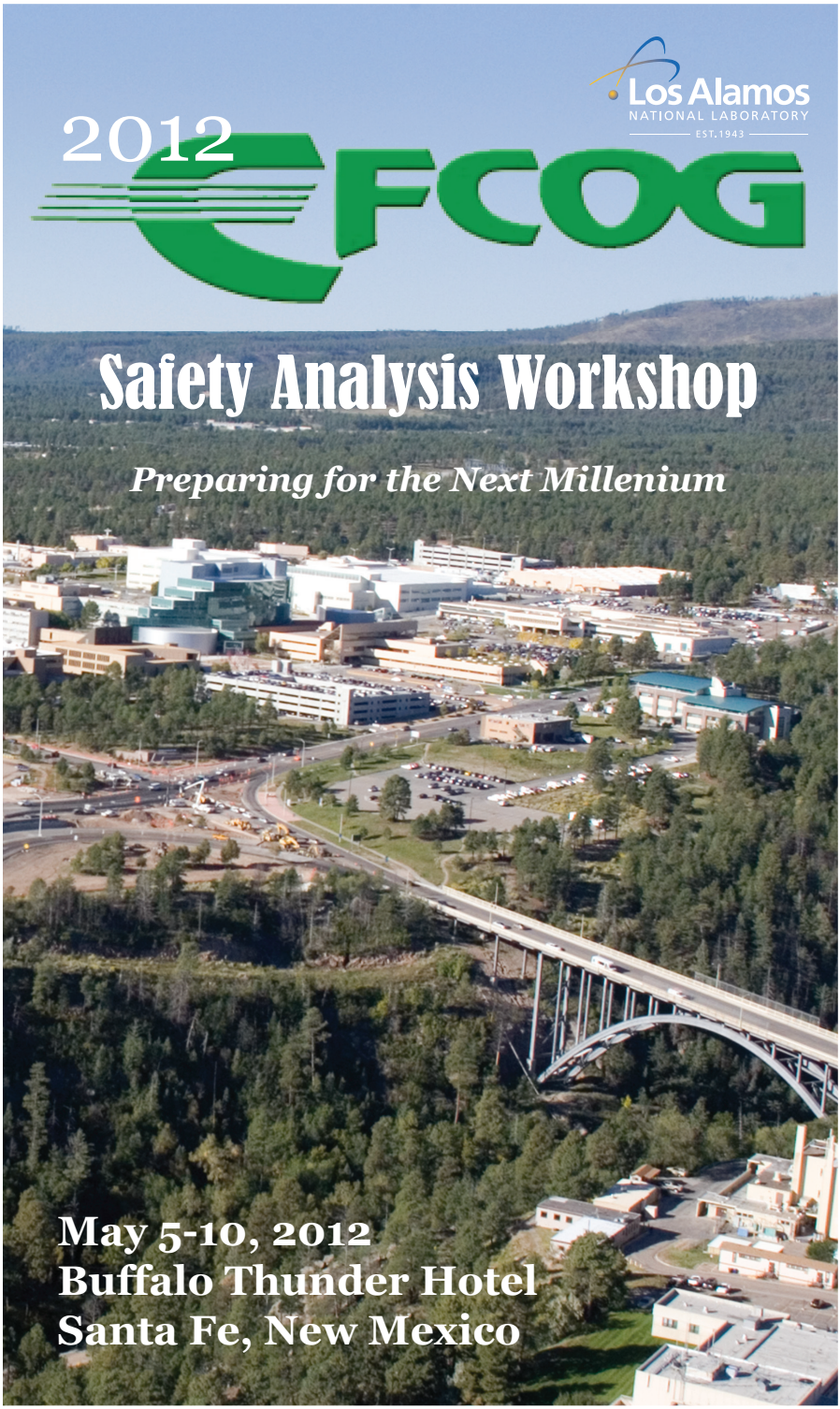
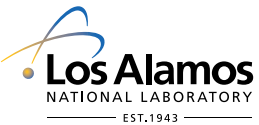

Los Alamos
NATIONAL LABORATORY
EST. 1943
2012
FCOG
Safety Analysis Workshop
Preparing for the Next Millenium
May 5-10, 2012
Buffalo Thunder Hotel
Santa Fe, New Mexico



LA-UR-12-20931

From Rob McKeehan, Safety Analysis Working Group Chair

Welcome to SAWG's 21st annual Safety Analysis Workshop.

These are challenging times. Perhaps they all are, but these seem especially so. It's a small nuclear community that we live and work in and that binds us all, for better or worse – be it contractor, DOE, oversight, commercial power, or international nuclear. As we have experienced these seemingly periodic and traumatic nuclear accidents – TMI, Chernobyl, and recently Fukushima – they grip the public's attention like nothing else, reinforce their fears, and place us all in the position of being an apologist. We have to wonder if this is indeed our Workshop's theme of preparing for the next nuclear millennium or merely a brief period of nuclear historical note. However, in looking back, in each one of these and other earlier nuclear accidents, as well as other searing tragedies in engineering – Bhopal, Challenger, "failsafe" blowout preventers – brought with them hard lessons learned and humility that we must do better, be more "what if," and not be so myopic on just meeting the regulations. So if there is any group in the DOE community that needs to accept the challenges of Fukushima and its predecessors, it is us, the Safety Analysis Working Group. Let's use this workshop as a stepping stone. I encourage you to engage in the subgroups, ask questions, listen, get trained, be open, step up, and discover at least one thing here to take back to your home site for implementation.

And not the least, enjoy your time and take advantage of this historic and fascinating Santa Fe/Los Alamos area. You might just rub shoulders with the ghost of Dr. Oppenheimer.

Carry on,

Rob McKeehan

SAWG chair

Rob McKeehan

Rob has over 20 years experience at Oak Ridge National Laboratory (ORNL) in leadership associated with the Laboratory's Facility Safety program. This program includes responsibilities for safety basis development and maintenance, the unreviewed safety question process, system engineering, and hazard categorization. For approximately three of these years he served in assignment as the facility manager of a nuclear Hazard Category 2 facility. Prior to working at ORNL, Rob worked for 17 years in the nuclear safety program for the Tennessee Valley Authority's power reactors. Rob has been involved with SAWG for most of his time while at ORNL and has served as chair of the Safety Basis subgroup and vice-chair of SAWG before assuming his current position as chair.

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Workshop Website

For the most up to date information, go to the workshop's website at https://sbts.lanl.gov/sawg2012/.

Workshop Sponsors

Every year the Safety Analysis Workshop is sponsored by strong leaders in the nuclear industry, and this year is no exception. It is with much appreciation and great thankfulness that we acknowledge their involvement and due diligence in pushing safety of the nuclear industry to new heights.

Workshop Host

Los Alamos NATIONAL LABORATORY EST.1943

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About EFCOG

The Energy Facility Contractors Group (EFCOG) is a self-directed group of contractors of U.S. Department of Energy (DOE) facilities.

The purpose of the EFCOG is to promote excellence in all aspects of operation and management of DOE facilities in a safe, environmentally sound, secure, efficient, and cost-effective manner through the ongoing exchange of information and corresponding improvement initiatives.

The four primary objectives of EFCOG are:

- Promote, coordinate, and facilitate the active exchange of successful programs, practices, procedures, lessons learned, and other pertinent information of common interest that have been effectively utilized by DOE contractors and can be adapted to enhance operational excellence and cost effectiveness for continual performance improvement by other DOE contractors.

- Identify and address issues of common interest (redundant with scope). Focus on active personal exchanges of management and technical information among contractors (redundant with scope).

- Utilize interfaces with organizations such as, but not limited to, the Nuclear Energy Institute (NEI), Electric Power Research Institute (EPRI), Institute of Nuclear Power Operations (INPO), Training Resources and Data Exchanges (TRADE), Association for Excellence in Reactor Operations (AERO), Nuclear Security Information Exchange (NSIE), to promote cooperation and interchange information, as appropriate, and minimize duplication of efforts.

- Interact with DOE in ways that produce value-added change for both DOE and the contractor community.

www.efcog.org

EFCOG Membership

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Waste Control Specialists
Wastren Advantage, Inc.
Wastren-EnergX Mission Support, LLC
Water Management, Inc.

(March 26, 2012)

About SAWG

The Safety Analysis Working Group (SAWG) is a working committee whose intent is to facilitate the objectives of the Energy Facility Contractors Group (EFCOG) as related to the particular area of Safety Analyses. The purpose of the working group is to promote excellence in the Department of Energy safety analyses programs through information sharing and application of lessons learned to other facilities.

The objectives of the SAWG are to:

- Operate within the framework of EFCOG.
- Provide planning and actions necessary to see that the overall objectives of the EFCOG come to fruition in the area of safety analyses.
- Promote, coordinate, and facilitate the active exchange of successful safety analyses programs, practices, procedures, lessons learned, and other pertinent information of common interest on safety analyses, which have been effectively utilized by M&O contractors.
- Promote training on safety analyses by sharing of management and technical information among contractors through mechanisms such as workshops, sub-working groups, and seminars.

www.efcog.org/wg/sa

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CR - Company Representative

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CR

General Workshop Information

Registration Hours – The registration desk, located on the Promenade of Buffalo Thunder, will be open for registration and information:

- Friday, May 4, 3:00 to 9:00 pm
- Saturday through Thursday, May 5-10, 7:00 am to 5:00 pm

Session Breaks – Breaks are scheduled for each morning and afternoon of the workshop. Refer to the workshop schedule for times.

Meals – Buffalo Thunder offers several dining options to suit your needs. Please refer to their website for details: www.buffalothunderresort.com/santa-fe-dining/index.cfm. A hosted lunch is provided on Tuesday.

Workshop Reception – Funding permitting, a welcoming reception will be held Monday evening, beginning at 6:00 pm.

Location of Meetings and Training Sessions – All meetings and training sessions will take place in the Buffalo Thunder. Meeting and training room assignments are provided in this booklet. The latest information on the workshop schedule, including any changes in paper presentations, is available at the registration desk.

Transportation – Transportation will be provided for those registered for the Bradbury Science Museum tour for Monday afternoon and for the Radiological Laboratory/Utility/Office Building (RLUOB) tour on Thursday. Tours will leave from the registration table.

Online Information –
https://sbts.lanl.gov/sawg2012/index.shtml


Hilton
Santa Fe Golf Resort & Spa at Buffalo Thunder
The Location:
20 Buffalo Thunder Trail
Santa Fe, New Mexico
United States 87506
Tel: (505) 455-5555
Fax: (505) 455-0200
Escalator to Casino
Events
POJOAQUE BOARDOOM
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Retail Space
MAIN ENTRANCE
LOCATOR MAP
BUFFALO THUNDER
Exit 177
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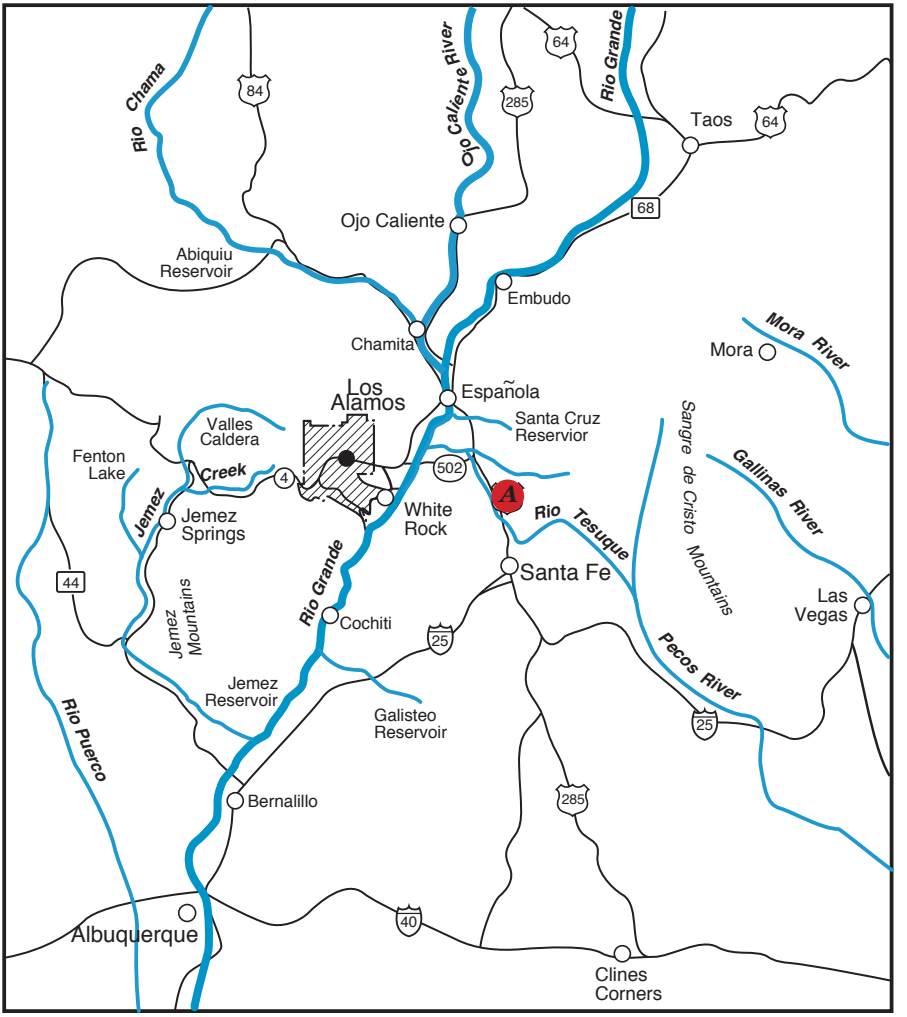
RESORT & CASINO®


