and other DOE sites and national laboratories, U.S. Nuclear Regulatory Commission contractor reports, other data entry can be implemented as determined by the user.

This paper discusses development of the Reliability Data Toolkit methodology, representative trial use applications, and plans for extending the Toolkit in the near-term future.

4:45 pm: ESO-Based Online Reliablity Estimation Method for Nuclear Reactors, Zhe Dong, Miao Liu, Zhiwu Guo, Xiaojin Huang, Chao Guo, Qianqian Jia *(Tsinghua Univ)*

Since nuclear power still fulfills more than 11% of world electricity demand nowadays, the flexible operation of nuclear reactors can be positive to improve the penetration level of intermittent renewable energy (IRE) sources. Meanwhile, the operational flexibility of nuclear reactors results in the frequent variation of reactor process variables such as the neutron flux, fuel temperature and primary coolant pressure and temperature, which may fasten the degradation of equipment, and further leads to the necessity of developing online reliability monitoring methods for nuclear reactors in the context of fast and deep IRE penetration. To calculate the reliability of a nuclear reactor, it is necessary to known its failure rate. In this paper, the failure rate is assumed to be determined by a function of uncertainties caused by exterior disturbances and interior degradation, which transfer the estimation of failure rate to that of uncertainties. Then, an extended state observer (ESO) for estimating these uncertainties is newly proposed, which is represented in the natural coordinates and can provide globally bounded estimation for both unmeasurable state-variables and uncertainties induced by the disturbances of reactivity and secondary coolant temperature. The convergence of this ESO is analyzed based on Lyapunov direct method and the dissipation structure of nuclear reactors. This ESO is then applied to the state and disturbance observation of a nuclear heating reactor (NHR). Numerical simulation results verify the theoretical analysis, and illustrate the relationship between ESO parameters and the failure rate of a nuclear reactor. The operational reliability is further calculated based on the failure rate given by this ESO, which realizes the online reliability monitoring. Finally, a numerical simulation verification experiment is performed, and the corresponding results show the feasibility of this ESO-based operational reliability online estimation method.

Uncertainty Quantification

Chair: James E. O'Brien *(DOE)* **Location:** Blue Topaz **Time:** 3:30-5:10 pm

3:30 pm: Alternative Approach for Defining Truncation Limits, Ricky Summitt *(EPM)*

Model quantification is becoming a concern as the PRA is utilized increasingly for real time activities. In many cases this is driven by the need to reach ever diminishing levels of cut set truncation to reach an arbitrary convergence standard.

This paper will present an alternative approach to defining the truncation limits such that the model results can be considered adequately refined while yielding faster run times. The approach maintains the traditional truncation for the model of record but for applications uses the concept of completeness based on the insights obtained and the risk metrics associated with basic components in the model. An approach to convergence to define the expectation for higher level of truncation when supporting plant applications is also provided. The essence of completeness revolves around the ability to capture insights rather than count cut set values.

3:55 pm: Quantification of the Uncertainty Due to State-of-Knowledge Using ROAAM+ Framework for Nordic BWRs, Sergey Galushin, Dmitry Grishchenko, Pavel Kudinov *(KTH)*

ROAAM+ framework for Nordic BWR is a further development of ROAAM ideas, where development and application of the framework is based on iterative processes of refinement of knowledge, where risk analysis is used as a guiding tool in identification of the major sources of uncertainty. ROAAM+ framework employs extended treatment of safety goals, where both "possibility" and "necessity" of containment failure are considered in the analysis. One of the most important features of the ROAAM+ treatment is risk quantification in different "state-of-knowledge" situations, e.g. where complete, partial or no probabilistic knowledge available.

In order to assess the importance of the missing information, i.e. when no probabilistic knowledge is available, distributions of epistemic modelling parameters are considered as uncertain and sampling in the space of possible probability distributions of these parameters is performed.

The main goal of this work is to demonstrate an approach for risk quantification in different state-of-knowledge situations, and evaluate the effect of selection of the distribution families and parameters characterizing distributions on risk analysis results.

4:20 pm: An Approach to Fire Probabilistic Risk Assessment Modeling, Uncertainty Quantification and Sensitivity Analysis, P. Boneham, G. Georgiev, P. Guymer *(Jacobsen Analytics Ltd)*

As the nuclear industry pursues risk-informed and performance-based initiatives, the identification, characterization and evaluation of uncertainties is part of the process of using fire probabilistic risk assessment (FPRA) results in a risk-informed framework. Historically, while uncertainties have been recognized and identified in the detailed fire progression modeling performed for FPRAs, there have been some limitations and simplifications in their characterization and evaluation.

The consensus FPRA practice has been to address uncertainties using conservative modeling assumptions in many cases, rather than performing explicit quantification. Conservatism has been a by-product of this approach. In particular, the bounding, conservative, approach has been used to address uncertainties related to fire scenario frequencies and associated fire damage.

This paper describes a more complete method for the performance of detailed fire progression modeling within a FPRA, including quantitative uncertainty modeling. The method was developed by Jacobsen Analytics during the performance of FPRAs performed in support of NFPA 805 and subsequently enhanced and expanded as part of a project funded by the Electric Power Research Institute. The integration of uncertainty quantification as a fundamental part of the method removes the need for conservative, bounding, approaches to the selection of point values in the fire progression models.

Uncertainty Quantification Continued

The method includes systematic identification and characterization of parameter and modelling uncertainties. Uncertainties remaining after preprocessing are then propagated using single and two loop Monte Carlo simulations.

The simulation model provides several benefits: a) minimization of conservatism by eliminating the need for simplifying/bounding assumptions; b) the use of more flexible and detailed modelling; and c) more accurate representation of correlation of parameter uncertainties in fire initiators, fire growth, and suppression models.

The method is able to generate importance rankings for input uncertainties to the fire induced damage state analyses, based on how strongly each individual input uncertainty affects the resulting fire induced damage state uncertainty. In the future, insights about the importance of input uncertainties might be used to inform research efforts or to suggest plant changes to minimize the effect of these uncertainties.

Based on work performed by Jacobsen Analytics Ltd on behalf the Electric Power Research Institute, Inc. EPRI Project Manager A. Lindeman.

4:45 pm: **Understanding and Effectively Managing Conservatisms and Safety Margin In Safety Analysis—Non-Reactor Nuclear Facility Example**, Steven Krahn *(Vanderbilt Univ)*, Mohammad Modarres *(Univ of Maryland)*, James O'Brien *(DOE)*

This paper describes research performed into practical applications for understanding and managing conservatisms in safety analysis. This is a follow-on effort to research performed by the authors on methods and practices for understanding and managing conservatisms in safety analysis. It reviews conservatism that are incorporated in performing elements of safety analyses including in: the identification of hazards, the evaluation of the potential events that could cause the release of hazardous materials, and the evaluation of the potential frequency and consequences of the release. The paper then evaluates areas where some of the conservatisms could be reduced without impacting the calculated margin of safety by improved data and analysis and the reduction of uncertainty that are associated with the data or analysis model. Finally two specific areas are evaluated as examples where improved analysis or data that reduces uncertainty can result in a better understanding of the level of conservatism in the analysis and how the level of conservatism could be reduced without reducing the margin of safety of the facility.

Risk-Informed R&D Prioritization: Near-Term and Long-Term Needs–Panel

Cochairs: Nathan Sui *(NRC/RES)*, F. Ferrante *(EPRI)* **Location:** Emerald Salon Two **Time:** 3:15-5:45 pm

The prioritization of a nuclear organization's research and development (R&D) activities is a classic problem of decision making under uncertainty. The decision making process, typically involving decision makers and supporting staff within different organizational entities and levels, must consider R&D portfolio options whose potential benefits, costs, and risks are usually multi-dimensional, uncertain, and valued by different stakeholders differently. The recent increasing emphasis on the use of risk information (particularly related to public health and safety, but also including enterprise risks) has the potential to change current prioritization approaches.

The two panels, one addressing near-term R&D and the other addressing long-term R&D, will involve a discussion of current approaches, lessons, and challenges from the viewpoints of a variety of organizations. The discussion will start with a brief statement by each panelist to provide some initial thoughts. These remarks will then be followed by facilitated discussion involving the panel and the audience. It is expected that the panels will inform ongoing efforts by a number of organizations to improve their prioritization processes.

