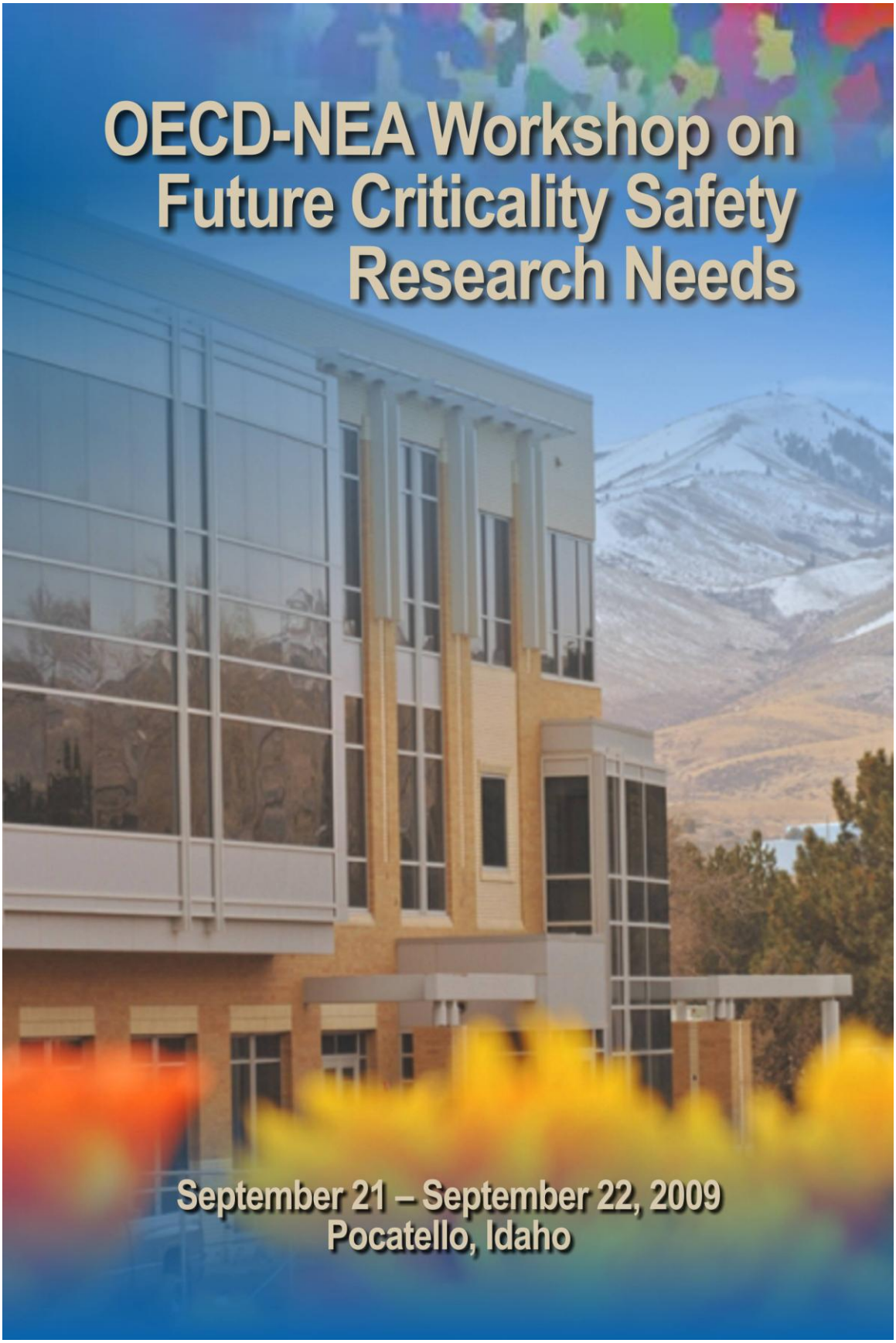
OECD-NEA Workshop on Future Criticality Safety Research Needs



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Pocatello, Idaho

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OECD-NEA Workshop on Future Criticality Safety Research Needs

Nuclear criticality safety is a key support function required for continued safe operation of current nuclear facilities as well as for anticipated future reactor and fuel cycle facilities that will present new and interesting challenges. These challenges will include the development of safety procedures for handling larger throughput of traditional uranium and plutonium inventories as well as the likely need to more comprehensively address the safe handling and storage of higher actinides such as americium and curium.

The primary purpose of the workshop is to help the international nuclear criticality safety community identify future criticality safety research needs so it will be better prepared to respond to those needs as future nuclear energy systems are developed. All sessions will be conducted by leading international nuclear experts. Topics include:

- Status of Development of Innovative Fuel Cycles for Future Nuclear Energy Systems
- Current Status and Expectations on Progress of Fuel Cycle Issues
- Oriented Basic Research Needs for Criticality Safety Study
- Decision Making Support for Criticality Safety Assessment
- Analysis of Criticality Issues Related to Waste Management and Disposal
- Status and Perspectives of Critical Experiments

Day 1 Schedule

Monday, September 21, 2009

7:30	Registration			
8:00	**Opening Remarks** Welcome: George Imel Welcome: David Nigg Keynote Speaker: Jerry McKamy Purpose and Goals: Véronique Rouyer			
8:45	**Session I: Status of Development of Innovative Fuel Cycles for Future Nuclear Energy Systems** Co-Chairs: Edward Fujita, Patrick Cousinou *Fuel cycle innovations, including fuel fabrication, transportation, and storage; issues associated with innovative nuclear systems, fuel cycles, reprocessing methods, as well as criticality issues, including handling of large quantities of higher actinides.*			
	Philip Finck 8:45 – 9:15	Pascal Anzieu 9:15 – 9:45	Tomozo Koyama 9:45 – 10:15	
10:15	Morning Break			
10:45	Discussion: Research needs and most probable innovations related to the advanced fuel cycle.			
12:00	Lunch Break – Hosted by Studsvik Speaker – Pierre d'Hondt, SCK			
13:30	**Session II: Current Status and Expectations on Progress of Fuel Cycle Issues** Co-Chairs: Calvin Hopper, Jose Conde *Industry perspectives on the current fuel cycle, such as increased U^{235} enrichment beyond 5 wt.%; new types of burnable poisons, such as erbium; material issues, such as retrieval, repacking, or transportation; and practical needs for criticality safety assessment, staff skills, and training.*			
	Marissa Bailey 13:30 – 13:50	Michel Doucet 13:50 – 14:10	Kouji Hiraiwa 14:10 – 14:30	Lon Paulson 14:30 – 14:50
14:50	Discussion: Main innovations related to the current fuel cycle.			
15:30	Afternoon Break			
15:50	**Session III: Oriented Basic Research Needs for Criticality Safety Study** Co-Chairs: Dennis Mennerdahl, Alain Santamarina *Codes, nuclear data, covariance data, and the need for new integral experiments of improved differential nuclear data.*			
	Massimo Salvatores 15:50 – 16:10	Richard McKnight 16:10 – 16:30	Alain Santamarina 16:30 – 16:50	
16:50	Discussion: Requirements for state-of-the-art criticality codes, nuclear data, covariance data and integral experimental needs.			
17:30	Adjourn			
18:45	Reception: Hosted by AREVA Speaker: Arthur Vailas, ISU Dinner: Sponsored by DOE/INL/ISU Speaker: Finis Southworth, AREVA			

Day 2 Schedule

Tuesday, September 22, 2009

7:30	Registration				
8:00	**Session IV: Decision Making Support for Criticality Safety Assessment** Co-Chairs: Cecil Parks, Jens-Christian Neuber *Methods of validation for criticality calculations (bias uncertainty analysis) via comparison with integral experiments, requirements for integral experiments, and approaches for scenario development.*				
	Tatiana Ivanova 8:00 – 8:20	Jens-Christian Neuber 8:20 – 8:30	Bradley Rearden 8:30 – 8:40	Christophe Venard 8:40 – 8:50	Patrick Cousinou 8:50 – 9:20
9:20	Discussion: Requirements and methods for state-of-the-art validation of criticality calculations for current and advanced fuel cycles and integral experimental needs.				
10:00	Morning Break				
10:30	**Session V: Analysis of Criticality Issues Related to Waste Management and Disposal** Co-Chairs: Michaele Brady-Raap, Jim Gulliford *Direct disposal of SNF, long term storage, HLW, legacy waste, decommissioning, and retrieval; codes for burn-up credit calculations, the need for assay data, realistic modeling of waste matrices, databases for relevant code validation, development of new standards/safe limits for common types of fissile waste streams, development of risk-informed approach to assessment of waste, NDA techniques for fissile waste, etc.*				
	Philip Wheatley 10:30 – 10:50	Robert Kilger 10:50 – 11:10	Peter Wood 11:10 – 11:30	John Wagner 11:30 – 11:50	
11:50	Discussion: Waste disposal challenges and possible ways to meet those challenges.				
12:30	Lunch Break – Sponsored by DOE/INL/ISU Speaker – TBD				
14:00	**Session VI: Status and Perspectives of Critical Experiments** Co-Chairs: Blair Briggs, Yoshinori Miyoshi *Availability and flexibility of experimental facilities for critical experiments and other experimental and measurement needs.*				
	Nichole Ellis 14:00 – 14:20	Anatoly Tsibulya 14:20 – 14:40	Hervé Glandais 14:40 – 15:00	Yuichi Yamane 15:00 – 15:20	
15:20	Discussion: Main innovations related to the current fuel cycle.				
16:00	Afternoon Break				
16:30	**Session VII: Summary and Conclusions** Véronique Rouyer				
17:30	Adjourn				

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Present Status on the Development of Advanced Reprocessing Technology for FBR Spent Fuel and Related Criticality Safety Design Issues: Tomozo Koyama

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United States Department of Energy Critical, Subcritical, and Fundamental Physics Measurements—Perspective and Status: Nichole Ellis

Experimental Programs at the BFS Facility Aimed for Validation of Criticality Safety for Plutonium Utilization: Anatoly Tsibulya

JAEA's Effort and New Program for Criticality Safety of Nuclear Fuel Facility: Yuichi Yamane

The Ten Year Mission and Vision for the DOE Nuclear Criticality Safety Program

MCKAMY, Jerry N., U.S. Department of Energy

Dr. Jerry N. McKamy is the Director of the Facilities and Operations Division (NA-172.1) within the National Nuclear Security Administration (NNSA) Office of Nuclear Safety and Operations, NA-17. He manages a staff responsible for the ~$1.6B/yr Readiness in Technical Base and Facilities (RTBF) Operations and Facilities budget that maintains readiness for all NNSA sites. Dr. McKamy is also the DOE Nuclear Criticality Safety Program (NCSP) Manager that establishes the criticality safety research and infrastructure maintenance program for the DOE. He received his Ph.D. in experimental nuclear astrophysics from Ohio State University (1982).

For the past decade the Department of Energy (DOE) Nuclear Criticality Safety Program (NCSP) has been chartered with maintaining the technical infrastructure supporting nuclear criticality safety.

The DOE NCSP mission is to provide sustainable expert leadership, direction, and the technical infrastructure necessary to develop, maintain, and disseminate the essential technical tools, training, and data required to support safe, efficient fissionable material operations within the DOE. The NCSP Vision is to be a continually improving, adaptable, and transparent program that communicates and collaborates globally to incorporate technology, practices, and programs to be responsive to the essential technical needs of those responsible for developing, implementing, and maintaining nuclear criticality safety.

The mission and vision will be achieved by identifying and accomplishing a set of five-year programmatic goals in six broad technical program elements that support identified ten-year goals. The yearly implementation plans to accomplish these goals will be developed with the advice and assistance of experts appointed by the NCSP manager or working under charters approved by the NCSP manager. The six technical program elements are:

- Analytical Methods
- Information Preservation and Dissemination
- Integral Experiments
- International Criticality Safety Benchmark Evaluation Project
- Nuclear Data
- Training and Education

The criticality safety research goals of the NCSP in the areas of Integral Experiments, Analytical Methods, and Nuclear Data will be discussed. In addition, an update on the US-French collaboration to establish a joint critical experiments research capability in Valduc will be presented. The goal of the joint collaboration is to provide a general purpose super-prompt critical actinide solution assembly and a large multipurpose horizontal split table capability. Together these two systems will provide important data for future fuel cycles and waste disposal applications in conjunction with the existing DOE Critical Experiments Facility (CEF) at the Nevada Test Site.