MEETING SPACE ENTRANCE
MESA ROOM
A
B
MEETING SPACE ENTRANCE
PREFUNCTION
TERRACE ON THE JEMEZ
TEWA BALLROOM
A
BAY 3
B
C
BAY 2
(Meetings)
A
BAY 1
B
C
PREFUNCTION
A
CALDERA ROOM
B
A
BARRANCA ROOM
B
REGISTRATION & GENERAL INFORMATION
OFFICE
PREFUNCTION
SERVICE CORRIDOR
B
VISTA ROOM
A
PUEBLO BALLROOM
A
BAY 3
B
BAY 2
BAY 1
B
C
C
PREFUNCTION
LOADING DOCK
COAT RM
TO CASINO
PREFUNCTION
ESCALATOR
STARBUCKS
Events



A *Buffalo Thunder*

Things To Do in and Around Santa Fe

Bandelier National Monument
http://www.nps.gov/band/index.htm

Bradbury Science Museum
http://www.lanl.gov/museum/

Fuller Lodge
http://www.fullerlodgeartcenter.com/

New Mexico Tourism Department
http://www.newmexico.org/

Santa Fe Convention and Visitors Bureau
http://communityconventioncenter.com/index.html

Santa Fe Museums
http://maps.google.com/maps?f=q&source=s_q&hl=en&geocode=&q=Museums+near+Santa+Fe,+NM&sll=35.586968,-105.89653&sspn=0.274734,0.531464&gl=us&ie=UTF8&hq=Museums&hnear=Santa+Fe,+New+Mexico&z=12

Santa Fe Plaza
http://en.wikipedia.org/wiki/Santa_Fe_Plaza

Eight Northern New Mexico Pueblos
http://www.santafenm.info/pueblos.htm

Training

Hazard Analysis

Saturday May 5, 8 am to 5 pm

Don Siebe , Jim Tingey
Los Alamos National Laboratory

This one-day course focuses on general process and detailed process hazard studies. Techniques covered include the following: Checklist Analysis, PHA, What-If Analysis, What-If/Checklist Analysis, HAZOP Analysis, and FMEA. This course also provides the working knowledge and skills needed to perform a comprehensive assessment of facility hazards and to provide a qualitative risk perspective to help in decision making for risk reduction. These evaluation techniques may be used to perform hazard analyses at a DOE facility.

DOE STD 1189

Saturday May 5, 8 am to 5 pm

Richard Englehart, Longenecker and Associates

Brad Evans, Pacific Northwest National Laboratory

This one-day course provides the central ideas and themes of DOE-STD-1189 and conveys lessons learned from project management implementation of the Standard as needed by Safety Basis personnel. The course was developed to show how project management, engineering design, and safety analysis can be successfully implemented.

DOE STD 3009

Saturday May 5 & Sunday May 6, 8 am to 5 pm

John Farquharson, ABS Consulting
Julie Johnston, Energy Solutions

This two-day course is designed to provide safety analysts with the knowledge and skills needed to develop a non-reactor nuclear facility Documented Safety Analysis (DSA) in accordance with requirements of 10 CFR 830, DOE Safe Harbor Standard 3009-94, and concepts from DOE Guide 421.1-2, Implementation Guide for Use in Developing Documented Safety Analyses to Meet Subpart B of 10 CFR 830.

Accident Analysis

Sunday May 6, 8 am to 5 pm

James Kuropatwinski, Los Alamos National Laboratory
Lisa Pansoy-Hjelvik, Los Alamos National Laboratory

This one-day course is designed to provide safety analysts with the knowledge and skills needed to understand the formal quantification of a limited subset of accidents from the hazard evaluation of a nuclear facility. The course also covers the general process used to carry forward a limited subset of accidents that bound the envelope of accident conditions to which a facility operation could be subjected.

Fire Consequence Evaluation

Sunday May 6, 8 am to 5 pm

Allan Coutts
URS Safety Management Solutions

This one-day course provides the important concepts of fire analysis as needed by Safety Basis personnel.

Technical Safety Requirements

Monday, May 7, 8 am to 5 pm
James O'Neil
Los Alamos Site Office, NNSA

This one-day course provides safety analysts with the basic knowledge to understand Technical Safety Requirements (TSRs) in accordance with requirements of 10 CFR 830 and concepts from DOE Guide 423.1-1A in a mannerthat is acceptable for DOE/NNSA approval and assure that TSR controls could be implemented.

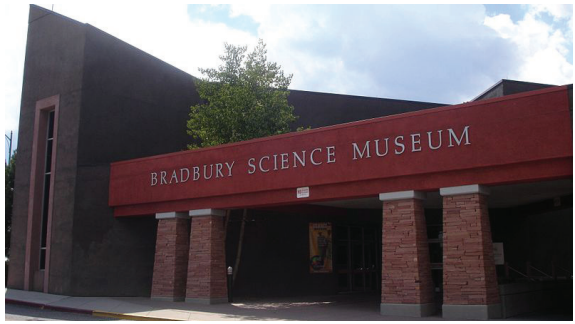
Plutonium Metallurgy

Monday, May 7, 8 am to 5 pm
Dr. Karl P. Staudhammer
Los Alamos National Laboratory

This one-day course provides the important concepts of plutonium science and metallurgy as needed by Safety Basis personnel.

Tour Information

Bradbury Science Museum



The Bradbury Science Museum was founded in 1963 and is named for the Laboratory's second director, Norris E. Bradbury, who served from 1945 to 1970. The Bradbury Science Museum's primary mission is to interpret Laboratory research, activities, and history to official visitors, the general public, and Laboratory employees; to promote greater public understanding of the Laboratory's role in national security programs; to assist the taxpaying public in making informed judgments in these matters; and to contribute to visitors' knowledge of science and technology and to improve the quality of math and science education in northern New Mexico.

Radiological Laboratory/Utility/Office Building

The Radiological Laboratory, Utility, and Office Building (RLUOB) consists of 19,500 square feet of radiological laboratory space, office space for 350 employees, and incident command and emergency response capabilities. It is the newest less-than Hazard Category 3 radiological facility at the Laboratory.

Plenary Speaker — Kevin Smith

Site Office Manager
Los Alamos Site Office
National Nuclear Security Administration
U.S. Department of Energy

Kevin W. Smith serves as Manager of the National Nuclear Security Administration's Los Alamos Site Office (LASO). As LASO Manager, he is responsible for administering the Los Alamos National Laboratory (LANL) Management and Operations (M&O) contract and all Federal activities on the site. Smith leads a Federal team of about 175 employees who oversee management, security, quality assurance, environment, health, safety, nuclear non-proliferation, and national security missions at the LANL.

Smith's career spans 36 years with a number of senior leadership positions with the Department of Defense and the Department of Energy. He came to LASO from NNSA's Y-12 Site Office where he served as Deputy Manager. Just before arrival he served as the Acting Manager of NNSA's Kansas City Plant.

He joined DOE and the Senior Executive Service in 2004, when he served as the Assistant Manager for the Nuclear Material Stabilization Project at the Savannah River Site. In that capacity he was responsible for two nuclear chemical reprocessing facilities, processing and storing special nuclear materials, spent nuclear fuel, and the Savannah River National Laboratory.

Prior to joining DOE, Smith served in the U.S. Air Force in a number of capacities to include the Director of Safety for Air Combat Command where he was responsible for Air Combat Command's flying, weapons, and industrial safety programs for 100,000 personnel worldwide. In addition he served as an F-16 Fighter

Squadron Commander, the Deputy J3 for U.S. Forces Korea, and Commander of 49th Operations Group at Holloman Air Force Base, New Mexico. He has flown F-4, F-16, and F-117 Stealth Fighter aircraft.

A native of Olympia, Washington, Smith is a graduate of the U.S. Air Force Academy with a bachelor's degree in physics. He also holds a bachelors degree in mathematics and a master of science degree in management from Troy State University. He and his wife, Dorothy, live in Los Alamos.

Plenary Speaker — Charlie Anderson

Associate Director
Nuclear and High Hazard Operations
Los Alamos National Laboratory

Charlie Anderson has 30 years of experience with increasing responsibilities in executive, technical, operations, program, and project management of nuclear materials disposition, nuclear materials production, nuclear waste management, and nuclear and coal fired power generation programs with Los Alamos National Lab, Nuclear Management Partners, URS corporation, Washington Group International,, the Department of Energy and the Tennessee Valley Authority. Served in senior management positions as a key member of these organizations with responsibilities in leadership, management, oversight, strategic planning, and execution of startups, transitions, and problematic programs. Responsibilities have included annual budgets of $6.2 Billion and workforces of 3,000 Federal and 30,000 contractor personnel. Work has typically involved bridging technical, business, and political agendas to solve problems in order to achieve programmatic and operational goals and objectives. Performance has always demonstrated a focus on progress and personal commitment. Have lead key cross cutting initiatives most of these organizations, which includes commercial, government, international, and government contractor.

Currently, Associate Director of Nuclear & High Hazard Operations at Los Alamos National Lab. Previously, General Manger of Nuclear Management Partners, the Management & Operations contractor consortium for the Sellafield operations in the northwest of England. Within the consortium, responsibility included serving as Business Unit Vice President with the Washington Division of the URS Corporation. Prior to joining URS, previously held the

position of Principal Deputy Assistant Secretary of Environmental Management, with the Department of Energy in Washington, DC, from 2005 through 2007. Provided innovative and sound leadership in a number of management positions at the Savannah River Site for the Department of Energy including Deputy Site Manager, Assistant Manager for High Level Waste; Director, Office of Defense Nuclear Nonproliferation; Director, Nuclear Materials Disposition; Director, Reactors and Spent Fuel Division; Director, High Level Waste Programs Division; and Director, Liquid Waste Division. In 1990, transferred from the Tennessee Valley Authority. Initially started career with the Tennessee Valley Authority as a construction project engineer at the Yellow Creek Nuclear Plant construction site. Relocated to the corporate engineering office in several system engineering positions which lead to Browns Ferry Nuclear Plant leading the system engineering efforts for a three unit operating nuclear plant. Last position with the Tennessee Valley Authority was Special Projects Manager at Browns Ferry Nuclear Plant with responsibility for the site program management of multi-discipline, site-wide problem recovery projects.

Plenary Speaker — Pam Horning

EFCOG Sponsoring Director for SAWG
Vice President
Babcock & Wilcox Technical Services Group

Ms. Horning has 28 years of experience in the nuclear industry associated with commercial nuclear operations, naval nuclear core manufacturing and DOE nuclear weapons production and operations. Her experience includes assignments including BWXT's commercial nuclear manufacturing facility, Rocky Flats, the Y-12 Nuclear Weapons Complex (NWC), and Lawrence Livermore National Security (LLNS). Ms. Horning is experienced in the operational and safety performance at commercial and DOE facilities that comprise complex, high consequence nuclear and national security operations. She currently holds the position of Vice-President with Babcock & Wilcox – Technical Services Group.

During her career, Ms. Horning has held a variety of leadership positions in Quality Assurance, Engineering, Operations, Emergency Management, and Project and Program Management. Ms. Horning holds a Bachelor of Science degree in Chemical Engineering from Purdue University and Masters of Engineering Administration from George Washington University.

Ms. Horning has been a member of EFCOG since 2002 and actively champions the objectives of EFCOG. She held the position of EFCOG Chair from 2006 to 2010. She is the EFCOG Sponsoring Director for the Safety Analysis Working Group and has previously sponsored the Energy and Infrastructure Management Working Group.

Plenary Speaker — Robert C. Nelson

Chief Safety Officer
Office of Safety Management
Office of Environmental Management

Dr. Nelson has over 44 years of experience in the areas of safety analysis, risk management, reactor safety and licensing, nuclear weapon safety, nuclear safety, criticality safety, radiation safety, environmental management and restoration, and space nuclear power and propulsion programs. His experience and expertise include project management, regulatory compliance, risk assessments, safety assessments, management assessments, and operational readiness reviews. His experience includes involvement with DOE, DOD, NRC, IAEA, and NASA. His experience includes management of safety analysis for DOD, NRC and DOE facilities, independent review of safety and environmental documentation and activities for the Office of River Protection Vitrification Plant and Tank Waste Remediation System (TWRS) and Richland Operations Office Spent Nuclear Fuel (SNF) Program at the Hanford Site, serving as lead technical advisor and chairman for over twenty (20) contractor and DOE readiness reviews at multiple sites including Buildings 559 and 707 at Rocky Flats, Project W460 at the Plutonium Finishing Plant at Hanford, the B696 Waste Facility at Lawrence Livermore National Laboratory, the Remote Handled Waste Facility at the West Valley Demonstration Site, the TRU Waste Processing Center at Oak Ridge and many others. Dr. Nelson served as the lead executive officer for licensing and start-up of DOD reactors (including liaison with NRC and DOE), reviewed proposed new reactor designs for the Department of the Air Force, and has provided launch safety and risk assessments for space nuclear payloads to the Executive Office of the President (Office of Science and Technology Policy) for the Ulysses, Galileo, Cassini, Pluto

New Horizons, and Mars Science Laboratory missions. Dr. Nelson supported measurement of intrinsic radiation associated with the U. S. nuclear weapon complex and served as the lead test recovery officer for all initial engineering and development tests associated with the ground, air, and sea launched cruise missile programs. His additional experience includes development and management of a large personnel dosimetry system, management of major radioactive material licenses under the NRC, safety review work for the development of space and terrestrial reactor systems, and preparation or review of numerous safety analysis reports.

Presently Dr. Nelson serves as Chief Safety Officer for Assistant Secretary for Environmental Management, Deputy Assistant Secretary Safety, Security, & Quality Programs, Office of Safety Management.