Panelists:
Near-Term:
Michael Cheok *(NRC/RES)*
Greg Krueger *(NEI)*
Cornelia Spitzer *(IAEA)*
Mohammad Modarres *(UMD)*
Estelle Sauvage *(EDF)*

Panelists:
Long-Term:
Robert Budnitz *(LBNL, retired)*
Ali Mosleh *(UCLA)*
Sal Golub *(DOE)*
Kelli Voelsing *(EPRI)*

Dynamic PSA Standard Development Initiation–Panel

Chair: Tunc Aldemir *(Ohio State)* **Location:** Emerald Salon One **Time:** 8:00-9:30 am

Main activities to initiate the development of a dynamic PSA standard will be described and discussed. These include identification of the technical challenges with understanding previous and ongoing efforts, ASME PRA standards and ASME/ANS standards for advanced reactors PRA, and definition of the scope of the dynamic PSA standard.

Panelists: Askin Guler Yigitoglu *(ORNL)*
Tunc Aldemir *(Ohio State)*
Zachary K. Jankovsky *(SNL)*
Martina Kloos *(GRS)*

Human Reliability and Human Factors—II

Chair: Rachel E. Vail *(AECOM Technical Services)* **Location:** Emerald Salon Two **Time:** 8:00-9:40 am

8:00 am: Organizational Factors in PRA: Twisting Knobs and Beyond, S. Peters, S. Morrow, S. Dennis, J. Lane *(NRC)*, Z. Ma *(INL)*, N. Siu *(NRC)*

Organizational problems have long been viewed within portions of the probabilistic risk assessment (PRA) community as important and perhaps even dominant contributors to nuclear power plant risk. This view has been reinforced by recent operational experience, most notably the Fukushima Dai-ichi reactor accidents. Attempts to explicitly incorporate organizational factors in a PRA context have a similarly long history. Nevertheless, it appears that there remains a lack of consensus regarding the extent to which such factors are already implicitly treated in PRA models, which PRA model elements need to be adjusted and to what extent, and what new PRA model elements are needed (e.g., to treat dependencies introduced by organizational structures, processes, and behaviors). This paper provides a perspective on these matters that is informed by operational experience and recent organizational research, as well as ongoing PRA-oriented development efforts documented in the literature. The paper does not suggest a "correct" approach to the treatment of organizational factors, but seeks to provide some considerations intended to help ongoing and future research efforts.

8:25 am: HRA and Dependency Analysis Insights from a Dominion Energy Model Update Project, Nicole Waugh, Allen Moldenhauer, Thomas John *(Dominion Energy Services, Inc.)*

The ASME/ANS Standard defines Human Reliability Analysis (HRA) as a structured methodology to identify potential human failure events (HFEs) and to estimate the probability of those events using data, models, and expert judgement. HRA is essential to include in the PRA to represent the as-built, as-operated power plant. During a recent Dominion Energy model update for Surry Power Station, an extensive update scope was developed for the HRA. The Surry post-initiator HRA scope included converting from SPAR-H methodology to EPRI HRA Calculator methodologies (HCR/ORE/THERP, CBDTM/THERP, etc), revising timing and scenario parameters for many HFEs, adding new post-initiator HFEs as part of the system fault tree and event tree development, and updating the dependency analysis using the HRA Calculator dependency module. The conversion and revision of the post-initiator HRA was time consuming and extensive, which then led to thousands of new combinations in the HRA dependency analysis. The ASME/ANS Standard requires that any potential dependencies among human failure events in the same accident sequences be assessed for the degree of dependency because the failure of one task or HFE can influence the likelihood of failure of a subsequent HFE, and if dependencies are not addressed then the risk may be underestimated. In general, the HRA dependency analysis methodology uses the following steps: 1. Identify cutsets with multiple HFEs, 2. Assess degree of dependence, 3. Address dependence in the PRA model quantification. Assessing the degree of dependence includes considering the use of a minimum joint HEP and how that will affect the dependence in the PRA model. Traditionally, a minimum joint HEP was applied to dependency analyses results, however the application of a minimum joint HEP has the possibility of skewing and artificially inflating risk metrics and risk insights. Referencing EPRI TR 3002003150, Dominion took the approach to assess a minimum joint HEP and document a sensitivity, and then remove the minimum joint HEP from the final dependency analysis results in the PRA model in an effort to retain the "best estimate" results from the dependency analysis and PRA model. Taking this approach, thousands of new combinations were include in the recovery rules and the PRA model results, leading to additional quantification time and failed quantifications at low truncation limits due to insufficient memory. This approach then expanded to the revision of the configuration risk model and led to undesired long quantification times in the configuration risk model. This paper will describe the HRA and dependency analysis methodology and outline the benefits as well as providing lessons learned from the Surry model update project.

8:50 am: Human Performance Modeling and Experimentation for Control Rooms, Monifa Vaughn-Cooke, Janell Joyner, Benjamin Knisely *(Univ of Maryland)*

Human operators interacting with complex control systems face safety-critical risks associated with stress and cognitive load, which can compromise human and system performance outcomes. In control rooms, the operator must be constantly vigilant and take appropriate actions while interacting with data management and control systems. These systems primarily rely on sensory cues to alert the operator of required interventions and provide decision support. Human factors design guidance is currently specified for control rooms and digital human-system interfaces (HSIs). However, there is currently a gap empirically linking the human factors control room design guidance to human performance outcomes. A human performance modeling and simulation study was performed to determine the effectiveness of alarm cueing strategies (visual, tactile, auditory) through measurement of neurophysiological predictors of cognitive load and human performance outcomes (accuracy, timing, and perceived workload). A Virtual Reality CAVE was used to simulate monitoring and decision making tasks. A cognitive task analysis was performed for the experiment and integrated with error classification models to further analyze the cognitive processes associated with error and propose design risk mitigation strategies. The results of this research will inform HSI best design practices for control rooms to reduce the risks associated with cognitive load, and thus improve operator performance and system safety.

9:15 am: An Illustrative, Interview-Based Risk Framework for Treatment of High-Stress Human Actions in Multiunit Nuclear Power Plant Accidents, Yinan Cai, Michael W. Golay *(MIT)*

Human Reliability and Human Factors—II Continued

Multiunit risk assessment has drawn growing attention after Fukushima Accident. Based on our review of operation experiences, shared systems, shared human management, physical proximity of units and electricity disturbance are significant multiunit risk contributors. In order to further understand multiunit dependencies in Fukushima Accident, we conducted interviews with TEPCO engineers about their experiences with the accident. The evidence from the interviews shows that accident propagation and human related events are most important risk contributors. In order to mitigate the accident, the operators need to evaluate the urgency of multiple units, diagnose the working status of the units and plant and implement response strategies. All of these tasks have to be finished under high mental stress and high situational uncertainty. Moreover, restored FLEX electricity and cooling systems are vulnerable to accident propagation, as due to hydrogen explosion from neighboring units. Given the time-dependent nature of these interaction, simply extending a single unit PRA to the whole site is not sufficient; new methods to quantify multiunit risk are needed.

In this paper, we provide an illustrative example of an interview-based multiunit PRA framework that addresses accident propagation and human-related events. A two-unit site with simplified safety systems is used to illustrate the approach. In the accident scenario of this two-unit site, all cooling methods except for FLEX are assumed to be unavailable for both units. Operators need to restore FLEX cooling in order to protect the units. Meanwhile, the restoration process of FLEX cooling for one unit can be disrupted by debris from a hydrogen explosion of the other unit. Except for physical interaction of two units due to propagation of hydrogen explosion debris, human-related dependencies of units are modeled as well. Specifically, both units are assumed to experience the same delay because of the shared management of the site. Meanwhile, onsite resources are assumed to be limited. The effects of resource distribution among units on success probabilities are analyzed in this paper. It turns out that there exists an optimal resource distribution when maximizing the probability of both units being successful. Results of this work provide theoretical base for modeling and decision making in future severe accident mitigation.

