Canadian Nuclear Society
Société Nucléaire Canadienne

in cooperation with the
International Atomic Energy Agency

14th International Conference on CANDU Fuel
"Powering a Greener Future with Advanced Fuel Cycles"
Hilton Meadowvale
Mississauga, Ontario, Canada
2019 July 21-24

Conference Program

The 14th International Conference on CANDU Fuel will be chaired by SNC-Lavalin Nuclear. The mandate of this conference is to provide the best forum for exchange of information by Canadian and international experts in the area of nuclear fuel, involved in design, fabrication, operation, R&D, modelling, safety analysis and regulation.

Sponsors

We would like to acknowledge and thank the organizations listed below which have made outstanding contributions to the success of the CNS 14th International Conference on CANDU Fuel and to the enjoyment of the attendees and their guests through their generous sponsorship.

SNC-Lavalin Nuclear

Canadian Nuclear Safety Commission (CNSC)

International Atomic Energy Agency (IAEA)

Kinectrics Inc.

Stern Laboratories Inc.

Canadian Nuclear Society (CNS)

Program Sessions

- International Experience with CANDU Fuel (Plenary Sessions #1 and #2)
- Fuel Design and Development (Technical Session #M1)
- Fuel Modelling and Computer Code Development (Technical Session #M2)
- Accident Tolerant Fuel and Novel Materials for CANDU Fuel (Plenary Session #3 and Technical Session #W2)
- Spent Fuel Management (Technical Session #T1)
- Fuel Fabrication (Technical Session #T2)
- Fuel Performance and Operating Experience (Technical Session #T3)
- Fuel Safety and Operational Margin Improvement (Technical Session #T4)
- Fuel Bundle Thermalhydraulics (Technical Session #W1)
- Special Interest Topics (Plenary Sessions #4)

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The abstracts are only included inside the digital version of Conference Program.

Conference Officials

Organizing Committee Members

Dr. Masoud Shams, Conference Chair (SNC-Lavalin Nuclear)
Mr. Scott Froebe, Treasurer (SNC-Lavalin Nuclear)
Dr. Ben Rouben, Advisor (12 & 1 Consulting)
Dr. Paul Chan, Advisor (RMC)
Dr. Ki-Seob Sim, Advisor (IAEA)

Technical Program Committee Members (alphabetical)

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Girma Chassie (SNC-Lavalin Nuclear)
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Wenjie Zhu (SNC-Lavalin Nuclear)
Zhen Xu (SNC-Lavalin Nuclear)

Conference Information

Special Events

Sunday Opening Reception & Registration, Exhibition opens
2019 July 21
7:00 PM – 9:30 PM, Room - Greenwich and Compass

Group Picture: Monday 12:00 PM - 12:15 PM.

Monday Luncheon
2019 July 22
12:15 PM, Hazel McCallion C&D

Monday Technical Tour
2019 July 22
3:30 PM (departure)
Location: Sheridan Park Reactor Control Room Mock-Up and Sheridan Park Engineering Lab.

Tuesday Luncheon
2019 July 23
12:00 (Noon), Hazel McCallion C&D

Tuesday Conference Banquet and Entertainment
2019 July 23
6:00 PM (bus departure, dinner and show starts at 7:30 pm)
Location: *Medieval Times Dinner and Tournament*

VIP Queen Royal Package: Priority Castle Access, VIP First Row All Section Seating or Second Row in Center Section, Priority Seating Access, VIP Lanyard, and Framed Entrance Group Photo.

If you prefer you can drive directly to the Castle, city parking is $13.

Wednesday Luncheon
2019 July 24
1:05 PM, Hazel McCallion C&D

Local Attractions and Activities
Several popular attractions are listed at the following website:
https://cns-snc.ca/events/fuel2019/

If you would like to take part in a spousal program, the organization committee will be happy to arrange one.

There are Niagara Falls and Winery tours that can be arranged with an additional fee.

Conference Registration
Registration is required for all attendees and presenters. Badges are required for admission to all events. The Full Conference Registration fee includes one (1) copy of the Conference Proceedings, conference program, conference souvenir, conference lunches and coffee breaks and one (1) ticket each to the conference banquet and the conference technical tour.

Conference Proceedings
The Conference Proceedings will be available after the Conference.

NOTES:
Additional tickets for the conference banquet can be purchased in advance on-line up to 3 days before the event.

Hot breakfast is served each day for all registrants from 7:00 am to 8:30 am at Hazel McCallion C&D room. Breakfast Speaker Tables with session chairs are arranged.

Parking is free for all conference registrants.

Session Format:
Plenary Session: 20-minute presentation and 10 minutes for questions.
Technical Sessions: 20-minute presentation and 5 minutes for questions.
All presentations must be loaded during breakfast and lunch periods. Please bring your power point presentations using a memory card to the conference.

Conference Schedule – At a Glance

Sunday, 2019 July 21	**(7:00 PM – 9:30 PM)** Registration and Reception, Exhibitions open

Monday, 2019 July 22

Registration and Breakfast	**(7:00 AM – 8:15 AM)** Room - Hazel McCallion C&D
Opening Ceremony	**(8:15 AM – 8:30 AM)** Welcoming Addresses Room - Hazel McCallion A&B
Plenary Session #1	**(8:30 AM – 10:00 AM)** International Experience with CANDU Fuel I Room - Hazel McCallion A&B
Coffee Break	**(10:00-10:30 AM)**
Plenary Session #2	**(10:30 AM – 12:00 PM)** International Experience with CANDU Fuel II Room - Hazel McCallion A&B
Group Picture	**(12:00 PM – 12:15 PM)**
Lunch	**(12:15 PM – 1:30 PM)** Room - Hazel McCallion C&D
Technical Session #M1	**(1:30 PM – 3:30 PM)** Fuel Design and Development Room - Hazel McCallion A
Technical Session #M2	Fuel Modelling and Computer Code Development Room - Hazel McCallion B
Coffee Break	**(3:10 PM)**
Technical Tour	**(3:30 PM – 6:30 PM)** Sheridan Park Reactor Control Room Mock-Up & SP Engineering Lab

Tuesday, 2019 July 23

Breakfast	**(7:00 AM – 8:30 AM)** Room - Hazel McCallion C&D
Plenary Session #3	**(8:30 AM – 10:00 AM)** Accident Tolerant Fuel and Novel Materials for CANDU Fuel Room - Hazel McCallion A&B
Coffee Break	**(10:00 AM)**

Conference Schedule – At a Glance

Technical Session #T1	**(10:20 AM – 12:00 PM)** Spent Fuel Management Room - Hazel McCallion A
Technical Session #T2	Fuel Fabrication Room - Hazel McCallion B
Lunch and Tributes	**(12:00 PM – 1:30 PM)** Room - Hazel McCallion C&D
Technical Session #T3	**(1:30 PM – 5:10 PM)** Fuel Performance and Operating Experience Room - Hazel McCallion A
Technical Session #T4	Fuel Safety and Operational Margin Improvement Room - Hazel McCallion B
Coffee Break	**(3:10 PM)**
Conference Banquet Dinner	**(6:00 PM – 10:00 PM)** Medieval Times Dinner and Tournament

Wednesday, 2019 July 24

Breakfast	**(7:30 AM – 9:00 AM)** Room - Hazel McCallion C&D
Plenary Session #4	**(9:00 AM – 10:30 AM)** Special Interest Topics Room - Hazel McCallion A&B
Coffee Break	**(10:30 AM)**
Technical Session #W1	**(11:00 AM – 1:05 PM)** Fuel Bundle Thermalhydraulics Room - Hazel McCallion A
Technical Session #W2	Accident Tolerant Fuel and Novel Materials for CANDU Fuel Room - Hazel McCallion B
Lunch	**(1:05 PM – 2:30 PM)** Room - Hazel McCallion C&D
Closing remarks	**2:30 PM**

Schedule of Technical Sessions

Monday, 2019 July 22, 8:15 AM -8:30 AM,
Room - Hazel McCallion A&B
Opening Ceremony

08:15 Opening remarks

Monday, 2019 July 22, 8:30 AM-12:00 PM
Plenary Session #1 & 2, Room - Hazel McCallion A&B
International Experience with CANDU Fuel
Chairs: Vali Tavasoli (CNSC)
Ki-Seob Sim (IAEA)

08:30 IAEA Activities on Nuclear Fuel Engineering for 2019-2021 and Overview of Coordinated Research Project on Reliability of High Power, Extended Burnup and Advanced PHWR Fuels, *Ki Seob Sim (IAEA)*

09:00 Experiences on Indigenous PHWR Fuel Manufacturing, Technology Developments and gearing up for forthcoming expansion of Nuclear Power Program in India, *Dinesh Srivastava* (*CEO of Nuclear Fuel Complex, India)*

9:30 CANDU Fuel Evolution, Past, Current and Future Challenges, *Vali Tavasoli (Director of Physics and Fuel, Canadian Nuclear Safety Commission, CNSC)*

10:00 Coffee Break

10:30 IAEA TECDOC on Pressurized Heavy Water Reactor fuel safety criteria for operational states and accident conditions, *Michel Couture (CNSC), Ho Chun Suk (CNSC) and Ki Seob Sim (IAEA)*

11:00 PHWR Fuel Activities in Argentina in the 2016-2018 period, *Luis Alvarez, A. Bussolini, J. P. Medina, P. Trípodi (Commission of National Atomic Energy, CNEA, Argentina)*

11:30 Proposed Fuel Modeling Requirements in LOCA Safety Analysis in Korea, *Joosuk Lee and Young-Seok Bang (Korea Institute of Nuclear Safety, KINS, Korea)*

Monday, 2019 July 22, 12:00 PM-1:30 PM
Lunch, Hazel McCallion A&B

12:00 - 12:15 Group Picture, location TBD

Monday, 2019 July 22, 1:30 PM-3:30 PM
Technical Session #M1, Room - Hazel McCallion A
Fuel Design and Development
Chairs: Steve Goodchild (OPG)
Ross Lewis (BP)

13:30 Mid-Plane Bearing Pad Height Adjustment Impact Assessment and Manufacturing Experience, *Erin Middaugh and Todd Daniels (Ontario Power Generation, OPG)*

13:55 Fuel Fitness for Service Pressure Limit for PHTS Static Pressure Test, *Jonathan Judah, Richard Scrannage, and Steve Goodchild (OPG), Farzin Abbasian, Gordon Hadaller and Campigotto Mario (Stern Lab)*

