Canadian Nuclear Society
Société Nucléaire Canadienne
in cooperation with the
International Atomic Energy Agency

11th International Conference on CANDU Fuel

"Flexible Fuel for a Greener Future"
"Souplesse du cycle du combustible pour un avenir plus propre"

Sheraton Fallsview Hotel and Conference Centre
Niagara Falls, Ontario
2010 October 17 – 20

Conference Program

Following the highly successful tenth conference held at Ottawa in 2008, the 11th International Conference on CANDU Fuel brings together international experts of the nuclear fuel industry involved in design, R&D, fabrication, operation, fuel 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 11th International Conference on CANDU Fuel and to the enjoyment of the attendees and their guests through their generous sponsorship.

AECL

Cameco Corporation

General Electric Canada Inc.

Power Workers' Union

Stern Laboratories Inc.

Kinectrics Inc.

Ontario Power Generation

University of Ontario Institute of Technology

Program Sessions

- International Experience with CANDU Fuel
- Special Interest Topics
- Fuel Design
- Advanced Fuel Cycles
- Fuel Modelling and Computer Code Development
- Fuel Performance, Reliability and Operating Experience
- Fuel Safety & Operational Margin Improvement
- Fuel Fabrication
- Spent Fuel Management

Honourary Chair

Joseph Lau (AECL)

Organizing Committee Members

Steve Palleck, Conference Chair (AECL)
Margaret Bates (AECL)
Daniel Dai (AECL)
Scott Froebe (AECL)
Ki-Seob Sim (AECL)
Asif Waheed (AECL)
Karen White (AECL)

Technical Program Committee Members

Ki-Seob Sim, Technical Program Chair (AECL)
Peter Boczar (AECL, retired)
Krishna Chakraborty (AECL)
Lawrence Dickson (AECL)
Fern Doria (AECL)
Samir Girgis (AECL)
Holly Hamilton (AECL)
Zhang He (AECL)
Victor Inozemtsev (IAEA)
John Montin (AECL)
Patrick Reid (AECL)
Masoud Shams (AECL)
Shouze (George) Xu (AECL)
Zhen Xu (AECL)

Special Events

Sunday Opening Reception & Registration
2010 October 17
7:00 PM – 9:30 PM, Sheraton Fallsview Hotel

Monday Luncheon
2010 October 18
12:00 (Noon), Sheraton Fallsview Hotel Hennepin Ballroom

Luncheon Speaker: D. Cox, Director, Safety Eng & Licensing, AECL

Monday Technical Tour
2010 October 18
4:15 PM
Location: Sir Adam Beck Generating Station

Tuesday Luncheon
2010 October 19
12:00 (Noon), Sheraton Fallsview Hotel Hennepin Ballroom

Luncheon Speaker: A. Thorne, Vice-President, Fuel Services Division, Cameco

Tuesday Conference Banquet
2010 October 19
6:00 PM, Skylon Tower

Keynote Speaker:
J. Hopwood, Vice-President, Product Development, AECL

Wednesday Luncheon
2010 October 20
12:50 PM, Sheraton Fallsview Hotel Hennepin Ballroom

Luncheon Speaker: P. Ottensmeyer, Professor Emeritus, University of Toronto

Local Attractions and Activities

Family-friendly Niagara Falls is a favourite international tourist destination. Several popular attractions are listed at the following website: ***http://www.cns-snc.ca/events/11th-international-conference-candu-fuel/***

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 on CD, conference program, booklet of abstracts, conference lunches and coffee breaks and one (1) ticket each to the conference banquet and the conference technical tour.

NOTE:
Additional tickets for Monday, Tuesday and Wednesday Technical Sessions, the Conference Banquet and/or the Conference Technical Tour, may be purchased in advance or at the CNS Registration Desk at the Sheraton Fallsview Hotel.

Conference Proceedings

The Conference Proceedings will be available on CD at the time of the Conference. Late papers will be made available on the Conference website for a limited period of time after the Conference.