Future Direction of Nuclear Fuel Cycle Research in the United States

FINCK, Phillip J., Idaho National Laboratory

- *Director, AFCI Office of Technical Integration*
- *INL Associate Laboratory Director for Nuclear Science & Technology*
- *Fellow of the American Nuclear Society*

Dr. Finck received his doctorate in nuclear engineering at MIT in 1982, and also holds an MBA from the University of Chicago. He was a mechanical engineer at Novatome, a reactor in France, from 1983 to 1986, and was involved in the safety and design of fast reactors, including France's Superphénix. In 1986, he joined the staff at Argonne National Laboratory (ANL) in neutronics methods development for the Integral Fast Reactor concept, and later for the New Production Reactor. In 1991, he became the lead for neutronics analyses for Experimental Breeder Reactor-II. In 1993, he joined the French Atomic Energy Commission, where he was head of the Reactor Physics Laboratory at the Cadarache Center, with activities in light-water reactors and liquid-metal reactors, criticality safety, fuel cycle physics and nuclear data. In 1995, he was elected to chair the European Nuclear Data Project. In 1997, he rejoined ANL, where he was Associate Director of the Technology Development Division. He has led activities in the Advanced Accelerator Applications program since 2000, and has been heavily involved in transforming the program from accelerator-based to reactor-based transmutation. In 2003, he was named ANL Deputy Associate Laboratory Director for Engineering Research. On April 11, 2006, he was named ANL Associate Laboratory Director for Applied Science and Technology, where he was responsible for coordination of all nuclear energy-related activities at Argonne, including Advanced Fuel Cycle Initiative and Generation-IV programs, and development of new initiatives. On October 19, 2006, Dr. Finck joined the Idaho National Laboratory (INL) and was named Associate Laboratory Director for Nuclear Science & Technology. He is also the Director of the Advanced Fuel Cycle Initiative Technical Integration Office.

The future direction of nuclear fuel cycle research in the United States involves a renewed emphasis on evaluating broad sets of fuel cycle options and a shift to development of fundamental understanding of key fuel cycle processes. In order to implement this shift, investments in new analytical tools and research infrastructure will be required and increased coordination between government programs, universities, and industry will be necessary. The shift is also driving redefinition of research priorities related to advanced transmutation systems development, advanced nuclear fuel development, separations process development, waste forms development, and other elements of the U.S. Fuel Cycle Research and Development Program. These changes will be discussed in the context of broader domestic priorities associated with the need for reductions in carbon dioxide emissions and the need for increases in energy independence.

Present Status on the Development of Advanced Reprocessing Technology for FBR Spent Fuel and Related Criticality Safety Design Issues

KOYAMA, Tomozo, Japan Atomic Energy Agency

- *1980: Graduated from the University of Tokyo, Nuclear Engineering, and entered PNC. Engaged in development of FBR reprocessing technologies and design of FBR reprocessing facility.*
- *1984: Engaged in coordination of PNC-DOE Joint Criticality Data Development Program*
- *1986: PNC-DOE Collaboration on Fast Reactor Fuel Cycle Technology Development*
- *1991: Engaged in safety design of the Recycle Equipment Test Facility, especially in criticality safety design and licensing*
- *1997: Investigated the cause of the fire and explosion incident at the Bituminization Demonstration Facility of Tokai Reprocessing Plant.*
- *2000: General Manager in charge of development of FBR reprocessing technologies.*
- *2008: Unit Director of advanced reprocessing unit of advanced nuclear system research and development directorate.*

The Japan Atomic Energy Agency (JAEA) launched the “Fast Reactor (FR) Fuel Cycle Technology Development (FaCT)” Project in cooperation with Japanese electric utilities in 2006.

The FaCT project seeks to adopt innovative technologies by 2010, and, based on R&D associated with these technologies, the conceptual design of demonstration and/or commercial facilities by 2015. It would then be possible to achieve development targets such as safety and reliability, sustainability (environmental protection, waste management, and efficient utilization of nuclear fuel resources), economic competitiveness and nuclear non-proliferation. This program proposes commercialization by 2015.

In the FaCT project, the advanced aqueous reprocessing system (NEXT: New Extraction System for TRU Recovery) has been developed, which was selected as the most promising concept for commercialization. The NEXT process is composed of the following process and equipment: (1) disassembly and shear, (2) dissolution with continuous dissolver, (3) crystallization with continuous crystallizer for pre-recovery of excessive uranium, (4) U/Pu/Np co-recovery with centrifugal contactors, (5) MA recovery with extraction chromatography, and (6) salt-free process for waste reduction.

Criticality safety features will be included in the design of each unique piece of required equipment, such as continuous dissolver, centrifugal contactors, and slab tanks for intermediate vessels.

In envisioning the future fuel cycle, Japan will face the need to increase equipment capacity to enable reprocessing 500–800 tons of FBR spent fuel per year due to increased circulation of plutonium with FBR. This issue will be discussed at this workshop.

Regulatory Perspective on the Nuclear Fuel Cycle and Needs for Criticality Safety Assessments

BAILEY, Marissa, U.S. Nuclear Regulatory Commission (USA)

Marissa G. Bailey is the Deputy Director for the Special Projects and Technical Support Directorate, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards (NMSS). Her responsibilities include safety and licensing reviews of major fuel cycle facilities, including the proposed mixed oxide fuel fabrication facility and development of a regulatory framework for spent fuel reprocessing. She has been with the NRC since 1989 and has a B.S. in Nuclear Engineering from the University of Maryland.

The U.S. Nuclear Regulatory Commission (NRC), Office of Nuclear Material Safety and Safeguards, is responsible for regulating various aspects of the civilian nuclear fuel cycle: fuel fabrication and development; transportation of nuclear materials, including fissile material; reactor spent fuel storage; and management and disposal of spent fuel and high-level radioactive waste. The Office is also responsible for regulation and licensing of recycling technologies intended to reduce the amount of waste to be disposed through geologic disposal and to reduce proliferation concerns. The NRC regulates these activities to ensure adequate protection of public health and safety and the environment, and one of its strategic outcomes is to prevent the occurrence of any inadvertent criticality events.

To this end, the NRC staff conducts rigorous reviews of fuel cycle designs and operations proposed by licensees and applicants, including the methods for ensuring criticality safety. The NRC continually seeks to identify and resolve potential criticality safety issues, including those with generic implications and those arising from new technologies or methods. Additional research or data, as well as specific staff skills and training, may be needed to adequately address such issues.

Nuclear Renaissance: New Challenges for Criticality Safety Concerns

DOUCET, Michel, AREVA NP Fuel Sector DSBU

M. Doucet – Senior Expert – AREVA NP Fuel Sector
Michel DOUCET joined FRAMATOME (former AREVA NP) in 1989 after 14 years with Belgonucleaire in Belgium where he was busy with Neutronic codes and methodology as well as criticality analysis for the MOX fuel fabrication plant. During his time with FRAMATOME (1989–1995), he worked for the development of the SCIENCE Neutronic code package. Since 1995, he has been with the Fuel Sector Design and Sales Business Unit where he handles the criticality safety analysis to support the AREVA NP fuel factories (FBFC – CERCA) and the fresh fuel shipping cask for UO2 and MTR materials. He managed the training and mentoring of young individuals for 10 years.

Nuclear energy has made a comeback and is clearly one of the most efficient ways to fight the greenhouse effect.

With the nuclear renaissance we observe:

- New Nuclear Power Plant (NPP) buildings
- Fuel manufacturing plants, opportunities, and production capabilities
- New fuel development
- New need of shipping casks.

Generation III reactors fuel management will lead to better use of nuclear fuel; nevertheless, enrichment is still limited today to 5% ^{235}U. Improvements and optimization of burnable poisons management is one of the major tasks for the fuel vendors. However, for criticality purposes (fabrication, transport), these burnable poisons are excluded for criticality safety assessments and analyses.

AREVA NP is to be ready to deliver a large number of fuel assemblies for the next generation nuclear power plant by 2010 (i.e., the EPR). Improved production means some renewal of existing tools. Criticality safety is a main concern to be accountable for in the global redesign of the different processes.

Fuel assembly transport is also a major item—there are 241 fuel assemblies in EPR reactor cores, and turnover time will not be sufficient to make several transports knowing that overseas travel takes a long time. A new fleet of fresh fuel shipping casks has to be developed and operational by 2011.

Today's criticality specialists face a heavy workload and important issues with manpower and skills. For a decade, the AREVA NP Fuel Sector has made efforts to challenge this situation. Training and mentoring young individuals has been key to this issue's resolution and success.

This presentation will develop three specific items:

- Higher efficiency of fuel fabrication plants to meet new demand
- Need for a global AREVA shipping cask fleet
- Training and mentoring young individuals.

Progress of Above 5% Enrichment Fuel Development in Japan

HIRAIWA, Kouji, Institute of Applied Energy / Nuclear Power Engineering Center (IAE); Toshiba Corporation / System Design & Engineering Center

1982: Graduated from Master Nuclear Engineering Course, Nagoya University
1982–1989: Researcher of BWR Nuclear Fuel Development Section in NAIG
1989–1998: Researcher of BWR Nuclear Fuel Development Section in Toshiba
1998–1999: Senior Researcher in Toshiba
2000–2003: Senior Specialist of Next Generation Reactor Development in Toshiba
2004–2008: Chief Specialist of Next Generation Reactor Development in Toshiba
2009: Senior Researcher of The Institute of Applied Energy

1. Scope of development for above 5% enrichment fuel in Japan's NGR project.

In 2008, Japan began to develop a next-generation light water reactor (NGR). The development of above 5% enrichment fuel is one of the major concepts of the NGR. Since the design for the NGR requires 70GWd/t in discharge average burnup with a 24-month cycle operation, the required enrichment becomes above 5%.

2. Status of study of criticality safety issue above 5% enrichment fuel.

After the first two years of development, we plan to evaluate the technological impacts, countermeasures, and fuel cycle economy of the introduction of 10% enrichment to current fuel cycle infrastructure. Criticality safety, shielding radiation, and heat removal are the typical technical impacts in fuel cycles. Regarding the front-end cycle, we found that there is a narrow margin of criticality limit on current fabrication facility in the study if above 5% enrichment uranium is introduced. Regarding the back-end cycle, especially radiation from gamma, beta and alpha in some reprocessing process becomes two or three times larger than 45GWd/t fuel process.

Poison reactivity credit by dilute burnable poison is the new concept for countermeasure of criticality limit of above 5% enrichment, such as fuel fabrication and fresh fuel transport. By introduction of dilute poison credit, some existing equipment in the front-end facility, such as the fuel storage vessel in fuel fabrication facility, becomes usable as is. This means the reduction of renewal of equipment.

Typical poisons are erbia (ErO_2) and gadolinia (GdO_2). The fundamental applicability of the dilute poison credits has been evaluated in this development.

Some applicability of dilute erbia had been already evaluated in other METI (Ministry of Economy, Trade, and Industry - Japan) projects of above 5% enrichment. The typical required content of erbia in 10% enrichment uranium becomes 3.7%. Some physical property impacts, such as melting temperature and capability of uniform mixing of the low contents of erbia in liquid solution, have been evaluated in this development.

Some applicability of dilute gadolinia, such as reactor core performance, had been presented in some papers. The required typical content of gadolinia becomes 50 ppm in 6% enrichment of UO_2, and 300 ppm in 10% enrichment of UO_2 because of very large thermal absorption cross section compared with

erbia. An above 5% enrichment UO2 fuel with “minimal-content gadolinia (MCG)” has been called out in the past papers, and the intrinsic issue of MCG is the impact of inhomogeneous mixing of very dilute poison compared with erbia. In case of the powder mixture of the particle size of UO2 between one micro meter and two micrometers, we found several percentage of gadolinia should be increased in this study.

3. Plan of criticality safety experiment with critical assembly facility.

Critical experiments for homogeneous solution systems of the above-5wt% fuel have been performed with STACY facility of JAEA. We are planning critical experiments for heterogeneous systems of the above-5wt% fuel rods under this NGR project.