DOE Accident Analysis: Uncertainty, Conservatism and Testing to Reduce Conservatism

Chair: Kevin R. O'Kula *(AECOM Technical Services)* **Location:** Emerald Salon Three **Time:** 8:00-9:30 am

8:00 am: Review of Airborne Release Fraction Used for Toxicological Free-Fall Liquid Spills Based on DOE-HDBK-3010 Requirements, William H. Slagle, Patrick J. Snouffer, Robert L. Hanson *(Bechtel National, Inc.)*

Recent analysis for a spill from height, using the Ballinger equation, was done to the requirements specified in DOE-HDBK-3010-94 (DOE Handbook, Airborne Release Fractions/Rates and Respirable Fractions for Non-reactor Nuclear Facilities); however, part of the requirements are not clear. The guidance specified in Section 3.2.3.1 points to a previous note described in Section 3.2.2.3.2, of the handbook, for a "gross density" assumption for determining which Airborne Release Fraction (ARF) and corresponding Respirable Fraction (RF) values to use. The wording from Section 3.2.2.3.2 is as follows:

"For the sake of simplicity, a gross density distinction is made for determining which ARF and RF values to use. Any solution containing heavy metal salts where the liquid alone has a density in excess of ~1.2 g/cm3 is considered a "concentrated heavy metal solution" for assigning ARF and RF values (i.e., 1E-3 and 0.4). Any solution containing heavy metal salts where the solution alone has a density less than ~1.2 g/cm3 is considered an "aqueous solution" for assigning ARF and RF values (i.e, 2E-3 and 1.0)."

This assumption has been translated into the following guidance: if the density (or specific gravity - SpG) is less than or equal to 1.2 g/cm3 (SpG = 1.2), then the solution is aqueous per Section 3.2.3.1 and the Ballinger equation results are

"potentially non-conservative for low-density aqueous solutions modelled on the uranine data. In order to determine a bounding ARF for this subset of solutions, the ARF value calculated from the model [Ballinger Equation] are multiplied by a factor of 3 (empirical observation). This factor does not apply to the other types of solutions."

However, the subsequent sections in DOE-HDBK-3010-94 (Section 3.2.3.2 and 3.2.3.3) provide the ARF / RF for Free-Fall of Slurries and Viscous Solutions using the Ballinger equation without the factor of 3 applied to the results. More than half of the data points for both the slurries and viscous solutions have densities (or SpG) less than or equal to 1.2 g/cm3 (1.2). In addition, the source document for the data used in DOE-HDBK-3010-94 Section 3.2.3.1 does not define a low density solution threshold nor does it include a factor of 3 for low density solutions when using the Ballinger equation. This criteria is only provided in the DOE handbook without a supporting basis for the 1.2 g/cm3 threshold and an explanation of the basis for the use of the factor of 3.

The basis for these important factors needs to be understood by the analyst in order to perform a proper safety analysis. The authors reviewed the original data sources and evaluated the data (namely the density or specific gravity) and the basis for the application for the factor of 3 to understand the degree of conservatism that results from its application.

It is not disputed that the factor of 3 would yield conservative results, but these are artificial results that could lead to other issues:

- *Over-conservatism in the aqueous consequence results could drive a facility to add safety controls that are not needed and could potentially drive the cost of the facility up and slow or delay the construction time.*
- *If controls were not available to offset these overly conservative consequences, a pseudo residual risk would exist that would have to be acceptable to the regulator.*
- *The controls included because of these over-conservatisms would make the facility more difficult to operate.*
- *Understanding the basis of this data will allow for more judicious use of safety margins.*

TECHNICAL SESSIONS - 8:00 AM

DOE Accident Analysis: Uncertainty, Conservatism and Testing to Reduce Conservatism Continued

8:30 am: Estimation of the Conservatism in the Free-Fall Spill Source Term Correlation for High-Level Waste, Kevin R. O'Kula *(AECOM Technical Services)*, John E. McAllister *(HukariAscendent)*

Accident analyses for most nonreactor nuclear facilities operated by U.S. Department of Energy (DOE) contractors and U.S. Nuclear Regulatory Commission (NRC) fuel cycle licensees apply airborne release fractions (ARFs) and respirable fractions (RFs), mostly based on the same sets of experiments and resulting correlations. These correlations, documented in DOE-HDBK-3010-94 and NUREG/CR-6410, provide bounding correlations for different accident types. The correlation for a frequently postulated accident scenario is a free-fall spill of high-level waste (HLW) or radioactive liquid material from postulated breached tanks and pipes is an empirical model of ARF and droplet size distribution, and is referred to as the Ballinger correlation. It is based on the experimental work conducted in the 1980s by Ballinger and others. The experiments used small-volume (125 cm3 –1,000 cm3) test samples and short spill distance heights (1m – 3m) and aqueous solutions with densities of ~1.0 g/cm3 (uranine) to ~1.3 g/cm3 (uranyl nitrate hexahydrate). The major mechanism for radionuclide release is through the impact of liquid waste onto solid surfaces, and is a function of free-fall spill height for liquid spills of constant viscosity. When applied to free-fall spill scenarios involving liquid waste densities and spill heights postulated in many Design Basis Accident analyses, the Ballinger experimental correlation can result in estimated ARFs for elevated spills that may be overly conservative. However, application of alternative methods is often problematic and difficult to technically justify from test and model conditions to those that would apply to the bounding spill assumed in many nuclear nonreactor facilities. This paper estimates the conservatism inherent from two perspectives: (1) assuming that the terminal velocity of the liquid waste solution can increase to values greater than the terminal velocity of the liquid, and (2) comparing the Ballinger free-fall spill correlation to newer Pacific Northwest National Laboratory (PNNL) correlation for a more dispersive mechanism, that of a low-pressure, spray release. The first perspective suggests that the elevated spill ARF for liquid density ranging from 1.0 g/cm3 to approximately 1.3 g/cm3 (near that of high-solids liquid waste) will increase excessively by nearly the square of the spill height above 5.1m – 7.4m, depending on the density of the spilling solution. For the second perspective, a comparison with a contemporary spray release correlation, published in PNNL-22415, shows a factor of three to nine smaller ARF depending on the assumed area of the low-pressure breach, specifically from 1E-04 m2 to 8E-03 m2. Overall, this paper concludes that the Ballinger correlation based on the small volume, limited height experimental data is reasonably conservative (approximately a factor of three) for spill heights from 3m to 5m, but that excessive conservatism may result with heights greater than approximately 5m.

9:00 am: Fire-Induced Pressure Response and Failure Characterization of PCV/ SCV/ 3013 Containers, Ray Sprankle, Stony Reid *(SRNS)*

The Savannah River Site currently stores and manages oxide material in DOT Type B shipping packages tested to withstand transportation fires. The common type 9975 package typically consists of an outer container, and three inner layers: the Secondary Containment Vessel (SCV), the Primary Containment Vessel (PCV), and innermost, the 3013 container. The SCV and PCV are robust stainless steel screw lid containers with O-ring seals, while the 3013 is a welded, two-layer SS container. None of these inner containers have been tested to withstand fires. There is concern that an inner container exposed to a facility fire after removal from the shipping package could potentially reach high pressure. If a container fails in a high pressure condition, a much higher Airborne Release Fraction would result in higher radiological consequences. To address this, SRS has subcontracted with Sandia National Laboratories to perform fire testing of these containers. Bounding fire tests, to ASTM E1529 criteria, are being conducted while monitoring external flux, temperature, and container pressure in real time. Three phases of testing are currently planned. Phase 1 will determine bounding external conditions involving five different configurations. Phase 2 will evaluate various internal conditions (plastic, fill, moisture, etc.) Phase 3 will test 3013s to the bounding conditions determined in Phases 1 and 2. All work is being done to NQA-1 safety class quality assurance criteria. A DOE complex-wide Technical Advisory Committee of industry experts has been established to ensure comprehensive vetting. The subcontract is funded through Phase 1. Six PCVs have been modified and shipped to SNL, with an expected testing date in September, 2018.

A published paper is not expected to be available prior to April, 2019. However, Phase 1 testing is expected to be complete and preliminary results available. Phase 2 testing is expected to be in progress. A PowerPoint presentation will outline the project for conference participants.