Conference Schedule – At a Glance

Sunday, 2010 October 17	**(7:00 PM – 9:30 PM)** Registration and Reception
Monday, 2010 October 18 Opening Ceremony	**(8:30 AM – 9:00 AM)** Welcoming Address Room - Oakes South
Plenary Session #1	**(9:00 AM – 12:00 PM)** International Experience with CANDU Fuel Room - Oakes South
Coffee Break	**(10:30 AM)**
Lunch	**(12:00 PM – 1:30 PM)** Hennepin Ballroom
Technical Session #M1	**(1:30 PM – 4:25 PM)** Fuel Design Room - Oakes North East
Technical Session #M2	Fuel Modelling and Computer Code Development I Room - Oakes South
Technical Session #M3	Fuel Performance, Reliability and Operating Experience I Room - Oakes North West
Coffee Break	**(2:45 PM)**
Technical Tour	**(4:00 PM)** Sir Adam Beck Generating Station
Tuesday, 2010 October 19 Plenary Session #2	**(8:30 AM – 10:00 AM)** Special Interest Topics Room - Oakes South
Coffee Break	**(10:00 AM)**
Technical Session #T1	**(10:20 AM – 12:00 PM)** Advanced Fuel Cycles Room - Oakes North East
Technical Session #T2	Fuel Modelling and Computer Code Development II Room - Oakes South
Technical Session #T3	Fuel Fabrication I Room - Oakes North West
Lunch	**(12:00 PM – 1:30 PM)** Hennepin Ballroom
Technical Session #T4	**(1:30 PM – 3:10 PM)** Spent Fuel Management Room - Oakes North West

	(3:30 PM – 5:10 PM)
Technical Session #T5	Fuel Fabrication II Room - Oakes North West
	(1:30 PM – 5:10 PM)
Technical Session #T6	Fuel Modelling and Computer Code Development III Room - Oakes South
Technical Session #T7	Fuel Safety and Operational Margin Improvement Room - Oakes North East
Coffee Break	**(3:10 PM)**
	(6:00 PM)
Conference Banquet Dinner	Location: Skylon Tower, Niagara Falls, Ontario

Wednesday, 2010 October 20	**(8:30 AM – 11:45 PM)**
Technical Session #W1	Fuel Modelling and Computer Code Development IV Room - Oakes South
Technical Session #W2	Fuel Performance, Reliability and Operating Experience II Room - Oakes North West
Coffee Break	**(10:10 AM)**
	(11:45 PM – 12:15 PM)
Plenary Session #3	CANDU Fuel Cycle Options Room - Oakes South
Lunch	**(12:20 PM – 3:00 PM)** Hennepin Ballroom
Adjournment	**3:00 PM**

Monday, 2010 October 18, 8:30 AM -9:00 AM,
Room - Oakes South
Opening Ceremony

08:30 Opening, *S. Palleck (Conference Chair)*
08:40 Welcoming Address, *J.H. Lau (Honorary Conference Chair)*

Monday, 2010 October 18, 9:00 AM-12:00 PM
Plenary Session #1, Room - Oakes South
International Experience with CANDU Fuel
Chair: M. Gabbani (GE Hitachi),
V. Inozemtsev (IAEA)

09:00 Consolidating Indigenous Capability for PHWR Fuel Manufacturing in India, *R.N. Jayaraj (Nuclear Fuel Complex)*

09:30 Load Following Tests on CANDU Type Fuel Elements in TRIGA Research Reactor of INR Pitesti, *G. Horhoianu [1], S.J. Palleck [2] and D. Ionescu [1] ([1] Institute for Nuclear Research), ([2] AECL Sheridan Park)*

10:00 CANDU-6 Fuel Bundle Fabrication and Advanced Fuels Development in China, *Wang Jun (China North Nuclear Fuel Corporation) (To be presented by D. Burton, Fuel Manufacturing Inc., Cameco)*

10:30 Coffee Break

11:00 Development and Commercial Implementation of CANFLEX Fuel in Korea, *J.H. Park and J.Y. Jung (Korea Atomic Energy Research Institute)*

11:30 CNSC Approach to the Review of Nuclear Fuel System Design for New Nuclear Power Plants, *H-C. Suk and M. Couture (Canadian Nuclear Safety Commission)*

Monday, 2010 October 18, 12:00 PM-1:30 PM
Lunch, Hennepin Ballroom

12:30 Luncheon Speaker - D. Cox (Director, AECL)

"Repair of the NRU Reactor Vessel –Technical Challenges in an Unplanned Forced Outage Situation"