Dr. Nelson has served on multiple standards committees for the American Nuclear Society and Health Physics Society. His main contribution was on the ANSI/ANS 15 Standards Committee for Research and Test Reactors.

Dr. Nelson has authored over 20 manuscripts and papers on multiple topics with respect to nuclear safety, radiation safety, and fundamental radiation biophysics topics.

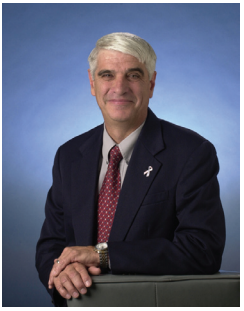
Plenary Speaker — Don F. Nichols

Associate Administrator for Safety and Health
Chief of Defense Nuclear Safety
National Nuclear Security Administration

Dr. Don F. Nichols is serving as the Associate Administrator for Safety and Health and Chief of Defense Nuclear Safety in the National Nuclear Security Administration (NNSA). His organization enables the NNSA mission in the areas of nuclear and occupational safety, directly supporting the Administrator and senior managers throughout the NNSA enterprise. He has held positions in the Federal government over the past thirty years, focusing on nuclear safety since 2001. He joined NNSA in 2005 as one of the charter members of the office of the Chief of Defense Nuclear Safety. Previously, he served for four years on the technical staff at the Defense Nuclear Facilities Safety Board, providing safety oversight of nuclear activities across the Department of Energy complex.

Dr. Nichols retired from the Air Force in 2001. He began his military service as an enlisted aircraft mechanic before being commissioned in 1987. He subsequently served in a variety of positions in the Air Force space nuclear reactor and nuclear weapons programs. He served two years as director of the Air Force TOPAZ space nuclear reactor assessment program, three years performing nuclear weapon design physics analysis at Los Alamos National Laboratory, nine months leading the Air Force cruise missile nuclear weapons program, and two years as a senior staff member for the Nuclear Weapons and Counterproliferation Agency at the Pentagon. He holds a Doctor of Philosophy degree in engineering physics and a Masters of Science degree in nuclear engineering from the Air Force Institute of Technology, and a Bachelors of Science degree in nuclear engineering from the Pennsylvania State University.

Luncheon Speaker — Sidney M. Gutierrez



Director and Chief of Safety for Environment, Safety, Health and Emergency Management
Sandia National Laboratories

Safety and the Space Shuttle - an Astronaut's Prospective

Sid is currently the Director of Radiation Protection, Waste Management and Environment, Safety and for Sandia. He was formerly Director of the Nuclear Energy & Global Security Technologies Center, the Systems Assessment and Research Center and Manager of the Proliferation Assessment Program. The program includes satellites, space payloads, airborne and ground sensors, ground stations, vulnerability assessments, and analysis tools.

He holds a BS in Aeronautical Engineering – Distinguished Graduate - from the United States Air Force Academy and an MA in Management from Webster University. He currently is a Member of the Board of Directors of Roadrunner Food Bank, Goodwill Industries of New Mexico, Rocket Crafters, Inc., the Air Force Scientific Advisory Panel, and the New Mexico Spaceport Authority.

He is retired from the Air Force at the rank of Colonel where he flew the F-15, F-16, T-38 and many other aircraft while serving as a fighter pilot, test pilot, and instructor pilot. He joined NASA as an astronaut in 1984. On his first mission, Sid served as pilot on the crew of STS-40 Spacelab Life Sciences (SLS-1), a dedicated space and life sciences mission. On his second mission, Sid served as Commander of STS-59 Space Radar Laboratory (SRL-1), part of Mission to Planet Earth.

Friday, May 4, 2012

Time	Event	Location
3:00 - 9:00	Registration	Promenade

Training Schedule

Saturday, May 5, 2012

Time	Event	Location
7:00 - 5:00	Registration/Information	Promenade
8:00	Hazard Analysis Training	Tewa 3A
	DOE STD 1189 Training	Tewa 3B
	DOE STD 3009 Training	Tewa 3C
~9:30	Break	
10:00	Training (continued)	
11:30	Lunch (on your own)	
1:00	Training (continued)	
~2:30	Break	
3:00	Training (continued)	
5:00	End of day	

Sunday, May 6, 2012

Time	Event	Location
7:00 - 5:00	Registration	Promenade
8:00	Accident Analysis Training	Tewa 3A
	Fire Consequence Training	Tewa 3B
	DOE STD 3009 Training	Tewa 3C
~9:30	Break	
10:00	Training (continued)	
11:30	Lunch (on your own)	
1:00	Training (continued)	
~2:30	Break	
3:00	Training (continued)	
5:00	End of day	

Workshop Schedule

Monday, May 7, 2012

Time	Location	Location
7:00 – 5:00	Registration/information	Promenade
	Sponsor exhibition booths	
8:00	Unreviewed Safety Question subgroup meeting	Barranca B
	Technical Safety Requirements (training)	Caldera A
	Plutonium Metallurgy (training)	Caldera B
	Safety Software working group	Vista A
	Accident Analysis Subgroup Meeting	Barranca A
~9:30	Break	
10:00	Training (continued)	
	Safety Basis subgroup meeting	Barranca A
	Criticality Safety subgroup meeting	Barranca B
11:30	Lunch (on your own)	
12:30	LANL tour leaves	
1:00	Hydrogen Safety interest group	Barranca B
~2:30	Break	
3:00	Meetings (continued)	
5:00	End of day	

Workshop Schedule

Tuesday, May 8, 2012

Time	Event	Location
7:00 – 5:00	Registration/information	Promenade
	Sponsor exhibition booths	
	* private meeting(s) (ask to reserve)	Barranca B
8:00	Meet and Greet	Promenade
	Posters	
9:00	**Plenary Session**	Pueblo 2,3
	Kevin Smith, Manager of Los Alamos Site Office, NNSA	
9:20	**Charlie Anderson**, Associate Director, Los Alamos National Laboratory	
9:45	**Pamela Horning**, Vice-President, Babcock & Wilcox - Technical Services Group	
10:20	**Bob Nelson**, Chief Safety Officer, Office of Safety Management, Office of Environmental Management	
10:50	**Don Nichols**, Associate Administrator for Safety and Health, Chief of Defense Nuclear Safety - NNSA	
11:30	Lunch (provided by the generosity of our sponsors)	Tewa 3
1:00	Posters (continued)	
	Accident Analysis I (paper session)	Caldera A
	Hazards Analysis (paper session)	Caldera B
	Safety Analysis and Controls (paper session)	Barranca A
~2:30	Break (authors present posters)	
3:00	Paper sessions (continued)	

Session Schedule

Tuesday, May 8, 1:30 - 5:00 pm, Caldera A

Session 1: Accident Analysis I Chair: Jerry Hansen	
1:00	**Jason P. Andrus, Dr. Chad L. Pope** Derivation of Accident-Specific Material at Risk Equivalency Factors
1:30	**Richard L. Garrett** One System Integration Project Team
2:00	**John Hargreaves** Propagation Limits of Burning Embers Lofted by Buoyant Plumes
~2:30	**Break**
3:00	**Andrew M. Vincent III, Scott Elliott** Dispersion Modeling Analysis for SRS
3:30	**Jorge Schulz, Thomas R. McDonnell** Adapting Dispersion Software to DOE Standard 3009
4:00	**Eric P. Hope, Scott Elliott, and Luke A. Wiencek** Comparison of Parameters for Modeling Tritium Dispersion

Tuesday, May 8, 1:00 pm

Derivation of Accident-Specific Material at Risk Equivalency Factors

Lead Author: Jason P. Andrus
Additional Author: Dr. Chad L. Pope
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Corresponding Author e-mail: Jason.Andrus@inl.gov

A novel method for calculating material at risk (MAR) dose equivalency developed at the Idaho National Laboratory (INL) now allows for increased utilization of dose equivalency for facility MAR control. This method involves near-real time accounting for the use of accident and material specific release and transport. It utilizes all information from the committed effective dose equation and the five factor source term equation to derive dose equivalency factors which can be used to establish an overall facility or process MAR limit. The equivalency factors allow different nuclide spectrums to be compared for their respective dose consequences by relating them to a specific quantity of an identified reference nuclide. The ability to compare spectrums to a reference limit ensures that MAR limits are in fact bounding instead of attempting to establish a representative or bounding spectrum which may lead to unintended or unanalyzed configurations. This methodology is then coupled with a near real time material tracking system which allows for accurate and timely material composition information and corresponding MAR equivalency values. The development of this approach was driven by the complex nature of processing operations in some INL facilities. This type of approach is ideally suited for facilities and processes where the composition of the MAR and possible release mechanisms change frequently but in well defined fashions and in a batch-type nature.

Tuesday, May 8, 1:30 pm

One System Integration Project Team

Richard L. Garrett
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The WTP Project and Tank Farm Project are moving jointly into a new phase of preparations for the commissioning and operation of the Waste Treatment Plant (WTP). This has been achieved through the establishment of a joint One System Organization Integrated Project Team (IPT) of Bechtel National, Inc. (BNI) and Washington River Protection Solutions (WRPS). Collectively, the attributes of the integrated One System approach increase the likelihood of achieving early LAW operation and Initial Plant Operations.

The One System IPT is a major step in developing and executing the programs that will be critical to successful waste feed delivery and WTP startup. It will combine WTP and WRPS capabilities in a mission-focused model that is clearly defined, empowered, and cost-efficient. This promotes an alignment of requirements and project definition, regulatory strategy, technical issue resolution, schedule integration, and commissioning planning.

The One System IPT will be focused on meeting consent decree commitments, lowering costs and risks and accelerating completion of the overall DOE River Protection Project mission.

Tuesday, May 8, 2:00 pm

Propagation Limits of Burning Embers Lofted by Buoyant Plumes

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Objective of Work Being Reported: Paper will discuss new theoretical work for modeling the trajectories of lofted burning embers resulting from small-scale ground fires.

Relationship of the Work to the Overall Interests of DOE Safety Analysis: This paper provides a theoretical basis for predicting the trajectory limits of burning embers lofted by small-scale wildfire-generated buoyant plumes. This theory allows for calculation of minimum safe standoff distances for foliage and other combustible materials relative to DOE hazardous materials sites.

Results of Work: This paper discusses modeling burning embers as spheres, cylinders, and disks; calculation of ember burnout rates and mass extinction times; and predicted propagation distances of embers lofted by buoyant plumes of heated air resulting from small wildfires (less than or equal to 40 MW). The influence of advecting winds on burnout rates and propagation distances is examined.

Tuesday, May 8, 3:00 pm

Dispersion Modeling Analysis for SRS

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A plan is being executed to address the dispersion parameters and deposition velocity (DV) issues at SRS. Actions within this plan include evaluation of meteorological data collection, evaluation of meteorological data normalization methods (e.g. EPA-454), development of SRS specific deposition velocities for particulates and tritium oxide (waters), evaluation of dispersion coefficient options (i.e. Tadmor-Gur, Briggs, etc.), and evaluation of surface roughness values appropriate for onsite and offsite receptors. This data is used in the Melcor Accident Consequence Code System, version 2 (MACCS2) computer code to support the sites Documented Safety Analyses. The initial phase of work is the development and agreement on analysis parameters and methods to be used in future dose calculations. The analysis and results to date will be described.

Tuesday, May 8, 3:30 pm

Adapting Dispersion Software to DOE Standard 3009

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Objective: Enable use of MACCS2 to generate 95th percentile overall atmospheric dispersion factors in accordance with the statistical treatment described in NRC Regulatory Guide 1.145 as required by DOE-STD-3009-94.

Relationship to DOE Safety Analysis Interests: Appendix A, section A.3.3, of DOE-STD-3009-94 states that the 95th percentile χ/Q should account for variations in distance to the site boundary as a function of direction and be consistent with the statistical treatment of χ/Q values described in regulatory position 3 of NRC Regulatory Guide 1.145. MACCS2 does not process the χ/Qs consistent with Regulatory Guide 1.145. MACCS2 determines directionally independent χ/Qs for different percentiles at a specific distance; thus, there is no way of obtaining the direction-dependent 95th percentile χ/Q without post-processing.

Results: The paper will outline the methodology in which MACCS2 expanded outputs for each hour of 10 years of site meteorological data and each distance interval were post-processed in an Excel spreadsheet. A spreadsheet was also used to independently validate the MACCS2 version 2.5 ATMOS module in accordance with DOE Quality Assurance Order DOE O 414.1C and Safety Software Guide DOE G 414.1-4.

In the course of this effort, two software errors were discovered in MACCS2. The first error was in the treatment of dispersion coefficients from a lookup table, and the second involved the method of accounting for plume meander. The paper will describe the errors and the evaluation that determined that the potential impact on existing WTP accident analyses is negligible.

Tuesday, May 8, 4:00 pm

Comparison of Parameters for Modeling Tritium Dispersion

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Consequences calculated for accident analysis scenarios involving large tritium releases (greater than one million Curies) are influenced by the selection of input variables. These include, among others, dispersion coefficients, chemical form, plume buoyancy and deposition velocity. When analyzing a deposition velocity for appropriate use, several processes are involved that can be described by an aggregate plume depletion velocity. The value of the aggregate factor selected for plume depletion may have enough impact to ultimately influence controls required for the facility. This paper includes a brief discussion of the implications of input variable selection and factors that make up an aggregate depletion velocity for tritium oxide. Conversion of the initial tritium gas source term by several energetic events into tritium oxide is evaluated by comparing the dose from a tritium gas release to the dose for an equivalent tritium oxide release. Consequence values for a unit release considering different levels of plume depletion using Gaussian diffusion codes are compared. The impact of various dispersion coefficients on the final consequence values at an offsite receptor location is presented.