TECHNICAL SESSIONS - 10:00 AM

Understanding of the Overall Risk Profile: Multiunit Context and Risk Aggregation Topics–Panel

Chair: Cornelia Spitzer *(IAEA)* **Location:** Emerald **Time:** 10:00 am-12:00 pm

The results and insights of risk assessment form an indispensable basis for risk-informed decision making, by providing comprehensive information on the overall risk profile. Understanding the overall risk profile allows decision makers to evaluate the priority, effectiveness and appropriateness of safety related decisions. Meanwhile, extension of the single NPP mindset to the multiunit level acknowledges the need for a more comprehensive understanding of various risk contributors, their aggregation and safety related decision making. The objective of the panel discussion is to share experiences and discuss challenges and benefits in the area of risk assessment in multiunit context and issues related to the aggregation of various risk contributors.

Panelists: Patricia Dupuy *(IRSN)*
Attila Bareith *(Nuclear Safety Research Inst)*
Zoltan Kovacs *(RELKO Ltd Eng & Consulting Services)*
Fernando Ferrante *(EPRI)*
Nathan O. Siu *(NRC)*

PSA 2019 Conference Grid by Day

Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
4/28/2019	Sunday	8:00	18:00	Topaz/Opal Prefunction	Registration	
4/28/2019	Sunday	8:00	18:00	Topaz/Opal/Emerald Prefunction	Exhibits (Set-up)	
4/28/2019	Sunday	9:00	12:00	Yellow Topaz & Blue Topaz	**1. Dynamic PSA Workshop—I** a.ADS (University of California at Los Angeles) b.ADAPT (Sandia National Laboratories)"	Professor Tunc Aldemir, The Ohio State University
4/28/2019	Sunday	12:00	13:00		Lunch on your Own	
4/28/2019	Sunday	13:00	16:30	Yellow Topaz & Blue Topaz	**2. Dynamic PSA Workshop—II** a. RAVEN (Idaho National Laboratory) b. PyCATSHOO (Electricité de France)	Professor Tunc Aldemir, The Ohio State University
4/28/19	Sunday	13:00	17:30	Opal 1	**3. SAPHIRE Workshop**	James K. Knudsen, Idaho National Laboratory
4/28/2019	Sunday	15:00	15:30	Topaz/Opal Prefunction	Mid-Afternoon Break	
4/28/2019	Sunday	18:00	21:00	Courtyard (Weather Permitting) or Emerald	**PSA 2019 Arrival Meet & Greet**	

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4/28/2019	Sunday	8:00	18:00	Topaz/Opal Prefunction	Registration	
4/28/2019	Sunday	8:00	18:00	Topaz/Opal/Emerald Prefunction	Exhibits (Set-up)	
4/29/2019	Monday	7:00	18:00	Topaz/Opal Prefunction	Registration	
4/29/2019	Monday	7:00	8:00	Opal 1	Continental Breakfast - Monday Chairs, Presenters, and Panelists	
4/29/2019	Monday	8:00	9:00	Opal 1	Spouse/Guest Breakfast	
4/29/2019	Monday	7:00	18:00	Opal 2	Speaker/Presentation Ready Station	
4/29/2019	Monday	7:00	18:00	Topaz/Opal/Emerald Prefunction	Exhibits	
4/29/2019	Monday	8:00	10:00	Emerald	**Monday Opening Plenary (OP1)** **a. Welcome Greeting - General Chair Kevin O'Kula** **b. Welcome from the Mayor Pro Tem, Peter Shahid, City of Charleston, South Carolina** **c. Welcome from Professor Mohammad Modarres, Technical Program Chair** **d. Keynote Presentation, Safe Enough? WASH-1400 and Its Legacy, Mr. Thomas Wellock, U.S. Nuclear Regulatory Commission Historian** **e. Presentation from Professor Mohammad Modarres to WASH-1400 Authors** **f. Remarks from WASH-1400 Authors: Richard Denning, Joseph Murphy, and Ian Wall**	Kevin O'Kula, AECOM TS Professor Mohammad Modarres, University of Maryland
4/29/2019	Monday	10:00	10:30	Topaz/Opal/Emerald Prefunction	Mid-Morning Break	
4/29/2019	Monday	10:30	12:15	Emerald	**MPS Opening Plenary Panel:** **27102 PRA Knowledge Management: Preserving Data and Information** **Professor Mohammad Modarres, Moderator**	Professor Mohammad Modarres, University of Maryland
4/29/2019	Monday	12:30	13:45	Emerald	**PSA 2019 Luncheon - Daniel Churchman, Fleet Engineering Director for SNC** **Sponsored by Westinghouse**	
4/29/2019	Monday	13:45	15:25	Emerald Salon One	**MTS1 Multi-Unit PSA and Risk Integration—I (4 papers)** 27149 Development of Multi-Unit PSA Model for the Case Study of the IAEA Project 26718 Seismic Correlation Modeling in Multi-Module PRAs 27126 Use of Risk Insights in the Practical Implementation of Integrated Risk-Informed Decision-Making Framework 26900 An Approach to Developing an Integrated Site Probabilistic Risk Assessment (PRA) Model	Cornelia Spitzer, IAEA
4/29/2019	Monday	13:45	15:15	Emerald Salon Two	**MTS2 Internal Events—I (3 papers)** 26641Simplified Structural Steel Analysis to Support Assumption of Loss of One Column for Building Structural Integrity 27035 An Approach for Apportioning Fire Scenario Frequencies to Induced Initiating Events 27151 Characterization of Interruptible and Growth Fires for Nuclear Power Plant Applications	Jeff Mitman U.S. NRC
4/29/2019	Monday	13:45	15:15	Emerald Salon Three	**MTS3 Working Group & International Program Insights (3 papers)** 27105 Summary of the results of 4th Mandate of PSA Working Group Established by Co-operation Forum for VVER Regulators 27409 BWROG Insights based on PRA Peer Review F&O Closure Workshops 26949 PWROG SAMG Implementation Lessons Learned	Dennis Henneke, GE-Hitachi