Monday, 2010 October 18, 1:30 PM-4:00 PM
Technical Session #M1, Room - Oakes North East
Fuel Design
Chair: M. Shams (AECL), J.H. Park (KAERI)

13:30 Technical Basis for ACR-1000 Fuel Acceptance Criteria, *A. Sun and M. Tayal (AECL Sheridan Park)*

13:55 CNSC Approach to the Review of Reactor Core Thermalhydraulic Design for New Nuclear Power Plants, *Y. Guo, N. Hammouda, H. Tezel and M. Couture (Canada Nuclear Safety Commission)*

14:20 Coolant Solubility of Burnable Neutron Absorbing Material: A Thermodynamic Treatment in Support of Advanced CANDU Reactor Fuel, *A.S. Blackier[1], E.C. Corcoran[1], M.H. Kaye[2], B.J. Lewis[1] and W.T. Thompson[1] ([1] Royal Military College of Canada), ([2] University of Ontario Institute of Technology)*

14:45 Coffee Break

15:10 Internal Gas Pressure and Circumferential Ridge Sheath Strains in CANFLEX-ACR Fuel, *A. Gill and D. Evens (AECL Sheridan Park)*

15:35 Selection of Instruments Used for Vibration Measurement of Fuel Bundles in a Pressure Tube under CANDU Reactor Operating Conditions, P. Alavi, K. Chakraborty and D. Gossain *(AECL Sheridan Park)*

Monday, 2010 October 18, 1:30 PM-4:00 PM
Technical Session #M2, Room - Oakes South
Fuel Modelling and Computer Code Development I
Chair: S. Livingstone (AECL),
D.N. Sah (Bhabha Atomic Research Centre)

13:30 Numerical Verification of Equilibrium Chemistry Software within Nuclear Fuel Performance Codes, *M.H. Piro, B.J. Lewis, W.T. Thompson (Royal Military College of Canada), S. Simunovic and T.M. Besmann (Oak Ridge National Laboratory)*

13:55 Modelling CANDU Fuel Element and Bundle Behaviour for In- and Out-Reactor Performance of Intact and Defective Fuel, *K. Shaheen, A.D. Quastel, J.S. Bell, B.J. Lewis, W.T. Thompson and E.C. Corcoran (Royal Military College of Canada)*

14:20 Review of High Temperature Thermochemical Properties and Application in Phase-Field Modelling of Incipient Melting in Defective Fuel, *M.J. Welland, B.J. Lewis and W.T. Thompson (Royal Military College of Canada)*

14:45 Coffee Break

15:10 Framework for Sheath Hydriding Model for Defective Nuclear Fuel, *G. Bruni (Oxford University, UK), B.J. Lewis and W.T. Thompson (Royal Military College of Canada)*

15:35 A Method to Model Hydrogen Precipitation, *L. Lai, Z. Xu and Q. Jiang (AECL Sheridan Park)*

Monday, 2010 October 18, 1:30 PM-4:00 PM
Technical Session #M3, Room - Oakes North West
Fuel Performance, Reliability and Operating Experience I
Chair: A.M. Manzer (PROMATION Nuclear), N. Saibaba (Nuclear Fuel Complex)

13:30 Wet Sipping System at Wolsong-1, *J.Y. Park, J.C. Shin, Y.C. Kim, C.H. Park, T.Y. Choi, C.J. Park (Korea Nuclear Fuel) and A.M. Manzer (AECL Sheridan Park)*

13:55 Ultrasonic Inspection of CANDU Fuel Bundles for Water Ingress in Compromised Fuel Element Sheaths, *W.H. Huang, T.W. Krause and B.J. Lewis (Royal Military College of Canada)*

14:20 Progress in Developing an On-Line Fuel-Failure Monitoring Tool for CANDU Reactors, *S. Livingstone (AECL Chalk River Laboratories), B.J. Lewis (Royal Military College of Canada), M. Ip (Bruce Power), F.C. Iglesias and A. Fitchett (CANDESCO)*

14:45 Coffee Break

15:10 A General Model for Predicting Coolant Activity Behaviour for Fuel-Failure Monitoring Analysis, *A. El-Jaby, B.J. Lewis, and W.T. Thompson (Royal Military College of Canada), F.C. Iglesias (CANDESCO) and M. Ip (Bruce Power)*