Session Schedule

Tuesday, May 8, 1:30 - 5:00 pm, Caldera B

Session 2: Hazards Analysis Chair: Julie Johnston	
1:00	**Don Alsbrooks** Gamma Irradiation Facility (GIF) 60Co Sealed Sources Transfer Operation
1:30	**Mr. William C. Walker** Characterization Of Radioisotope Inventory In The Building 9204-3 Actinide Facility
2:00	**Mark E. Wong** Primary Hazard Screening for ISMS Implementation and Safety Basis Classification
~2:30	**Break**
3:00	**Peter S. Ebey, Jeff Hatchell, Randy Janke, James Jeffries, Mark Kobi, Dan Mangan, Jennie Richardson** A "Rulebook" approach to Hazard Analysis (HA) development with application to a HA program
3:30	**J.C. Laul** Receptor Height: 1.5 m in Dispersion Modeling
4:00	**Terry Foppe, J.C. Laul** Minimum Beryllium TQ (Metal or Oxide) for a Low Chemical Hazard Categorization at MDA-B

Tuesday, May 8, 1:00 pm

Gamma Irradiation Facility (GIF) 60Co Sealed Sources Transfer Operation

Safety Basis Analyst, Don Alsbrooks
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In an effort to reduce the amount of non-certified 60CO sealed sources (pins) stored at the GIF so that the facility can be downgraded to a radiological facility, a sealed source transfer operation at the GIF will be removing approximately 10k curies of non-certified 60CO sealed sources from the GIF.

The GIF sealed source transfer operation involves moving a Department of Transportation (DOT) transfer cask into the GIF High Bay. A cask insert will be removed from the DOT transfer cask and placed in the GIF pool. The 10k curies of non-certified 60CO sealed sources will be placed into the cask insert. The cask insert will be removed from the GIF pool, then placed into the DOT transfer cask. The DOT transfer cask will be sealed and removed from the GIF High Bay.

This operation provided some unique problems to consider when developing the hazard analysis for the safety basis supplement that had to be submitted to the Sandia Site Office for approval. With a much reduced MAR and different transfer cask that previously evaluated, the operation provided unique challenges in the hazard analysis that should provide interesting solutions. The safety basis supplement has yet to be approved by the Sandia Site Office.

Tuesday, May 8, 1:30 pm

Characterization Of Radioisotope Inventory In The Building 9204-3 Actinide Facility

Mr. William C. Walker
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The Actinide Facility, located in Building 9204-3 (a Manhattan Project-era facility), ceased isotope separation activities in the 1970's and glove box operations in the 1990's. Until recently, the defined facility radioactive material inventory was limited to small quantities of unused source materials. In 2008, a comprehensive effort was initiated to determine the overall radioactive material inventory of the Actinide Facility in support of the development of an updated Hazard Categorization for the facility.

The determination of the facility radioactive material inventory involved the execution of a characterization plan that involved the use of (i) intrusive sampling and (ii) non-intrusive analysis (i.e., in-situ gamma spectrometry and radiological survey data) throughout the Actinide Facility. A detailed inventory profile from the intrusive sampling effort was used as a baseline profile of alpha-emitters for insertion into the data obtained from the non intrusive analysis. Additionally, the isotopic data obtained from the in-situ gamma spectrometry analyses was refined through decay-chain analysis to identify the appropriate parent-progeny isotopic relationships that should be accounted for in the inventory.

The inventory characterization effort was concluded in 2011. The comprehensive inventory results allowed for the implementation of a risk management strategy focused on processes and items with the highest radioactive material inventory. This included both inventory reduction and hazard management relative to the inventory present.

Tuesday, May 8, 2:00 pm

Primary Hazard Screening for ISMS Implementation and Safety Basis Classification

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Sandia National Laboratories (SNL) uses a computerized Primary Hazard Screening (PHS) process on all new and ongoing work it undertakes. The screening helps identify the scope of the work and provides a series of questions, developed in conjunction with subject matter experts, to identify hazards and requirements associated with the work. The PHS also combines logic in the software with requirements for human evaluations to determine Safety Basis Hazard Classification for SNL. This paper discusses the PHS process, the use of PHS at SNL, the methodology for determining Safety Basis Hazard Classifications, and the challenges of the process.

Tuesday, May 8, 3:00 pm

A "Rulebook" approach to Hazard Analysis (HA) development with application to a HA program

Peter S. Ebey, Jeff Hatchell, Randy Janke[1], James Jeffries[2], Mark Kobi[3], Dan Mangan2, Jennie Richardson
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[1]LASEA, [2]Hukari, [3]Vector

A novel rulebook method is being used at LANL to ensure consistency and completeness of HA development. A problem that arises in the performance of software based qualitative HAs is a lack of consistency between analysis of hazard scenarios of the same type, across different accident families, and between individual analysts. A set of semi-quantitative rules was developed and applied to the analysis of DSA hazard scenarios to develop a qualitative HA. The HA is being developed using proprietary software (i.e. eCHAP©), but this rulebook approach is not restricted to that software and would improve any HA regardless of chosen documentation method. The semi-quantitative rules assign unmitigated consequence and frequency (and risk) bins consistent with DOE-STD-3009 for each hazard scenario. The consequence rules (for worker, collocated, public) are based on quantity and form of MAR and energy sources, while the frequency rules are based on scenario details and human and equipment reliability data. The rules guide crediting of administrative and engineered controls to estimate consequence and frequency reduction and assign mitigated consequence and frequency (and risk) bins. Additional rules include risk matrices, clarifications, and definitions. This work is performed at LANL and supported by the U.S. DOE: contract DE-AC52-06NA25396. LA-UR-11-000576.

Tuesday, May 8, 3:30 pm

Receptor Height: 1.5 m in Dispersion Modeling

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LA-UR-12-00025

EPIcode (chemicals) and HOTSPOT (radionuclides), approved codes in DOE-HQ Toolbox, recommends a receptor height (ht) of 1.5 m by default, which is typical for "chest ht and breathing zone" near the ground service. However, MACCS2 and other codes use zero meter receptor ht, and yield a highly conservative value at short distances. A ground level receptor ht means that a receptor is lying on the ground, which is not the typical case at all. A conservative value is not necessarily realistic.

A sensitivity analysis at 0 m and 1.5 m receptor ht for beryllium (Be) and chlorine (Cl2) was performed by EPIcode. Emergency Preparedness personnel are interested in concentrations at 30 m (Alert) and 100 m (SAE) for protective actions. DOE sites mostly use 100 m for facility chemical hazard categorization (CHC) in non-nuclear areas. Zero ht yields values about 144 times higher than using 1.5 m receptor ht at 30 m, which is highly unrealistic from the field measurements. Chlorine is highly toxic (PAC-3; 58 mg/m3) and a receptor would face a life threatening value at 30 m with 0 m ht than with 1.5 m ht. Likewise, for Be exposure a worker would face a life threatening value at 30 m with 0 m ht than with 1.5 m ht. At 100 m, zero ht yields values 60% higher than with 1.5 m ht, which is again quite high. At 300 m, the values are about the same within 6%. Beyond 400m, receptor ht is not that important. By using 1.5 m receptor ht, one can increase TQs by 60%, which can help significantly in the CHC.

At LANL in February 2011, there was a release of Be powder/dust (~

16 lb) during excavation in Enclosure #9 for restoration of a 1940s landfill known as Material Disposal Area B (MDA-B). An Industrial Hygienist set up an air monitoring equipment on a table (1.3 m ht) 10 m away from the source within the enclosure. Total airflow passed through was 271 liter in 2.5 hr (1.81 L/min) at atmospheric pressure. The analysis showed 0.44 ug/m3 (4.4E-4 mg/m3) inside Enclosure #9, which is very low concentration. This further strongly supports 1.5 m receptor ht to be used for modeling. Details of the spill analysis, enclosure area, exhaust system, air flow exchange, velocity, and technical basis of 1.5 m ht will be presented at the meeting.

Tuesday, May 8, 4:00 pm

Minimum Beryllium TQ (Metal or Oxide) for a Low Chemical Hazard Categorization at MDA-B

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LA-UR-12-00026

During excavation of a 1940's landfill, Material Disposal Area (MDA) B, that had received hazardous and radiological contamination, on February 22, 2011, operators excavating a trench (15 ft W x 15 ft deep) in Enclosure #9 observed a significant quantity of Beryllium (Be) in 16 glass (mason) jars. A conservative estimate of 20 lb was used as a bounding amount to evaluate the impact of the discovery on MDA B's chemical hazard categorization (CHC). MDA-B was categorized as a Low CHC, based on 170 chemicals, including one lb of Be inventory. At LANL, a facility is considered to be a Low hazard, if the chemical quantity, based on consequences, is below the PAC-3 TQ at 100 m, a Moderate if the quantity exceeds the PAC-3 TQ at 100 m, and a High if the quantity exceeds the PAC-3 TQ for the public at the site boundary (SB). LANL justified adjusting the TQs for Be upward based on data summarized by Mishima et al that presented lower ARF/RF values for encased Be metal than those used earlier.

The SB distance is 20 m which presents a unique situation as compared to 100 m. The TQs of a chemical based on PAC-3 value at 20 m for the public are about 9 times lower than the TQs at 100 m. EPIcode, an approved code in DOE "Toolbox", was used for spill, fire, and explosion scenarios to evaluate consequences assuming 90% Be metal and 10% BeO as conservative. Due to the about two order of magnitude difference between the ERPG-3 values for Be oxide (11.1 mg/m3) and Be metal (0.1 mg/m3) in the TQ for determining a High (public) or Moderate (100 m) CHC, a parametric analysis was performed for each form of material varying from no

oxide to 100% oxide up to no metal to 100% metal. An Explosion (6 lb TNT) was determined to be the bounding scenario limiting MAR, because of its high ARF/RF.

Using a hazard index (HI) approach, results show that 100% Be metal yield a TQ of 32 lb and 100 % BeO yield TQ of 50 lb to maintain a Low CHC. Other % distributions of Be metal and oxide lie in between. The Low CHC was also verified by evaluating an aircraft crash impact plus fire. NNSA/LASO approved new limit of 32 lb for Be metal. This exercise provides important information in establishing an operational procedure to limit the amount that can be exposed in future excavation in order to maintain Low CHC. The findings and results from modeling will be presented at the meeting.

Session Schedule

Tuesday, May 8, 1:30 - 5:00 pm, Barranca A

Session 3: Safety Analysis and Controls Chair: Kevin Carroll	
1:00	**Gary Coleman, Jerry Gnoose** Dynamic Radiological Inventory Control at the Tank W-1A Remediation Project
1:30	**Mrs. Brenda Hawks** Control Selection Techniques Employed for D&D Projects with Emphasis on Nuclear Criticality Safety Controls
2:00	**Boyd D. Christensen** Safety Basis for the Irradiated Materials Characterization Laboratory
~2:30	**Break**
3:00	**James A. McCormick** WIPP Preliminary Documented Safety Analysis For TRUPACT-III
3:30	**John Williams, Alan Ramble** Maintenance of Passive Design Features
4:00	**Rebecca N. Bell** Evaluation of the Impact to the Safety Basis of Research Conducted in Production Facilities at Y-12

Tuesday, May 8, 1:00 pm

Dynamic Radiological Inventory Control at the Tank W-1A Remediation Project

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Tank W- 1 A is a remediation project located within the main campus of the Oak Ridge National Laboratory (ORNL). Tank W- 1 A was installed in the 1950s and removed from service in 1986. While in service, Tank W- 1A collected wastes from several high radiation level analytical facilities at ORNL. During its operation, transfer lines to Tank W-1A leaked, causing soil and groundwater contamination in the vicinity of the tank. Remediation activities currently within the scope of the Tank W-IA Removal Action includes removal of the contaminated soil, Tank W-IA itself, and the concrete saddles and pad that support the tank. Daily activities and container tracking include packaging, non-destructive assay, lid changes, loading and shipping – each with varying inventory restrictions protected by Specific Administrative Controls.

The Tank W-1A Detailed Inventory Spreadsheet (TWDIS), an Excel spreadsheet has been constructed to control and track the dynamic inventory of Tank W-1A boxes of soil and/or drums of TRU waste staged within the Tank W-1A property boundaries. The TWDIS contains warning notes for the user when the container count exceeds 80% of the appropriate DSA limit; therefore the user

has advanced warning to prevent inadvertent exceedance of the respective DSA limit.

The spreadsheet is a cost-effective yet rigorous method for maintaining the protected assumptions listed in the Documented Safety Analysis and flowed into the Technical Safety Requirements.

Tuesday, May 8, 1:30 pm

Control Selection Techniques Employed for D&D Projects with Emphasis on Nuclear Criticality Safety Controls

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When nuclear facilities are in the end phase of their life cycle, and will soon be dismantled, the existing engineered safety features will not always be available or comprehensive in controlling many of the worker hazards encountered during decommissioning. It is anticipated that these systems may at some point physically interfere with further disposition activities and require removal before all hazardous materials can be fully removed. To appropriately select the controls for Deactivation and Decommissioning (D&D) activities requires an understanding of the potential work hazards and the controls that will be present. The correct selection of the suite of controls including Nuclear Criticality Safety will ensure that the work can proceed safely while maintaining an awareness of the final state of the project.

The methodology used to select controls is founded on the principles of DOE STD-3009-94, DOE G 423.1-1A, DOE-STD-1120-2005, and DOE-STD-1186. Engineered systems are selected when available over administrative controls (AC), passive over active, preventative over mitigative, control closest to the hazard, etc. When an engineered system is not available or not as effective in controlling the hazard as an AC, an AC may be selected. In selecting the most appropriate control set for the remaining facility mission, the safety and operations personnel weigh the level of hazard; the effectiveness

of available controls; reliability; remaining life of available engineered system; potential safety benefit of installing or upgrading the existing structure, systems and components (SSCs); the costs associated with upgrade; and maintenance actions. With Nuclear Criticality Safety (NCS) Controls, the qualified Nuclear Criticality Safety Engineer must evaluate the work activities and the available controls to select the best overall package for the various activities.