14:20 Royal Military College of Canada Contribution to IAEA CRP# T12027 "Use of Neutron-Absorbers to Improve CANDU Reactor Operating Margins, *Paul Chan (Royal Military College of Canada, RMC)*

14:45 Impact of Design Change from 37R to 37M on CANDU Fuel Bundle Vibration Characteristics, *Gong Cheng, Krishna Chakraborty, Girma Chassie, Zhen Xu, Akash Gill, Masoud Shams (SNC-Lavalin Nuclear), Guodong Ye and Mingjun Chen (CNNO, China)*

15:10 Coffee Break

Monday, 2019 July 22, 1:30 PM-3:30 PM
Technical Session #M2, Room - Hazel McCallion B
Fuel Modelling and Computer Code Development
Chairs: Paul Gillespie (Kinectrics)
Mukesh Tayal (AECL - Retired)

13:30 ELESTRES 1.3 Computer Code for Modelling the Thermal, Mechanical and Micro-Structural Behaviour of CANDU Fuel Element under Normal Operating Conditions, *Girma Chassie and Masoud Shams (SNC-Lavalin Nuclear)*

13:55 Development of ELOCA 2.3 Computer Code for Modelling the Performance of CANDU Fuel Element under Postulated Accident Conditions, *Girma Chassie, Alan Jiang, Akash Gill, Hazen Fan and Masoud Shams (SNC-Lavalin Nuclear)*

14:20 SOURCE IST 2.0 Benchmarking Against Production Data, *A.I. Popescu, A. Cziraky, J. Sun, H. Hasanein, T. Danniels, H. Albasha and P. Gillespie (Kinectrics)*

Schedule of Technical Sessions

14:45 Applications of Mesoscale Models for Nuclear Fuel and Structural Components, *Michael Welland, Andrew Prudil, Evan Thomas and Eric Tenuta, (CNL)*

15:10 Coffee Break

Monday, 2019 July 22, 3:30 PM – 6:30 PM (latest)
Technical Tour

15:30 Shuttle buses depart from the hotel
(First come, first served – please register at the desk in advance)

16:15 Technical Tour of Sheridan Park: Reactor Control Room Mock-Up and SP Engineering Lab.

Two groups, maximum 25 persons in each group. You can drive to the site by yourself if you prefer.

Reactor Control Room Mock-Up (duration ~30 min)

- An overview of the systems and capabilities of SNC-Lavalin's digital control room mock up
- Discussion of how the mock-up is used for human factors training in a simulated real-world reactor control room environment
- An overview of how the mock-up can be configured to re-produce the control room environment of various reactor types

Sheridan Park Engineering Lab (duration ~45 min)

- Overview of SNC-Lavalin's manufacturing, assembly and testing capabilities
- View and discuss the Zone 3 Tooling Maintenance facility
- Tour of the machine shop, fabrication & welding, metrology lab, NDE & testing lab
- Note that safety shoes are required for this tour. A limited supply of safety shoes is available.

Tuesday, 2019 July 23, 8:30 AM-10:00 AM
Plenary Session #3, Room - Hazel McCallion A&B
Accident Tolerant Fuel and Novel Materials for CANDU Fuel
Chairs: Mikael Jolkkonen (Royal Inst. of Tech Sweden)
Richard Holt (Queens University)

08:30 Accident Tolerant Fuels: on Potential Options for CANDU in Light of International Activities for LWRs, *Ho Chun Suk, Michel Couture and Wade Grant (Canadian Nuclear Safety Commission, CNSC).*

09:00 Towards industrial scale manufacture of Uranuim Nitrides (UN) fuel for CANDU reactors, *Janne Wallenius (CEO of LeadCold Reactors, Sweden), Yulia Mishchenko (KTH, Sweden), Mikael Jolkkonen (KTH, Sweden) and Daniel Laurin (Promation Nuclear)*

09:30 Accident Tolerant Fuel for Application in CANDU and Small Modular Reactors, *Jerzy Szpunar (University of Saskatchewan)*

10:00 Coffee Break

Tuesday, 2019 July 23, 10:20 AM-12:00 PM
Technical Session #T1, Room - Hazel McCallion A
Spent Fuel Management
Chairs: Mihaela Ion (*NWMO*)
Seongki Lee (KEPCO)

10:20 Elemental Composition of Unirradiated CANDU Fuel, *Kelly Liberda, Helen Leung, Paul Gierszewski and Liana Orlovskaya (Nuclear Waste Management Organization, NWMO)*

10:45 Fission Product Core Inventory Estimation using ORIGEN-S Code, *W.J. Zhu, L.Y. Huang and H.Z. Fan (SNC-Lavalin Nuclear)*

11:10 Natural Convection in Closed Irradiated Fuel Bay (IFB) racks after a Loss of Coolant Event, *Derek Logtenberg (CNSC), P Chan and E Corcoran (RMC)*

11:35 Planning of Spent Fuel Integrity Evaluation Technology for Dry Storage, *Seongki Lee and Manseok Do (KEPCO Nuclear Fuel, Korea)*

Tuesday, 2019 July 23, 10:20 AM-12:00 PM
Technical Session #T2, Room - Hazel McCallion B
Fuel Fabrication
Chairs: Dale Clark (CAMECO)
Jordon Brown (BWXT)

10:20 Strategies in Mitigating Stress Corrosion Cracking of Zircaloy-4 Fuel Sheathing: A Case for Polysiloxane Coating, *M. Farahani, P.K. Chan, E.C. Corcoran, R. Hameed and T. Torkelson, (RMC)*

10:45 Best-estimate Plus Uncertainty Analysis of CANDU Fuel Reliability using Manufacturing Data and Simulated Core Data, *Jason Song and Paul Chan (RMC), Mahesh Pandey (Cameco)*

Schedule of Technical Sessions

11:10 RMCC SLOWPOKE Reactor Core Fabrication and Refueling, *Justin Spencer, P Chan, B Lewis, C. Thiriet, A Bergeron, S. Livingstone and S. Yue (CNL)*

11:35 The Microstructure and Thermal Conductivity of Spark Plasma Sintered ThO2, *Linu Malakkal, A Prasad, J. Ranasinghe, E. Jossou, B. Szpunar, L. Bichler and Jerzy Szpunar (University of Saskatchewan)*

Tuesday, 2019 July 23, 12:00 PM-1:30 PM
Lunch, Hazel McCallion C&D

12:45 Luncheon – Tributes and Announcements

Tuesday, 2019 July 23, 1:30 PM-5:10 PM
Technical Session #T3, Room - Hazel McCallion A
Fuel Performance and Operating Experience
Chairs: Al Manzer (AECL-Retired)
Steve Palleck (AECL - Retired)

13:30 Developments in Post-Irradiation Examination Hot Cell Equipment on CANDU fuel at Chalk River Laboratories (CNL), *Jeffrey Armstrong, S. Audette, C. Buchanan, J DeVreede, J. Olfert and P. Sullivan (CNL)*

13:55 Irradiation Performance of PHWR Fuel at Extended Burnup, *Prerna Mishra, B.N Rath, Ashwini Kumar, V.P Jathar, U. Kumar, H.N. Singh, P. K. Shah, R.S. Sriwastaw, J.S. Dubey, G.K. Mallik, J.L. Singh, P.G. Jaison and S. Kannan (Bhabha Atomic Research Centre, Trombay, India)*

14:20 Guidelines for Achieving Excellence in CANDU Fuel Performance, *Ben Wong, Andrew Fitchett , Ross Rock, and Paul Gillespie (Kinectrics), Ross Lewis and Philipe Paquette (BP), Todd Daniels (OPG)*

14:55 Unit 2 RTS PHTS Hot-Conditioning: Results of Investigations to Assess Fitness for Service Expectations for OPG 37M Fuel Bundles, *Jonathan Judah, Richard Scrannage, and Steve Goodchild (OPG), Farzin Abbasian, Gordon Hadaller and Mario Campigotto (Stern Lab)*

15:10 Coffee Break

15:30 Analysis of the PIE Results to Determine/Confirm Primary Root Cause for Failed Pickering B Fuel Element, *Krishna Chakraborty, Akash Gill, Zhen Xu (SNC-Lavalin Nuclear), Todd Daniels, Erin Middaugh and Mariana Dobrean (OPG)*

15:55 Novel Tool and Methodology used for Prioritizing Suspect Defected Fuel at Pickering Nuclear Generating Station, *Shahab Dabiran (OPG)*

16:20 CANDU Failed Fuel Detection and Location System, *Jeffrey Arndt and Michael Heibel (Global Technology Office, Westinghouse, USA)*

16:45 CANDU Fuel Engineer's Manual - Current Development Status and Future Plans, *P. Gillespie (Kinectrics)*

Tuesday, 2019 July 23, 1:30 PM-5:10 PM
Technical Session #T4, Room - Hazel McCallion B
Fuel Safety and Operational Margin Improvement
Chairs: Relu Istrate (CERNAVODA)
Sungmin Kim (KHNP)

13:30 Safety Analysis Results of the 37M During Transition Core in Wolsong, *Sungmin Kim (Korea Hydro & Nuclear Power, KHNP).*

13:55 Stylized Study using FEAST Computer Code on Fuel Bundle End Plate Deformation Subject to Sustained High Temperature, *Z. Xu, L.Y. Huang and H.Z. Fan (SNC-Lavalin Nuclear)*

14:20 Fuel Behaviour Modelled for Severe Accident Progression and Consequence Assessment using MAAP-CANDU Computer Code, H.Z. Fan, T. Nguyen and L. Comanescu (*SNC-Lavalin Nuclear)*

14:45 The Use of Burnable Neutron Absorbers to Mitigate the Effect of Coolant Voiding on CANDU 37-Element Fuel, *M. Couture, P Chan and H Bonin (RMC)*

15:10 Coffee Break

15:30 SOURCE IST 2.0 Validation Against Fast and Slow Power Ramp Tests, *Adam Cziraky, Adrian Popescu and John Sun (Kinectric).*

15:55 Bundle Slumping Temperature Observed in Experiments of Bundle Deformation Subject to Sustained High Temperature, *H.Z. Fan, W.J. Zhu, L.Y. Huang and V. Lau (SNC-Lavalin Nuclear)*

16:20 Finite Element Analysis of Pressure Tube Creep and Bundle Deformation of a CANDU Fuel Bundle, *Kyuhwan Lee, Diane Wowk and Paul Chan (RMC)*

16:45 Sensitivity Analyses for Fuel Behaviour During a Large Break in the Primary Heat Transport Circuit, *Catalin Zalog and Emanoil-Relu Istrate (CERNAVODA NPP, Romania)*

Schedule of Technical Sessions

Tuesday, 2019 July 23, 6:00 PM - 10:00 PM
Banquet Dinner and Entertainment
Location: Medieval Times Dinner and Tournament

The show length is approximately two hours. Doors open 75 minutes before the show at 7:30 PM. The visitor's experience begins as guests enter the entertainment complex inspired by an 11th century, European-style Castle. There is a Museum of Torture, which features reproductions of torture instrument used during the Middle Ages. Serfs and wenches attend to every need as guests enjoy a four-course feast in true medieval, pre-silverware fashion. The pace quickens as the Tournament of Games begins. The valiant knights on horseback face their competition in games of skill, such as the ring pierce, flag toss, and javelin throw. The new story composed by Dr. Daniel May who directed and recorded the composition in Kiev with the National Symphony Orchestra of Ukraine.