15:35 In-Reactor Degradation of Fuel and Cladding in Fuel Pins Operated with Weld Defects, *P. Mishra, V.P. Jathar, J.L. Singh, D.N. Sah, P.K. Shah, J.S. Dubey and S. Anantharaman (Bhabha Atomic Research Centre)*

Monday, 2010 October 18, 4:15 PM – 7:00 PM (latest)
Technical Tour

16:15 Shuttle buses departure from the hotel
(First come first serve)

16:45 Sir Adam Beck Generating Station
Group 1: 4:45 – 5:45 PM
Group 2: 5:00 – 6:00 PM
Group 3: 5:15 – 6:15 PM
Group 4: 5:30 – 6:30 PM
Maximum 30 persons for each group

Tuesday, 2010 October 19, 8:30 AM-10:00 AM
Plenary Session #2, Room - Oakes South
Special Interest Topics I
Chair: F. Iglesias (CANDESCO)

08:30 Short, Medium and Long Term Consequences of Inadequate Defect Fuel Management, *J.G. Roberts (Cantech Associates Ltd), R. Nashiem, M. McQueen and G. Ma (Bruce Power)*

09:00 Some Aspects on Security and Safety in a Potential Transport of a CANDU Spent Nuclear Fuel Bundle, in Romania, *G. Vieru (Institute for Nuclear Research)*

09:30 Overview of the UNENE/COG/NSERC Industrial Research Chair in Nuclear Fuel at the Royal Military College of Canada, *B.J. Lewis, W.T. Thompson and E.C. Corcoran (Royal Military College of Canada)*

10:00 Coffee Break

Tuesday, 2010 October 19, 10:20 AM-12:00 PM
Technical Session #T1, Room - Oakes North East
Advanced Fuel Cycles
Chair: Noel Harrison (AECL)

10:20 Post-Irradiation Examination of MOX Fuel with Varying Pu Homogeneity, *N. Harrison, T. Janathasing and F.C. Dimayuga (AECL Chalk River Laboratories)*

10:45 Post-Irradiation Examination of CANDU Fuel Bundles Fuelled with $(Th,Pu)O_2$, *M. Karam, F.C. Dimayuga and J. Montin (AECL Chalk River Laboratories)*

11:10 Safety Assessment on NUE Fuel Bundles for Demonstration Irradiation in a CANDU Reactor, *H.Z. Fan, Z. Li, N. Barkman, C. Cottrell and A. Ranger (AECL Sheridan Park)*

11:35 Post-Irradiation Examination of DUPIC Fuel, *M. Karam, N. Harrison and J. Montin (AECL Chalk River Laboratories)*

Tuesday, 2010 October 19, 10:20 AM-12:00 PM
Technical Session #T2, Room - Oakes South
Fuel Modelling and Computer Code Development II
Chair: D. Oh (AECL)

10:20 Iodine Stress Corrosion Cracking and Possible Mitigation Strategy for CANDU Fuel, *M.R. Kleczek, K. Shaheen, W.T. Thompson, B.J. Lewis (Royal Military College of Canada) and F.C. Iglesias (CANDESCO)*

10:45 Measurements of Elastic Modulus in Zr Alloys for CANDU Applications, *Z.L. Pan, N. Wang and Z. He (AECL Chalk River Laboratories)*

11:10 An Improved UO_2 Thermal Conductivity Model in the ELESTRES Computer Code, *G.G. Chassie, M. Tochaie and Z. Xu (AECL Sheridan Park)*

11:35 Structural and Mechanical Properties of Urania Oxidized to U_3O_8, *B. Szpunar, J.A. Szpunar (University of Saskatchewan), D. Oh (AECL Sheridan Park)*

Tuesday, 2010 October 19, 10:20 AM-12:00 PM
Technical Session #T3, Room - Oakes North West
Fuel Fabrication I
Chair: K. Chakraborty (AECL)

10:20 PHWR Fuel Fabrication with Imported Uranium – Procedures and Processes, *R.V.R.L.V. Rao, Sheela, A. Rameswara Rao, G.V.S. Hemantha Rao and R.N. Jayaraj (Nuclear Fuel Complex)*

10:45 Use of Servo Controlled Weld Head for End Closure Welding, *S.K. Pathak, D.S. Setty, A. Rameswara Rao, G.V.S. Hemantha Rao and R.N. Jayaraj (Nuclear Fuel Complex)*