An Overview of GEH NCS Fuel Cycle Activities

PAULSON, Lon E., GE Hitachi Nuclear Energy

Manager, Nuclear Safety and Radiation Protection
GE Hitachi Nuclear Energy
Mr. Paulson has over 20 years' practical nuclear engineering experience involving radiological engineering, reactor physics, and nuclear criticality safety. He holds a B.S. in NE from Kansas State University (1984) and an M.S. in NE from North Carolina State University (1991). Work history includes 4 years' experience at commercial LWR, 3 years' experience at a DOE site, and 15 years' experience leading nuclear safety engineering at GE Hitachi Nuclear Energy (GEH) - Wilmington site. Mr. Paulson currently provides nuclear safety technical support of diverse GEH nuclear business units including GNF-A, Global Laser Enrichment (GLE), Vallecitos Nuclear Center (VNC), and GEH Canada divisions. He has authored/co-authored numerous publications on program/technical matters pertaining to nuclear criticality safety and nuclear packaging. He holds two U.S. patents as a co-inventor.
ANS member since 1984; served on NCSD Executive, Program, and Education Committees; has organized or chaired several NCS technical sessions/panel sessions; currently an active member of ANSI/ANS-8.1 and ANSI/ANS-8.26 working groups.

The GEH nuclear criticality safety (NCS) function, headquartered in Wilmington, NC, remains actively engaged in advancements to the nuclear fuel cycle. In addition to BWR fuel manufacturing, recent GEH emphasis to become more vertically integrated into front-end (enrichment) and back-end (reprocessing) fuel cycle technologies has had a dramatic impact on the NCS function. Required internal research in general physics areas, required nuclear criticality safety evaluations, Monte Carlo computational methods expansion, critical benchmark validation methodology improvements, nuclear packaging re-licensing efforts, quantitative risk assessments, nuclear cross-section data set development, and domestic/international licensing needs *collectively* present significant real-world NCS staff challenges. As the landscape of the GEH nuclear business growth opportunities continues to evolve, so does the required depth of NCS knowledge and technical expertise. This presentation provides a brief "snapshot" of select NCS efforts to support the GEH nuclear fuel cycle and provides insight to potential areas in which additional nuclear criticality safety research is needed.

The existing GNF-A fuel manufacturing operation (SNM-1097) recently underwent a license renewal. USNRC and industry expectations for validation of neutron transport methods used in nuclear criticality safety also evolved. As a result, the internal GEMER Monte Carlo code critical benchmark validation was updated, areas of applicability (AOAs) expanded, and a more sophisticated statistical methodology was established (to rigorously derive computational bias and associated bias uncertainty). Another area of focus has been to expand our analytic tool chest, including formal validation of MCNP5 using ENDF/B-VII cross-section library. The GEH subsidiary, Global Laser Enrichment (GLE), nuclear criticality safety function assisted in the design, licensing, and commissioning of a third-generation laser enrichment process. An experimental "Test Loop" was licensed under SNM-1097; separate GLE commercial facilities (CF) design, license application (LA), and ISA summary has also been performed. The traditional uranium system's subcritical limits were expanded from 5 to 10 weight percent U-235 enrichment in support of this effort.

Compound physics and chemistry associated with UF6 operations has also been internally advanced. The GEH Nuclear Energy Canada business has also entered the traditional enriched uranium fissile operation regime via planned future LEU fuel manufacture in support of advanced Candu reactor designs. The Peterborough operation was augmented with new nuclear criticality safety staff, and training was provided at GEH HQ in Wilmington, NC. Formal CNSC licensing efforts in support LVRF/ACR fuel manufacture at the Peterborough facility are underway. The NCS staff also assisted with criticality accident excursion modeling to support the planned Environmental Assessment (EA) for the facility. Finally, computational studies and subcritical limit derivations in support of advanced fuel cycles have begun as part of GEH involvement in the Global Nuclear Energy Partnership (GNEP) program. GEH NCS is evaluating a proposed nuclear fuel recycling center (NFRC) in which spent LWR or LMR nuclear fuel is recycled into a usable metal fuel for fast nuclear reactors using pyroprocessing technology. Pyroprocessing involves understanding the reactivity behavior of U-235, Pu-239, and Pu-241 metal systems to design effective controls. Collectively, these NCS fuel cycle integration activities have significantly expanded our traditional work scope, methods, and practices. This presentation provides an overview of these NCS technical support activities and identifies potential areas where additional NCS research and collaboration is needed.

Use of Covariance Data for Reactor Applications

SALVATORES, Massimo, CEA-Cadarache

- *Presently Senior Scientific Advisor at CEA, ANL, and INL*
- *Policy Director of the Generation-IV International Forum*
- *Leader of International Expert Groups and Projects in various areas of nuclear data for applications.*

Nuclear data uncertainties still have a significant impact on the assessment of the performances of most innovative nuclear systems, in particular on their waste transmutation potential and on their associated fuel cycle characteristics. The paper will give an overview of recent applications of covariance data and general sensitivity analysis tools to the assessment of integral parameters uncertainties and derived nuclear data target accuracy requirements. Indications will be given on possible strategies to meet these requirements.

How Covariance Data Helped Destroy Wall Street (and my Retirement Account) – And Can It Do The Same for Criticality Safety?

MCKNIGHT, Richard D., Argonne National Laboratory

Dr. McKnight has more than 35 years experience in reactor analysis, critical experiment design and analysis, criticality safety analysis, code validation, sensitivity analysis, benchmark modeling, and nuclear data testing. He is Section Manager for Criticality Safety in the Nuclear Systems Analysis Department of the Nuclear Engineering Division at Argonne. He chaired the Data Validation Committee of the Cross Section Evaluation Working Group for over 20 years; is one of the U.S. representatives to the NEA/NSC Working Group on International Data Evaluation Cooperation; is a member of the NEA/NSC International Criticality Safety Benchmark Evaluation Project Working Group, and currently chairs the Nuclear Data Advisory Group for the U.S. Department of Energy Nuclear Criticality Safety Program.

The role of covariance data in the recent demise of the markets and the role of covariance data in current criticality safety practice will be discussed. The current state of the art for analytical methods, nuclear data, integral experiments, and V&V for criticality safety will be reviewed with an emphasis on identifying the priority needs and future directions in each of these areas.

Status of JEFF-3.1.1 and ERANOS-2.1 Code System Validation for Fuel Cycle Calculation of Fast Reactors

SANTAMARINA, Alain, CEA

Prof. Alain Santamarina, Director of Research

The aim of the paper is to present the European JEFF-3.1.1 Library, the ERANOS-2.1 code system, and their capabilities to predict the fuel cycle behaviour of FBR applications by the analysis of various experiments in fast neutron spectra: reaction rates, criticality, PIE of irradiated separated-isotope samples, and reactivity worth measurements of minor actinides.

The JEFF-3.1.1 Nuclear Data Library is the latest version of the Joint Evaluated Fission and Fusion Library. The complete suite of data was released in 2008 and contains general-purpose nuclear data evaluations compiled at the NEA Data Bank in cooperation with several laboratories in NEA Data Bank member countries. JEFF-3.1.1 also contains radioactive decay data, activation data, and fission yields data. It combines the efforts of the JEFF and EFF working groups who have contributed to this combined fission and fusion file. The library contains neutron reaction data, incident proton data, and thermal neutron scattering law data in the ENDF-6 format.

The new release of the European Reactor Analysis Optimized Code System, ERANOS-2.1, has been developed and validated to establish a suitable basis for reliable neutronic calculations of current and advanced FBR cores of the GEN-IV International Forum. The latest version of the ERANOS code and data system, ERANOS-2.1, contains all of the functions required for reference and design calculations of the fuel cycle behaviours of Liquid Metal Fast Reactors (LMFRs), with extended capabilities for treating advanced reactor fuel subassemblies and cores of Gas Cooled Fast Reactors (GCFRs).

The analysis of several types of experiments with JEFF-3.1.1, associated with ERANOS-2.1 or/and continuous-energy Monte Carlo TRIPOLI4, shows the reliability of these calculation tools for criticality calculations and fuel inventory prediction. The first series of experiments used is critical core experiments, such as CIRANO, made in zero-power reactors like MASURCA or ZPPR. The second series is dedicated to fuel cycle behaviours: PIE experiments of fuel pins (TRAPU experiment) and separated-isotope samples irradiation (PROFIL experiments), both in LMFBR reactor PHENIX. The third series of experiments involves reactivity worth measurements of actinides in various spectra within the OSMOSE oscillation experiment in the MINERVE reactor.

From this qualification work, some required improvements on nuclear data are highlighted along with the need for new specific integral experiments.

Uncertainty Assessment for Criticality Safety Studies: An Overview of Techniques

IVANOVA, Tatiana, Institut de Radioprotection et Sûrete Nucléaire (IRSN)

T. Ivanova, PhD, is an R&D staff member in Criticality Assessment Study and Research Department at IRSN. Prior to joining the IRSN, she worked as senior researcher for IPPE, where her responsibilities included uncertainty assessments for SFR design studies, validation of the nuclear cross-section library, design, evaluation of integral experiments performed on BFSs and KOBRA facilities, and development of tools for uncertainty quantification and code validation. She joined IRSN in 2005 and has since been involved in calculation, analysis, and evaluation of integral experiments and the development of S/U capabilities for validation of criticality codes. Dr. Ivanova is a participant in ICSBEP and IRPhEP and chairman of the OECD/NEA EG Uncertainty Analysis for Criticality Safety Assessment.

To establish subcritical limits for a design system containing fissile materials, it is required to validate the criticality code to be used. Uncertainties inherent in the validation study are caused by several factors that can be grouped into the following classes: numerical approximations in the neutron transport code, uncertainties in basic nuclear data, and uncertainty of the representative experiments. A key point of the validation procedure is to propagate the uncertainties and any bias of the calculation method to the calculated k_{eff} of the design system.

The variation of criticality safety assessment rules that are adopted in different countries results in diverse approaches for validation of criticality computations. Rigor of the techniques currently used is sometimes constrained by engineering judgment when selecting the experiments and establishing the area of their applicability. Definition of bias and the bias uncertainty often employs simple statistical analysis. These limitations make the predicted accuracy reliant on subjective judgments that make it difficult to predict the bias for systems that have no or few similar benchmark experiments.

Improvements in methodology and tools for criticality validation should make it possible to optimize safety margins, i.e., accurately predict the k_{eff} bias and the bias uncertainty and provide better validation of criticality calculations. This becomes even more relevant, considering that new concepts are to be developed for future generation fuel cycles. There certainly will be a need not only to achieve an adequate level of criticality safety but also to achieve the economic optimization of large-scale, long-term materials handling operations.
The Expert Group on Uncertainty Analysis for Criticality Safety Assessment (EG UACSA) was established within the OECD/NEA Working Party on Nuclear Criticality Safety in December 2007 to exchange experiences from different countries in uncertainty assessment for criticality safety studies. Since its creation, the EG has been focused upon the techniques for criticality code validation.

A report providing a description of state-of-the-art methodologies and results of benchmark—exercises aimed at testing the capabilities of these methodologies to predict the k_{eff} bias and the bias uncertainty will be written by EG participants.

Based on contributions to the report, this workshop presentation will summarise how different methodologies address these and other relevant questions:

- What degree of correlation between the experiment and the application is necessary to validate the application area?
- What parameters are to be chosen for quantification of the correlation?
- How many experiments are needed to verify an application?
- How are correlations between the experimental uncertainties taken into account?
- How are the above discussed uncertainties propagated to the final results of the validation study?
- How is convergence of the validation procedure determined?

The presentation will also focus upon the necessity to provide and use in practice more formal validation procedures to better face the coming nuclear renaissance. It will discuss how and which validation procedure should serve in defining the experimental needs for the advanced fuel cycle and in designing integral experiments.

Hierarchical Bayesian Methods for Criticality Safety Assessment

NEUBER, Jens Christian, AREVA NP GmbH, Dept. NEEA-G

- *MSc in Physics.*
- *Worked in PSA/PRA of nuclear fuel processing and reprocessing facilities, propagation of pollutants with off-gas, waste air and water, criticality safety analysis and control.*
- *Has worked for Siemens/AREVA since 1989.*
- *Group leader criticality safety analysis since 1990.*
- *Chair of the Criticality Safety Committee of the German Society of Standardization (DIN).*
- *Member of KTA working groups dealing with criticality safety.*
- *Member of WPNCS and most of their expert groups since 1998.*
- *Member of the IAEA burn-up credit consultancy group.*

Criticality Safety Analysis uncertainty evaluation requires accountability for all uncertainties and correlations in the parameters describing the nuclear fuel system (NFS) to be analyzed, the parameters characterizing the benchmarks selected to estimate the bias of the applied calculation procedure, and the nuclear data (ND) involved in the NFS to be analyzed and the selected benchmarks. This is conveniently accomplished by using hierarchical Bayesian methods for the bias estimation procedure.