Wednesday, 2019 July 24, 9:00 AM-10:30 AM
Plenary Session #4, Room - Hazel McCallion A&B
Special Interest Topics
Chairs: Janne Wallenius (LeadCold Reactors, Sweden)
Jeffrey Arndt (Westinghouse, USA)

09:00 Fuel Recycling for Canada's Short- and Long-Term Fuel Independence and Security, *Peter Ottensmeyer (University of Toronto)*

09:30 Thermalhydraulics Role in the Advanced PHWR Fuel Development, *Jun Yang (Canadian Nuclear Labs)*

10:00 CANDU Owners Group Nuclear Safety Research and Development on the CANDU Fuel, *Wei Shen(COG)*

10:30 Coffee Break

Wednesday, 2019 July 24, 11:00 AM-1:05 PM
Technical Session #W1, Room - Hazel McCallion A
Fuel Bundle Thermalhydraulics
Chairs: Yujun Guo (CNSC)
Gord Hadaller (Stern Lab)

11:00 Post-Dryout Heat Transfer Phenomenon and Prediction Methods in CANDU Fuel Bundles, *Yujun Guo and Naj Hammouda (CNSC)*

11:25 Methodology for Investigating Subchannel Coolant Flow in a 43-element Fuel Bundle, *Caitlyn Cavanagh-Dollard, Paul Chan and Diane Work (RMC)*

11:50 The Effect of Azimuthal Heat Conduction on Heater Surface Temperatures under Post Dryout Conditions, *Rick Fortman (Stern Lab)*

12:15 Critical Heat Flux and Post-Dryout Experiments Using the Modified 37-Element Fuel Simulation in Water with a 6.8% Crept Flow Channel, *Gordon Hadaller, Rick Fortman and Jay Snell (Stern Lab)*

12:40 Full-Scale Critical Heat Flux Experiment for Plutonium-based Mixed Oxide Advanced Fuel Bundle Design, *Lanqin Yuan, Jun Yang, Bruce Addicott, Vinson Gauthier and Matthew Dickerson (CNL)*

Wednesday, 2019 July 24, 11:00 AM-1:05 PM
Technical Session #W2, Room - Hazel McCallion B
Accident Tolerant Fuel and Novel Materials for CANDU Fuel
Chairs: Markus Piro (UOIT)
Richard Scrannage (OPG)

11:00 Microstructural and Thermophysical Characterization of Pure and Mo-doped UO2 Pellets, *Murali K. Tummalapalli, L. Malakkal, E. Jossou, J. A Szpunar, A. Prasad and L. Bichler (University of Sask.)*

11:25 First Principles Study on Thermal Conductivity of U3O8, *J. Ranasinghe, L. Malakkal, B. Szpunar, E. Jossou and J.A. Szpunar (University of Saskatchewan)*

11:50 Structural Stability of SiC, *Barbara Szpunar and J.A. Szpunar (University of Saskatchewan)*

12:15 Progress in Experimental and Computational Investigations of Molten Fluoride Salt Thermodynamics for Small Modular Reactors, *Bernard Fitzpatrick, D. Hallatt, K. Lipkina, R Murphy-Snow, P Bajpai, Max Poschmann and Markus Piro (UOIT)*

12:40 Accident-Tolerant Fuel Review and Potential Applications to CANDU Reactors, *Zhen Xu, Masoud Shams (SNC-Lavalin Nuclear)*

Wednesday, 2019 July 24, 1:05 PM-3:00 PM
Lunch, Hazel McCallion C&D

2:30 Closing Remarks

Monday, 2019 July 22, 8:30 AM - 12:00 PM

Plenary Session #1, 8:30 AM - 10:00 AM

International Experience with CANDU Fuel I

IAEA ACTIVITIES ON NUCLEAR FUEL ENGINEERING FOR 2019-2021 AND OVERVIEW OF COORDINATED RESEARCH PROJECT ON RELIABILITY OF HIGH POWER, EXTENDED BURNUP AND ADVANCED PHWR FUELS

Ki Seob Sim (IAEA)

IAEA activities on nuclear fuel engineering planned for 2019-2021 are introduced. An overview of the IAEA's Coordinated Research Project on Reliability of High Power, extended Burnup and Advanced PHWR fuels is provided.

EXPERIENCES ON INDIGENOUS PHWR FUEL MANUFACTURING, TECHNOLOGY DEVELOPMENTS AND GEARING UP FOR FORTHCOMING EXPANSION OF NUCLEAR POWER PROGRAM IN INDIA

Dinesh Srivastava (Nuclear Fuel Complex, India)

Pressurized Heavy Water Reactors (PHWR) are the backbone of Indian nuclear power program at present scenario. Nuclear Fuel Complex (NFC), an industrial organization under Government of India has been playing a pioneering role to focus on front end nuclear fuel cycle, meeting the enhanced target and successfully envisage the future requirement by the nuclear industries in India.
Since its inception in early 1970's, NFC has expertise multi-faceted technical area, become mature, self-sufficient on manufacturing technology of natural UO2 fuel fabrication. After thorough study, NFC overcame technological challenges and successfully demonstrated large scale production of consistent quality fuel pellets from the raw materials of various chemical forms received from indigenous sources. Introduction of many innovative state-of-art technologies for various processes, equipment mechanization and automation lead to ergonomic advantages of the plants. Continual analysis and optimization of the process parameters resulted in improvement of overall process recovery with higher productivity, environmental and radiological safety.
NFC is presently gearing up for enhancing the production activity manifold to meet the forthcoming requirement as per the present expansion plan of nuclear power especially of PHWR program by Government of India.
The paper discusses about cutting edge technology innovation on various processes, equipment modernization, automation etc.

CANDU FUEL EVOLUTION, PAST, CURRENT AND FUTURE CHALLENGES

Vali Tavasoli (CNSC)

CANDU fuel design is approaching its 70th year since its first in-reactor loading at NPD in the 1950s. The design has evolved over the years as the industry gained more experience with CANDU fuel operation, as challenges emerged and as reactors faced Heat Transport System aging issues in later years. Safe storage of used fuel bundles has also evolved

with the introduction of dry storage facilities. Moreover, as part of the Fukushima follow-up actions, CNSC and the nuclear industry reviewed the safety of CANDU reactor operation and improved the mitigation measures to further decrease the possibility of having large-scale fuel failures following a severe accident in CANDU operated facilities. Finally, the research and development aimed at understanding fuel behaviour under transient conditions led to an evolution in safety analysis, resulting in the development of new acceptance criteria to demonstrate effectiveness of systems that contribute to level-two and level-three defence-in-depth, development of better validated and better understood computational toolset, and ability to more accurately model fuel transient behaviour. The AECB and CNSC regulatory expectation was a contributor to this evolution via creation of Generic Action Items and setting of compliance requirements. These are major achievements if one takes into account the size of the CANDU community and the level of international cooperation.

Recognizing past achievements, the presentation will discuss current challenges in the operation of the CANDU reactors as well as in the modelling of fuel behaviour under accident conditions. It will also look at the international activity to improve fuel behaviour in severe accident conditions and possible ways our industry could benefit from it.

To obtain a copy of the abstract's document, please contact us at cnsc.info.ccsn@canada.ca or call 613-995-5894 or 1-800-668-5284 (in Canada). When contacting us, please provide the title and date of the abstract.

Monday, 2019 July 22, 10:30 AM - 12:00 PM

Plenary Session #2, 10:30 AM - 12:00 PM

International Experience with CANDU Fuel II

IAEA TECDOC ON PRESSURIZED HEAVY WATER REACTOR FUEL SAFETY CRITERIA FOR OPERATIONAL STATES AND ACCIDENT CONDITIONS

Michel Couture (CNSC), Ho Chun Suk (CNSC) and Ki Seob Sim (IAEA)

The light water reactor fuel safety criteria have been reviewed several times, mainly by the Organization for Economic Cooperation and Development/Nuclear Energy Agency Fuel Safety Working Group since the late 1990s, in order to reflect the up-to-date knowledge on fuel performance and the influence of design improvements to pellets, cladding materials and fuel assembly structures. Unlike the light water reactor fuel, a systematic review of the pressurized heavy water reactor (PHWR) fuel safety criteria used in IAEA Member States has not been performed, although PHWR countries have invested significant efforts into developing fuel safety criteria and methods that allow the use of fuel computer programs to assess the margins.

This paper provides a summary of review results, to be published as an IAEA TECDOC, on PHWR fuel safety criteria for operational states and accident conditions currently used in IAEA Member States (Argentina, Canada, China, India, South Korea, Pakistan and Romania).

PHWR FUEL ACTIVITIES IN ARGENTINA IN THE 2016-2018 PERIOD

Luis Alvarez, A. Bussolini, J. P. Medina, P. Trípodi (CNEA, Argentina)

The Argentine Nuclear System is composed of four organizations with clearly defined functions: the National Atomic Energy Commission, responsible for development tasks and fuel engineering, CONUAR, responsible for manufacturing the nuclear fuels, NASA which is in charge of the operation of the nuclear power plants and ARN that is the nuclear regulatory authority. In the period 2016-2018, the fuel activities of these organizations were, in some extension, influenced by the activities related to the Refurbishment and Life Extension Project of the Embalse Nuclear Power Plant.