The Documented Safety Analysis/Technical Safety Requirement contains a balance between SSCs, specific administrative controls, and programmatic controls that are not specific, but important to the safety analysis. Programmatic controls typically provide significant defense in depth (e.g., ignition source/hot work control(s) that address the majority of initiating events). The results of the control selection have allowed the D&D effort to eliminate several costly SSCs and replace them with administrative controls which has greatly reduced the project costs while not impacting safety.

Tuesday, May 8, 2:00 pm

Safety Basis for the Irradiated Materials Characterization Laboratory

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The Battelle Energy Alliance (BEA) at the Idaho National Laboratory (INL) is constructing a new building at the Materials and Fuels Complex (MFC) to support missions in material and fuel development.

The need for capability to analyze and characterize irradiated fuel and structural components is important to the fuel and component development programs at the INL. In the past this type of analysis was performed at the Hot Fuel Examination Facility (HFEF), the Analytical Laboratory (AL), and the Electron Microscope Laboratory (EML), all located at MFC. Given the range of new analytical equipment available, the IMCL will provide a flexible environment in which to utilize a wider variety of equipment and analyses than has been previously available at the MFC.

In accordance with Department of Energy (DOE) Order 413.3A, "Program and Project Management for the Acquisition of Capital Assets", safety must be integrated into the design process of new nuclear facilities or modifications to existing DOE Hazard Category 1, 2, and 3 nuclear facilities. DOE-STD-1189-2008, Integration of Safety into the Design Process" provides expectations for identifying hazards and accidents early in the design and incorporating

appropriate controls to mitigate those hazards throughout the design and construction process.

This paper discusses the process and methods used to successfully integrate nuclear safety into the design of the IMCL such that design and administrative controls are utilized to provide a framework for safe facility operation. Nuclear safety work to date includes a Safety Design Strategy (SDS) report, Preliminary Hazard Analysis (PHA), Conceptual Safety Design Report (CSDR), and a draft Preliminary Documented Safety Analysis (PDSA). Supporting documentation includes a Fire Hazards Analysis (FHA) and a number of dose consequence analyses supporting the hazard and accident analysis.

As a result of early involvement from the nuclear safety analysis team, the project is moving forward with construction with safety requirements clearly identified. This will help reduce the risk of unforeseen expenses associated with project delays.

Tuesday, May 8, 3:00 pm

WIPP Preliminary Documented Safety Analysis For TRUPACT-III

WIPP Nuclear Safety Manager, James A. McCormick
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Several U.S. Department of Energy (DOE) sites have contact-handled (CH) transuranic (TRU) waste inventories that are packaged in large boxes. These boxes are too large to be accommodated by the shipping packages currently licensed for CH-TRU waste shipment (i.e., TRUPACT-II and HalfPACT). The TRUPACT-III shipping package has been designed and developed primarily for the transportation of these large boxes.

When evaluated to DOE Standard 1189-2008, Integration of Safety into the Design Process, the modification of the Waste Isolation Pilot Plant facility to accommodate the receipt and emplacement of TRUPACT-III packages was determined to be a major modification. As a result, a Preliminary Documented Safety Analysis (PDSA) was developed.

Prior to developing the PDSA, a Safety Design Strategy (SDS) that authorized limited procurement activities was developed by URS and approved by DOE Carlsbad Field Office (CBFO). This authorization for limited approval allowed the procurement of equipment and components to proceed prior to approval of the PDSA.

The approved PDSA only authorized the facility modifications. The use of the TRUPACT-III at the WIPP was not authorized until the PDSA requirements were integrated into the approved WIPP DSA.

Challenges throughout the project included:

• Timely submittal of the PDSA – Although driven by an accelerated schedule, the PDSA could not be completed until enough of the

design was finalized, and

• Maintaining configuration control – At the time the PDSA was being developed, one DSA revision (Rev. 2) was awaiting CBFO approval and another (Rev. 3) was being routed for comments. The PDSA could not be approved until Rev. 2 was approved.

Tuesday, May 8, 3:30 pm

Maintenance of Passive Design Features

John Williams, Xron
Alan Ramble, CHPRC

Evaluation performed at the Hanford Waste Encapsulation and Storage Facility raised question whether a credited passive design feature could be relied upon to perform its intended safety function in the absence of in-service inspections or tests. Concern was raised that environmental degradation effects could render the passive design feature unusable. DOE O 433.1B, Maintenance Management Program for DOE Nuclear Facilities, specifies that the Nuclear Maintenance Management Program must include, "The process for conducting inspections to evaluate aging-related degradation and technical obsolescence to determine whether the performance of SSCs is threatened." CHPRC at Hanford is implementing a process for defining maintenance requirements that ensure the credited safety function will be available over the life of the facility considering the potential effects of operational and environmental degradation. Also this approach enhances information presented in the documented safety analysis to ensure these maintenance and surveillance activities are appropriately identified within the safety basis.

Tuesday, May 8, 4:00 pm

Evaluation of the Impact to the Safety Basis of Research Conducted in Production Facilities at Y-12

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Objective: This paper describes the unique challenges to maintaining the integrity of the safety basis during the conduct of nuclear research in existing production facilities at the Y-12 National Security Complex. It reviews lessons learned in the performance of advanced radiation measurement evaluations performed as part of the Nuclear Materials Identification System (NMIS) and Advance Portable Neutron Imaging System (APNIS) in operating facilities at Y-12.

Relationship to Overall Interests of DOE Safety Analysis: Section 203(d)(3) of 10 CFR 830 Subpart B requires the contractor responsible for a hazard category 1, 2, or 3 DOE nuclear facility to implement the DOE approved USQ procedure in situations where there is a: (1) Temporary or permanent change in the facility as described in the existing documented safety analysis; (2) Temporary or permanent change in the procedures as described in the existing documented safety analysis; (3) Test or experiment not described in the existing documented safety analysis. This paper shares how Y-12 production facilities ensure the integrity of the safety basis by evaluating the conduct of nuclear experiments prior to them being performed in operating facilities.

Results: The unique challenges of ensuring the integrity of the safety basis, while utilizing neutron generators to evaluate nuclear materials and components, resulted in the development of good

business practices. The information collected has use in applications such as nuclear safeguards, arms control, nonproliferation and counter terrorism.

Benefits of Work to Mission of Sponsoring Organization: The dissemination of these good business practices and lessons learned allow other sites to benefit from the methods used at Y-12.

Workshop Schedule

Wednesday, May 9, 2012

Time	Event	Location
7:00 – 5:00	Registration/information	Promenade
	Sponsor exhibition booths	
	* private meeting(s) (ask to reserve)	Barranca B
8:00	Accident Analysis II (paper session)	Caldera A
	Hazards Categorization (paper session)	Caldera B
	Regulatory Topics (paper session)	Barranca A
~9:30	Break	
10:00	Paper sessions (continued)	
11:30	Lunch (on your own)	
1:00	Accident Analysis III (paper session)	Caldera A
	Software and Quality Topics (paper session)	Caldera B
	Criticality Safety and USQ Topics (paper session)	Barranca A
~2:30	Break	
3:00	Changes to DOE Order, Standards, and Guides (panel)	Tewa 3
5:30	End of day	
6:00	Steering Committee dinner	Vista A

Session Schedule

Wednesday, May 9, 8:00 - Noon, Caldera A

Session 4: Accident Analysis II Chair: Mukesh Gupta	
8:00	**Nathan Cathey, Brad Evans, Pete Lowry, Steve Maheras** Deposition Velocity impact on the PNNL Radiochemical Processing Laboratory
8:30	**Nicholas J. Schira, Dennis R. Armstrong, David C. Thoman, and Elizabeth A. R. Henley** Deposition Velocity Estimation with the GENII2 Computer Code
9:00	**John Hargreaves** Modeling of Explosive Buoyant Plumes of Natural Gas
~9:30	**Break**
10:00	**Kevin R. O'Kula, M.G. Wentink** Early Lessons Learned from Risk Applications on DOE Nonreactor Nuclear Facilities
10:30	**Roger Lanning** Understanding DOE-HDBK-3010 Without Becoming an Accident Analyst
11:00	**John Wang** Alternatives of MACCS2 in LANL Dispersion Analysis for Onsite and Offsite Doses

Wednesday, May 9, 8:00 am

Deposition Velocity impact on the PNNL Radiochemical Processing Laboratory

Nathan Cathey, Nuclear Safety and Facility Authorization
Pacific Northwest National Laboratory
Brad Evans, Pete Lowry, Steve Maheras
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This presentation will discuss the application of DOE's Office of Health, Safety, and Security Safety Bulletin 2011-02, Accident Analysis Parameter Update to the Radiochemical Processing Laboratory (RPL) operated by the Pacific Northwest National Laboratory. One unique aspect is that the estimated doses to the Maximally Exposed Offsite Individual (MOI) and the Collocated Worker (CW) currently do not challenge the offsite evaluation guideline or onsite guidelines such that safety class or safety significant controls are required for the MOI and CW. The current accident analysis assumes a bounding inventory and the MOI is located approximately 570 m northeast of the RPL.

In determining the approach to addressing the issues outlined in Safety Bulletin 2011-2, one concern was that applying an overly conservative deposition velocity might drive new controls. Therefore the three different options described in Safety Bulletin 2011-2 were evaluated to determine which approach to apply. Factors such as resources required to execute a specific option and potential impact to the existing accident analysis and resulting control set were considered. A first estimation of potential impact was made by using the default deposition velocity of 0.1 cm/s. Based on that, a final approach was chosen. This presentation will describe the overall results of the analysis and also potential impacts on the Emergency Preparedness Program.

Wednesday, May 9, 8:30 am

Deposition Velocity Estimation with the GENII2 Computer Code

Nicholas J. Schira; Dennis R. Armstrong;
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In 2010, the Department of Energy (DOE) Chief of Nuclear Safety and Office of Health, Safety and Security (HSS), with the support of industry experts in atmospheric sciences and accident dose consequences analysis, performed detailed analyses of the basis for the deposition velocity (DV) values used in the MACCS2 computer code. As a result of these analyses, DOE concluded that the MACCS2 default DV values of 1 centimeter/second (cm/s) for unfiltered/unmitigated releases and 0.1 cm/s for filtered/mitigated releases may not be reasonably conservative for all DOE sites and accident scenarios.

HSS recently issued Safety Bulletin 2011-02, Accident Analysis Parameter Update, recommending the use of the newly developed default DV, 0.1 cm/s for an unmitigated/unfiltered release. Alternatively site specific DV values can be developed using GENII version 2 (GENII v2) computer code.

Key input parameters for calculating DV values include surface roughness, minimum wind speed, particle size, and particle density. This paper will include reasonably conservative inputs, and a truncated parametric study.

In lieu of the highly-conservative recommended DV value (0.1cm/s) for unmitigated/unfiltered release, GENII v2 has been used to justify estimated 95th percentile DV values. Also presented here are atmospheric dilution factors (χ/Q values) calculated with the

MACCS2 code using the DV values form GENII v2, χ/Q values calculated directly with GENII v2, and a discussion of the benefits of each method.

This paper will give an overview of the process of calculating DV with GENII v2 including a discussion of the sensitivity of input parameters.

Wednesday, May 9, 9:00 am

Modeling of Explosive Buoyant Plumes of Natural Gas

John Hargreaves
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Objective of Work Being Reported: This paper discusses work done at LANL combining the buoyant plume theories of Hanna and Briggs with (1) Turner's method for calculating atmospheric dispersion and (2) meteorological measurements of Pasquill Atmospheric Stability.

Relationship of the Work to the Overall Interests of DOE Safety Analysis: This paper provides a theoretical basis for predicting the possibility of momentum-dominated or buoyancy-dominated plumes of natural gas reaching a DOE hazardous site while retaining an explosive concentration of natural gas in air. This method has been used at LANL to screen existing and proposed locations of natural gas lines near to LANL nuclear facilities.

Results of Work: This paper discusses a synthesis of theory and measurements used at LANL to determine the hazards of subterranean natural gas lines sited proximate to nuclear material facilities (CMR, PF-4, CMRR, and TWF). Hanna and Briggs Gaussian Bent-Plume theory is used show if natural gas plume can travel to a LANL nuclear building. Standard atmospheric dispersion modeling is then employed using Pasquill Stability Classes, Turner air concentration data, and Slade power law approximations to calculate the volume percent concentration of natural gas reaching the LANL nuclear facility. This result is then compared to multi-year compilations of meterological data to quantify the possibility and/or probability of a natural gas plume reaching a LANL nuclear facility while retaining an explosive concentration of gas in air.

Wednesday, May 9, 10:00 am

Early Lessons Learned from Risk Applications on DOE Nonreactor Nuclear Facilities

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Over the past two years, the Department of Energy has taken several actions to provide an infrastructure for providing appropriate controls and support for use of risk assessments and risk informed decision making as it applies to nuclear safety. These actions include establishing a Risk Assessment Technical Experts Working Group, revising its Nuclear Safety Policy to explicitly address the use and control of risk assessments, and developing a Draft Standard, "Development and Use of Probabilistic Risk Assessments in Department of Energy Nuclear Safety Applications," (December 2010). With respect to the draft standard, its purpose is "to provide guidance and criteria for a standard approach to utilization of probabilistic risk assessments (PRAs) in nuclear safety applications". In particular, it states, "DOE's nuclear safety decision-making processes can be supplemented and strengthened through application of quantitative and probabilistic risk assessment methodologies; such methodologies may be useful in aiding the evaluation of alternatives that comply with DOE nuclear safety requirements, supporting the USQ process, augmenting traditional safety assessment methods, evaluating changes to DOE safety requirements, and in general, enhancing the quality, transparency, and credibility of analytical results and decisions that are made". Given the publication of the draft standard, significant perspectives can be gained by reviewing past and current efforts where full-

scope risk analyses were or are being applied to nonreactor nuclear facilities. In this regard, the objectives of this paper are to review past or ongoing risk analyses performed for DOE nonreactor nuclear facilities, and secondly, to identify the lessons learned from these studies as guidance to the current initiatives on risk-informing safety guidance and design, as well as identifying strengths and limitations of risk assessment to the prospective users reviewers, and regulators.