Step 1 of the procedure consists of drawing joint Monte Carlo (MC) samples on all the parameters characterizing all the benchmarks and on the ND involved in all the benchmarks by using probability density functions (PDFs) derived from the empirical data while taking the possible incompleteness of these data into account.

Step 2: Calculation of k_{eff}-values for each of the MC samples, thus obtaining a $n \times m$ matrix K of estimated bias values (n = number of MC samples, m = number of benchmarks).

Step 3: Calculation of sample means and sample covariance matrix from matrix K.

Step 4: Verification of a normal PDF model from which sample means and sample covariance matrix are observed.

Step 5: Drawing of an MC sample on the unknown covariance and then on the unknown benchmark means of the normal PDF model by making use of the distribution properties following from the PDF model for the sample covariance matrix and the sample means.

Step 6: Regression analysis of the sampled benchmark means using a linear model including a certain set of explanatory variables for the benchmarks (such as the TSUNAMI c_k values, fuel enrichment, moderation ratios, etc.).

Step 7: Drawing of an MC sample on the bias at those values of the explanatory variables which characterize the NFS to be analyzed. This step makes use of the distribution properties of the residual sum of squares and the estimates of the coefficients of the linear regression model.

It is necessary to repeat Steps 5 and 6 until a sufficient number of samples on the bias related to the NFS to be analyzed have been drawn. The distribution of the sampled bias values is then used to estimate an upper confidence limit of the bias related to some pre-given probability content (e.g., 95%).

The steps to evaluate the NFS are as follows:

Step 1: Same as Step 1 above in the case of the bias evaluation (MC samples on ND only required if not drawn for the benchmarks).

Step 2: Calculation of k_{eff} for each of the MC samples.

Step 3: Evaluation of the distribution of the sampled k_{eff}-values (estimation of an upper confidence limit).

These hierarchical procedures take into account not only all the uncertainties due to the NFS, the benchmarks, and the ND, but also all the uncertainties due to the empirical data required for choosing PDFs, the finite number and the possible incompleteness of these data, and the fundamental variability due to the selection of PDFs required for evaluating the empirical data.

SCALE TSUNAMI Sensitivity and Uncertainty Analysis Capabilities and Data

REARDEN, Bradley T., Oak Ridge National Laboratory

Brad Rearden obtained his B.S., M.S. and Ph.D. in Nuclear Engineering from Texas A&M University. He is a senior R&D staff member and leader of the Nuclear Criticality Safety Methods Team in the Radiation Transport and Criticality Group at Oak Ridge National Laboratory. He is the lead developer of the TSUNAMI sensitivity and uncertainty analysis tools of SCALE.

SCALE 6.0 TSUNAMI provides a comprehensive suite of tools for sensitivity and uncertainty analysis, especially as applied to criticality code validation. The sensitivity of k_{eff} to all multigroup cross-section data can be easily generated on explicit three-dimensional (3D) models using KENO V.a or KENO-VI. Uncertainties in k_{eff} due to uncertainties in the cross-section data are quantified with the comprehensive SCALE 6 cross-section-covariance data library. Correlation coefficients are generated based on the shared uncertainty between application systems and benchmark experiments to quantify the similarity of the systems on a system-wide or nuclide-reaction specific level. These correlation coefficients can be used in trending analysis to quantify the bias and bias uncertainty for the application based on available benchmark experiments. Gaps in validation coverage can be identified, and an additional margin of safety can be quantified to account for potential biases due to validation gaps. Data from various types of experiments can be assimilated to quantify a bias and bias uncertainty using data adjustment techniques. With the data adjustment techniques, it is possible to quantify sources bias on an energy-dependent, nuclide-reaction specific level. Also, bias uncertainties due to gaps in available benchmark experiments can be quantified on a detailed level. Bias information for particular materials obtained from replacement-type experiments can be obtained from reactivity-difference sensitivities and incorporated in the data-adjustment process, providing a unique means to project measured biases from the use of a material in an experimental matrix to its use in an application system, which could be substantially different from the experiments.

Methodology Used by the CEA to Determine the Computational Bias and Associated Uncertainty Due to Nuclear Data

VENARD, Christophe, CEA

For the past 11 years, Mr. Venard has been working on the development and experimental validation of the French criticality CRISTAL package at CEA. From 1988 to1998, he worked on the fuel management and core design of the Superphenix Fast Breeder Reactor at AREVA.

French companies mainly use the CRISTAL package to perform their studies. CEA and IRSN developed CRISTAL in collaboration with AREVA NP Company. The CRISTAL experimental validation database contains more than 2000 experiments from the ICSBEP handbook and French experimental programs. The validation procedures exploit comparisons between the calculated results and the experimental measurements. Sometimes, the similarities between the design system and the experiments are not obvious, despite the large number of experiments. In this case, it is difficult to choose experiments for the validation and transposition of experimental validation results to the design system calculations.

To respond to this need, CEA is developing a tool to decide and validate which experiments will be integrated into the future major version of CRISTAL: R.I.B. (*Représensativité, Incertitudes, Biais*) by CEA. The R.I.B. tool linked with the CRISTAL experimental validation database allows:

- Definition of the experimental validation area corresponding to the design system
- Assessment of k_{eff} computational bias and its uncertainty because of the nuclear data uncertainties for the design system.

The first part of this presentation shows the CEA's methodology for assessing errors and uncertainties in nuclear data. With this methodology, the information from the criticality safety experiments is taken into account in the assessment of these parameters. The R.I.B. method is based on the results of experiments with respect to the studied application and on the statistical adjustment theory of the nuclear data. The second part describes the main features of the R.I.B. tool, a Java software.

Toward Optimized Criticality Safety Limits

COUSINOU, Patrick, Insitut de Radioprotection et de Sûreté Nucléaire (IRSN)

1972: Graduated from the Industrial Chemistry High School in Lyon
1977: PHD in Physical Chemistry of Materials: Institut National Polytechnique of Lorraine
1972-1979: Teacher: High School of Mines in Nancy - (Chemical processes and Thermo chemistry)
1979-1982: COGEMA engineer: in charge of criticality studies for the UP3 and UP2-800 reprocessing plants design
1982-1988: Head of health physics measurements in the La Hague plant
1988-1990: IPSN Engineer: Criticality Studies department
1990-1993: IPSN: Head of Criticality Safety review office
1993-2003: IPSN: Head of Criticality Studies department
2003: IRSN: assistant of the Director of Plants, Laboratories, Transports and Waste Safety Division, in charge of research activities and criticality safety issues.

As part of the demonstration of the criticality safety of the industrial operations concerning the nuclear fuel cycle and the fuels for future reactors, several open questions can be legitimately asked.

The answer to these questions is likely to have a very important impact on the design of the apparatuses and on the research of the optimal solutions with respect to safety and the industrial and economic requirements.

The nuclear criticality control remains founded on the *a priori* limitation of one or more operational parameters such as the mass of fissile material, the geometrical form and dimensions and the quantity of moderating materials; this stage of the choice of the criticality control modes is determining.

Moreover, to fix the acceptable limits of the control parameters, it is necessary to seek, among all the fissile mediums present in the normal and possible abnormal operating situations, leading to the most constraining limits; this stage leads to the definition of a “bounding” fissile medium called “reference” medium, which is used to carry out dimensioning calculations.

In the past, the choice of “simplified” reference fissile mediums was generally made in order to facilitate the demonstration of their “bounding” character and because of the lack of validated basic nuclear data for some minor isotopes or sufficiently precise physico-chemical knowledge.

Thus it would be of a great interest that the clinicians and the specialists in the physico-chemical processes implemented in the factories of the fuel cycle join in a step of global optimization and undertake required researches in order to justify the safety margins associated with optimized dimensioning.

Moreover, it is probable that such an optimization would go in the same direction as the fight against the proliferation, which constitutes one of the issues to be taken into account in the choice of the processes and the designs of the factories of the future civil fuel cycle.

From some examples, this paper aims at proposing tracks of research to answer these questions.

Criticality Modeling for the Safe Disposal of Spent Nuclear Fuel

WHEATLEY, Philip, Idaho National Laboratory

Mr. Wheatley is currently the Division Manager for Environmental and Natural Resources at Idaho National Laboratory (INL). He is also a registered professional engineer who has practiced with increasing positions of responsibility over the past 33 years at INL. He has made significant contributions to technical programs in the areas of reactor safety analysis, low-level radioactive waste management, and most recently, spent nuclear fuel management. Mr. Wheatley has been responsible for a broad range of programs dealing with spent nuclear fuel and nuclear materials.
Most recently, Mr. Wheatley was part of the Department of Energy (DOE) spent nuclear fuel program within DOE - Environmental Management. This includes ensuring that all needed analyses and information were available for the Yucca Mountain License Application.
Mr. Wheatley currently directs the efforts of INL staff engaged in technology development for environmental remediation and nuclear materials management.

The United States continues to debate the safe disposal of spent nuclear fuel and other fissile materials currently being stored by DOE and commercial nuclear utilities. It is likely that nuclear materials will be stored for at least the next 20 years as repository development continues. Longer storage times will increase the pressure to increase the storage density of materials to minimize storage costs. Increased storage density will also require increased accuracy when performing criticality calculations supporting transportation. In addition, the geologic media for disposal is once again in question as the United States investigates new disposal sites. DOE is also looking at alternatives for disposal of fissile materials. All of these developments will require criticality modeling efforts and benchmark data to gain acceptance by safety and regulatory agencies.

Criticality Issues of Direct Disposal of SNF and Fissile Waste

KILGER, Robert, Gesellschaft für Anlagen- und Reaktorsicherheit GRS mbH

Dipl.-Phys. Dr. rer. nat., thesis at University of Ratisbon in 2000; since 2003 he has worked with GRS in the Nuclear Fuel Department as a Criticality Safety Expert; involved in safety assessments for German facilities for fuel fabrication, transportation, storage and disposal; co-author of the German Handbook for Criticality; joined NEA Working Group on BUC in 2004;mMember of German Working Group on Criticality Safety of DIN; has been in charge of several studies on criticality safety for direct disposal of SNF in Germany.

Criticality safety for disposal of fissile material and SNF is a challenge in several respects. First, the demonstration of sub-criticality in the operational phase of the final repository is a “classical” task of criticality safety, even if new tracks like burn-up credit application are being pursued. In general, criticality safety of the SF canisters or waste packages has to be demonstrated for handling and storage under normal and accidental conditions. Burn-up of SNF as well as neutron absorbing materials may be considered appropriately.

Second, the criticality analysis for the post-closure phase is a much more challenging exercise, which requires co-operation and expertise from different areas, such as geology, geo-chemistry, criticality, and perhaps probabilistic analysis. Important parameters determining the extent of such analyses are, for instance, the characteristics of the waste packages, the fissile content of the waste or SNF, the geologic conditions (host rock of the repository, long term behavior of the system) and the time frame to perform the analysis. Deterministic as well as probabilistic methods may be applied, depending on the type of approach.

The third issue of disposal criticality analyses is the determination of possible consequences of a hypothetically assumed critical system in the post—closure phase of a repository. The objective of such an analysis, which may be performed additionally, is to evaluate possible effects on the stability of the barrier system of the repository.

From these tasks, requirements on the codes and data used for the analysis arise if the analyses are to be done for a licensing application of a planned disposal facility. Special attention has to be given on validation of nuclear data of material mixtures, which are typical for geologic disposal (e.g., salt rock, clay, and crystalline rock). Important nuclides in this context are Cl, Si, Al, Fe, Ti, Ca, Mg, Mn, S and P. Furthermore, comparison of codes for nuclide inventory and criticality calculation is an important issue. Regarding the validation of codes for critical excursion analysis under geologic conditions, systems providing a very low reactivity supply may be of special interest.

In the presentation, these aspects will be discussed in more detail. Examples from criticality analysis for direct disposal of SNF in Germany will be presented. Possibilities for improvement of code validation and specific nuclear data for disposal criticality analysis will be discussed.