Among these activities one of the most important performed by CONUAR and CNEA was the development of equipments, procedures, engineering and qualification for the production of fuel pellets from UO2 powder obtained by the ADU method, specifically for uranium depleted pellets required for the start up of the plant. Usually the production of fuel pellets in Argentina is made from ex AUC powder.

At the same time, works continued on the mechanical stability of the fuel pellets and the methods to evaluate it. Also on the elimination of Beryllium from the CANDU fuel assembly line and the replacement of graphite coating as a method to prevent the occurrence of defects induced by PCI-SCC.

Regarding the performance of the fuels, the failure rate in the two NPP in operation remained stable, only presenting isolated defects. Although progress was made in the definition of a failure analysis methodology with the participation of NASA, CONUAR and CNEA, the main drawback to establish corrective measures to reduce the failure rate is the difficulty in identifying the root cause.

In the case of the Atucha-2 NPP, an alternative fuel design was developed and verified, including out of pile long lasting endurance tests. This new design may also allow, in the future, the use of slightly enriched uranium in Atucha-2 as it is in Atucha-1. At present, verifications in accident conditions are being performed in a dedicated Freon Loop built for this purpose.

PROPOSED FUEL MODELING REQUIREMENTS IN LOCA SAFETY ANALYSIS IN KOREA

Joosuk Lee and Young-Seok Bang (Korea Institute of Nuclear Safety, KINS, Korea)

The current discharge burnup of PWR fuel in Korea has reached almost twice as great as expected when the fuel was used firstly in 1979. Accordingly, fuel has experienced significant embrittlement even in use of advanced fuel materials. Under this circumstance, revision of ECCS acceptance criteria in Korea is being prepared. The proposed criteria have fuel modeling requirements such as the consideration of inner surface cladding oxidation, factorization of thermal properties of ZrO_2 and crud layer, and consideration of fuel relocation, dispersal and coolable geometry change. In this paper, reason for the selection of these requirements and their impacts on LOCA safety analysis are described.

Monday, 2019 July 22, 1:30 PM-3:30 PM

Technical Session #M1

Fuel Design and Development

MID-PLANE BEARING PAD HEIGHT ADJUSTMENT IMPACT ASSESSMENT AND MANUFACTURING EXPERIENCE

Erin Middaugh and Todd Daniels (OPG)

The fuel bundle bearing pads are designed to allow the fuel bundle to slide inside the fuel channel and ensure that local fuel temperatures do not compromise the pressure tube. Through experimentation, it was found that adjusting the minimum mid-plane bearing pad height improved critical heat flux along with the safety analysis margins for design basis accidents. The impact on fuel safety, fuel design, fuel performance or operations was assessed and no adverse impact was found as a result of this change. Manufacturing changes were implemented and adjusted fuel bundles were successfully manufactured. All adjusted bundles passed the bent tube gauge test which ensures the bundles will pass through the rolled joint region of the pressure tube. The midplane bearing pad was successfully adjusted to match the height of the outboard bearing pads.

FUEL FITNESS FOR SERVICE PRESSURE LIMIT FOR PHTS STATIC PRESSURE TEST

Jonathan Judah, Richard Scrannage, Steve Goodchild (OPG), Farzin Abbasian, Gordon Hadaller and Campigotto Mario (Stern Lab)

The reference plan for the Return-to-Service of Darlington Unit 2 after refurbishment schedules a pressure test of the Primary Heat Transport System (PHTS) main circuit after manual loading of the first charge of fresh fuel bundles, and prior to initial Hot Conditioning and Approach to Critical. The required pressure for the Darlington PHTS pressure test exceeds the existing 37M fuel design limit of 12 MPa.

This paper provides a review and assessment of industry operating experience, fuel manufacturer's data, contracted vendor assessments and recent out-reactor testing commissioned by OPG. For the return to service fresh fuel load in Unit 2, a fuel Fit-for-Service (FFS) PHTS pressure limit of 14.2 MPa for 2 hours at a maximum temperature of 60°C can now be supported with no additional activities required to ensure fuel FFS. A design centered pressure limit of 12.7 MPa can also be supported for fuel elements built to the design limits.

ROYAL MILITARY COLLEGE OF CANADA CONTRIBUTION TO IAEA CRP# T12027 "USE OF NEUTRON-ABSORBERS TO IMPROVE CANDU REACTOR OPERATING MARGINS"

Paul Chan (RMC)

This work proposes a novel modification of the existing CANDU fuel element to improve reactor operating margins and to relax the constraints on fuel management posed by tight operating power limits. The intent is to dope the existing CANDU fuel bundle with minute amounts of Burnable Neutron Absorbers to reduce the magnitude of the refuelling power ripples.

The design work was performed with Winfrith Improved Multigroup Scheme (WIMS) AECL 3.1, Monte Carlo N-Particle code (MCNP 6) and AECL Reactor Fuelling Simulation Program (RFSP), three modeling codes widely used in the Canadian nuclear industry and at the Royal Military College of Canada.

This work is of value since it will facilitate the: 1) mitigation of the xenon-free effects and reactivity peaking due to plutonium in fresh fuel bundles, 2) improvement in operating margins (by removing fueling transients) while at the same time providing more safety margins by mitigating end-flux peaking and in-core LOCA, and 3) collaboration between the university, nuclear industry and IAEA while developing highly qualified personnel for the industry.

Impact of Design Change from 37R to 37M on CANDU Fuel Bundle Vibration Characteristics

Gong Cheng, Krishna Chakraborty, Girma Chassie, Zhen Xu, Akash Gill, Masoud Shams (SNC-Lavalin Nuclear, Guodong Ye and Mingjun Chen (CNNO, China)

Vibration responses of CANDU fuel bundles under flow induced excitations are an important performance factor considered in the fuel design. The design change of CANDU fuel from 37R to 37M can potentially modify the fuel vibration behaviour. In this paper, the dynamic vibration characteristics of fuel bundles are compared between the two fuel designs in a finite element modal analysis. In addition, the effect of the design change on the spacer pad fretting wear is investigated via finite element simulations on the spacer pad work rate in the axial flow and cross-flow regions. The simulation results indicate no appreciable impact of the CANDU fuel design change from 37R to 37M on the fuel bundle dynamic characteristics and spacer pad fretting wear.

Monday, 2019 July 22, 1:30 PM-3:30 PM

Technical Session #M2

Fuel Modelling and Computer Code Development

ELESTRES 1.3 COMPUTER CODE FOR MODELLING THE THERMAL, MECHANICAL AND MICRO-STRUCTURAL BEHAVIOURS OF CANDU FUEL ELEMENT UNDER NORMAL OPERATING CONDITIONS

Girma Chassie and Masoud Shams (SNC-Lavalin Nuclear)

The ELESTRES (ELEment Simulation and sTRESses) computer code models the thermal, mechanical and micro structural behaviour of CANDU® fuel element under normal operating conditions. The main purpose of the code is to calculate fuel temperatures, fission gas release, internal gas pressure, fuel pellet deformation, and fuel sheath strains in fuel element design analysis and assessments. The code is also used to provide initial conditions for evaluating fuel behaviour during high temperature transients.

ELESTRES is the industry standard toolset (IST) code for evaluating fuel performance and simulating the behaviour of CANDU fuels. The ELESTRES 1.3 computer code has been developed, based on an industry standard tool version of the code, through the implementation or modification to code models such as the fuel pellet thermal conductivity, the flux depression (radial power distribution in the fuel pellet), the fission gas release, the fission products diffusion to UO_2 grain boundaries, the two-dimensional heat transfer effect between the fuel pellet and the fuel sheath, the fuel sheath creep deformation, and and an automatic finite element meshing capability to handle various fuel pellet shapes.

The ELESTRES 1.3 code design and development was planned, implemented, verified, validated, and documented in accordance with the Candu Energy Inc. software quality assurance program, which meets the requirements of the Canadian Standards Association standard for software quality assurance CSA N286.7-99.

This paper presents an overview of the ELESTRES 1.3 code with descriptions of the code's theoretical background, solution methodologies, application range, input data, and interface with other analytical tools. Code verification and validation results, which are also discussed in the paper, have confirmed that ELESTRES 1.3 is capable of modelling important fuel phenomena and the code can be used in the design assessment of CANDU fuels.

DEVELOPMENT OF ELOCA 2.3 COMPUTER CODE FOR MODELLING THE PERFORMANCE OF CANDU FUEL ELEMENT UNDER POSTULATED ACCIDENT CONDITIONS

Girma Chassie, Alan Jiang, Akash Gill, Hazen Fan and Masoud Shams (SNC-Lavalin Nuclear)

The ELOCA (Element Loss Of Coolant Accident) computer code models the thermal and mechanical behaviour of CANDU® fuel element under postulated accident conditions. Given the power history of a reactor transient, and the fuel boundary condition histories (i.e., coolant temperature and pressure, sheath to coolant heat transfer coefficient), the ELOCA code calculates the thermo-mechanical response of a fuel element during the transient. ELOCA calculations include: expansion, contraction, cracking, and melting of the fuel, variations in the element internal gas pressure, changes in the fuel/sheath heat transfer, deformation of the sheath, chemical reaction of Zr with H_2O and UO_2, and beryllium assisted cracking of the sheath.

ELOCA is the industry standard toolset (IST) code for evaluating fuel performance under postulated accident conditions. The ELOCA 2.3 computer code has been developed, based on an industry standard tool version of the code, through the implementation of bug fixes and code enhancements that have been identified since the last major code release.

The ELOCA 2.3 code design and development was planned, implemented, verified, and documented in accordance with the Candu Energy Inc. software quality assurance program, which meets the requirements of the Canadian Standards Association standard for software quality assurance CSA N286.7-99. The ELOCA 2.3 validation activities are being performed by following Candu`s software quality assurance program.

This paper presents an overview of the ELOCA 2.3 code with descriptions of the code’s theoretical background, solution methodologies, application range, and input data. The paper identifies the key model enhancements, and the implementation, the verification and the validation activities that are being performed.