The nuclear facilities selected for this review are the Defense Waste Processing Facility (DWPF) at the Savannah River, and the Waste Treatment and Immobilization Plant (WTP) at Hanford. The two studies are: (1) the full-scope, Probabilistic Safety Assessment (PSA) for DWPF performed in the 1990s (prior to startup); and the Quantitative Risk Analysis (QRA) of hydrogen events in WTP. While very similar to nuclear power plant PSA studies in that it quantified radiological impacts and risks, the DWPF PSA study was never implemented in the safety basis. The QRA is currently in progress and is being used to guide final design to inform the final design of WTP piping systems, but is not a radiological assessment.

The major lessons identified from these studies are:

- Design and operational vulnerabilities are more easily identified and prioritized on a relative basis
- Normal operation tends to dominate risk over abnormal and accident event risk
- The relative risk values are often more useful than the absolute risk values. If the latter are used to inform decision-making then it is critical to be cognizant of the uncertainty or retained margin.
- Prompt effects are not as important for nonreactor facilities, as are latent effects, contamination, and other long-term risks.
- Applicable reliability data for nonreactor facilities should be carefully considered, especially in the design phase. Caution should be exercised to identify applicable data to support nonreactor facility operational frequency analysis.

These and several additional observations will be discussed in the full paper. Use of these perspectives from nonreactor nuclear risk assessments can assist in promoting consistent use of risk assessment tools in throughout the DOE Complex, while recognizing the key lessons from past full-scope applications.

This discussion supports the theme of the Workshop, Preparing for the Millennium, by emphasizing how PRA and QRA can be used to inform the safety and design of today's DOE nuclear facilities, and potentially identify vulnerabilities not as readily found through deterministic approaches alone.

Wednesday, May 9, 10:30 am

Understanding DOE-HDBK-3010 Without Becoming an Accident Analyst

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This presentation provides an overview of DOE-HDBK-3010, commonly known as the "DOE Accident Analysis Handbook". The goal is to present the contents of DOE-HDBK-3010 in a conceptual way that may be more useful for hazards and safety analysts.

When working to DOE-STD-3009, accident analysis calculations often rely heavily on the values in DOE-HDBK-3010 given for airborne release fractions (ARF) and respirable fractions (RF) in accident scenarios. While accident analysts are responsible for the calculations that employ these ARF and RF values, it is important for hazards and safety analysts to understand what these values mean and how applicable they are to a given scenario. Recent DNFSB Recommendations at WTP have highlighted potential gaps in the DOE handbook that need to be communicated to the EFCOG community.

This presentation will briefly cover the accident analysis methodology, the contents and structure of DOE-HDBK-3010, the benefits and cautions when using the handbook, specific examples applying the handbook, a visual demonstration of ARF and RF values, and how WTP has responded to DNFSB questions on ARF and RF values.

Wednesday, May 9, 11:00 am

Alternatives of MACCS2 in LANL Dispersion Analysis for Onsite and Offsite Doses

John Wang, Ph.D.
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The purpose of hazard analyses and accident analyses in Los Alamos National Laboratory (LANL) is to establish the safety basis for LANL facilities and to comply with different regulations and integrated safety management requirements. In modeling atmospheric dispersion of accident release, one of the common statistical analysis methods at LANL is MACCS2, a DOE Safety Analysis Toolbox code. However, there are some limitations and shortfalls of MACCS2 for both onsite and offsite dispersion analyses. One of the often concerns is the over-conservatism in MACCS2 methodology and parameters. Alternative computer codes are investigated to provide more realistic calculations to support the safety basis analyses at LANL.

In the early development of Yucca Mountain Project (YMP), similar concerns were raised for the suitability of MACCS2 for onsite worker doses and ARCON96 was chosen to replace MACCS2. The control room habitability design and improved building wake algorithm of ARCON96 are particularly useful for the wide range of facility configurations in YMP. The flexible meteorological input of ARCON96 is also easier for analyses of long-term trends (95th percentile χ/Q). The application of ARCON96 in YMP licensing application clearly demonstrates the merit for onsite worker safety analyses in various complex configurations and accident scenarios.

For offsite general public, AERMOD could be a good candidate. AERMOD is a next generation air dispersion model based on planetary boundary theory, and is adopted by EPA as a preferred

model since 2005. AERMOD fully incorporates the PRIME building downwash algorithms, advanced depositional parameters, local terrain effects, and advanced meteorological turbulence calculations. Overall, the advanced capacity of AERMOD will provide more confidence in accuracy of offsite public doses.

Session Schedule

Wednesday, May 9, 8:00 - Noon, Caldera B

Session 5: Hazard Categorization Chair: Bruce Wilson	
8:00	**Chelise A. Van De Graaff, Dr. Chad Pope, Dr. J. Todd Taylor** Hazard Categorization Reduction via Nature of the Process Argument
8:30	**Raymond F. Sartor and Stephen A. Costigan** LANL Nuclear Facility Categorization
9:00	**Boyd D. Christensen** Hazard Classification of the Remote Handled Low-Level Waste Disposal Facility
~9:30	**Break**
10:00	**Nathan G. Cathey, Brad Evans, Andy Prichard, Art Stithem** Addressing Daughter Products as Part of a Facility's Inventory
10:30	**Kelsey L. F. Curran** ERPG, AEGL, and TEEL Comparison
11:00	**Mr. Jon C. Guy** Segmentation and Downgrade of the SRS C Reactor Spent Fuel Basin

Wednesday, May 9, 8:00 am

Hazard Categorization Reduction via Nature of the Process Argument

Chelise A. Van De Graaff
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This paper documents the Hazard Categorization (HC) and Criticality Safety Evaluation (CSE) for activities performed using an Inspection Object (IO) in excess of the single parameter subcritical limit of 700 g of U-235. By virtue of exceeding the single parameter subcritical limit and the subsequent potential for criticality, the IO HC is initially categorized as HC2. However, a novel application of the nature of the process argument was employed to reduce the HC from HC2 to less than HC3 (LTHC3).

The IO is composed of five separate uranium metal plates (nominally 100%) that total no greater than 3.82 kg of 20% U-235 enriched by mass. The IO is planned to be arranged in various configurations. As the IO serves as a standard for experimentation aimed at establishing techniques for detection of fissionable materials, it may be placed in close proximity to various reflectors, moderators, or both.

The most reactive configurations of the IO were systematically evaluated, and shown that despite the mass of U-235 and potential positioning near various reflectors and moderators, the IO cannot be assembled into a critical configuration. Therefore, the potential for criticality does not exist.

With Department of Energy approval, a Hazards Assessment Document with high-level (facility-level) controls on the plates negates the potential for criticality and satisfies the nature of the process argument to reduce the HC from HC2 to LTHC3.

Wednesday, May 9, 8:30 am

LANL Nuclear Facility Categorization

Raymond F. Sartor (SB-TS) and Stephen A. Costigan (RP-1)
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The U.S. Department of Energy (DOE) has defined four categories of nonreactor nuclear facility:

- Hazard Category 1 (HC-1),
- Hazard Category 2 (HC-2),
- Hazard Category 3 (HC-3), and
- Less-than-HC-3.

Although specific radionuclide threshold quantities have been established by DOE for the classification of HC-2 and HC-3 facilities, there are no quantitative thresholds to distinguish the less-than-HC-3 nuclear facilities from non-nuclear facilities. In fact, during the rulemaking process for 10 CFR 830, several comments requesting a threshold for nonreactor nuclear facilities were submitted. DOE disagreed with these requests, replying that the "nonreactor nuclear facility" definition, as written, is intended to cover all situations (other than nuclear reactors) with the potential to cause radiological harm.

Thus, it is the responsibility of the operators/managers of DOE facilities to identify which facilities are nuclear by the qualitative parameters provided. However, qualitative parameters require interpretation to be used. LANL has a proposal for defining the radiological limits below which "harm" to the worker cannot occur, thus establishing a quantitative method to calculate hazard threshold quantities and standardizing identification of nuclear facilities at the Los Alamos National Laboratory (LANL). This article will present the technical development and current status of the hazard threshold quantities.

Wednesday, May 9, 9:00 am

Hazard Classification of the Remote Handled Low-Level Waste Disposal Facility

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The Battelle Energy Alliance (BEA) at the Idaho National Laboratory (INL) is constructing a new facility to replace remote-handled low-level radioactive waste disposal capability for INL and Naval Reactors Facility operations.

Current disposal capability at the Radioactive Waste Management Complex (RWMC) will continue until the facility is full or closed for remediation (estimated at approximately fiscal year 2015). Development of a new onsite disposal facility is the highest ranked alternative and will provide RH-LLW disposal capability and will ensure continuity of operations that generate RH-LLW for the foreseeable future.

As a part of establishing a safety basis for facility operations, the facility will be categorized according to DOE-STD-1027-92. This classification is important in determining the scope of analyses performed in the safety basis and will also dictate operational requirements of the completed facility.

This paper discusses the issues affecting hazard classification in this nuclear facility and impacts of the final hazard categorization.

Wednesday, May 9, 10:00 am

Addressing Daughter Products as Part of a Facility's Inventory

Nathan G. Cathey
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Pacific Northwest National Laboratory (PNNL) uses a wide variety of radioactive materials in performing various research activities. Most radionuclides have decay chains where the parent radionuclide dose conversion factor bounds that of the daughter products, however, this is not the case in many instances. It is important to appropriately account for materials whose daughter products are more limiting with respect to HC-3 TQs at the time initial hazard categorization is performed.

The result of the latter case are Hazard Category 3 threshold quantities (HC-3 TQs) that are more restrictive than the values provided by DOE-STD-1027, and recognition that as the parent decays, higher dose equivalents are generated by the same initial inventory. The approach taken at PNNL is to develop a set of correction factors for these radionuclides and integrated them into the hazard categorization and radioactive material tracking processes.

This presentation will discuss an approach to address selected daughter products as part of a facility's radioactive material inventory. The radionuclides of concern are identified, the method of evaluating the decay chains, and factors comparing the daughter products to their parents are presented.

Wednesday, May 9, 10:30 am

ERPG, AEGL, and TEEL Comparison

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A comparison is performed on the exposure thresholds of Protective Action Criteria (PAC) for the Emergency Response Planning Guidelines (ERPGs), Acute Exposure Guideline Levels (AEGLs), and Temporary Emergency Exposure Limits (TEELs) of several hazardous chemicals. Background historical information and methods used to derive PAC thresholds for ERPG, AEGL, and TEEL values are researched, equated, and contrasted. Strengths and weaknesses are presented for using ERPG, AEGL, and TEEL threshold quantities for modeling, as input to industrial facility hazard classification purposes. Background and recommendations for use of ERPG, AEGL, or TEEL values for hazard classification are offered using modeling data.

Wednesday, May 9, 11:00 am

Segmentation and Downgrade of the SRS C Reactor Spent Fuel Basin

Mr. Jon C. Guy
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The 105·C Reactor Building is identified in Appendix K, "Facilities to be Decommissioned", of the Federal Facility Agreement (FFA 1993). However, parts of the building continue to store legacy moderator. This moderator cannot be dispositioned in a cost effective manner at this time. The 105-C Reactor facility is considered a Hazard Category 2 Nuclear facility. The 105-C reactor disassembly spent fuel basin is a subunit of the 105-C Reactor Building. The basin is no longer used to store nuclear fuel but does retain ~ 3 million gallons water. A proposal to dewater and grout the basin was put forth to eliminate the potential for ground water release. While this task could be accomplished within the current safety basis a proposal to segment and down grade the disassembly spent fuel basin was identified as a cost effective alternative. Segmentation of the disassembly basin and categorization as a Radiological Facility eased work control and design requirements in accordance with the reduced risk. This approach was well accepted by the DOE as a solution to reducing risk and maintaining the required safety basis posture for an existing Hazard Category 2 facility. A segmentation and downgrade of the C-Area disassembly spent fuel basin was approved and implemented in November of 2011. Work on the dewatering and grouting of the basin continues with the completion of all activities expected in September of 2012.

Session Schedule

Wednesday, May 9, 8:00 - Noon, Barranca A

Session 6: Regulatory Topics Chair: John Farquharson	
8:00	**Michael R. Greutman, Keith A. Voss** Exemption Request Use for One Time Processing of Radioactive Materials
8:30	**Carmen DeLong** Streamlining the Nuclear Power Reactor Licensing Process
9:00	**John A. Farquharson** Safety Basis Approaches – ISA vs. DSA – One Safety Analyst's Opinion
~9:30	**Break**
10:00	**Timothy S. Stirrup** Implementation of DOE Order 420.2C Safety of Accelerator Facilities
10:30	**Michael Hillman, Garrett Smith, and James O'Brien** Preliminary Results of the Department of Energy's Beyond Design Basis Events Evaluation Pilot Program
11:00	**Samuel Rosenbloom, Garrett Smith, and James O'Brien** Revision of DOE Standard 3009, Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analysis

Wednesday, May 9, 8:00 am

Exemption Request Use for One Time Processing of Radioactive Materials

Michael R. Greutman
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10 CFR 820 Subpart E, "Exemption Relief," provides the bases for requesting exemption relief from provisions of DOE Nuclear Safety Requirements at nuclear facilities, providing that the exemption is; (1) authorized by law, (2) would not present an undue risk to the public health and safety, the environment, or facility workers, and (3) is consistent with the safe operation of a nuclear facility.