11:10 Fabrication of Simulated Intermediate-Burnup ACR Fuel in the RFFL, *D. Woods, G. Cota-Sanchez and F.C. Dimayuga (AECL Chalk River Laboratories)*

11:35 Impact of Chemical Quality of Source Material on Sintering Recovery, *E. Venkatesam, T.K. Sinha and D. Pramanik (Nuclear Fuel Complex)*

Tuesday, 2010 October 19, 12:00 PM-1:30 PM
Lunch, Hennepin Ballroom

12:30 Luncheon Speaker - A. Thorne (Cameco)

Tuesday, 2010 October 19, 1:30 PM-3:10 PM
Technical Session #T4, Room - Oakes North West
Spent Fuel Management
Chair: P. Alavi (AECL)

13:30 CANDU Fuel Rod Behaviour During Dry Storage, *A.C. Marino (Comisión Nacional de Energía Atómica)*

13:55 Spent Fuel Storage Bay, Water Temperature Response under Full Reactor Discharge, *A. Gidi (SNC-Lavalin Nuclear) and A. Al-zubaidi (McMaster University)*

14:20 Delayed Hydride Cracking Properties of the Endplate Resistance Welds of CANDU Fuel Bundles, *G.K. Shek, B.S. Wasiluk (Kinectrics Inc.), J. Freire-Canosa (Nuclear Waste Management Organization) and T. Lampman (AMEC Nuclear Safety Solutions)*

14:45 FEAT 4.1: Modeling of Sheath Oxidation and Heat Flow in CANDU Fuel Elements, *Q. Jiang, L. Lai, Z. Xu and U.K. Paul (AECL Sheridan Park)*

15:10 Coffee Break

Tuesday, 2010 October 19, 3:30 PM-5:10 PM
Technical Session #T5, Room - Oakes North West
Fuel Fabrication II
Chair: A. Pant (Cameco Fuel Manufacturing)

15:30 Advances in the Manufacture of Clad Tubes and Components for PHWR Fuel Bundle, *N. Saibaba, S.K. Jha, B. Chandrasekhar, S. Tonpe and R.N. Jayaraj (Nuclear Fuel Complex)*

15:55 Resistance Welding Equipment Manufacturing Capability for Exports, *V.S. Sastry, Y.S. Raju, A.K. Somani, D.S. Setty, A. Rameswara Rao, G.V.S. Hemantha Rao and R.N. Jayaraj (Nuclear Fuel Complex)*

16:20 Automation in Inspection of PHWR Fuel Elements & Bundles at Nuclear Fuel Complex, *M.N.V. Viswanath, B. Kamalesh Kumar , K.S. Subramanian, A Lakshminarayna and A.V. Ramana Rao (Nuclear Fuel Complex)*

16:45 Beryllium Brazing Considerations in CANDU Fuel Bundle Manufacture, *J. Harmsen (Cameco)*

Tuesday, 2010 October 19, 1:30 PM-5:10 PM
Technical Session #T6, Room - Oakes South
Fuel Modelling and Computer Code Development III
Chair: Z. Xu (AECL), G. Horhoianu (INR)

13:30 Longitudinal Ridging Tests for ACR-1000 Fuel Sheathing, *M. Bates*, *S. Abbas, K. Chakraborty, X. Wang and Z. Xu (AECL Sheridan Park)*

13:55 LONGER: A Computer Program for Longitudinal Ridging and Axial Collapse Assessment of CANDU Fuel, *U.K. Paul, Z. Xu, S. Xu, X. Wang, K. Chakraborty (AECL Sheridan Park)*

14:20 FEAST 3.1: Finite Element Modeling of Sheath Deformation Such as Longitudinal Ridging and Collapse into Axial Gap, *X. Wang, Z. Xu, Y-S. Kim, L. Lai, G. Cheng and S. Xu (AECL Sheridan Park)*

14:55 ELESTRES 2.1 Computer Code for High Burnup CANDU Fuel Performance Analysis, *G.G. Chassie, Q. Jiang, H. Liu, M. Tochaie and Z. Xu (AECL Sheridan Park)*

15:10 Coffee Break

15:30 Development of a Rigidity Enhancement Factor (REF) Correlation for Use in AECL Fuel Codes, *S. Xu, Z. Xu and Q. Jiang (AECL Sheridan Park)*