SOURCE IST 2.0 BENCHMARKING AGAINST PRODUCTION DATA

A.I. Popescu, A. Cziraky, J. Sun, H. Hasanein, T. Danniels, H. Albasha and P. Gillespie (Kinectrics)

SOURCE IST 2.0 is the Industry Standard Toolset code for simulating fission product release from CANDU fuel under accident conditions. A new dataset, “Fission Gas Released (FGR) from production fuel 2001 – 2015” was prepared based on CANDU production fuel elements from 28-element and 37-element bundles, which were subjected to FGR measurements during post-irradiation examination. The dataset was used for SOURCE IST 2.0 predictions benchmarking. The results of the benchmarking are presented together with the modelling techniques and options used to simulate the FGR during normal operating conditions irradiation.

APPLICATIONS OF MESOSCALE MODELS FOR NUCLEAR FUEL AND STRUCTURAL COMPONENTS

Michael Welland, Andrew Prudil, Evan Thomas and Eric Tenuta (CNL)

Mesoscale phenomena, in which interfacial effects play a key role, are attracting increasing attention for their significance in determining the performance of nuclear fuels and structural components. Quantitative simulation of these phenomena is supported by modern computational approaches such as phase-field, and the included phase model. Such models offer direct integration with equilibrium thermodynamic treatments (CALPHAD-type) to enhance fidelity and thermodynamic self-consistency of predictions. This work summarises the fundamentals of mesoscale modelling, including the phase-field and included phase model and presents a selection of simulation results. Applications include over pressurised intragranular fission gas bubble migration and intergranular accommodation, long range percolation in Mixed Oxide fuels, multiphase Al-Mg interdiffusion for advanced research reactor fuel, zirconium hydride dissolution / precipitation hysteresis, and intergranular He bubble formation in Ni spacer springs.

Tuesday, 2019 July 23, 8:30 AM-10:00 AM

Plenary Session #3, 8:30 AM -10:00 AM

Accident Tolerant Fuel and Novel Materials for CANDU Fuel

ACCIDENT TOLERANT FUELS: ON POTENTIAL OPTIONS FOR CANDUS IN LIGHT OF INTERNATIONAL ACTIVITIES FOR LWRS

Ho Chun Suk, Michel Couture and Wade Grant (CNSC)

Accident-tolerant fuels for light-water reactors (ATFLs) have become a topic of international interest ever since the March 2011 events at the Fukushima Daiichi nuclear power plant in Japan. This paper reviews the definitions, objectives and status of ATFLs being developed in the world, and evaluates, from an accident tolerance perspective, possible evolutions of CANDU fuel design employing concepts of accident tolerant fuels.

The views expressed in this paper are those of the authors and do not necessarily reflect those of CNSC, or any part thereof.

TOWARDS INDUSTRIAL SCALE MANUFACTURE OF URANUIM NITRIDES (UN) FUEL FOR CANDU REACTORS

Janne Wallenius (LeadCold Reactors, Sweden), Yulia Mishchenko (KTH, Sweden), Mikael Jolkkonen (KTH, Sweden) and Daniel Laurin (Promation Nuclear)

The use of UN-15 fuel in CANDU reactors permits to increase the fuel average residence time by up to 60% This corresponds to a reduced cost for manufacture of fuel assemblies and waste packages, as well as a reduced wear on refueling machines. Moreover, it would provide a reduced magnitude of the positive power coefficient and increase the margin to fuel melting during overpower transients, potentially allowing for uprating of CANDU reactors operating at a reduced power.

In this contribution, we describe recent progress towards industrial scale production of UN fuel for CANDU reactors:

At KTH, UN is produced by direct ammonolysis of uranium tetrafluoride, at a scale of 10-100 grams. This process is amenable for industrial scale up and yields a highly pure nitride powder product suitable for pellet production.

Promation Nuclear develops approaches to automation of spark plasma sintering, aiming at improving production rates by an order of magnitude. The processes subject to automation include powder feed, graphite die replacement, pellet unloading, grafoil removal, final machining and individual pellet inspection.

ACCIDENT TOLERANT FUEL FOR APPLICATION IN CANDU AND SMALL MODULAR REACTORS

Jerzy Szpunar (Univ of Sask)

Development and use of accident tolerant fuel (ATF) in commercial light water reactors (LWRs), Generation IV nuclear reactors and small modular reactors (SMR) is studied extensively at present. This presentation will address research of our team in area of new high thermal conductivity composites based on urania, thoria and silicates and diborrite of uranium and thorium. Presented work is both experimental and theoretical.

We manufactured various (collaboration with UBC) types of composites with uranium and thorium, in also uranium carbides, silicates and borate and often obtained very significant improvement of thermal conductivity. Performed for the first time detailed microstructural analysis and comparative studies of influence of porosity and fission products on the thermal conductivity.

Our computational prediction were based on Density Functional Theory (DFT) and were focused on prediction of thermal conductivity, electronic properties and structural changes of materials for fuels at very high temperatures, and under irradiation and also in oxidation environment. DFT and molecular dynamics (MD) was used to predict the thermal conductivity and mechanical properties of defects and incorporation of xenon (Xe) and zirconium (Zr) fission products in studied in materials for fuel. Materials like UO_2-BeO, UO_2-SiC and uranium carbide, uranium silicates, uranium boride were studied. This work provides fundamental understanding of structure-property relationships under irradiation and with presence of fission products in fuels and can serve as an important data for future experimental efforts. The possible application of investigated fuels in Small Modular Reactors is considered. The calculations were performed within the framework of (DFT) +U approach, using Quantum ESPRESSO (QE) code.

Our research is also focus on fuel cladding and address the problem of severe environment of service in Gen IV nuclear reactors, where high temperature, or supercritical water oxidation and other factors may require the replacement of Zr alloys in many applications. We propose novel coatings to improve performance of Zr alloys and also investigated stainless steels, nickel alloys and Inconels as a possible replacement of Zr alloys. Microstructural design of materials for fuel cladding allowed us to improve resistance to supercritical water oxidation.

Tuesday, 2019 July 23, 10:20 AM - 12:00 PM

Technical Session #T1

Spent Fuel Management

ELEMENTAL COMPOSITION OF UNIRRADIATED CANDU FUEL

Kelly Liberda, Helen Leung, Paul Gierszewski and Liana Orlovskaya (NMWO)

Trace elements in CANDU fuel may be important for long-term waste management purposes, since they can produce long-lived radionuclides. Most of these are not addressed in the fuel and cladding technical specifications. Furthermore, the actual levels may be much less than the specification limits.

In this study, published data on unirradiated fuel elements were reviewed. In addition, 21 unirradiated fuel elements from across the history of the CANDU program were analyzed for the main and trace element composition of the fuel and cladding. The results will be summarized in this paper. They can be used to improve predictions of long-term radionuclide inventories in used CANDU fuel.

FISSION PRODUCT CORE INVENTORY ESTIMATION USING ORIGEN-S CODE

W.J. Zhu, L.Y. Huang and H.Z. Fan (SNC-Lavalin Nuclear)

Irradiation of CANDU®fuel in the reactor core produces hundreds of radioactive fission products and actinides. The ORIGEN-S code accounts for the effect of both irradiation and decay, and determines time related quantities or activities of elements or isotopes. The fission product inventory in the core is the key parameters applied in safety analysis under normal operating and accident conditions. The estimation methodology is developed for fission product inventory in the core based on fuel power and burnup distributions and utilise ORIGEN code from a single fuel cross-section to all the fuel bundles in the core. The core distribution of the inventories for major fission products are estimated for a typical Candu 6 reactor.

NATURAL CONVECTION IN CLOSED IRRADIATED FUEL BAY (IFB) RACKS AFTER A LOSS OF COOLANT

Derek Logtenberg, Paul Chan and Emily Corcoran (RMC)

A Loss of Coolant Accident (LOCA) in an Irradiated Fuel Bay (IFB) is an unlikely event with adverse consequences for the surrounding public, workers, and environment. After the near miss in Unit 4's Spent Fuel Pool at Fukushima, more efforts have been dedicated to understand the convection process during such an accident. This paper predicts the air temperatures of flows through CANDU® type fuel racks using analytical models and Computational Fluid Dynamics techniques by approximating the rack as an in-line tube bank. The results at low temperatures offer a useful comparison tool for IFB severe accident code development.

PLANNING OF SPENT FUEL INTEGRITY EVALUATION TECHNOLOGY FOR DRY STORAGE

Seongki Lee and Manseok Do (KEPCO Nuclear Fuel, Korea)

Spent fuel dry storage of CANDU fuel is a main management option worldwide. To do this end, one of key element is do establish the integrity evaluation scheme. However, compared to PWR fuel, CANDU fuel is in an immature status on technology infra such as data, model, code etc. Thus, feasible planning is important to build a reliable the evaluation frame. In this study, the strategy is to picture overall system for CANDU fuel management in Dry storage by bench-marking PWR's.

Tuesday, 2019 July 23, 10:20 AM - 12:00 PM

Technical Session #T2

Fuel Fabrication

STRATEGIES IN MITIGATING STRESS CORROSION CRACKING OF ZIRCALOY-4 FUEL SHEATHING: A CASE FOR POLYSILOXANE COATING

M. Farahani, P.K. Chan, E.C. Corcoran, R. Hameed and T. Torkelson (RMC)

Abstract The focus of this investigation is to elucidate numerous advantages afforded by Polysiloxane technology, a single-component, silicone-based alternative coating for protecting Zircaloy fuel sheaths in CANDU reactors from Stress Corrosion Cracking (SCC), due to the pellet-clad-interaction (PCI) and corrosive fission products. In particular the resiliency to corrosive and oxidative, radiation, high temperature, and mechanical stress environments, of commercially available PYROMARK1200®, is further assessed and presented. Preliminary results suggest that the PYROMARK1200®, due to its superior performance (as tested), has great potential as a viable alternative to replace the existing DAG154N Graphite and hence the ability to effectively combat the fuel failure for various reactor conditions (e.g., power ramps, increased linear power, etc.), and to further support higher burnups in advanced fuel cycles.

BEST-ESTIMATE PLUS UNCERTAINTY ANALYSIS OF CANDU FUEL RELIABILITY USING MANUFACTURING DATA AND SIMULATED CORE DATA

Jason Song, Paul Chan (RMC) and Mahesh Pandey (CAMECO)

A novel method for assessing the reliability of 37-element CANDU fuel was developed. The approach follows the principle of “best estimate plus uncertainty,” where probability of failure to meet a fuel performance criterion is adapted as a measure of safety with due consideration of uncertainties. In this study, fuel performance was predicted using the industry standard code, ELESTRESS. The outputs of the code were construed against failure criteria derived from industry norms to determine the probability of failure. The manufacturing parameters were adapted from data provided by Cameco, and the model core was adapted from Darlington nuclear generating station dataset.