Criticality Safety Research for a UK Geological Disposal Facility

WOOD, Peter, Nuclear Decommissioning Authority / Radioactive Waste Management Directorate

Peter Wood has a degree in Chemistry and a doctorate in electroanalytical chemistry from the University of Oxford. Peter joined the nuclear industry in 1980, conducting research into the performance of nuclear fuel in and out of reactor. He was seconded to the OECD Halden Reactor Project during 1991-92. Peter moved into waste management in 1994 and into the NDA (Nuclear Decommissioning Authority) in 2007. He is currently the Criticality Research Manager in the Research Group of RWMD (Radioactive Waste Management Directorate).

The Radioactive Waste Management Directorate (RWMD) of the UK's Nuclear Decommissioning Authority (NDA) is charged with the delivery of a Geological Disposal Facility (GDF).

The wastes and materials that might require management in a GDF include Intermediate Level Waste (ILW), High Level Waste (HLW), Spent Fuel, plutonium, and uranium. With the exception of the vitrified HLW, these all contain ton quantities of fissile radionuclides, principally U-235 and Pu-239.

Safety assessments and supporting studies have been conducted over the past twenty years, initially focusing on ILW, but recently being extended to other materials. The presentation will:

- Outline safety arguments for the disposal of wastes and materials containing fissile material;
- Describe the research program undertaken to understand criticality under repository conditions;
- Summarize the status of this research program and its reporting;
- For some outstanding areas of uncertainty, identify research needs and opportunities; for example, validation needs and benchmarking opportunities in order to build confidence in the models that have been developed to predicted the consequences of hypothetical criticality events; and
- Indicate how the understanding gained and model predictions are being applied in safety cases.

Postclosure Criticality: Basis for Exclusion and Role of Burnup Credit

WAGNER, John C., Oak Ridge National Laboratory / Nuclear Science & Technology Division

John Wagner is the Radiation Transport & Criticality group leader at Oak Ridge National Laboratory (ORNL). He received a B.S. in nuclear engineering from the University of Missouri-Rolla in 1992 and a M.S. and Ph.D. degrees from Pennsylvania State University in 1994 and 1997. His dissertation research focused on development of a new hybrid radiation transport method and parallel processing for Monte Carlo shielding calculations. Wagner spent two years with Holtec International, primarily performing criticality safety analyses for storage and transport of commercial spent fuel. Wagner joined ORNL in 1999, where he continued work in hybrid transport methods development and burnup credit criticality safety. In late 2006, Wagner became technical lead for postclosure criticality in support of the proposed repository at Yucca Mountain.

This presentation will describe the methodology and approach used to address the issue of criticality in the proposed repository at Yucca Mountain during the postclosure disposal time period. The United States Nuclear Regulatory Commission regulatory criteria will be reviewed relevant to postclosure criticality; the role of criticality will be explained within the overall repository performance assessment; the strategy will be described for preventing criticality via design features and waste form properties; and the numerous considerations will be reviewed relevant to criticality of spent fuel during disposal.

The most significant and effective measures for prevention of criticality in the repository include: multiple, redundant barriers that act to isolate the fissionable material from water (which can act as a moderator, corrosive agent, and transporter of fissile material); inherent geometry of the waste package internals and waste forms; presence of fixed neutron absorbers in the waste package internals; and fuel burnup for commercial SNF.

Within the probabilistic approach, criticality is considered an event, and the total probability of a criticality event during the disposal time period is calculated and compared against the regulatory criterion. The total probability of criticality includes contributions associated with both internal (within the waste package) and external (external to the waste package) criticality for each of the initiating events that could lead to waste package breach. Despite numerous and significant conservative analysis assumptions in the event sequences requisite to enabling criticality (i.e., analysis assumptions that increase the calculated probability of criticality), the probability of nuclear criticality during the postclosure performance period is determined to be below the regulatory threshold for inclusion in the repository performance assessment.

United States Department of Energy Critical, Subcritical, and Fundamental Physics Measurements—Perspective and Status

ELLIS, A. Nichole, Ellis Nuclear Engineering LLC

Nichole Ellis is currently an independent consultant working with the Department of Energy Nuclear Criticality Safety Program. Her responsibilities as part of the NCSP Management Team include being a liaison between various aspects of the Nuclear Criticality Safety Program, such as the Criticality Safety Support Group, Nuclear Data Advisory Group, EndUsers Group, and the Criticality Safety Coordinating Team. Additionally, Nichole is responsible for the development and coordination of the process for the United States Criticality Integral Experiment Requests and Critical and Subcritical Experiment Design Teams. Nichole also participates in the International Criticality Safety Benchmark Evaluation Project and the International Reactor Physics Evaluation Project.

The United States Department of Energy (DOE) Nuclear Criticality Safety Program (NCSP) is responsible for maintaining the fundamental capability to be able to perform critical, subcritical, and fundamental physics measurements, within the limits of its resources, to address the specific needs of the DOE United States Criticality Safety community. The capability to perform these various integral measurements supports safe, efficient fissionable material operations within DOE and is done by providing a sustainable infrastructure and a systematic, interactive process to assess, design, perform, and document integral criticality safety-related benchmark-quality experiments/measurements.

DOE previously performed the majority of critical, subcritical, and fundamental physics measurements at Los Alamos National Laboratory in Los Alamos, NM. However, in order to ensure continued safety and security measures at designated DOE measurement facilities, DOE shut down the critical, subcritical, and fundamental physics experiment operations at Los Alamos and safely transported the material and associated equipment to a new, more secure location at the Nevada Test Site (NTS). The new Critical Experiment Facility (CEF) is currently under construction at the Device Assembly Facility and has already resumed subcritical and fundamental physics measurements. The CEF is scheduled for completion in 2010, and once the new experiment facility is fully operational, multiple approved critical measurements are scheduled to commence at this facility. Other DOE facilities, such as those located at Sandia National Laboratory, are also available and used for performing critical, subcritical, and/or fundamental physics measurements as necessary to support safe, efficient fissionable material operations within DOE.

In order to facilitate integral measurement capabilities with DOE, a Critical Subcritical Experiment Design Team (C_EdT) process has been developed. The goal of the C_EdT process is to provide a viable method for identifying, designing, and approving all new critical, subcritical, and fundamental physics measurements. This process will ensure that an experiment requestor's nuclear data needs are well understood and met by integrating all capabilities of the NCSP to design and approve the requested measurements, including deciding which facilities within DOE are best suited to perform and document the requested measurements. Additionally, this new process is an interactive process that ensures continuous communications between all parties of the C_EdT. At the completion of the approved measurements, the measurement data

and evaluation of that data are provided to the Criticality Safety Community via the *International Handbook of Evaluated Criticality Safety Benchmark Experiments*.

This presentation will discuss the DOE NCSP's continued need for the ability to assess, design, perform, and document integral criticality safety-related benchmark-quality experiments/measurements; how the new CEF and other measurement facilities within the DOE are expected to perform and document the necessary requested measurements; how the new C_EdT process will ensure that all the measurements are properly assessed, designed, performed, and documented; and what current measurements are approved and/or are being performed by the DOE NCSP.

Experimental Programs at the BFS Facility Aimed for Validation of Criticality Safety for Plutonium Utilization

TSIBULYA, Anatoly, Institute of Physics and Power Engineering (IPPE)

Education

- *Moscow Engineering Physics Institute (MEPhI), Obninsk, Russia, Nuclear Engineering, MS, 1971*
- *Institute of Physics and Power Engineering (IPPE), Obninsk, Russia, Physics and Mathematics, PhD, 1980, Dissertation Topic: "Validation of System of Group Constants for Calculations of Fast Reactors"*

Current and Previous Positions

- *Junior Scientist, Scientist, Senior Scientist, IPPE, Obninsk, Russia, 1971–1987*
- *Head of the Laboratory of Cross Section Development for Calculations of Reactors, Shielding, and Safety, IPPE, Obninsk, Russia, 1987–1997*
- *Director of the IPPE Division – Institute of Nuclear Power Reactors, IPPE, Obninsk, Russia, 1997–present*

Field of Researches

- *Development of cross sections and codes for calculations of neutron-physical characteristics of fast reactors and nuclear fuel cycle*
- *Verification and validation of cross sections and codes*
- *Design and analysis of experiments on critical assemblies and power reactors*

Publications

- *Coauthor of 2 books: Group Cross Sections for Calculations of Reactors and Shielding (in Russian), 1981, Multigroup Approximation in the Neutron Transport Theory (in Russian), 1984, and over 80 referred publications*

The BFS critical facility was originally designed to investigate neutron-physics characteristics of sodium-cooled fast reactors. However, because of the plutonium utilization problem, the facility is now being used to investigate criticality safety issues for non-reactor applications. The first experiments under this program were performed in 1999–2000 at the BFS-79, -81 assemblies. Critical conditions simulating near autocatalytic chain reactions in an underground fissile material repository were measured. Various levels of water ingress into a fissile material-silicon dioxide medium were modeled. In 2003, a second series of experiments was performed at the BFS-93 assemblies in which the reactivity effects and power distributions were measured for water-moderated VVER-type fuel assemblies with replacement of standard fuel rods with vibrocompacted MOX-fuel. The third series of experiments was performed in 2004–2005 at the BFS-97, -99, and -101 assemblies. Critical parameters of systems with MOX-fuel and low hydrogen-to-fuel ratios that occur during fuel manufacture were modeled.

In Russia, a new plant has been designed for the pyrochemical technology of manufacturing of vibrocompacted MOX-fuel for the BN-800 reactor. Several additional critical experiments at the BFS facility are planned.

Results of Russian and U.S. critical experiments with plutonium and low H/Pu ratios revealed some contradictions. The report will show unexpected differences between the experimental data and the calculational results. These discrepancies do not allow the uncertainty of criticality estimations for these systems to be reduced below 1%. Additional experiments are planned to resolve

the discrepancies. In the pyrochemical technology plant project, damp MOX powder occurs when washing residual salts from the granulated fuel. To validate criticality calculations for transportation and storage of fresh fuel assemblies of the BN-800 reactor with MOX-fuel (~25% of plutonium in the highly enriched zone), experiments will be performed with the 80-cm-height vibrocompacted MOX-fuel rods (~400 rods). Low water content will be modeled by placement of a thin polyethylene displacer into the gaps between the fuel rods. High water content will be modeled by filling tanks containing MOX fuel rods with water while using the BFS core as a driver. To validate criticality calculations of the melt of salts with MOX fuel, reactivity measurements will be taken with CsCl, NaCl, and Na in the assemblies modeling the electrolyzer, and critical characteristics of assemblies containing the CsCl salt will be investigated.

JAEA's Effort and New Program for Criticality Safety of Nuclear Fuel Facility

YAMANE, Yuichi, Japan Atomic Energy Agency

Yuichi Yamane (B.S., Physics, 1990; M.S., 1992; and Ph.D., Applied Physics, 1995, Nagoya University, Japan) is an assistant principal researcher of criticality safety for the Japan Atomic Energy Agency (JAEA). His research interests include criticality accident analysis methods and subcriticality monitoring methods.

JAEA has constructed two types of cores, STACY and TRACY, for the criticality safety of domestic fuel processing facilities and has systematically collected criticality data of uranyl nitrate solution since 1995.

STACY, STAtic experiment Criticality facilitY, was used to measure the criticality amount of uranyl nitrate solution and to develop subcriticality monitoring methods. The experiments were done with a cylindrical or slab core, a coupled core, and an inhomogeneous core with or without a water reflector. The criticality height was measured for 10%-enriched uranyl nitrate solution with concentration under 500gU/L. A 6%-enriched uranyl nitrate solution and less than 500 fuel rods were used for the inhomogeneous core. The data are used as the benchmark data of ICSBEP and the second edition of the criticality handbook data collection. The subcriticality monitoring method using inverse kinetics or the noise method was developed.

TRACY, TRAnsient experiment Criticality facilitY, was used to collect the power, temperature, and pressure profile data for spontaneous or ramp insertion of up to 3% reactivity in order to understand criticality accident phenomena and to develop the evaluation method for its consequence. The 10%-enriched uranyl nitrate solution with 365 to 433 gU/L concentration was used in a 50-cm-diameter cylindrical tank with or without a 50-cm-thick water reflector. The one-point kinetics code AGNES was developed based on the data, and the information from the TRACY experiment helped decide the action at the JCO criticality accident and develop and perform the criticality accident evaluation of the MOX powder process.