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Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
4/29/2019	Monday	13:45	15:15	Yellow Topaz	**MTS4 Risk-Informed Decision-Making—I (3 papers)** 27036 Quantitative Risk Analysis Support to Decision-Making for New Systems 27097 Use of Risk Insights in the Practical Implementation of Integrated Risk-Informed Decision-Making Framework 27038 Identifying Key Factors Affecting the Performance of Decision-Making Tasks Included in SAMGs	Professor Michelle (Shelby) Bensi, University of Maryland
4/29/2019	Monday	13:45	15:15	Blue Topaz	**MTS5 Passive System Reliability (3 papers)** 26876 Decision Making for Active and Passive Safety Systems Alternative: Preliminary Assessment 27027 Interfacing Passive System Performance Degradation Initiated by Nuclear Power Plant Operator Action 27100 Bathtub Shaped Hazard Rate functions Change Points Determination; Hazard Graph's Properties	Curtis L. Smith, Idaho National Laboratory
4/29/2019	Monday	13:30	15:15	Opal 1	**MPS1 27078 Understanding and Managing Conservatisms and Safety Margins to Support Safety Decisions**	James E. O'Brien, NNSA, U.S. DOE
4/29/2019	Monday	15:15	15:45	Topaz/Opal/Emerald Prefunction	**Mid-Afternoon Break**	
4/29/2019	Monday	15:45	17:25	Emerald Salon One	**MTS6 Multi-Unit PSA and Risk Integration—II (4 papers)** 27060 A Review of Selected Multi-Unit PRA Issues 27127 A Method for Considering Numerous Combinations of Plant Operational States in Multi-Unit PSA Models 27411 Simplified Methodology for Multi-Unit PSA Model 27162 Methodological Approach for a Hydrological Hazards PSA for a Multi-Unit Multi-Source Site	Andrea Maioli, Westinghouse Electric Company"
4/29/2019	Monday	15:45	17:15	Emerald Salon Two	**MTS7 Internal Events—II (3 papers)** 27152 Modeling of Personnel Suppression in Nuclear Power Plant Applications 27158 Radiative Heat Flux Zone of Influence for Open Fires and Electrical Enclosures Fires 27188 The Effect of Pressurizer Heaters on Spurious Pressurizer Main Spray Initiation, MSO 36, Scenario in a Reference Plant Fire PRA	Sunil D. Weerakkody, U.S. NRC
4/29/2019	Monday	15:45	17:25	Emerald Salon Three	**MTS8 Risk-Informed Decision-Making—II (4 papers)** 27121 Applying Risk-Informed Decision-Making to the Acceptance Criteria for Evaluating Leak-Before-Break Analyses in Piping Which is Susceptible to Primary Water Stress Corrosion Cracking Degradation 27015 Condition-Based Probabilistic Safety Assessment for an Induced Steam Generator Tube Rupture 26956 Development of 3+ Level Probabilistic Safety Assessment Methodology and Application for Akkuyu Nuclear Power Plant	Fernando Ferrante, Electric Power Research Institute (EPRI)
4/29/2019	Monday	15:45	16:45	Yellow Topaz	**MTS9 Risk Management (2 papers)** 26402 Discussion of Risk Aggregation in Three Dimensions for Various Risk Hazards 27063 Insights from Risk-Related Implementation of Reactor Pressure Vessel Water Inventory Control (RPV WIC)	Gerald Loignon, SCANA (ret.)
4/29/2019	Monday	15:45	17:15	Blue Topaz	**MTS10 Extended Sequences (3 papers)** 27207 PSA Evaluation of the New Independent Feedwater System at Ringhals NPP in Sweden 26947 Power Supply and Mitigation System Consdierations for Extended Loss of All AC Power Events 28205 Modeling Fire-Induced Main Control Room Abandonment in PRA Fault Trees	Felix Gonzalez, U.S. NRC
4/29/2019	Monday	15:45	17:15	Opal 1	**MTS11 Criticality Safety Insights (3 papers)** 27043 Defining Realistic Conservatism in Nuclear Criticality Safety Analysis 27133 Criticality Safety Insights for a Nuclear Waste Process Using Hazard Analysis 27167 Estimating the Probability of Multiple Misloads in Spent Fuel Casks for Light Water Reactor Systems	Professor Robert Hayes, North Carolina State University, Assistant Chair Herbert Carl Benhardt, AECOM Technical Services
4/29/2019	**Monday**	**18:00**	**21:00**	**South Carolina Aquarium**	**PSA 2019 Opening Reception - Hosted by Jensen Hughes**	

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Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
4/30/2019	Tuesday	7:00	18:00	Topaz/Opal Prefunction	Registration	
4/30/2019	Tuesday	7:00	8:00	Opal 1	Continental Breakfast - Tuesday Chairs, Presenters, and Panelists	
4/30/2019	Tuesday	8:00	9:00	Opal 1	Spouse/Guest Breakfast	
4/30/2019	Tuesday	7:00	18:00	Opal 2	Speaker/Presentation Ready Station	
4/30/2019	Tuesday	7:00	18:00	Topaz/Opal/Emerald Prefunction	Exhibits	
4/30/2019	Tuesday	8:00	9:30	Emerald	**Tuesday Opening Plenary (OP2)** **PSA methodologies for external hazards at nuclear power plants: Current status and future developments - Plenary (II)** **Robert J. Budnitz, Lawrence Berkeley National Laboratory, (retired)**	Professor Mohammad Modarres, University of Maryland
4/30/2019	Tuesday	9:30	10:00	Topaz/Opal/Emerald Prefunction	Mid-Morning Break	
4/30/2019	Tuesday	10:00	12:00	Emerald	**27134 Seismic Multi-Unit PSA: Special Challenges and Opportunities**	Robert J. Budnitz, LBNL (retired)
4/30/2019	Tuesday	12:00	13:30	Topaz/Opal/Emerald Prefunction	**Buffet Luncheon**	
4/30/2019	Tuesday	13:30	15:00	Emerald Salon One	**TTS1 SMR and Advanced Reactor PSA (3 papers)** 27138 Severe Accident Source Terms for Small Modular SFRs 27388 Development of a Methodology for Early Integration of Saftety Analysis into Advanced Reactor Design 27198 Probabilistic Risk Assessment of a Single-Failure-Proof Crane for Small Modular Reactor Refueling Operations	Tom Morgan, ENERCON Services, Inc.
4/30/2019	Tuesday	13:30	15:10	Emerald Salon Two	**TTS2 External Events—I (4 papers)** 26969 Once upon a time, there was a total loss of ultimate heat sink... 27023 A Study on Probabilistic Risk Assessment Methodology of External Hazard Combinations, - Identification of Hazard Combination Impacts on Air Cooling Decay Heat Removal System 26670 Risk-Reduction Credit for Very Early Warning Fire Detection: From FAQ to Fiction 27110 Incorporation of Spatial Variability of Ground Motions in a Seismic Multi-Unit Probabilistic Risk Assessment	Robert J. Budnitz, Lawrence Berkeley National Laboratory (retired)
4/30/2019	Tuesday	13:30	15:10	Blue Topaz	**TTS3 Level 1 and 2 PSA—I (4 papers)** 27209 Accident Sequence Probability in PSA 27020 Modeling Hydrogen Explosion in Level 1 PSA 27009 Practical Application of the Loss of Offsite Power Recovery Analysis using the Convolution Methodology 27229 Simplified/harmonized PSA: a generic modeling framework applied to precursor analysis	Jeff Gabor, Jensen Hughes
4/30/2019	Tuesday	13:30	15:00	Yellow Topaz	**TTS4 Digital I&C, Software Reliability, and Cyber Risk (3 papers)** 27106 Comparative Application of Digital I&C Modeling Approaches for PSA 27154 MODEL BASED RELIABILITY ANALYSIS OF DIGITAL I&C OF THE HOISTING EQUIPMENT IN NUCLEAR FACILITIES 26999 Development of Cyber-Attack Complexity Evaluation Model for Cyber Security of Nuclear Power Plants	Professor Tunc Aldemir, The Ohio State University
4/30/2019	Tuesday	13:30	15:00	Opal 1	**TTS5 Internal Events & Common Causes—III (3 papers)** 27056 Internal Flooding PRA Refinement by Partitioning of Pipe Rupture Frequencies 27021 Evaluation of Common Cause Failure by an Initiating Event for Multi-unit using Bayesian Belief Network 27224 Development of Inter Unit CCF Methods for Multi-Unit PSA	Rick Summit, EPM Inc.

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Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
4/30/2019	Tuesday	13:00	15:00	Emerald 3	**27275 Advancing HRA Technology: Short-Term and Long Term Needs**	Paul Amico, Jensen Hughes
4/30/2019	Tuesday	15:00	15:30	Topaz/Opal/Emerald Prefunction	**Mid-Afternoon Break**	
4/30/2019	Tuesday	15:30	17:15	Emerald Salon One	**27233 Learning from Experienced Nuclear Events: The Role of Precursor Analysis**	Wolfgang Kröger, ETH Zürich
4/30/2019	Tuesday	15:30	17:10	Emerald Salon Two	**TTS6 External Events—II (4 papers)** 27120 Integrating External and Internal Event Hazard Models at Nuclear Power Plants 27192 A Preparatory Study on Systematically Considering Combinations of External Events in the Design Basis and the Probabilistic Safety Assessment of NPP Paks 27058 Lessons Learned from Recent Seismic Risk Evaluations Including Probabilistic Risk Assessments to Support Regulatory Actions 27146 A new method to allocate combination probabilities of correlated seismic failures into CCF probabilities	Zoltan Kovacs, RELKO Ltd.
4/30/2019	Tuesday	15:30	17:00	Emerald Salon Three	**TTS7 Level 1 and 2 PSA—II (3 papers)** 26664 Assessment for SRV line break 27088 General Screening Criteria for Loss of Room Cooling in PRA Modeling 27074 On Assessing the Risk Related to Consequential Steam Generator Tube Rupture Events in Nuclear Power Plants 27096 Release Category Characterization; Towards a More Realistic Method	Gabriel Georgescu, IRSN
4/30/2019	Tuesday	15:30	17:10	Yellow Topaz	**TTS8 Risk-Informed Regulation—I (4 papers)** 27181 Office for Nuclear Regulation Risk Informed Regulatory Decision Making 27285 Assessing the Impact of TSTF 505 Initiative 4B Risk-Informed Completion Times on Baseline Risk 27059 Insights from Review of Seismic Probabilistic Risk Assessments in the Context of 10 CFR 50.69 26662 Risk-Deformed Regulation: What Went Wrong with NFPA 805	Susan Cooper, U.S. NRC George Flanagan, ORNL
4/30/2019	Tuesday	15:30	17:10	Opal 1	**TTS9 Low Power Risk, Accident Management and Emergency Planning (4 papers)** 27044 Power restoration timescales and probabilities: new data and a general theory 27029 Low Power Shutdown PRA Modelling Challenges and Recommendations 26124 NuScale's Emergency Planning Zone Methodology 27065 FLEX Equipment Reliability Data	Bruce Morgen, EPM Inc.
4/30/2019	Tuesday	15:15	17:15	Blue Topaz	**30118 PRA Standard Update**	Andrea Maioli, Westinghouse Electric Company
4/30/2019	Tuesday	17:30	21:30	Charleston Harbor	**Dinner Harbor Cruise Aboard Spirit Line Cruise Ship "Lowcountry"**	