15:55 Computation of Actinide Pourbaix Diagrams at 298K and 550K (U, Np, Pu, Am, Cm – H_2O), *M.H. Piro, G.M.F. Bruni, B.J. Lewis, W.T. Thompson (Royal Military College of Canada), F.C. Iglesias (CANDESCO), M.A. Guoping, R. Nashiem (Bruce Power) and J.G. Roberts (Cantech Associates Ltd.)*

16:20 2009 Creep Correlations of Zircaloy for ACR-1000 Fuel, *A. Sun, M. Tayal, P. Reid and S. Abbas (AECL Sheridan Park)*

16:45 BEAM 1.7: Development for Modelling Fuel Element and Bundle Buckling Strength, *G. Cheng, S. Xu, Z. Xu and U.K. Paul (AECL Sheridan Park)*

Tuesday, 2010 October 19, 1:30 PM-5:10 PM
Technical Session #T7, Room - Oakes North East
Fuel Safety and Operational Margin Improvement
Chair: T. Daniels (OPG), A. Waheed (AECL)

13:30 Fuel Behaviour and Fission-Product Release in the Power Pulse 1 Experiment, *R.S. Dickson, A.I. Belov, M.D. Gauthier, R.T. Peplinskie and C.A. Buchanan (AECL Chalk River Laboratories)*

13:55 Fuel Safety Analysis Following Feeder Break Accident for Refurbished Wolsong 1, *J. Jung and J.H. Park (Korea Atomic Energy Research Institute)*

14:20 Methodology for Fission Product Release Calculations During an ACR-1000 End-Fitting Failure Event, *W. Zhu, F.L. Huang and Z. Bilanovic (AECL Sheridan Park)*

14:45 ACR-1000 End-Temperature Peaking Analysis under Postulated Accident Conditions, *F. Gao, Q.M. Lei and Z. Bilanovic (AECL Sheridan Park)*

15:10 Coffee Break

15:30 CATHENA Modelling of ACR-1000 Fuel Handling Events, *M. Maltchevski, H.Z. Fan and F. Huang (AECL Sheridan Park)*

15:55 Prediction of Power-Ramp Defects in CANDU Fuel, *P. Gillespie, S. Wadsworth (AMEC Nuclear Safety Solutions) and T. Daniels (Ontario Power Generation)*

16:20 Validation Summary for REDOU Code, *D. Oh, E. Zariffeh (AECL Sheridan Park), R. Dickson (AECL Chalk River Laboratories) and S. Girgis (AECL Sheridan Park)*

Tuesday, 2010 October 19, 6:00 PM-9:00 PM
Banquet Dinner

Location: Skylon Tower

20:00 Key Note Address - J. Hopwood (AECL)

Wednesday, 2010 October 20, 8:30 AM-11:45 AM
Technical Session #W1, Room - Oakes South
Fuel Modelling and Computer Code Development IV
Chair: S. Girgis (AECL), C. Marino (CNEA)

08:30 BOW 4.0: Modeling Deformations of Fuel String with Channel Ageing Effects, *S. Xu, Z. Xu, G. Cheng, L. Lai (AECL Sheridan Park), S. Yu and D. Wen (Ryerson University)*

08:55 Deformation of Unirradiated CANDU Fuel Elements under Bending Loads, *J. Freire-Canosa (Nuclear Waste Management Organization), A. Popescu, and T. Lampman (AMEC Nuclear Safety Solutions)*

09:20 Numerical Model for Thermal and Mechanical Behaviour of a CANDU 37-Element Bundle, *L. Jiang, K. MacKay (Martec Limited) and R. Gibb (Canadian Nuclear Safety Commission)*

09:45 Modeling of Transient Dynamic Bundle Deformation Using Time Integration Scheme, *S. Xu, Y-S. Kim and Z. Xu (AECL Sheridan Park)*

10:10 Coffee Break

10:30 Modeling of Fuel Bundle Deformation at High Temperatures: Requirements, Models and Steps for Consideration, *S. Xu, Z. Xu, H. Fan (AECL Sheridan Park) and T. Nitheanandan (AECL Chalk River Laboratories)*