RMCC SLOWPOKE REACTOR CORE FABRICATION AND REFUELING

Justin Spencer, Paul Chan, Brent Lewis, Catherine Thiriet, Andrew Bergeron, Steve Livingstone and Shuwei Yue (CNL)

The Safe LOW-POwer Kritical Experiment (SLOWPOKE) reactor at the Royal Military College of Canada (RMCC) has been used for educational and research purposes since 1985. The core is nearing the point where the addition of beryllium shims can no longer compensate for the reactivity losses associated with burnup. Canadian Nuclear Laboratories (CNL) and RMCC have initiated a project to refuel the reactor. This will involve a range of activities including fabrication of a new core, removal of the spent core, and recommissioning of the reactor. The work, particularly the uranium metal to oxide conversion and fuel fabrication process, is described in this paper.

THE MICROSTRUCTURE AND THERMAL CONDUCTIVITY OF SPARK PLASMA SINTERED THO2

Linu Malakkal, Anil Prasad, Jayangini Ranasinghe, Ericmoore Jossou, Barbara Szpunar, Lukas Bichler and Jerzy Szpunar (Univ of Sask)

Thorium dioxide (ThO_2) is proposed to play a vital role in the worlds future energy needs and is considered a better and safer alternative to the currently used nuclear fuel Uranium dioxide (UO_2) not only because it is more abundant but also due to better thermo-physical properties, chemical stability, proliferation resistance, higher burnup, more stable waste form and reduction of plutonium inventory. However, one of the major problems in the front end of the thorium fuel cycle is associated with the preparation of dense ThO_2 pellets using the conventional sintering technique. The currently used conventional sintering techniques need higher sintering temperature (>2000 oC) and the longer time to prepare ThO_2 pellets. Therefore, we explore the possibility of non-conventional spark plasma sintering technique in overcoming the disadvantages of the conventional sintering technique. Therefore, in this work we carry out a systematic study on the effect of sintering parameters on the microstructure, density and thermal conductivity of spark plasma sintered ThO_2. The pellets were prepared by varying the sintering temperature, pressure and hold time and further characterized using various techniques. XRD analysis indicated the formation of thoria and ruled out the formation of any reaction products or intermetallic. The electron back scattered diffraction (EBSD) study has revealed a large dependence of sintering parameter on the grain size of the sintered pellets and possibility of control over the grain size of the pellets. The density measurement using the Archimedes principles have confirmed that extremely dense pellets of ThO_2 are possible with a lower sintering temperature in a short time. Furthermore, using the laser flash technique the thermal conductivity of various samples with different porosity were analysed indicating a significant dependence of thermal conductivity on the porosity of the pellets.

Tuesday, 2019 July 23, 1:30 PM – 5:10 PM

Technical Session #T3

Fuel Performance and Operating Experience

DEVELOPMENTS IN POST-IRRADIATION EXAMINATION HOT CELL EQUIPMENT ON CANDU FUEL AT CHALK RIVER LABORATORIES (CNL)

Jeffrey Armstrong, Shane Audette, Craig Buchanan, Javin DeVreede, Jeffrey Olfert and Patrick Sullivan (CNL)

Canadian Nuclear Laboratories (CNL) continues to update and develop equipment to improve post-irradiation examination (PIE) techniques on irradiated fuel. Development of PIE equipment supports ongoing CANDU fuel examinations at the CNL Hot Cells in Chalk River, including fuel surveillance and defect root cause examinations. This paper focuses on recent developments in equipment specific to supporting CANDU fuel PIE. Techniques discussed include: • Visual examination improvements to portability, reliability and image quality during non-destructive examination (NDE) • Ongoing development of small cladding defect detection using ultrasonic testing (UT) • Development of a longitudinal fuel cutting tool • Implementation of modern low-magnification microscopes for destructive examination (DE) • Updated gas volume measurement and collection system • Ongoing development of an eddy current scanning system on zirconium cladded fuel.

IRRADIATION PERFORMANCE OF PHWR FUEL AT EXTENDED BURNUP

Prerna Mishra, B.N Rath, Ashwini Kumar, V.P Jathar, U. Kumar, H.N. Singh, P. K. Shah, R.S. Sriwastaw, J.S. Dubey, G.K. Mallik, J.L. Singh, P.G. Jaison and S. Kannan (Bhabha Atomic Research Centre, Trombay, India)

There is a worldwide trend to extend the discharge burnup of fuel assemblies in nuclear power reactors. Normally, natural UO2 fuel is discharged from PHWRs at a bundle average burnup of ~ 7000 MWd/tU. To accumulate high burnup experience and evaluate the performance of PHWR fuel, few natural UO2 fuel bundles were irradiated for extended periods up to the burnup of ~25,000 MWd/tU. Detailed post irradiation examination of one of these fuel bundles (No. 145530) was carried out to generate data on their performance with respect to fuel restructuring, swelling, fission gas release, cladding corrosion, cladding strain and ductility etc. The details of PIE carried out on the extended burnup fuel bundle and results are presented in this paper.

GUIDELINES FOR ACHIEVING EXCELLENCE IN CANDU FUEL PERFORMANCE

Ben Wong, Andrew Fitchett , Ross Rock, and Paul Gillespie (Kinectrics), Ross Lewis and Philipe Paquette (BP), Todd Daniels (OPG)

Fuel element failures contaminate the primary heat transport system (PHTS) and increase the consequential occupational radiation exposures to station staff. Therefore, it is important for each CANDU utility to implement strategies to achieve and maintain the industry goal of zero fuel element failures. This paper summarizes seven key attributes necessary for

achieving and sustaining failure free fuel performance in CANDU reactors. Characteristics of each attribute are described to help define CANDU utility-specific standards and expectations.

UNIT 2 RTS PHTS HOT-CONDITIONING: RESULTS OF INVESTIGATIONS TO ASSESS FITNESS FOR SERVICE EXPECTATIONS FOR OPG 37M FUEL BUNDLES

Jonathan Judah, Richard Scrannage, and Steve Goodchild (OPG), Farzin Abbasian, Gordon Hadaller and Mario Campigotto (Stern Lab)

The reference plan for the Return-to-Service of Darlington Unit 2 after refurbishment schedules the Primary Heat Transport System (PHTS) Hot-Conditioning after manual loading of the first charge of fresh fuel bundles, after PHTS pressure testing and prior to initial Approach to Critical. CANDU refurbishment operating experience indicates that PHTS Hot Conditioning after fuel loading carries a risk that fuel bundle surfaces might become coated with magnetite and other depositing or precipitating materials during the Hot Conditioning evolution. These deposits, if they occur, may put the fuel bundles outside the presently analyzed fuel operating envelope.

This paper provides a review and assessment of industry operating experience, contracted vendor assessments, recent out-reactor testing commissioned by OPG, and expert assessments related to the Hot Conditioning evolution.

These efforts provide a reliable assessment of the risks and benefits of the planned evolution, and have assisted in the formulation of the strategy for the Darlington Unit 2 PHTS Hot Conditioning evolution to mitigate a similar risk to fuel performance.

ANALYSIS OF THE PIE RESULTS TO DETERMINE/CONFIRM PRIMARY ROOT CAUSE FOR FAILED PICKERING B FUEL ELEMENT

Krishna Chakraborty, Akash Gill, Zhen Xu (SNC-Lavalin Nuclear), Todd Daniels, Erin Middaugh and Mariana Dobrean (OPG)

Failed fuel element from Pickering B station was shipped to CNL for post-irradiation examination with the objective of determining the primary root cause of fuel failure. The failed element exhibited defects with elevated deuterides. Leak testing and dye penetrant tests of the remainder of the sheath sections did not reveal any new defect location(s). Sheath ID examination of the remaining sheath sections revealed several areas of interest that were followed up with metallographic examination. After completion of the non-destructive and destructive examinations, Candu Energy performed defect root cause analysis. Analysis included studying potential parameters/areas in manufacturing, operations and design that could affect fuel performance and cause defect. The primary defect site was located and finite element analysis using FEAST code was performed to confirm the root cause for failure.

NOVEL TOOL AND METHODOLOGY USED FOR PRIORITIZING SUSPECT DEFECTED FUEL AT PICKERING NUCLEAR GENERATING STATION

Shahab Dabiran (OPG)

A software toolset, XeCorr, has been developed to expedite the discharge of fuel defects by quantifying the correlation between bundle powers and Xe-133 activity in the coolant. XeCorr has been used at PNGS, successfully assigning a high

priority status to the fuel channels containing defected bundles.
This paper discusses testing of XeCorr on two periods of historic fuel defect operation at the Pickering Nuclear Generating Station. The tool successfully assigned a high priority status to historic and subsequent fuel channels containing defected fuel bundles. Defect removal efforts at Pickering have been improved by the implementation of this toolset.

CANDU FAILED FUEL DETECTION AND LOCATION SYSTEM

Jeffrey Arndt and Michael Heibel (Global Technology Office, Westinghouse, USA)

The detection and identification of a leaking fuel bundle position inside a CANDU-style reactor during operation is currently a very complex process. The determination of exactly what fuel bundle(s) have failed is imprecise and time consuming. A novel Failed Fuel Detection and Location System (FFDLS) is presented. The system is based on the capabilities of prompt responding SiC detectors with gamma radiation energy sensitivity suppression. The fundamental concept of the technology is to create an array of detectors using the Vanadium In-Core Instrument (ICI) assemblies in spare or non-utilized openings. Using this array of detectors, the location of the failed fuel bundle can be triangulated based on relative changes in measured detector signal intensity.

CANDU FUEL ENGINEER'S MANUAL - CURRENT DEVELOPMENT STATUS AND FUTURE PLANS

Paul Gillespie (Kinectrics)

The CANDU Fuel Engineer's Manual is an invaluable resource for those working in the CANDU industry, and particularly for those working in the fuel area. Original development began in the early 1990s with contributions from numerous senior experts from across the industry. The manual continues to be developed and expanded, including conversion in recent years to an online resource from the original paper version. In this paper, an overview is provided of the manual contents, purpose and future development plans.