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Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
5/1/19	Wednesday	7:00	13:00	Topaz/Opal Prefunction	Registration	
5/1/19	Wednesday	7:00	8:00	Opal 1	Continental Breakfast - Wednesday Chairs, Presenters, and Panelists	
5/1/19	Wednesday	8:00	9:00	Opal 1	Spouse/Guest Breakfast	
5/1/19	Wednesday	7:00	13:00	Opal 2	Speaker/Presentation Ready Station	
5/1/19	Wednesday	7:00	18:00	Topaz/Opal/Emerald Prefunction	Exhibits	
5/1/19	Wednesday	8:00	9:30	Emerald	**Wednesday Opening Plenary (OP3)** International perspective of ongoing and future PSA activities at the IAEA and its Member States - Plenary (III) Ms. Cornelia Spitzer Safety Assessment Section, Division of Nuclear Installation Safety, Department of Nuclear Safety and Security, International Atomic Energy Agency	Professor Mohammad Modarres, University of Maryland
5/1/19	Wednesday	9:30	10:00	Topaz/Opal/Emerald Prefunction	**Mid-Morning Break**	
5/1/19	Wednesday	10:00	12:00	Emerald Salon One	**WTS1 State-of-the-Art Consequence Analysis (SOARCA)/Uncertainty Analysis (4 papers)** 27145 State-of-the-Art Reactor Consequenece Analyses Project: Uncertainty Analyses for Station Blackout Scenarios 27164 State-of-the-Art Reactor Consequenece Analyses Project: Insights on Accident Progression and Source Term 27165 State-of-the-Art Reactor Consequenece Analyses Project: Insights on Offsite Consequences 27166 State-of-the-Art Reactor Consequenece Analyses Project: Insighs on Methodologies	Jeff Gabor, Jensen Hughes Carl Mazzola, PEC
5/1/19	Wednesday	10:00	12:00	Emerald Salon Two	**WTS2 Dynamic PSA—I (3 papers)** 27012 Integration of Recoveries into Dynamic Event Trees: A Case Study 27051 India-United States Collaboration on Advanced Dynamic Reliability Modeling 27055 Code Surrogate Development for Dynamic PRA 27142 Applications of Evidence Theory to Issues with Nuclear Weapons	Professor Tunc Aldemir, The Ohio State University
5/1/19	Wednesday	10:00	12:00	Emerald Salon Three	**WTS3 Level 1 and 2 PSA—III (4 papers)** 27419 Extension of a Level 2 PSA Event Tree Based on Results of a Probabilistic Dynamic Safety Analysis (Dynamic PSA) of Induced Steam Generator Tube Rupture (SGTR) 27108 Source Term Analysis for PWR ISLOCA Using MAAP5 27161 Recent Developments on a Level 1 PSA for a Research Reactor 27218 A Source Term Evaluation in a SGTR accident sequence using the MELCOR code	John E. McAllister, HukariAscendent
5/1/19	Wednesday	10:00	11:30	Opal One	**WTS4 Risk-Informed Regulation—II (3 papers)** 27408 Demonstration of NEI 18-04 RIPB Guidance for non-LWR Licensing Basis Development 27338 NRC Decommissioning Rulemaking 27136 PRA Maintenance and PRA Upgrade	Antonios Zoulis, U.S. NRC
5/1/19	Wednesday	10:00	12:00	Blue Topaz	**27155 Insights from Advanced and Small Modular Reactor PRA Development**	Sarah Bristol, NuScale Power
5/1/19	Wednesday	10:00	12:00	Yellow Topaz	**WTS5 Plant & Site Level PSA Applications—I (4 papers)** 27241 Insights from a WGRISK Activity on the Status of Site-Level PSA Developments 27034 TMRE Implementation Experience at Duke Energy and Southern Nuclear Company Pilot Plants 27049 Incorporation of Surveillance Frequency Control Program Risk Evaluations in PRA Models 27144 A Simplified Probabilistic Model for Flywheel Integrity Using “”R”	Zhegang Ma, Idaho National Laboratory
5/1/19	Wednesday	12:00	20:00	Opal 1	**ASME/ANS RA-S-1.3 Level 3 PRA Standard Working Group Meeting**	
5/1/19	Wednesday	12:00	22:00	PSA 2019 Mid-Week Break	**Tours to Plant Vogtle, High Level Waste Facilities at Savannah River Site, Hunley Submarine Museum, Charleston Walking Tours and time on your own to explore Charleston**	

PSA 2019 Conference Grid by Day

Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
5/2/19	Thursday	7:00	13:00	Topaz/Opal Prefunction	Registration	
5/2/19	Thursday	7:00	8:00	Opal 1	Continental Breakfast - Thursday Chairs, Presenters, and Panelists	
5/2/19	Thursday	8:00	9:00	Opal 1	Spouse/Guest Breakfast	
5/2/19	Thursday	7:00	13:00	Opal 2	Speaker/Presentation Ready Station	
5/2/19	Thursday	7:00	12:00	Topaz/Opal/Emerald Prefunction	Exhibits	
5/2/19	Thursday	8:00	10:00	Emerald	**Thursday Opening Plenary (OP4)** **PSA Research and Education at Universities: History, Impact, Challenges, and Future Outlook - Plenary (V)** **Professor Ali Mosleh, Moderator, (Garrick Institute of the Risk Sciences, UCLA)** **Panel: George Apostolakis (MIT Emeritus Prof.)** **Mohammad Modarres (Prof. UMD)** **Ali Mosleh (Prof. UCLA)** **Akira Yamaguchi (Prof. Univ. of Tokyo)** **Katrina Groth (Assistant Prof. UMD)** **Tunc Aldemir (Ohio State)** **Wolfgang Kröger (ETH Zürich)**	Professor Ali Mosleh, Garrick Institute of Risk Sciences, UCLA
5/2/19	Thursday	10:00	10:30	Topaz/Opal/Emerald Prefunction	Mid-Morning Break	
5/2/19	Thursday	10:30	12:30	Emerald	**27231 Perspectives on Nuclear Safety Since the Three Mile Island Event: Learning from the Past 40 Years (VI)** **Dr. Robert A. Bari, Chair, Brookhaven National Laboratory, Senior Physicist Emeritus, (retired)** **Dr. Robert J. Budnitz, Lawrence Berkeley National Laboratory, (retired)** **Dr. Robert E. Henry, Fauske and Associates, Inc. (retired)** **Dr. Roger J. Mattson, Consultant**	Robert A. Bari, Brookhaven National Laboratory, (retired)
5/2/19	Thursday	12:30	13:30	Topaz/Opal/Emerald Prefunction	Buffet Luncheon	
5/2/19	Thursday	13:30	15:00	Emerald Salon One	**ThTS1 Dynamic PSA—II (3 papers)** 27099 Simulation Based Dynamic Event Tree Analysis 27117 Mutual Integration of Classical and Dynamic PRA 29590 Dynamic Probabilistic Risk Assessment with PyCATSHOO: The Case of the Emergency Power Supply of a Nuclear Power Plant	Zachary Jankovsky, Sandia National Laboratories
5/2/19	Thursday	13:30	15:10	Emerald Salon Two	**ThTS2 Human Reliability Analysis and Human Factors—I (4 papers)** 27260 Models for Human Performance Improvement 27062 Human Reliability Analysis Quantification Guidance for Main Control Room Abandonment Scenarios in Fire PRAs: What's New and When Can Existing Methods Be Used? 27128 The Use of Expert Judgment to Support Human Reliability Analysis of Implementing FLEX Equipment 27081 Human Action Dependency Development in the Age of Automation	Mary Presley, Electric Power Research Institute
5/2/19	Thursday	13:30	15:10	Opal 1	**ThTS3 Level 3 PSA (4 papers)** 27032 Level 3 PSA Application for Akkuyu Nuclear Power Plant 27217 Ingestion Dose Evaluation in A Food Chain Model for Consequence Analysis 27129 Development and Status of the ASME/ANS RA-S-1.3 Level 3 PRA Standard 25805 A Focused Sensitivity Study on the Key Input Parameters Important to Long-Term Level 3 PSA Metrics	Nathan Bixler, Sandia National Laboratories
5/2/19	Thursday	13:30	15:10	Yellow Topaz	**ThTS4 Safety Goals,Risk Metrics, and Guidance Updates (4 papers)** 27037 Re-Evaluating the Current Safety Goal Policy 27125 Technical Evaluation of the Margins Between Established Risk Goals and Health Objectives for Nuclear Power Plants 27213 Application of Qualitative Importance Measures 27150 Overview of the Society of Fire Protection Engineers (SFPE) latest Engineering Guide to Fire Risk Assessments, 2nd Edition	S. Tina Ghosh, U.S. NRC