Lowering the total cost of nuclear electricity demands the longer periods of operation and higher uranium enrichment, so the feasibility study of the fuel cycle system of over 5% enriched uranium fuel becomes much more important. It is also necessary that the criticality safety study include burn-up credit for fuel processing, transportation, reprocessing, and storage.

According to the experiment program for research on nuclear criticality safety of fuel cycle and development of advanced light water reactors, the conceptual design of critical core conditions using fuel rod assemblies is ongoing. Using the modified experimental core, the fundamental critical characteristics and kinetic parameters of heterogeneous cores with wide moderating ratio and fuel conditions will be studied. For the licensing procedure, safety designs for the reactivity control system have been performed. Typical calculation results of basic properties for the planned core will be presented.

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AREVA NP Inc. is a designated subcontractor in the BEA contract to manage and operate the Idaho National Laboratory. Also, AREVA Enrichment Services is designing, licensing, and constructing a uranium enrichment facility eighteen miles west of Idaho Falls. This facility will be operated by AREVA, providing up to 6 Million SWUs to the US enrichment market annually, and will begin operation in 2014. AREVA NP Inc. is also working with Advanced Energy Holdings, Inc., seeking to build an EPR power plant in Idaho. Idaho is key to the future of nuclear energy in the US, and key to a many of AREVA's activities. AREVA is also pleased to support the growing strength of academic programs in Idaho universities in support of nuclear power.

Build long-term relationships with our clients by anticipating their needs and delivering economically and socially valuable science and technology.

Energy. Battelle is leading a global energy revolution to diversify sources of energy, ensure reliability of the energy system, and maximize productivity for economic advantage. From leading major government and industrial initiatives like FutureGen and coal conversion to developing commercially viable fuel cells and energy storage "beyond batteries," Battelle's science and technology base and dedicated leadership are forging a powerful future—with energy for all.

Health & Life Sciences. Beyond sequencing the human genome, Battelle is tapping assets in proteomics, advanced computational biology, and medical device development to take life sciences to the next level. We are building the science and technology foundation for major changes in medicine, healthcare, and bio-energy..

National Security & Defense. Battelle is applying unmatched expertise and unique facilities to deliver homeland security solutions. From detection and protection against weapons of mass destruction to emergency preparedness/response and protection of critical infrastructure, we are working with industry and government to integrate policy, operational, technological, and logistical parameters that will secure a safe future.

Laboratory Management. High impact managers of scientific resources, Battelle delivers the highest return on research dollars. Serving on the management teams of four premier Department of Energy National Labs whose staff are redefining the sciences of energy, materials, security, and biological systems, we are trustworthy stewards of the most powerful scientific instruments in the world.

Community & Education. A strong partner in building tomorrow's technology workforce, Battelle follows our founder's mandate daily: invest in educating men and women who will become the innovators of the future. We are leading STEM education initiatives—from a million dollar donation to the National Society of Black Engineers to building Metro High School—that leverage the power of business/government/educator partnerships for students.

The Mission and Vision of the U.S. Department of Energy's Nuclear Criticality Safety Program (NCSP)

The NCSP mission is to provide sustainable expert leadership, direction, and the technical infrastructure necessary to develop, maintain, and disseminate the essential technical tools, training, and data required to support safe, efficient fissionable material operations within the DOE.

The vision is for the NCSP is to be a continually improving, adaptable, and transparent program that communicates and collaborates globally to incorporate technology, practices, and programs to be responsive to the essential technical needs of those responsible for developing, implementing, and maintaining nuclear criticality safety.

The NCSP Mission and Vision will be achieved by identifying and accomplishing a set of five-year programmatic goals in six broad technical program elements that support identified ten-year goals. The yearly implementation plans to accomplish these goals will be developed with the advice and assistance of experts appointed by the NCSP manager or working under charters approved by the NCSP manager. The six technical program elements are:

- Analytical Methods
- Information Preservation and Dissemination
- Integral Experiments
- International Criticality Safety Benchmark Evaluation Project
- Nuclear Data
- Training and Education

INL today reports up through the DOE Headquarters Office of Nuclear Energy, and serves as the nation's lead laboratory for nuclear energy research and development. INL conducts a wide range of agency-supporting activities:

Nuclear Science. INL is the leading laboratory in basic nuclear and radiological science research and applications. Both DOE and non-DOE customers utilize the expertise and assistance of INL's leading nuclear scientists to address critical needs.

Nuclear Reactor Design, Development, Operations & Safety. With nearly sixty years of experience in nuclear reactor and nuclear materials processing, plant design, operations and decommissioning, INL has internationally recognized expertise to conduct nuclear reactor R&D.

National Security. INL leverages its signature capabilities in wireless and communication systems, process control and cybersecurity, UAV platforms and sensors along with its complex, secure and remote facilities to provide comprehensive critical infrastructure testing and technology development to DOE, DHS, other government agencies and industrial partners.

Fossil Energy, Energy Efficiency, and Renewable Energy. With its significant science and engineering capability in fundamental energy and materials science, transportation, industrial efficiency, energy efficiency, building technologies, and fossil energy, INL conducts targeted R&D for DOE's Fossil Energy, and Energy Efficiency and Renewable Energy programs. INL is one of three lead laboratories for the DOE Geothermal Program and is responsible for the program's geoscience research component. INL is lead laboratory for engineering support to the DOE Hydropower Program wherein it applies engineering expertise to issues such as fish mortality, impact on aquatic environments, water quality and land use.

Science. The scientific reputation and credentials of INL researchers play a key role in accomplishing basic science activities. Broad research disciplines represented at INL include earth sciences and environmental engineering, biotechnology, physical systems modeling, systems engineering, intelligent automation and remote systems, applied engineering, materials processing, chemical separations and processing, and sensing and diagnostics.

Idaho State University is a public doctoral research and teaching university supported by the citizens of the State of Idaho. It was established in 1901 in Pocatello, Idaho as the Academy of Idaho. Today the university consists of seven colleges providing degrees in a wide variety of fields. Idaho State University has a student population of 14,000. It is Idaho's center for education in the health professions and is the only university in the state offering degrees in nuclear engineering at all levels from Bachelor to Doctorate level.

The Nuclear Engineering and Health Physics Departments offer degrees at all levels. Currently there are over one hundred undergraduates and seventy-nine graduate students in these programs. Idaho State University has its own AGN 201 reactor on the Pocatello campus for lab classes. The Idaho Accelerator Center provides opportunities for R & D in nuclear physics applications in materials science, biology, homeland and national security to the University, industry and national laboratories. Nuclear Engineering and Health Physics each sponsor a wide variety of extracurricular and professional development activities through student organizations and special field trips. Recent employers of our graduates include AREVA NP, INPO, Energy Northwest, Bechtel, Duke Power, and Battelle.

Idaho State's proximity to the Idaho National Laboratory (INL) and its new Center for Advanced Energy Studies (CAES) provides nuclear engineering and health physics students with unique educational experiences. Students have access to graduate research opportunities, frequent guest lectures by top nuclear scientists, as well as summer and academic-year internships for both undergraduates and graduate students. For the faculty, contacts with the INL and CAES keep them abreast of current research in the nuclear field and offer opportunities for collaborative research.

Idaho State partners with the Idaho National Laboratory to sponsor special educational opportunities such as the 2009 first annual Modeling, Experimentation and Validation Summer School. In 2008 ISU and INL jointly sponsored a Verification and Validation for Nuclear Systems Analysis Workshop. In 2005 ISU and INL hosted the first World Nuclear University Summer Institute.

NUCLEAR ENERGY AGENCY

ORGANISATION FOR ECONOMIC
CO-OPERATION AND DEVELOPMENT

The Nuclear Energy Agency (NEA) is a specialised agency within the Organisation for Economic Co-operation and Development (OECD), an inter-governmental organisation of industrialised countries, based in Paris, France. The mission of the NEA is to assist its Member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for the safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. To achieve this, the NEA works as: a forum for sharing information and experience and promoting international co-operation; a centre of excellence which helps Member countries to pool and maintain their technical expertise; a vehicle for facilitating policy analyses and developing consensus based on its technical work.

The NEA's current membership consists of 28 countries, in Europe, North America and the Asia-Pacific region. Together they account for approximately 85% of the world's installed nuclear capacity. Nuclear power accounts for almost a quarter of the electricity produced in NEA Member countries. The NEA works closely with the International Atomic Energy Agency (IAEA) in Vienna—a specialised agency of the United Nations—and with the European Commission in Brussels. Within the OECD, there is close co-ordination with the International Energy Agency and the Environment Directorate, as well as contacts with other directorates as appropriate.

Australia	France	Japan	Slovak Republic
Austria	Germany	Korea	Spain
Belgium	Greece	Luxembourg	Sweden
Canada	Hungary	Mexico	Switzerland
Czech Republic	Iceland	Netherlands	Turkey
Denmark	Ireland	Norway	United Kingdom
Finland	Italy	Portugal	United States

Studsvik™ Scandpower

Studsvik Scandpower, Inc. is a recognized world leader in the development and application of complex computer codes which are widely used for the design, analysis, licensing, operational support, and core monitoring of Light Water Reactors (LWRs). These computer codes are currently used by more than fifty utilities and fuel vendors throughout the world in the analysis of more than 125 commercial nuclear power plants.

The suite of Studsvik Scandpower reactor analysis codes has been developed continually over the last 20 years—currently representing more than 250 person-years of development effort. These codes have been applied by utilities and fuel vendors to more than 500 cycles of power plant operations. The resulting validation against operational data from nuclear plants around the world has established the Studsvik Scandpower code system as the premier LWR analysis package available in the world today. The codes currently used in the Studsvik Scandpower Core Monitoring System (CMS) include CASMO and HELIOS, SIMULATE-3, SIMULATE-3K, SIMULATE-3R, GARDEL, and SNF.

LOCAL AREA INFORMATION

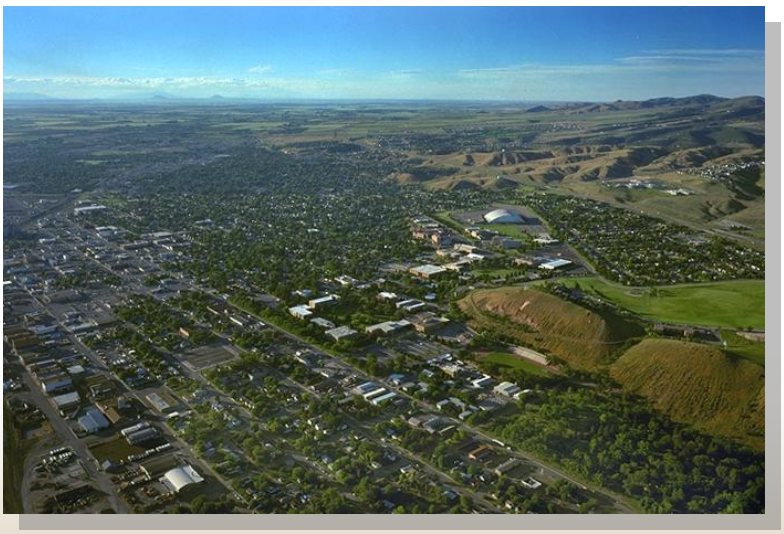
- **About Pocatello**
- **Local Area Attractions**
- **Restaurants**
- **Maps**

About Pocatello

Pocatello, also known as the “Gateway to the Northwest,” was founded in 1889. It is also nicknamed the “Gate City.” It was named after a Chief of the Shoshone-Bannock Native American Tribes whose land was donated to the city for the railroad. Pioneers, settlers, and gold miners passed through the Portneuf Gap just south of Pocatello while traveling on the Oregon Trail, the wagon trail to the Far West. Inevitably, stage lines and the railroad quickly followed, and the city of Pocatello began to flourish.

Pocatello has been listed in the Financial Times as one of the top ten Micro Cities of the Future. It was also listed as the 2nd Best Economic Potential and 5th in Best Quality of Life. Pocatello has also been recognized by Forbes as a “Best Small Place for Business” six consecutive years. Primary Relocation and World ERC ranked Pocatello in the top ten small places to raise a family.

Pocatello’s population is 52,000 and its suburb of Chubbuck, which is separated only by a city-limits sign, has a population of 10,000.