Tuesday, 2019 July 23, 1:30 PM – 5:10 PM

Technical Session #T4

Fuel Safety and Operational Margin Improvement

SAFETY ANALYSIS RESULTS OF THE 37M DURING TRANSITION CORE IN WOLSONG

Sungmin Kim (Korea Hydro & Nuclear Power, KHNP)

As a CANDU reactor ages, the operating margin has eroded due to an aging phenomena such as pressure tube creep, which increases bypass coolant flow to the fuel bundle in the channel. To mitigate this problem, KHNP decided to load the modified 37-elements bundle into the Wolsong NPP site. The modified 37-elements bundle was developed by Canadian utilities, which is small difference with existing 37-elements fuel. As the licensing process, KHNP performed the demonstration irradiation and safety analysis. Based on DI test and safety analysis, the behaviour of the modified 37-elements bundle was nearly same as the existing fuel bundles. The objective of this paper is to compare the results of the safety margins with 37R full core, transition core, 37M full core.

STYLIZED STUDY USING FEAST COMPUTER CODE ON FUEL BUNDLE END PLATE DEFORMATION SUBJECT TO SUSTAINED HIGH TEMPERATURE

Z. Xu, L.Y. Huang and H.Z. Fan (SNC-Lavalin Nuclear)

CANDU fuel bundle is structurally assembled with four components: end plates, end caps, sheath tubes and beryllium brazed joints. FEAST computer code has been used in fuel design to ensure that the end plates can perform well and fit for service, retaining fuel bundle configuration with insignificant deformation. In this paper, FEAST code is used in a stylized study on onset of significant structural deformation of the end plates subject to a sustained high temperature. In the study, the nonlinear properties of Zircaloy as the structural material of the fuel bundle with respect to the temperature have been taken into account.

FUEL BEHAVIOUR MODELLED FOR SEVERE ACCIDENT PROGRESSION AND CONSEQUENCE ASSESSMENT USING MAAP5-CANDU COMPUTER CODE

H.Z. Fan, T. Nguyen and L. Comanescu (SNC-Lavalin Nuclear)

CANDU reactors are designed and operated with safety systems and features for multi level defence in depth for public safety. For beyond design basis accidents, deterministic safety analysis with the best estimation approach is needed to provide accident progression and consequences to use in probabilistic safety assessment. As a modular accident analysis program, the MAAP5 CANDU computer code has modelled CANDU fuel in details from its normal operating state to core debride bed forming and relocation states to determine the severe core damage progression time, which form the basis for emergency mitigating equipment supporting severe accident management.

THE USE OF BURNABLE NEUTRON ABSORBERS TO MITIGATE THE EFFECT OF COOLANT VOIDING ON CANDU 37-ELEMENT FUEL

Mark Couture, Paul Chan and Hugues Bonin (RMC)

A study of the mitigating effects that burnable neutron absorbers have on coolant voiding in CANDU 37-element fuel was conducted. The purpose of this study was to determine if safety margins may be improved from the addition of a small amount of burnable neutron absorbers in the fuel through the simulation and analysis of a large loss of coolant accident. Target burnable neutron absorber concentrations of 120 mg Gd_2O_3 and 300 mg Eu_2O_3 have been shown to dampen the refueling and plutonium transients with negligible impact on the fuel burnup. The lattice code WIMS-AECL was used to compute macroscopic cross sections which were passed to the diffusion code RFSP to model a CANDU reactor undergoing a primary heat transport system reactor inlet header break. Burnable neutron absorbers Gd_2O_3 and Eu_2O_3 were added to the CANLUB layer of specific fuel pins and core power distributions were computed for comparison against the natural uranium fuel case for a fresh core. Preliminary results show a reduction in the core enthalpy change when burnable neutron absorbers are added, along with a reduction in the side-to-side power oscillation during the transient. The reduction in enthalpy change implies that the positive void reactivity in CANDU reactors during loss of coolant accidents could be mitigated through the addition of burnable neutron absorbers in the fuel. Further research is to be conducted with the modeling of various core configurations including reactor start up and on-power refueling equilibrium flux shapes along with the development of a fully coupled physics and thermohydraulic model in order to more accurately predict enthalpy fluctuations throughout a loss of coolant accident with and without burnable neutron absorbers.

SOURCE IST 2.0 VALIDATION AGAINST FAST AND SLOW POWER RAMP TESTS

Adam Cziraky, Adrian Popescu and John Sun (Kinectric)

SOURCE IST 2.0 is the Industry Standard Toolset code for simulating fission product release from CANDU fuel under accident conditions. Validation of SOURCE IST 2.0 against the OECD dataset NEA-1705/02, was performed. This experimental dataset contains fission gas measurements for AGR fuel which underwent NOC irradiation and subsequent fast or slow power ramp transients in the Halden HBWR experimental reactor. The selected tests are representative of CANDU in-reactor transients where relatively low fission gas release is expected and extends the validation basis of SOURCE IST 2.0. The results of the validation are presented together with the modelling techniques and options used to simulate fission gas release during these power ramp events.

BUNDLE SLUMPING TEMPERATURE OBSERVED IN EXPERIMENTS OF BUNDLE DEFORMATION SUBJECT TO SUSTAINED HIGH TEMPERATURE

H.Z. Fan, W.J. Zhu, L.Y. Huang and V. Lau (SNC-Lavalin Nuclear)

CANDU industry has many years of activities in research and development related to bundle deformation. The available experiment data are useful for qualitative illustration of aspects of extreme bundle degradation and have useful indications for threshold temperatures of onset of bundle deformation. The experimental studies enhance theoretical understanding and experiment knowledge bases regarding fuel bundle behaviour under high temperature conditions, and enable the selection of the bundle slumping temperature for safety analysis in which fuel bundles are subject to a sustained high temperature.

FINITE ELEMENT ANALYSIS OF PRESSURE TUBE CREEP AND BUNDLE DEFORMATION OF A CANDU FUEL BUNDLE

Kyuhwan Lee, Diane Wowk and Paul Chan (RMC)

CANDU fuel bundles experience plastic deformations over time, and the horizontal configuration of the bundle in a crept pressure tube causes coolant to bypass the sagged lower half of the bundle. Parts of the bundle where the flow is limited may experience dryout earlier than expected due to reactor aging. Yet, work is being done to investigate the effects of permitting more creep for a more effective utilization of fuel bundles. This paper will introduce the preliminary stages of a finite element model development, targeting the addition of creep, thermal and gravity effects, and contact on a full bundle.

SENSITIVITY ANALYSES FOR FUEL BEHAVIOUR DURING A LARGE BREAK IN THE PRIMARY HEAT TRANSPORT CIRCUIT

Catalin Zalog and Emanoil-Relu Istrate (CERNAVODA NPP, Romania)

Following a large break initiation in the PHTS, the core flow decreases and some channels may become steam filled and others can experience stratified two-phase flow conditions. As a consequence, the reactor power is increasing until is shuted down on a trip parameter.

Until the fuel channels are refilled by Emergency Core Cooling System, some fuel elements are exposed for different time durations to steam cooling. The mismatch between the stored energy and the energy removed from the fuel element increases the internal fuel element gas pressure, whereas the rise in sheath temperature reduces the sheath strength. If the channel coolant pressure falls below the fuel element internal gas pressure, the sheath stress is increased. If fuel temperature becomes high enough, sheath failure can occur, releasing fission products.

Wednesday, 2019 July 24, 9:00 AM - 10:30 AM

Plenary Session #4

Special Interest Topics

FUEL RECYCLING FOR CANADA'S SHORT- AND LONG-TERM FUEL INDEPENDENCE AND SECURITY

Peter Ottensmeyer (University of Toronto)

The advent of small modular reactors (SMRs) brings an entirely new requirement of fuel procurement to Canada. The normal fuel requirements of SMRs to achieve operational neutron equilibrium necessitates enrichment of the fissile fuel composition. Canada has no enrichment facilities, making Canada dependent on foreign powers, i.e. nuclear weapons nations, for such enriched fuel. As the only economical alternative for short- and long-term fuel independence Canada can and must adopt recycling capabilities. Then Canada's current 60,000 tonnes used CANDU fuel can already furnish safe enriched fissile starting material for SMRs equivalent to at least a total 24,000 MWe fast-spectrum reactors.

THERMALHYDRAULICS ROLE IN THE ADVANCED PHWR FUEL DEVELOPMENT

Jun Yang (CNL)

Thermalhydraulics studies play an important role during the course of advanced fuel development, from conceptual design to full-core implementation in a commercial power reactor. New fuel designs must meet safety requirements. Fuel elements in the bundle must transfer heat to the flowing coolant without experiencing dryout on any element, under all anticipated operating conditions throughout its operational life. The qualification of a new fuel bundle design requires thermalhydraulics testing and calculations using analysis tools to confirm the adequacy of the thermal margin before in-core implementation (for either irradiation in a test reactor or in-reactor demonstration in a power reactor). This paper provides an overview of the advanced pressurized heavy-water-moderated-and-cooled reactor (PHWR) fuel development program at Canadian Nuclear Laboratories (CNL), and identifies the knowledge gaps from a thermalhydraulics perspective.

CANDU OWNERS GROUP NUCLEAR SAFETY RESEARCH AND DEVELOPMENT ON THE CANDU FUEL

Wei Shen (COG)

In Canada, safe operation of CANDU reactors is supported by a full-scope program of nuclear safety Research and Development (R&D) in key technical areas. Basically, the R&D program required for due diligence in CANDU safety is dominated by COG (CANDU Owners Group). The current COG R&D Program comprises one Strategic R&D program and five base R&D programs such as Fuel Channels, Safety & Licensing (S&L), Health, Safety & the Environment, Chemistry, Materials & Components and IST. The S&L R&D Program addresses issues related to the safety design basis and safe operating envelope of existing nuclear plants, and has a strong focus on supporting the resolution of outstanding generic safety and licensing issues. There are four disciplines under the S&L R&D Program: Containment and Severe

Accidents, Fuel and Fuel Channels & Fuel NOC (Normal Operating Condition), Physics, and Thermalhydraulics. This paper presents an overview of the CANDU safety R&D which currently conducted by the COG (CANDU Owners Group). In particular, the nuclear safety R&D on the CANDU fuel is discussed. The paper concludes by identifying directions for the future nuclear safety R&D.