10:55 CFD Simulations of the Single-Phase and Two-Phase Coolant Flow of Water inside the Original and Modified CANDU 37-Element Bundles, *F. Abbasian, G.I. Hadaller, and R.A. Fortman (Stern Laboratories Inc)*

11:20 FEED 1.6: Modelling of Hydrogen Diffusion and Precipitation in Fuel Bundle Zircaloy Components, *L. Lai, Z. Xu, Q. Jiang, G. Cheng and S. Xu (AECL Sheridan Park)*

Wednesday, 2010 October 20, 8:30 AM-11:45 AM
Technical Session #W2, Room - Oakes North West
Fuel Performance, Reliability and Operating Experience II
Chair: H. Fan (AECL), H-C. Suk (CNSC)

08:30 CNSC Fuel Oversight Programme, *A. El-Jaby, K. Conlon, W. Grant and M. Couture (Canadian Nuclear Safety Commission)*

08:55 Pickering NGS B – Unit 6 Strategy to Remove Fuel Defects during Operation at High Power, *M. Dobrean and E. Sorin (Pickering NGS, Ontario Power Generation)*

09:20 Fuel Defect Investigation at CERNAVODA NGS in Romania, *E. Suk, A.M. Manzer (AECL Sheridan Park), P.J. Valliant, J. Pyoli (AECL Chalk River Laboratories), S. Holostencu, N. Baraiaru, D. Stanila and A. Gisnac (SNN)*

09:45 Assessing the Impact of Increased Transport Time between Failed Fuel and Delayed Neutron Detectors, *A. Fitchett, F. Iglesias (CANDESCO), B.J. Lewis and K. Shaheen (Royal Military College of Canada)*

10:10 Coffee Break

10:30 Examination of Irradiated PHWR Fuel Pins Subjected to Isothermal Heating at 700-900°C inside the Hot Cells, *D.N. Sah, Prerna Mishra, P.M. Satheesh, Sunil Kumar and S. Anantharaman (Bhabha Atomic Research Centre)*

10:55 Preliminary Assessment of Noble Gas Bundle Tagging Using a Partial Krypton Backfill, *B. Pollack, B.J. Lewis and D. Kelly (Royal Military College of Canada)*

11:20 Thermal Aspects of Alternative Fuels for Use in Supercritical Water-Cooled Nuclear Reactors, *L. Grande, W. Peiman, B. Villamere, A. Rodriguez-Prado, S. Mikhael, L. Allison and I. Pioro (University of Ontario Institute of Technology)*

Wednesday, 2010 October 20, 11:45 AM-12:15 PM
Plenary Session #3, Room - Oakes South
CANDU Fuel Cycle Options
Chair: S. Palleck (AECL)

11:45 Unique Fuel Cycle Capabilities of CANDU, *C. Cottrell, R. Bodner and S. Kuran (AECL Sheridan Park)*

12:45 Closing Remark, *S. Palleck (Conference Chair)*

Wednesday, 2010 October 20, 12:50 PM-3:00 PM
Lunch, Hennepin Ballroom

13:30 Luncheon Speaker - P. Ottensmeyer (Professor Emeritus, University of Toronto)

"CANDU Used Fuel 'Waste' in Canada: A $36 Trillion Energy Resource in Fast Reactors"

Monday, 2010 October 18, 9:00 AM-12:00 PM
Plenary Session #1
International Experience with CANDU Fuel

Consolidating Indigenous Capability for PHWR Fuel Manufacturing in India

R.N. Jayaraj (Nuclear Fuel Complex)

Since inception of Nuclear Power Programme in India greater emphasis was laid on total self-reliance in Fuel manufacturing. For Pressurized Heavy Water Reactors (PHWRs), which forms a base for the first stage of the programme, an integrated approach was adopted encompassing different areas of expertise – Design, Construction and Operation of PHWRs; Heavy Water production and Fuel Design & Manufacturing technologies. For the first PHWR constructed about 35 years back with the Canadian collaboration, known as Rajasthan Atomic Power Station (RAPS), half the core requirement of fuel was met from the fuel manufactured for the first time in India. Since then the fuel production capabilities were enhanced by setting up an industrial scale fuel manufacturing facility – Nuclear Fuel Complex (NFC) at Hyderabad, India during early ‘70s. NFC has been continuously expanding its capacities to meet the fuel demand of all the PHWRs constructed and operated by Nuclear Power Corporation of India Limited (NPCIL).