Local Attractions

Mystique Theater

Masters of Magic – A World Class Gala

Showcases 3 internationally acclaimed magicians from Las Vegas and Hollywood's Magic Castle
158 East Chubbuck Road
Prices range from$10.60 to $47.70
Box Office M-F (11-5): (208) 238-8001

Idaho Museum of Natural History
Located on the campus of Idaho State University
Adults $5.00, Children $2.00
Tuesday thru Saturday 10-5

Local Golf Courses

Highland Golf Course (208) 237-9922
Riverside Golf Course (208) 232-9515

Restaurants

Within Walking Distance From Hotel

Applebee's Neighborhood Grill
1411 Bench Road (208) 637-0135

Chang's Garden
1000 Pocatello Creek Road (208) 234-1475

Pantry Coffee Shop
1555 Pocatello Creek Road (208) 233-2200

Perkins Restaurant and Bakery
1600 Pocatello Creek Road (208) 233-0006

Pier 49 San Francisco Pizza
1000 Pocatello Creek Road (208) 234-1414

Sandpiper Restaurant & Lounge
1400 Bench Road (208) 233-1000

Sizzler
1000 Pocatello Creek Road (208) 233-1547

Eating on Campus

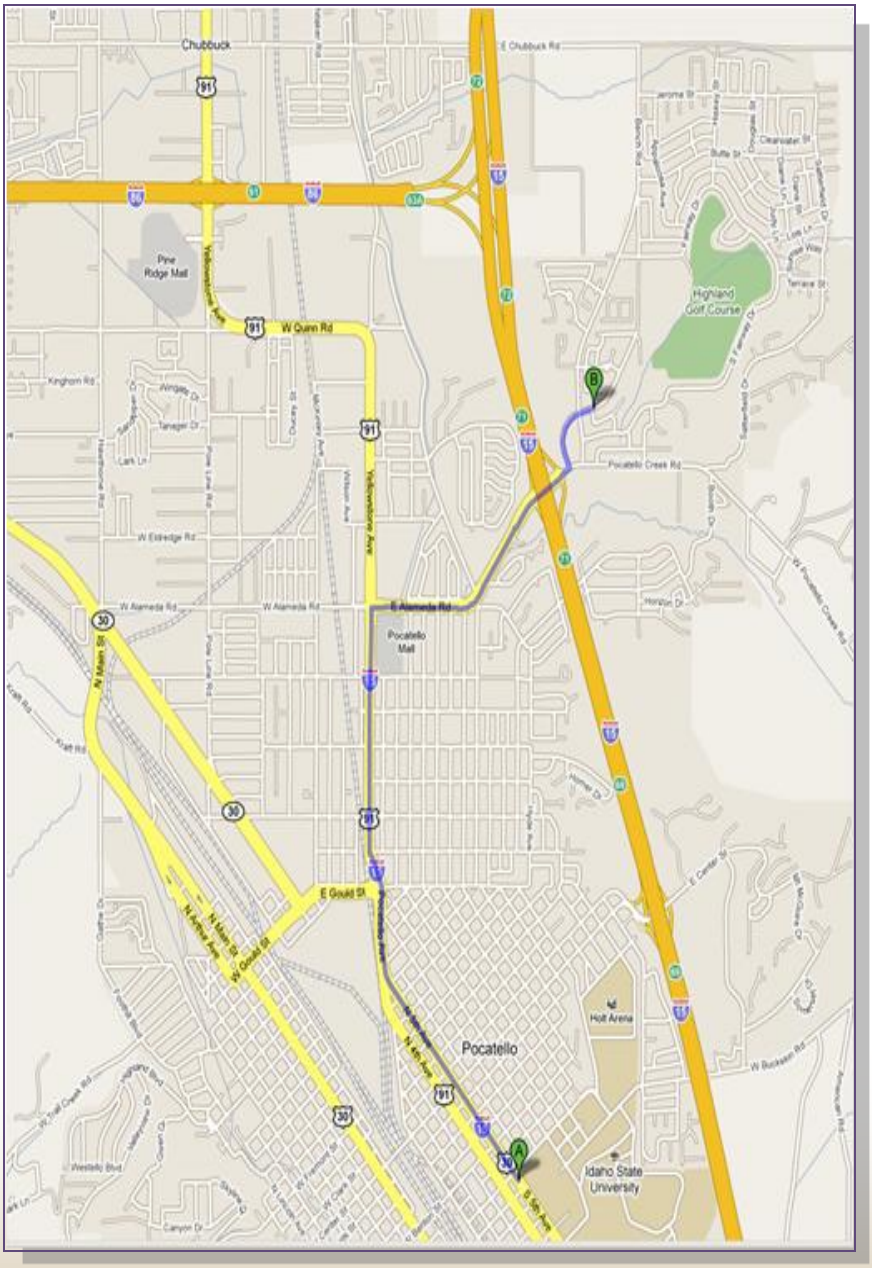
Chartwells
Pond Student Union Building Food Court, M – Th
7:00 – 17:00, F 7:00 – 14:00

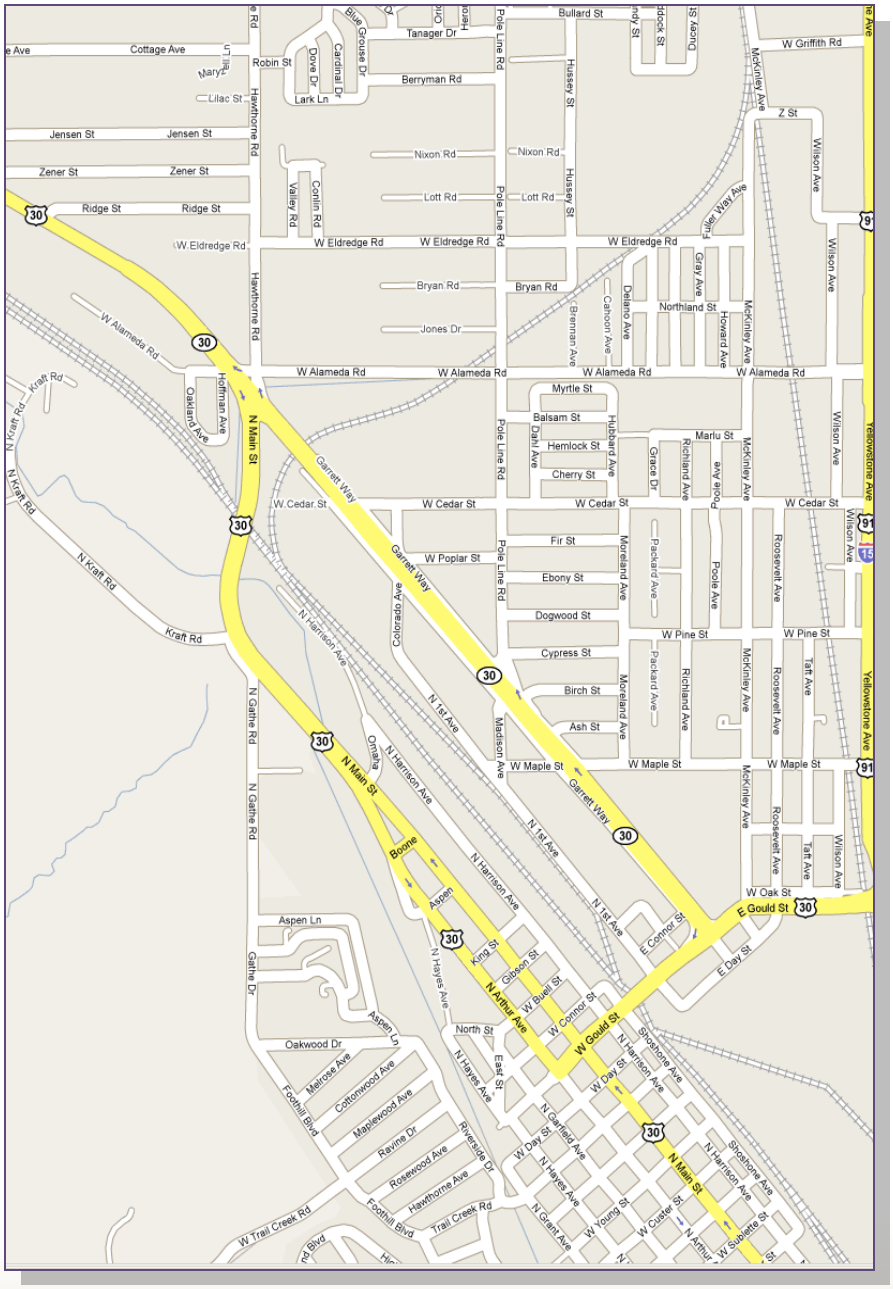
Coyote Jack's Grill
Rendezvous Complex, M – F 8:30 – 19:30