SNL received transuranic waste drums from the Lovelace Respiratory Research Institute (LRRI). Five of the drums that exceeded HC-3 Threshold Quantities were to be processed through an existing HC-3 Nuclear Facility. None of the LRRI drums were expected to necessitate opening. However, upon further characterization, it was determined that these drums would need to be repackaged and/or have HEPA filters replaced.

The drums contained highly respirable, high activity alpha emitting materials. Engineered radiological control features were needed to safely process these drums. The existing nuclear facility did not have the appropriate hazard controls, whereas an existing radiological facility did have these controls (gloveboxes with HEPA-filtered exhaust with secondary and building confinement). However, it was not approved to process materials exceeding HC-3 quantities. Upgrading the existing HC-3 nuclear facility for the purpose of processing five HC-3 drums would have been costly, and the drums would not have been able to be shipped to WIPP in time to meet regulatory commitments. The approval of an exemption provided relief from the provisions of 10 CFR 830, Subpart B, and allowed for the safe processing of the drums, while meeting other regulatory commitments.

Wednesday, May 9, 8:30 am

Streamlining the Nuclear Power Reactor Licensing Process

Carmen DeLong
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The nation is facing a monumental challenge in meeting the need for dramatic increases in electrical power while reducing the total impact of power production on the environment. The private sector is responding to that challenge, as is the Nuclear Regulatory Commission (NRC), with certified designs. The potential for reducing the time to first kilowatt is great if both industry and the Government are able to establish new ways of doing business.

Meeting the nation's power needs is estimated to require at least 100 plants of 1,000 megawatts each by the year 2030. Neither industry nor the government can hope to succeed at this critical challenge without standardization of design and implementation of a streamlined licensing process.

There is a limited and shrinking pool of technical experts to address private sector design, operation, and licensing needs. The process must be streamlined in order to achieve success over the next 10 years. Currently the NRC requires at best 30 months to complete a license review, however, it can be reduced to 20 months with minor process enhancements and by tracking the journey of each application. Without such marked changes in process, the NRC cannot expect to keep pace with the private sector efforts.

Implementation of an efficient tracking methodology within the NRC is crucial to supply data to form the basis for process improvement. These data, if collected and managed properly, can dramatically reduce the licensing effort on both the NRC and contractors.

Wednesday, May 9, 9:00 am

Safety Basis Approaches – ISA vs. DSA – One Safety Analyst's Opinion

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The Department of Energy (DOE) nonreactor nuclear facilities and the private industry nuclear fuel cycle facilities (regulated by the Nuclear Regulatory Commission [NRC]) have similar hazards and potential accidents. However, their safety bases are regulated by different (1) federal agencies, (2) regulations, and (3) (although somewhat similar) hazard and accident analysis techniques. Generally, the "bottom line" of safety bases documentation is to (1) assess the risk vs. an established standard and (2) identify a set of credited controls to ensure the risk is appropriately managed. This paper provides an objective overview of the two approaches (Integrated Safety Analysis [ISA] vs. Documented Safety Analysis [DSA]). The DOE approach (DSA) is more of a consequence-based approach while the NRC approach (ISA) is more of a layer of protection analysis (LOPA) approach.

A fictitious nonreactor nuclear facility is used to illustrate the similarities and differences between the approaches and the techniques are linked back to other techniques used in nonnuclear industries. General guidance referenced in both regulations is from the Center for Chemical Process Safety's (CCPS's) "Red Book" (Guidelines for Hazard Evaluation Procedures).

Wednesday, May 9, 10:00 am

Implementation of DOE Order 420.2C Safety of Accelerator Facilities

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An update to the Department of Energy (DOE) Executive Order (Order) on the, "Safety of Accelerator Facilities" (420.2C) was issued on July 21, 2011. The most significant change in the update is the inclusion of exempt accelerators into the Order. Historically, smaller local, non-complex accelerators were exempt from the Order; however, the recent update now includes these traditionally exempt accelerators. The intent of this change is to compel contractors pull together a comprehensive list of accelerator equipment and document the determination of applicability to the Order. This paper discusses how Sandia National Laboratories (SNL) is implementing changes to the updated Order and addressing the inclusion of these traditionally exempt accelerators.

Wednesday, May 9, 10:30 am

Preliminary Results of the Department of Energy's Beyond Design Basis Events Evaluation Pilot Program

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This paper presents the initial results from the ongoing Beyond Design Basis Event Evaluation Pilot Program. The pilot program was initiated in January 2012 to evaluate the need for improvements in guidance and criteria for beyond design basis events (BDBEs) in the following areas:

- Evaluation of BDBEs in the facility safety analysis.
- Walk down of safety systems for potential vulnerability to natural phenomena hazards events focused on evaluating margins to BDBEs.
- Accident management strategies (and the resources necessary for implementing such strategies) for BDBEs, including events that may affect multiple facilities and site infrastructure.

Wednesday, May 9, 11:00 am

Revision of DOE Standard 3009, Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analysis

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This paper summarizes the current revision of DOE Standard 3009, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analysis. The paper will discuss the revision process and key changes made in the current draft revision in the following areas:

- Hazard Assessment Process
- Accident Analysis
- Worker Safety Controls
- Public Safety Controls
- Criticality Controls
- Safety Management Programs
- Implementation Methodology for Compliance with 10 CFR 830, Nuclear Safety Management

Session Schedule

Wednesday, May 9, 1:00 - 2:30 pm, Caldera A

Session 7: Accident Analysis III Chair: Andrew Vincent	
1:00	**Douglas M. Gerstner, Cliff B. Davis** Thermal-Hydraulic Analysis Results of a Seismically-Induced Loss of Coolant Accident Involving Experiment Out-of-Pile Loop Piping at the Idaho National Laboratory Advanced Test Reactor
1:20	**M. R. Yeung, J. E. McAllister Jr., J. A. Fishel, D. K. Allison, E. A. Henley, T. C. Campbell, H. Hutchins, S. P. Graham, S. Chow, K. A. Harris, A.G. Mohiuddin, G. E. Dorfler, S. M. Lonchar** Saltstone Facility Analyses
1:40	**M. R. Yeung, D. C. Thoman, K. H. Barbour** A CST Waste Tank Accident Analysis for a Post-Seismic Explosion Event
2:10	**Jeff Woody, Terry Foppe** Clarification of Safety Basis Topics from DOE-STD-5506-2007
~2:30	Break

Wednesday, May 9, 1:00 pm

Thermal-Hydraulic Analysis Results of a Seismically-Induced Loss of Coolant Accident Involving Experiment Out-of-Pile Loop Piping at the Idaho National Laboratory Advanced Test Reactor

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The Advanced Test Reactor (ATR) is a Department of Energy (DOE)-owned test reactor whose principal function is to provide a high neutron flux for testing reactor fuels and other materials. The reactor also provides other irradiation services such as radioisotope production. The reactor has a design thermal power of 250 MW with a nominal operating pressure of 360 psig at the top of the core and a nominal maximum reactor outlet temperature of 170°F. The ATR and its support facilities are located at the ATR Complex of the Idaho National Laboratory (INL).

The ATR Updated Final Safety Analysis Report (UFSAR) evaluates the safety basis of the facility to meet the requirements of DOE 0 5480.23, "Nuclear Safety Analysis Reports," and 10 CFR 830, "Nuclear Safety Management." The format and content of the UFSAR follows the NRC Regulatory Guide, RG 1.70, "Format for Safety Analysis Reports," and as such, the design basis events considered in the UFSAR are similar to those considered for the commercial reactor industry. Those events considered include external events, and specifically for this analysis, seismic events.

The ATR UFSAR has fully evaluated the impact of seismic events on the primary coolant system (PCS) and has shown conclusively that the consequences of seismically-induced loss of coolant accidents (LOCAs) meet the ATR plant protection criteria (PPC). However, an ATR Unreviewed Safety Question (USQ) determination has identified that a seismic event involving the ATR experiment out-of-pile loop piping has not been fully addressed in the ATR accident analysis.

An extensive analysis was conducted to address the ATR USQ by analyzing the seismic break and leakage in all four existing experiment standard in-pile Tubes (SIPTs), one large IPT (LIPT), and one additional SIPT scheduled for future installation. The seismically-induced experiment loop LOCA is postulated to occur simultaneously with the currently analyzed seismically-induced LOCA involving the ATR PCS. The combined effect of a seismically-induced LOCA involving both the PCS and out-of-pile loop piping was evaluated using the RELAP5, ATR SINDA, and SINDA-SAMPLE codes to ensure that the ATR PPC are satisfied. This paper summarizes the challenges associated with the analysis and the final conclusions for continued safe operation of ATR.

Wednesday, May 9, 1:20 pm

Saltstone Facility Analyses

M. R. Yeung, J. E. McAllister Jr., J. A. Fishel, D. K. Allison, E. A. Henley, T. C. Campbell, H. Hutchins, S. P. Graham, S. Chow, K. A. Harris, A.G. Mohiuddin, G. E. Dorfler, S. M. Lonchar
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Modifications to the Savannah River Site (SRS) Saltstone Production Facility (SPF) are being implemented and include the installation of Salt Solution Receipt Tanks (SSRT), construction of new Salt Disposition Units (SDUs), and changing the function of the Waste Concentrate Tank (WCT). A broad overview of the facility is presented and the locations of the various analyses for the SSRTs, SDU, and WCT are shown. The analyses in the presentation are chosen to show the diversity of the accident scenarios that should be considered to demonstrate safety in a large complex facility. Transient thermal analyses were completed to demonstrate that the SSRT heat up rate was acceptably low after design changes were implemented. A second class of safety analyses is the ventilation/ flammability studies that were completed for the SSRTs and WCT. Significant efforts were completed in a third class of analyses to determine the CLFL characteristics of the headspace in the SDU 2. A minimum-time grout pour schedule with ventilation of the headspace to fill the SDU was developed to maintain the headspace above the grout below a CLFL value of 60%. This parametric study varied the ventilation schedule, the temperature inside the SDU, the concentration of Isopar® L, and a minimum daily grout pour size. Additionally, a dose consequence calculation for an explosion event in the SDU 2 headspace was completed. Last, the technique for determining the source term of the salt-solution volume-equivalent of aerosols in the vapor space of SDUs due to splashing from operational pours is presented.

Wednesday, May 9, 1:40 pm

A CST Waste Tank Accident Analysis for a Post-Seismic Explosion Event

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An analysis has been conducted to define a bounding tank to represent the 51 waste storage tanks at the Concentrate Storage and Transfer (CST) Facility of the Savannah River Site (SRS) for the seismic follow-on explosion event. This bounding tank is used to perform the accident analysis to investigate the consequence of the explosion for two scenarios. Analytical models have been developed to determine the hydrogen concentration in the tank vapor space as a result of radiolysis generation and sudden release of the hydrogen trapped in the sludge and saltcake. This analysis also investigates the consequences caused by hydrogen deflagration and detonation. The results of the analysis indicate that the offsite dose of such an event is limited to approximately 2.1 rem.

Wednesday, May 9, 2:10 pm

Clarification of Safety Basis Topics from DOE-STD-5506-2007

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DOE-STD-5506-2007, Preparation of Safety Basis Documents for Transuranic (TRU) Waste Facilities, was issued in April 2007. This Standard provides analytical assumptions and methods, as well as hazard controls to be used when developing safety basis documents for TRU waste facilities, supplementing the applicable 10 CFR 830 Subpart B "safe harbor method" such as DOE-STD-3009-94, Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses.

As with any new guidance document, questions typically arise regarding specific implementation challenges or intent of the Standard. The authors of this paper were also contributors to the development of Standard 5506 and support the review of TRU waste safety basis documents at numerous DOE TRU waste sites. These authors have been periodically contacted and requested to offer their personal opinions in response to questions from DOE sites upgrading their TRU safety basis documents.

This paper will share the authors' experiences in implementing the Standard and responding to questions about the Standard. A few examples include:

- Anomalies with the statistical Material-at-Risk (MAR) approach (e.g., situations that can skew data)

- Container deflagration source term factors related to contaminated plastics

- Clarifications to damage Ratios (DRs) used in certain accidents and containers
- Application of the risk ranking Table 6.2-2 for hazard control selection
- Reconciling 25 rem Evaluation Guideline discussions in DOE-STD-5506 vs. DOE-STD-1189

DOE field sites can benefit from a discussion of ongoing communications and experiences related to the standard. This will also foster discussion of shared site experiences.

Session Schedule

Wednesday, May 9, 1:00 - 2:30 pm, Caldera B

Session 8: Software and Quality Topics Chair: Mike Greutman	
1:00	**Raymond F. Sartor, John H. C. Wang** Requirements and Guidance for Generating MACCS2 Meteorological Data Files
1:20	**David L. Louie, Larry L. Humphries, and Randall O. Gauntt** MELCOR 2.1 Leak Path Factor Assessments and Guidance
1:40	**Larry L. Humphries, John Reynolds, and Randall O. Gauntt** MELCOR 2.1 Software Quality Assurance Program
2:10	**Megan Deising Houchin** Y-12 Safety Basis Annual Update Process & Quality Improvements
~2:30	Break

Wednesday, May 9, 1:00 pm

Requirements and Guidance for Generating MACCS2 Meteorological Data Files

Raymond F. Sartor (LANL) and John H. C. Wang (URS)
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At LANL, the MELCOR Accident Consequence Code System Version 2 (MACCS2) computer code is used in the atmospheric dispersion calculations to determine the 95th percentile X/Q value. MACCS2 is advantageous in that it is a DOE Toolbox code, readily available, and contains models for deposition, plume meander, and plume buoyancy. To calculate the 95th percentile X/Q, one or more data files of the annual local meteorological data are required by MACCS2. LANL Safety Basis analysts recently developed new MACCS2 data files from the data records of the four meteorological towers operated by the Environmental Data and Analysis group.