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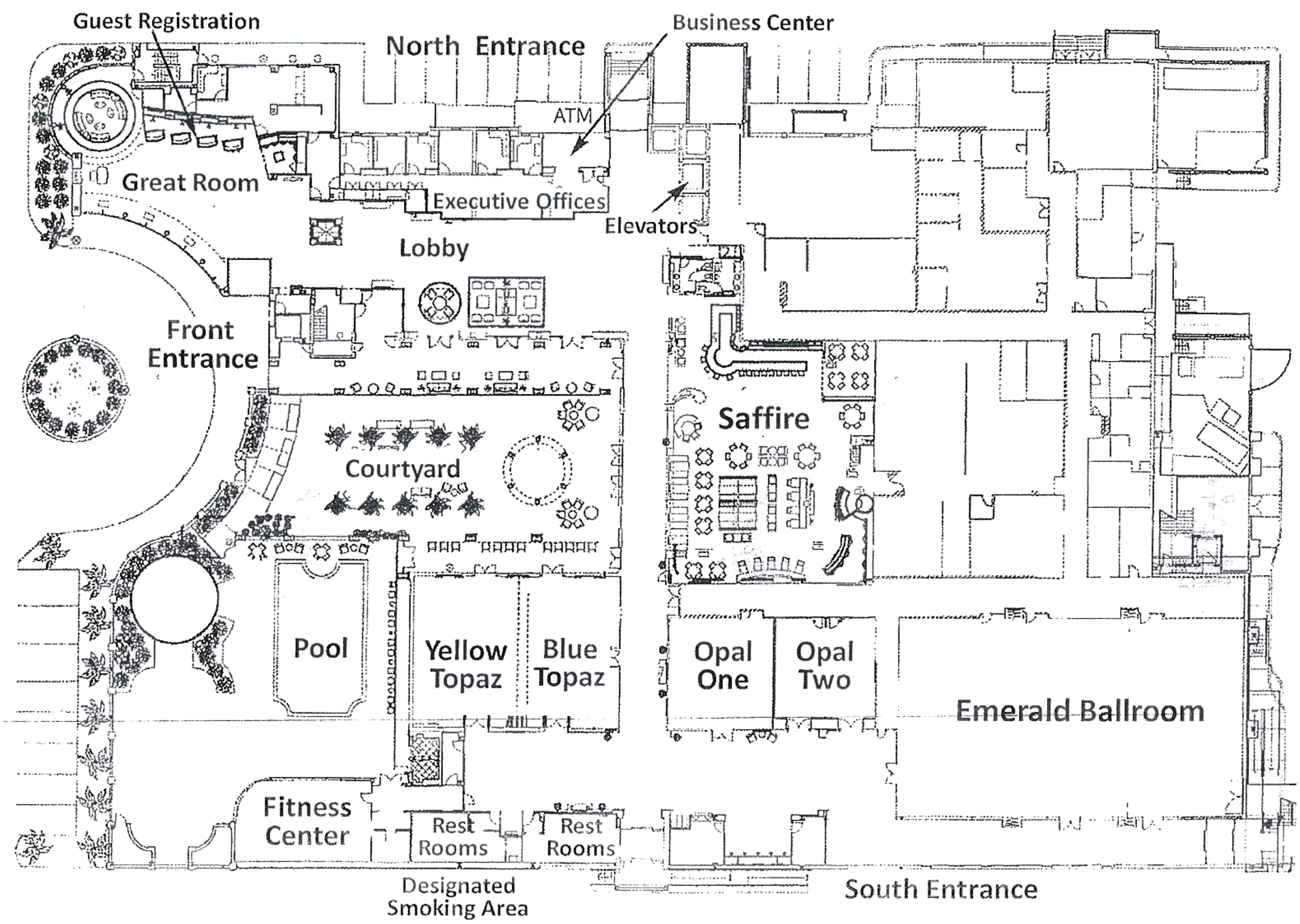
Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
5/2/19	Thursday	13:30	15:10	Blue Topaz	**ThTS5 Reliability Estimation and Data Analysis—I (4 papers)** 27108 ANALYSIS OF LOSS-OF-OFFSITE-POWER EVENTS 1987-2017 27095 Pilot Application of SACADA Database for Feed and Bleed Operator Action 27013 A complex network analysis for balanced design verification 27184 A Guidance for the Scoping and the Frequency of a PRA Data Update	Richard H. (Chip) Lagdon, Bechtel National Inc.
5/2/19	Thursday	13:30	15:10	Emerald Salon Three	**ThTS6 Plant & Site Level PSA Applications—II (4 papers)** 27109 Complex Modeling for Surveillance Test Interval Extensions 27057 Application of Electrical Power Recovery in the South Texas Project (STP) PRA Model 27077 Whole-site Risk Characterization Approaches in Canada: Regulatory and Technical Challenges 27014 Reliability analysis of a dynamic system using Petri net and comparison with Smart Component Methodology	Marina Röwekamp, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS)
5/2/19	Thursday	15:00	15:30	Topaz/Opal/Emerald Prefunction	Mid-Afternoon Break	
5/2/19	Thursday	15:30	17:00	Emerald Salon One	**ThTS7 Dynamic PSA—III (3 papers)** 27173 ATF: A Dynamic PRA Comparison 27169 Preliminary Methodology and Scenarios for Integrated Safety/Security Dynamic Probabilistic Risk Assessments 27016 A Dynamic Safety Margins Estimation with a Limited Number of PWR Large Break LOCA Simulations	Valentin Rychkov, EDF
5/2/19	Thursday	15:30	17:00	Opal 1	**ThTS8 Education, Training, and Knowledge Management (3 papers)** 27104 Communicating PRA Concepts to Non-Practitioners 26981 RISKAUDIT and CNEN Cooperation on Probabilistic Safety Analysis	Maeley K. Brown, AECOM Technical Services
5/2/19	Thursday	15:30	17:10	Emerald Salon Three	**TTS9 Advances in Computer Tools and HRA (4 papers)** 27418 The GRS Source Term Prognosis Software FaSTPro for PWR and BWR Spent Fuel Pools 27156 New Functions and Features Associated with EPRI HRA Calculator Version 5.2(R) 29625 SAPHIRE's Current “”State-of-Practice”” to Meet PRA Demands 27143 Human Reliability Dependency Analysis and Configuration Risk Management	Rachel E. Vail, AECOM Technical Services
5/2/19	Thursday	15:30	17:00	Yellow Topaz	**ThTS10 Reliability Estimation and Data Analysis—II (3 papers)** 27130 Sodium Pump Performance in the NaSCoRD Database 27191 Development of a Reliability Data Toolkit for Component Analysis in Liquid Waste Nuclear Facilities 27124 ESO-based online reliability estimation method for nuclear reactors	John E. McAllister, HukariAscendent
5/2/19	Thursday	15:30	17:10	Blue Topaz	**ThTS11 Uncertainty Quantification (4 papers)** 27079 Alternative Approach for Defining Truncation Limits 27375 Quantification of the Uncertainty Due to State-of-Knowledge Using ROAAM + Framework for Nordic BWRs 27214 An Approach to Fire Probabilistic Risk Assessment Modeling, Uncertainty Quantification and Sensitivity Analysis 27107 Understanding and Effectively Managing Conservatisms and Safety Margin	James E. O'Brien, U.S. DOE
5/2/19	Thursday	15:15	17:45	Emerald Salon Two	**27070 Risk Informed R&D Prioritization: Near-Term and Long-Term Needs**	Nathan Siu, U.S. NRC
5/2/19	Thursday	18:30	21:00	Emerald	**PSA 2019 Banquet**	