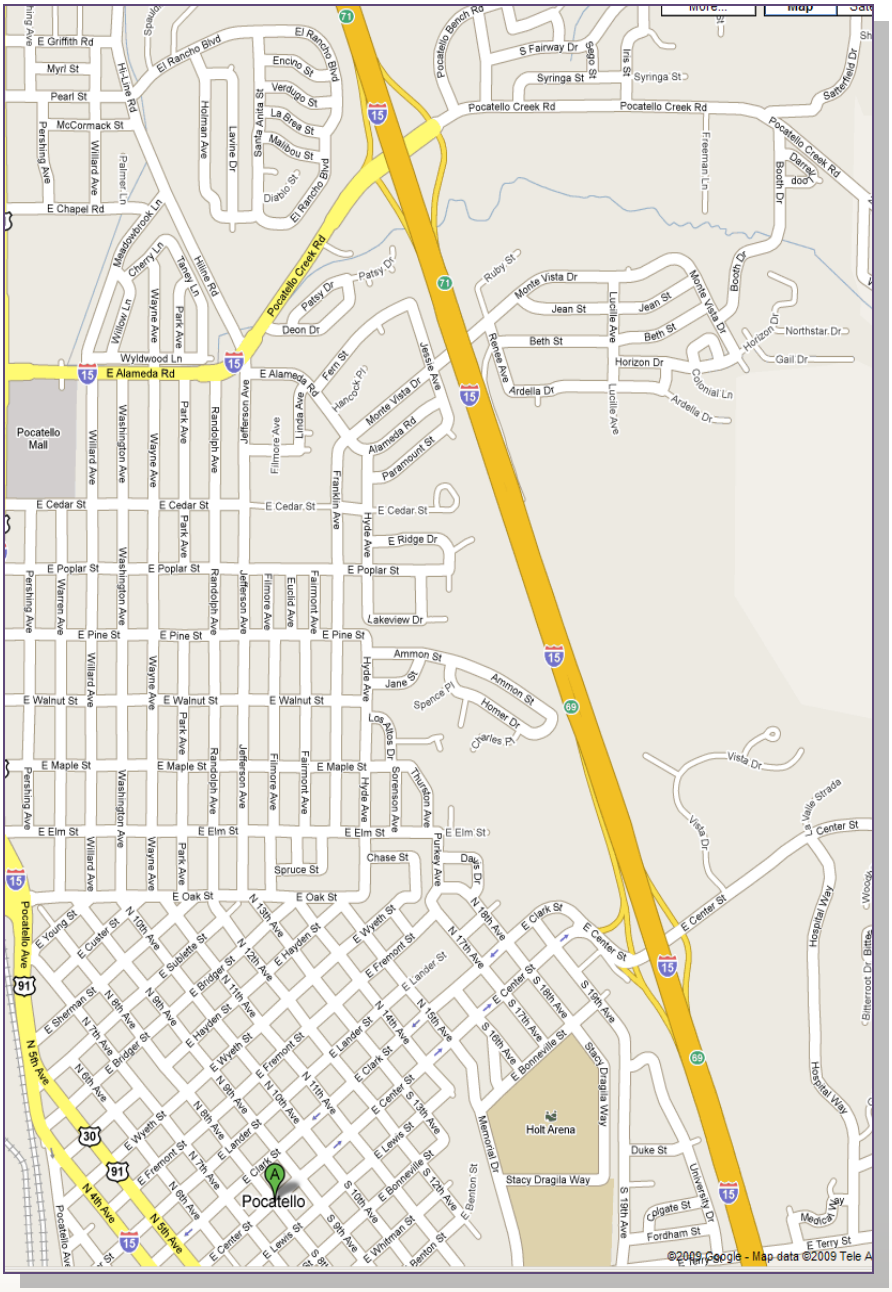
Food Court M – F 10:30 – 14:00

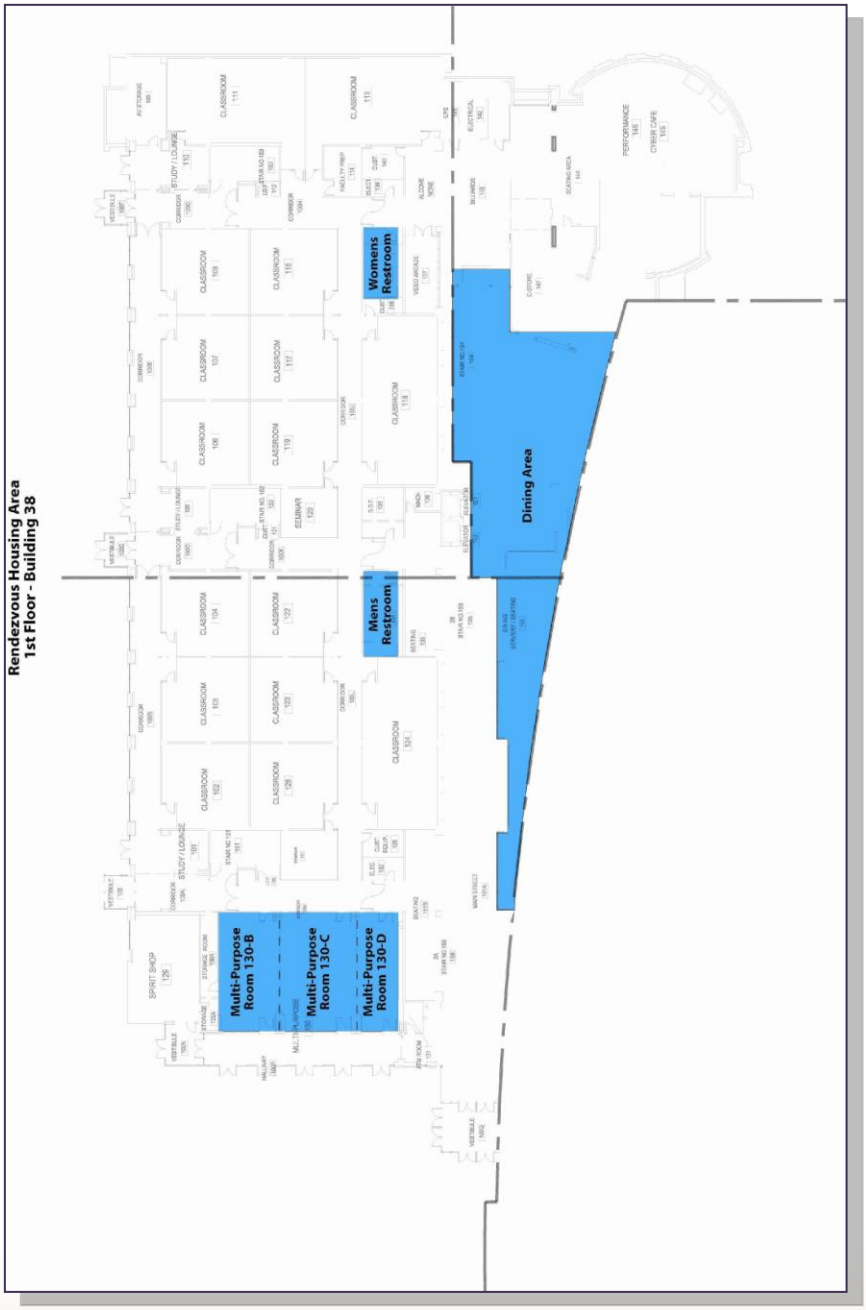
Convenience Store M – F 7:30 – 23:00

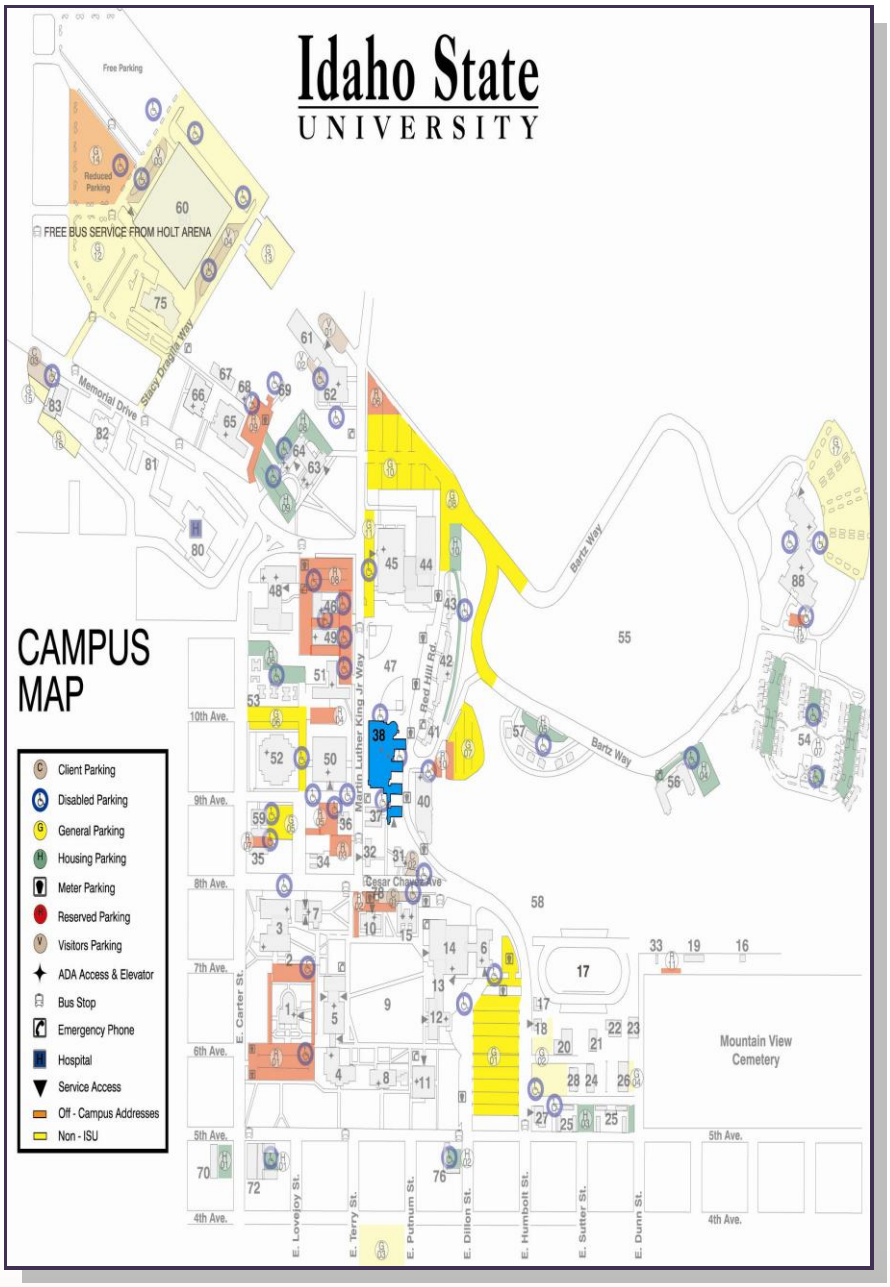
Driving Route Between AmeriTel Inn and Idaho State University




Cottage Ave
Jensen St
Zener St
Ridge St
W Eldredge Rd
W Alameda Rd
Kraft Rd
N Kraft Rd
Hoffman Ave
Oakland Ave
Hawthorne Rd
Robin St
Lilac St
Dove Dr
Cardinal Dr
Blue Grouse Dr
Lark Ln
Tanager Dr
Berryman Rd
Nixon Rd
Lott Rd
Bryan Rd
Jones Dr
Valley Rd
Conlin Rd
Pole Line Rd
Bullard St
Hussey St
Dusky St
Fuller Way Ave
Northland St
Calhoun Ave
Delano Ave
Branman Ave
Gray Ave
Howard Ave
McKinley Ave
W Griffith Rd
Z St
Wilson Ave
Yellowstone Ave
Myrtle St
Balsam St
Hemlock St
Cherry St
Dahl Ave
Hubbard Ave
Marlu St
Grace Dr
Richland Ave
Poole Ave
W Cedar St
Fir St
W Poplar St
Ebony St
Dogwood St
Moreland Ave
Packard Ave
Roosevelt Ave
W Pine St
Cypress St
Birch St
Ash St
Taft Ave
W Maple St
W Oak St
E Gould St
E Connor St
E Day St
Garrett Way
N Main St
N Gaithe Rd
Gaithe Dr
Aspen Ln
Oakwood Dr
Metrose Ave
Cottonwood Ave
Maplewood Ave
Ravine Dr
Rosewood Ave
Hawthorne Ave
Foothill Blvd
W Trail Creek Rd
Trail Creek Rd
Colorado Ave
N Harrison Ave
N 1st Ave
Madison Ave
Omaha
Boone
Aspen
King St
Gibson St
W Buell St
W Connor St
N Arthur Ave
W Gould St
North St
East St
N Hayes Ave
Riverside Dr
N Garfield Ave
W Day St
N Grant Ave
W Young St
W Custer St
W Sublette St
Shoshone Ave
30
91
15


Pocatello
Pocatello Mall
Holt Arena
E Alameda Rd
Pocatello Creek Rd
Stacy Dragila Way
Pocatello Ave
N 5th Ave
N 4th Ave
©2009 Google - Map data ©2009 Tele A


Rendezvous Housing Area
1st Floor - Building 38
Multi-Purpose Room 130-B
Multi-Purpose Room 130-C
Multi-Purpose Room 130-D
Mens Restroom
Womens Restroom
Dining Area


Idaho State
UNIVERSITY
CAMPUS MAP
Client Parking
Disabled Parking
General Parking
Housing Parking
Meter Parking
Reserved Parking
Visitors Parking
ADA Access & Elevator
Bus Stop
Emergency Phone
Hospital
Service Access
Off - Campus Addresses
Non - ISU
FREE BUS SERVICE FROM HOLT ARENA
Free Parking
Reduced Parking
Memorial Drive
Stacy Dragila Way
Martin Luther King Jr Way
Red Hill Rd.
Bartz Way
Cesar Chavez Ave
E. Carter St.
E. Loveloy St.
E. Terry St.
E. Putnum St.
E. Dillon St.
E. Humbolt St.
E. Sutter St.
E. Dunn St.
10th Ave.
9th Ave.
8th Ave.
7th Ave.
6th Ave.
5th Ave.
4th Ave.
Mountain View Cemetery

Emergency Contact Information

In Case of medical emergency dial: 911

Workshop Organization Committee:
Blair Briggs, INL – 208.757.8201
Dave Nigg, INL – 208.521.4722
George Imel, ISU – 208.705.2344
Karen Leibert, ISU – 208.520.9158
Lori Braase, INL – 208.569.2631

Local Hospital:
Portneuf Medical Center
651 Memorial Drive
Pocatello, ID 83201
208.239.1800

Located just off the ISU campus

OECD Planning Committee

Working Party on Nuclear Criticality Safety

Name	Email	Role
Rouyer, Véronique	Véronique.Rouyer@irsn.fr	WPNCS Chair
Rugama, Yolanda	Yolanda.Rugama@oecd.org	NEA Secretariat
All members of the WPNCS		

Workshop Organization

Name	Email	Role
Braase, Lori	lori.braase@inl.gov	INL Logistics
Briggs, J. Blair	j.briggs@inl.gov	INL Program
Imel, George	gimel@isu.edu	ISU Co-Chair
Leibert, Karen	leibkare@isu.edu	ISU Logistics Coordinator
Nigg, Dave	david.nigg@inl.gov	INL Co-Chair

Workshop Support

Name	Email	Role
Bragassa, Jodi	jodi.bragassa@inl.gov	Nuclear Program Planning
Cook, Bruce	bruce.cook@inl.gov	Finance
Grgich, Jodi	jodi.grgich@inl.gov	Program
Reagan, Desiree	desiree.reagan@inl.gov	Web Master
Sanders, Janaye	janaye.sanders@inl.gov	Finance
Townsend, Julia	julia.townsend@inl.gov	Security
White, Christine	christine.white@inl.gov	Graphics

Notes

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OECD-NEA Workshop on Future Criticality Safety Research Needs

September, 2009