Wednesday, 2019 July 24, 11:00 AM – 1:05 PM

Technical Session #W1

Fuel Bundle Thermalhydraulics

POST-DRYOUT HEAT TRANSFER PHENOMENON AND PREDICTION METHODS IN CANDU FUEL BUNDLES

Yujun Guo and Naj Hammouda (CNSC)

Post-dryout (PDO) heat transfer is the heat transfer regime downstream of the critical heat flux (CHF) location. PDO heat transfer is of great interest for CANDU fuel safety, as it is during this heat transfer regime that fuel-sheath temperatures start to rise rapidly.

The PDO heat transfer phenomenon in CANDU fuel bundles is very complex because of the geometrical configuration of CANDU fuel bundles and the non-uniform axial and radial power profiles. Because of this, predictions of PDO heat transfer in CANDU fuels are predominantly empirical. Presently, one-dimensional computer codes are used to predict PDO heat transfer in CANDU fuel bundles, using cross-sectional averaged empirical correlations. Such simplified empirical correlations are very limited in their application ranges.

This paper outlines the PDO heat transfer phenomenon in CANDU fuel bundles, the various prediction methods and their associated limitations, and the areas of potential improvement.

METHODOLOGY FOR INVESTIGATING SUBCHANNEL COOLANT FLOW IN A 43-ELEMENT FUEL BUNDLE

Caitlyn Cavanagh-Dollard, Paul Chan and Diane Work (RMC)

CANFLEX is a 43-element fuel bundle developed as a carrier for advanced fuel in CANDU reactors. The design includes critical heat flux buttons to increase dryout and safety margins, and two element diameters to reduce peak linear element ratings while maintaining bundle power. As part of a research initiative to support CANDU fuel development, the bundle was redesigned without buttons to make the bundle more economical by simplifying fabrication. This paper will present methodology for modelling steady-state, single phase coolant flow in 43-element fuel bundle subchannels under normal operating conditions. The subchannels nearest where the fuel bundle rests on the bottom of the pressure tube will be investigated to determine coolant flow characteristics and axial pressure differentials. Parametric studies will be done with bundles featuring increased bearing pad heights to determine if this design variation results in increased heat transfer performance.

THE EFFECT OF AZIMUTHAL HEAT CONDUCTION ON HEATER SURFACE TEMPERATURES UNDER POST DRYOUT CONDITIONS

Rick Fortman (Stern Labs)

The heat transfer mechanism of a CANDU fuel simulation changes from nucleate boiling to film boiling when the electric power supplied exceeds the critical heat flux (CHF) value. It is expected that in a heat-flux controlled system, the heater surface temperatures should rise suddenly due to the drastic reduction in heat transfer coefficient under film boiling. This phenomenon has not been observed during the post-dryout (PDO) testing of CANDU fuel simulations at Stern Laboratories Inc.

In fact, due to azimuthal heat conduction the temperature responses are subdued and reach high values at significant over initial dryout power value.

CRITICAL HEAT FLUX AND POST-DRYOUT EXPERIMENTS USING THE MODIFIED 37-ELEMENT FUEL SIMULATION IN WATER WITH A 6.8% CREPT FLOW CHANNEL

Gordon Hadaller, Rick Fortman and Jay Snell (Stern Labs)

Experiments were performed to measure the critical heat flux (CHF) performance of the 37-element fuel simulation with a modified (smaller) center element and a downstream skewed AFD using a new 6.8% maximum crept flow channel. Post Dryout (PDO) tests were also conducted with the 6.8% maximum crept flow channel

In general, the results are consistent and follow established trends. The daily repeats exhibited the gradual decrease with time as observed in previous campaigns and is attributed to the formation of crud on heater surfaces.

A consistency check of the data for all modified 37-element, downstream skewed AFD initial dryout power values was performed by normalizing the measured power with respect to the predicted un-crept power at the same conditions and plotted against pressure tube creep. The resulting trend is fairly linear justifying extrapolation of critical power estimates for flow channels with intermediate maximum creep to 6.8%.

FULL-SCALE CRITICAL HEAT FLUX EXPERIMENT FOR PLUTONIUM-BASED MIXED OXIDE ADVANCED FUEL BUNDLE DESIGN

Lanqin Yuan, Jun Yang, Bruce Addicott, Vinson Gauthier and Matthew Dickerson (CNL)

A critical heat flux (CHF) experiment for the advanced Plutonium (Pu)-based mixed oxide fuel (MOX) was performed at Canadian Nuclear Laboratories using a full-scale fuel string simulator of a 43-element bundle design. The fuel string was heated electrically with the uniform axial power profile and the radial power profile representative of the PU-based MOX fuel. The fuel string simulator was vertically oriented and cooled by R-134a refrigerant. The experiment revealed that the dryout powers of the MOX fuel are significantly lower than those of the natural uranium fuel at similar test conditions. A pre-test analysis was performed to evaluate the MOX fuel dryout power using a model to account for the effect of radial power profile on CHF. The pre-test calculations agree well with the dryout power measurements.

Wednesday, 2019 July 24, 11:00 AM – 1:05 PM

Technical Session #W2

Accident Tolerant Fuel and Novel Materials for CANDU Fuel

MICROSTRUCTURAL AND THERMOPHYSICAL CHRACTERIZATION OF PURE AND MO-DOPED UO2 PELLETS

Murali Krishna Tummalapalli, Linu Malakkal, Ericmoore Jossou, Jerzy A Szpunar, Anil Prasad and Lukas Bichler (University of Sask.)

Uranium dioxide (UO2) is the nuclear fuel of choice for several decades in the commercial nuclear reactors. The main drawback in the use of UO2 is the low thermal conductivity which degrades with an increase in temperature and radiation dosage. In this study, we attempt to improve the thermal properties of UO2 by the addition of Molybdenum (Mo) which has high thermal conductivity. High-dense pellets with theoretical density (TD) ranging from 95% to 99.5% were prepared by spark plasma sintering (SPS) technique. Phase identification using x-ray diffraction ruled out the formation of a secondary phase while scanning electron microscopic (SEM) and electron backscatter diffraction (EBSD) measurements were used to determine the microstructure and grain size of the UO2-Mo composite fuel. The laser flash analysis at a temperature range from 25 to 900 oC show that UO2-Mo composite has remarkably enhanced thermal conductivity which is a desirable characteristic for an accident tolerant fuel (ATF).

FIRST PRINCIPLES STUDY ON THERMAL CONDUCTIVITY OF U3O8

Jayangani Inoka Ranasinghe, Linu Malakkal, Barbara Szpunar, Ericmoore Jossou and Jerzy A. Szpunar (University of Sask.)

In this work, we study the ground state structural, elastic, vibrational properties and the lattice thermal conductivity (K_L) of alpha-U3O8 in the temperature region 300 K to 1000 K. The evaluation is carried out using density functional theory (DFT) along with the Boltzmann transport equation (BTE). First-principles simulation codes Quantum ESPRESSO (QE), Phonopy and ShengBTE have been used in the evaluation. This work shows large anisotropy of K_L along (100), (010) and (001) directions. Further, it shows that the optical phonon modes make a significant contribution to K_L compared to the acoustic modes. Predicted K_L varies from 1.57 W.m^-1.K^-1 to 0.49 W.m^-1.K^-1 when the temperature goes from 300 to 1000 K.

STRUCTURAL STABILITY OF SIC

Barbara Szpunar and J.A. Szpunar (University of Sask.)

The good mechanical, thermal, and chemical properties of SiC such as high stiffness, high hardness, high mechanical strength at high temperature, high thermal conductivity and melting point make SiC a candidate for various applications in nuclear industries. In particular it is proposed to be used as cladding since it has a very high melting point and much slower reaction with water than presently used zirconium alloys. SiC is also expected to be used as a component in

composite fuels, due to one order of magnitude higher thermal conductivity than traditionally used urania, Such composite fuel will have enhanced heat dissipation and this should delay/prevent fuel melting in case of loss of coolant accident.

In this work we evaluate the effect of defects on structural stability of SiC. We use first principles molecular dynamics as implemented in CASTEP Density Functional code. As reviewed previously [1], each of available potentials for classical molecular dynamics, which can handle larger defects has some deficiencies. Therefore our calculations, which are limited to 64 atoms with periodic boundary conditions provide important information about structural stability of SiC in the presence of defect over wide range of temperatures up to the melting point. We found that SiC covalent structure is very stable up to the melting temperature and in contrast to urania or thoria no sub-lattice premelting is observed.

[1] B. Szpunar, Malakkal L., Rahman J., Szpunar J.A., J. Am. Ceram. Soc., (2018) https://onlinelibrary.wiley.com/doi/abs/10.1111/jace.15712, Atomistic modeling of thermo-mechanical properties of cubic SiC. 101 (10) 4753-4762.

PROGRESS IN EXPERIMENTAL AND COMPUTATIONAL INVESTIGATIONS OF MOLTEN FLUORIDE SALT THERMODYNAMICS FOR SMALL MODULAR REACTORS

Bernard Fitzpatrick, Daniel Hallatt, Ksenia Lipkina, Ryan Murphy-Snow, Parikshit Bajpai, Max Poschmann and Markus Piro (UOIT)

Fluoride salts are fabricated and thermodynamically investigated to better understand and predict the behaviour of liquid fuel salt and fission products in a molten salt reactor. Differential scanning calorimetry, thermogravimetric analysis, and x-ray diffraction are performed to determine temperatures and enthalpies of phase transitions, along with the heat capacity and thermal stability. Experimental results are being used to support thermodynamic database development of various fluoride salts with fission products. Computational efforts include applying and enhancing the performance of the thermodynamic solver Thermochimica for multi-physics applications, including the coupling of thermodynamics with isotopics and thermal-hydraulics.

ACCIDENT-TOLERANT FUEL REVIEW AND POTENTIAL APPLICATIONS TO CANDU REACTORS

Zhen Xu, Masoud Shams (SNC-Lavalin Nuclear)

Accident-Tolerant Fuel (ATF) concept was initiated in the US after Fukushima Daiichi accident caused by tsunami in 2011. The purpose was to study, research and develop power reactor fuels that can tolerate loss of active cooling in the core for a considerably longer time period (during postulated accidents) compared to standard UO_2–Zr fuel. ATF is also expected to maintain or improve fuel performance during normal operations. The research in ATF has since expanded internationally.

This presentation focuses on review of current ATF concepts and designs and past relevant experiences, in an effort to better understand the advantages and disadvantages of those concepts/designs, and to identify potential applications in developing future CANDU® fuels, which may also improve operational margins.

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