PSA 2019 Conference Grid by Day

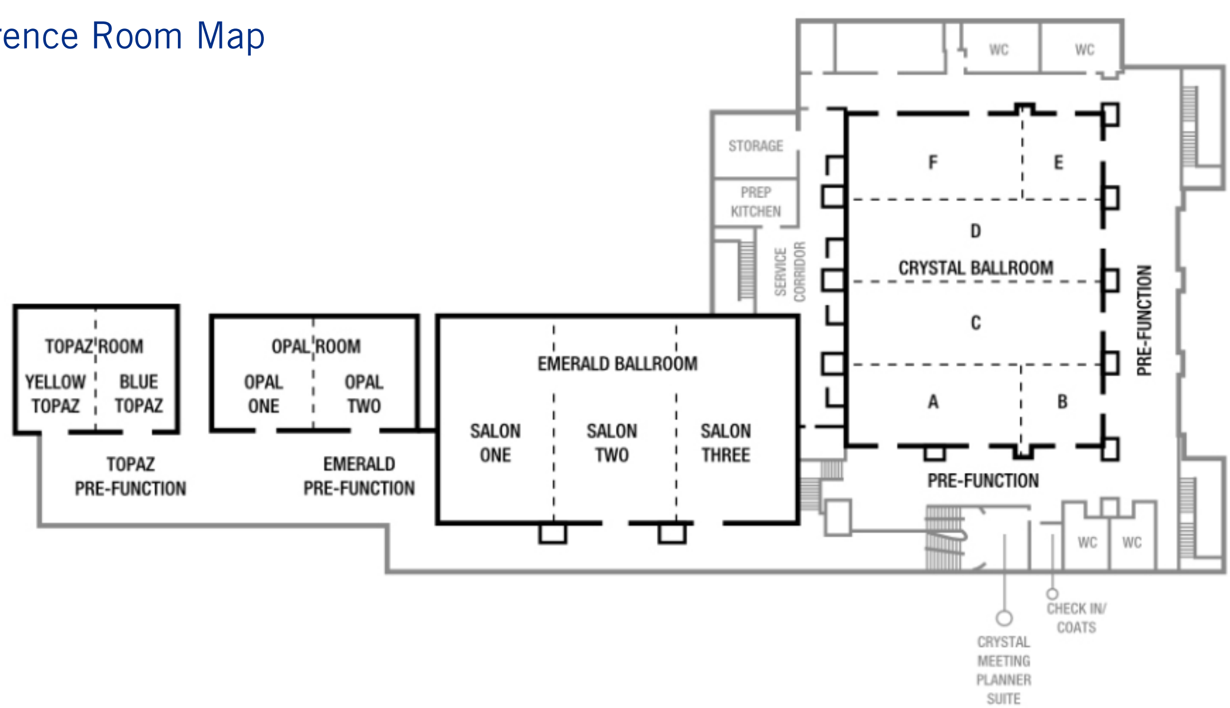
Date	Day	Start Time	Ending Time	Function or Event Space or Room	Session or Event	Session Chair or Lead Workshop Facilitator
5/3/19	Friday	7:00	12:00	Topaz/Opal Prefunction	Registration	
5/3/19	Friday	7:00	8:00	Opal 1	Continental Breakfast - Friday Chairs, Presenters, and Panelists	
5/3/19	Friday	8:00	9:00	Opal 1	Spouse/Guest Breakfast	
5/3/19	Friday	7:00	10:00	Opal 2	Speaker/Presentation Ready Station	
5/3/19	Friday	8:00	9:30	Emerald Salon One	**27263 Dynamic PSA Standard Development Initiation**	Professor Tunc Aldemir, The Ohio State University
5/3/19	Friday	8:00	9:40	Emerald Salon Two	**FTS1 Human Reliability Analysis and Human Factors—II (4 papers)** 27069 Organizational Factors in PRA: Twisting Knobs and Beyond 27028 HRA and Dependency Analysis Insights from a Dominion Energy Model Update Project 27228 Human Performance Modeling and Experimentation for Control Rooms 27448 An Illustrative, Interview-based Risk Framework for Treatment of High-Stress Human Actions in Multiunit Nuclear Power Plant Accidents	Rachel E. Vail, AECOM Technical Services
5/3/19	Friday	8:00	9:30	Emerald Salon Three	**FTS2 DOE Accident Analysis: Uncertainty, Conservatism & Testing to Reduce Conservatism (3 papers)** 29876 Review of Airborne Release Fraction Used for Toxicological Free-Fall Liquid Spills Based on DOE-HDBK-3010 Requirements 30161 Estimation of the Conservatism in the Free-Fall Spill Source Term Correlation for High-Level Waste 27089 Fire-Induced Pressure Response and Failure Characterization of PCV / SCV / 3013 Containers	Kevin O'Kula, AECOM Technical Services
5/3/19	Friday	9:30	10:00	Topaz/Opal/ Emerald Prefunction	Mid-Morning Break	
5/3/19	Friday	10:00	12:00	Emerald	**26403 Understanding of the Overall Risk Profile: Multiunit Context and Risk Aggregation Topics**	Cornelia Spitzer, IAEA
5/3/19	Friday	13:00	17:00	Topaz	**MACCS Workshop—I**	Nathan E. Bixler, Sandia National Laboratories
5/3/19	Friday	13:00	17:00	Opal	**WGRISK Meeting**	
5/4/19	Saturday	8:00	12:00	Topaz	**MACCS Workshop—II**	Nathan E. Bixler, Sandia National Laboratories

Hotel Layout

PSA 2019 Lobby Level



Conference Room Map

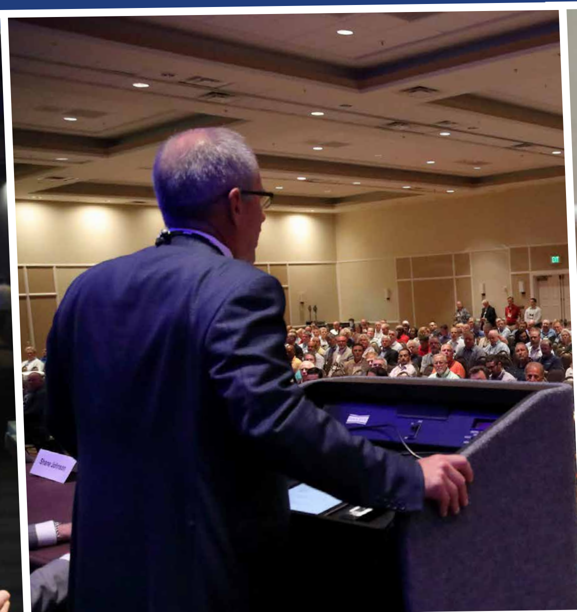


Save the date

ANS PSA 2021

17th International Topical Meeting on Probabilistic Safety Assessment and Analysis

September 26 – 30, 2021 | Columbus, Ohio



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