Appendix A of DOE Standard 3009 calls out NRC Regulatory Guide (RG) 1.23 as describing the acceptable means of generating the meteorological data upon which dispersion is based. Revision 1 (2007) of RG 1.23 identifies several requirements and methods for generating meteorological data files, but is insufficient by itself. Although not explicitly cited by DOE requirements and guidance, the following documents are appropriate, if not necessary, for generating the meteorological data files:

- EPA-454/R-99-005, Meteorological Monitoring Guidance for Regulatory Modeling Applications,
- ANSI/ANS-3.11-2005, Determining Meteorological Information at Nuclear Facilities, and
- NRC Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.

From these documents, this article will summarize the requirements and guidance regarding:

- Criteria for meteorological instruments,
- Data selection or averaging schemes,
- Completeness requirements for the original data,
- Data substitution to resolve omissions in the primary data,
- Treatments for calm conditions,
- Methods for determining the stability class, and
- Mixing layer heights.

Although LANL's work was specific to the MACCS2 computer code, the reported requirements and guidance will be applicable to creating meteorological data files for other computer codes using the Pasquill-Gifford dispersion model.

Wednesday, May 9, 1:20 pm

MELCOR 2.1 Leak Path Factor Assessments and Guidance

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MELCOR 2.1 is the current version of MELCOR developed at Sandia National Laboratories (SNL) for the Nuclear Regulatory Commission (NRC). Starting with version 2.0 and beyond, MELCOR is developed in the Fortran 95 programming language. In addition, MELCOR 1.8.5, which is listed in the current Department of Energy (DOE) Toolbox Tool specification for leak path factor (LPF) calculations, is no longer developed or maintained for code issues. SNL will continue to develop and assess the MELCOR 2.1code while improving code numerics and performance. We recommend adopting MELCOR 2.1 or later versions for LPF calculations to take advantage of new code development efforts.

Starting in MELCOR 2.0, the input format has changed dramatically. To help this transition, an input deck converter is available for translating inputs from MELCOR 1.8.6 to 2.1. The use of the converter and guidance/limitation of MELCOR 2.1 features for LPF applications will be provided in an appendix in the planned release of the Volume III (Demonstration Problems) of the MELCOR manuals. A summary of the guidance/limitation is given in this paper.

Finally, this paper discusses the assessment of MELCOR 2.1 using the sample problems as described in the current DOE MELCOR Computer Code Application Guidance for LPF in Documented

Safety Analysis (2004). The results of the comparison among MELCOR 1.8.5, 1.8.6 and 2.1 by using these sample problems show no significant difference. Additional assessments on MELCOR 2.1 features for the LPF applications were also performed and described in this paper.

Wednesday, May 9, 1:40 pm

MELCOR 2.1 Software Quality Assurance Program

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The MELCOR code is developed at Sandia National Laboratories (SNL) for the US Nuclear Regulatory Commission (NRC). As part of our ongoing software development program, Software Quality Assurance (SQA) is an integral piece of the development process. SNL has adopted a SQA program that focuses on reducing code error, improving documentation of all processes, automation of procedures to minimize cost and improve consistency, improvements of code user training and communication, and continuous integration of procedures into daily work processes.

This paper discusses ongoing improvements to the MELCOR SQA program at Sandia National Laboratories. A review of internal processes such as requirements development and management, code design management, verification and validation, development and lifecycle support, configuration management, measurement and analysis, and integrated product and teaming will be presented.

In May 2004, the U.S. Department of Energy (DOE) published a gap analysis report (DOE-EH-4.2.1.3-MELCOR-Gap Analysis) outlining areas where MELCOR SQA should be improved to resolve issues identified in the Defense Nuclear Facilities Safety Board recommendation 2002-1. This paper presents key improvements that satisfy those concerns.

Wednesday, May 9, 2:10 pm

Y-12 Safety Basis Annual Update Process & Quality Improvements

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Objective: This paper describes the Safety Basis document (specifically the DSA and TSR) annual update process used at Y-12. It reviews recent issues with document timeliness and quality and steps taken to mitigate these concerns in a "lessons learned" approach.

Relationship to Overall Interests of DOE Safety Analysis: Section 202(c) of 10 CFR 830 Subpart B requires the contractor to annually submit to DOE either the updated DSA or a letter stating that there have been no changes. B&W Y-12 has eleven nuclear facilities and submits annual updates one year from the date of the SER. This program is in response to several late submittals that occurred in previous years. Problems have also occurred in the quality of some of these submittals. This paper shares how Y-12 adheres to the above regulation and actions taken to improve document quality.

Results: The recent quality issues placed attention for the Facility Safety group to standardize the safety basis approach to changes. Focus groups, checklists, and standardization to a common computer platform were all solutions to improve the quality of these documents in addition to the use of timelines and stoplight metrics.

Benefits of Work to Mission of Sponsoring Organization: These quality improvements provide a smoother path to the Y-12 safety basis documents revision process. This paper allows other sites to gain lessons from the developments and methods used at Y-12.

Session Schedule

Wednesday, May 9, 1:00 - 2:30 pm, Barranca A

Session 9: Criticality Safety and USQ Topics Chair: Mark Mitchell	
1:00	**Phillip, B. Montgomery** Implementation of Expert USQDs in the DOE Complex
1:40	**Calvin M. Hopper, Sedat Goluoglu** DOE Nuclear Criticality Safety Engineer "Hands-On" Subcritical and Critical Experiments Training and Education Course
2:10	**Jeffrey W. Marr, Stephen A. Coffing, Michael R. Greutman** Lessons Learned from Development and Implementation of Electronic Unreviewed Safety Question (eUSQ) Software
~2:30	Break

Wednesday, May 9, 1:00 pm

Implementation of Expert USQDs in the DOE Complex

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The objective of this paper is to provide an update on the efforts across the DOE Complex to implement Expert USQDs (EUSQDs). Sites may find the compiled information helpful in their efforts to apply this methodology in their existing processes.

Many sites are actively streamlining the Unreviewed Safety Question (USQ) Process to improve its efficiency and reduce the burden. The EUSQD is one method, recently approved by DOE, to help streamline the process, reducing costs, yet remaining compliant with 10 CFR 830 and not diminishing quality. Although approved by DOE with a high bar to hurdle, many sites (both contractor and DOE) are interested in implementing it into their USQ Processes.

Because of variations in work & change control processes, some sites implementing the expert USQD methodology are tailoring the approach to more closely fit individual situations. Progress among the sites is explored and details of the programs will be shared where available. Contractor and DOE Assessments on implemented EUSQDs by both contractors and DOE have been completed and important lessons learned will be discussed.

Wednesday, May 9, 1:40 pm

DOE Nuclear Criticality Safety Engineer "Hands-On" Subcritical and Critical Experiments Training and Education Course

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The purpose of this presentation is to provide a description of the new US DOE Nuclear Criticality Safety Program (NCSP) "Hands-On" Subcritical and Critical Experiments Training and Education Course for Nuclear Criticality Safety Engineers (NCSE) in the DOE complex. The no-cost-to-student course is sponsored by the US DOE NNSA NCSP and replaces historic NCSE training programs at Los Alamos National Laboratory and then Lawrence Livermore National Laboratory. The course is designed to provide the elements of training and education to NCSEs that are not typically available or provided by the employers of NCSEs. The course is a comprehensive two-contiguous week course at various US DOE facilities. The need for the course was identified by the NNSA to develop consistency in the understanding and application of nuclear data/experiments, national consensus standards, US DOE regulations, orders, standards, guides as they apply to nuclear criticality safety programs in NNSA facilities, and the performance of nuclear criticality safety evaluations. The successful completion of the course by complete attendance, participation and passing of examinations is a part of the qualification of NNSA personnel with nuclear criticality safety oversight responsibilities and provides a common basis of training for contractor NCSEs.

Wednesday, May 9, 2:10 pm

Lessons Learned from Development and Implementation of Electronic Unreviewed Safety Question (eUSQ) Software

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Automation of nuclear safety processes has often been implemented to reduce effort, increase efficiency, increase productivity, and to improve the quality of the process. Over the past three decades, we have witnessed great strides in automating calculation-based processes in several areas such as dispersion, fire and explosion analysis, criticality evaluations, and accident/equipment failures. We have seen limited automation of paper-based processes. While the gains may be more limited than for calculation-based processes, the overall gains nevertheless can be sufficient to justify the expense of development and implementation.

The Sandia National Laboratories Safety Basis Department has completed development and implementation of an Electronic Unreviewed Safety Question (eUSQ) system to support the USQ process at Sandia. A specific goal for implementing this system was the reduction in observations and findings that have been documented in the past through software workflow controls. In addition to these workflow controls and allowing qualified individuals to prepare, review, and approve USQ evaluations, this system provides for access control, document control (including document retention and retrieval), and tracking of USQ evaluations.

During the development process, it was learned that development and implementation of the USQ process was not nearly as simple as automating an existing procedure. Several items were identified that required consideration and exploration including the choice of software platform, hardware resources, system availability, access control, tools for creation, storing, and retrieval of information, handling of electronic signatures, and project documentation. This paper discusses the lessons learned from the development and implementation of the eUSQ system.

Use of FRP Ductwork in a Nuclear Facility

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Objective: The objective of this paper is to determine the suitability for return to service of fiberglass reinforced plastic (FRP) ventilation ductwork in a nuclear facility.

Relationship to Overall Interests of U.S. Department of Energy Safety Analysis: Postulated fires are dominant scenarios in the safety analysis of nuclear facilities. A new process in a facility involves the return to service of fire retardant FRP ventilation ductwork. This study examines whether FRP ductwork can lose its properties with time and whether the fire retardant properties of the ductwork can be credited in the fire hazards analysis (FHA) and safety basis documentation for the facility.

Results: With proper design, analysis, and active fire protection, FRP ductwork can be successfully employed to mitigate the effects of a fire. However, the ductwork is not designed to prevent or stop a fire altogether. A set of good practices is developed for the operating organization to help meet Fire Protection Program requirements.

Benefits of Work to Mission of Sponsoring Organization: Understanding the fire retardant properties of FRP ductwork is beneficial in terms of analyzing the fire hazards associated with its use. This study ensures that the ductwork is appropriately analyzed in the FHA and safety basis documentation and that the ventilation system can be safely returned to service.

Panels

DOE Regulatory Panel
Chair: B. Evans
Wednesday 3:00 – 5:30, Tewa 3

DOE Office of Health, Safety, and Security, DOE/NNSA Program Offices, and DOE Field and Contractor representatives will discuss ongoing changes to DOE Directives and Standards, and the implementation of post-Fukushima actions, such as evaluation of beyond-design-basis events. This is planned as an interactive panel soliciting broad participation of those most affected by the recent or planned changes. Nominally a one-hour panel, but may be extended based on level of participation.

Use of Risk Applications for DOE Nonreactor Nuclear Facilities
Chair: Kevin O'Kula
Thursday 8:00 – 9:30, Caldera

The goal of the panel will be to review the status and developments in the DOE Complex since the draft December 2010 release of "Development and Use of Probabilistic Risk Assessments in Department of Energy Nuclear Safety Applications".

Specifically, within the context of DNFSB Recommendation 2009-1 on Risk Assessment Methodologies at Defense Nuclear Facilities, the panel session will provide observations and insights on the following:

1. Initial use of the draft Standard – successes, surprises, improvement areas.

2. Are there examples of implementation of the standard that serve as useful benchmarks?

3. Plans to revise and finalize the Standard reflecting on initial and anticipated use

4. Are there commercial applications from the nuclear industry, or the chemical process industry that could be used as models?

5. Will (or should) the standard be changed to reflect use of PRAs/ QRAs to address beyond design basis events?

6. How will the lessons learned from the Fukushima Dai-ichi event, and the DOE pilot program examining beyond design basis events potentially influence the Standard?

Atmospheric Dispersion Modeling in Safety Analysis
Chair: A. Vincent
Thursday 10:00 – 11:30, Caldera

The panel will address current issues and challenges in atmospheric dispersion modeling for site specific safety analysis application. This will include topics of deposition velocity as described in the recent HSS Safety Bulletin, codes such as MACCS and GENII, and choice and acceptance of conservatism's in particular parameters in modeling as well as within the overall modeling. It will discuss the various perspectives of the Department of Energy HQ, DOE field offices, contractors, and various experts. This will serve in conjunction with a number of papers to be presented in earlier sessions to feed an afternoon working session as support to DOE in establishing an integrated/coordinated approach for responding to the HSS Safety Bulletin and broader dispersion modeling issues.

Workshop Schedule

Thursday, May 10, 2012

Time	Location	Location
7:00	Registration/information	Promenade
	Sponsor exhibition booths	
	* private meeting (ask to reserve)	Barranca B
8:00	Risk Applications for DOE Non-reactor nuclear facilities	Caldera
	Nuclear Safety Research and Development Interest Group	Barranca B
	LANL tour leaves	
~9:30	Break	
10:00	Atmospheric dispersion modeling concerns	Caldera
11:30	Lunch (on your own)	
1:00	Atmospheric dispersion modeling workshop	Caldera
5:00	End of day	

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