Presently, fifteen PHWR 220 MWe units and two PHWR 540 MWe units are in operation and one more PHWR 220 MWe unit is in advanced stage of commissioning in India. While continuously engaged in the manufacture of fuel for these reactors, NFC has been upgrading the production lines with new processes and quality assurance systems. In order to multiply the production capacities, NFC has embarked on developing indigenous capability for design and building of special purpose process equipment for Uranium dioxide powder production, pelletisation and final assembly operations. Some of the equipment having state-of-the-art features includes dryers/furnaces for UO_2 powder, presses/ sintering furnaces for pelletisation and resistance welding equipment/ machining stations for assembly operations. In addition, several campaigns were taken over the years for manufacturing PHWR fuel bundles containing reprocessed Uranium, Thoria and slightly enriched Uranium.

The paper summarises various actions taken that resulted in successfully consolidating indigenous capability for PHWR fuel manufacturing in India.

Load Following Tests on CANDU Type Fuel Elements in TRIGA Research Reactor of INR Pitesti

G. Horhoianu (Institute for Nuclear Research), S.J. Palleck (AECL Sheridan Park) and D. Ionescu (Institute for Nuclear Research)

Two load following (LF) tests on CANDU-type fuel elements were performed in the TRIGA Research Reactor of INR Pitesti, where the tests were designed to represent fuel in a CANDU reactor operating in a load following regime. In the first LF test the designated "78R" fuel element successfully experienced 367 power cycles, mostly between 23 and 56 kW/m average linear power. In the second LF test, developed under INR-AECL co-operation, the fuel element designated as "ME01" withstood 200 power cycles from 27 to 54 kW/m average linear power, as well as additional ramps due to reactor trips and restarts during the test period. This experimental program is ongoing at INR Pitesti. Both LF tests were simulated with finite element computer codes in order to evaluate Stress Corrosion Fatigue (SCF) of the cladding arising from expansion and contraction of the pellets. New LF tests are planned to be performed in order to establish the limits and capabilities for CANDU fuel in LF conditions. This paper presents the results of the LF tests performed in the INR TRIGA Research Reactor compared with the analytical assessment for SCF conditions and their relation to CANDU fuel performance in LF conditions.

CANDU-6 Fuel Bundle Fabrication and Advanced Fuels Development in China

Wang Jun (China North Nuclear Fuel Corporation)

In recent years, China North Nuclear Fuel Corporation (CNNFC) has introduced several modifications to the manufacturing processes and the production line equipment. This has been beneficial in achieving a very high level of quality in the production of fuel bundles. Since 2008 CNNFC has participated in a multi party project with the goal of developing advanced fuels for use in CANDU reactors. Other project team members include the Nuclear Power Institute of China (NPIC), Third Qinshan Nuclear Power Company (TQNPC) and Atomic Energy of Canada Ltd (AECL). This paper will present the improvements developed during the manufacture of natural fuel bundles and advanced fuels.

Development and Commercial Implementation of CANFLEX Fuel in Korea

J.H. Park and J.Y. Jung (Korea Atomic Energy Research Institute)

Since the early 1990's, Korea Atomic Energy Research Institute (KAERI) and Atomic Energy of Canada Limited (AECL) have cooperated to develop, verify, and demonstrate an advanced CANDU fuel, called a CANFLEX-NU (CANdu FLEXible-Natural Uranium) fuel, which has been specially designed so as to compensate the deterioration of the heat transfer rate in a crept pressure tube and also to enhance the safety margin of it by a reduction of the maximum linear element power of a fuel bundle. Now, the CANFLEX-NU fuel is ready to be commercialized in CANDU-6 because its design and demonstration irradiation have been completed in both Korea and Canada.

Currently, some CANDU plants are being refurbished as they are approaching the end of their life span. One of the refurbished CANDUs is Wolsong unit 1 in Korea. It has been operating for 26 years and the operating power before its shutdown for refurbishment was less than 90% of its full power because of a reduction of the margin for the ROP trip set point. One of the most appropriate ways, up-to now, to overcome such a power de-rating due to a crept pressure tube is the introduction of the CANFLEX-NU fuel into a CANDU.

Korea is now seriously considering the commercialization of the CANFLEX-NU fuel in the Wolsong units, especially units 2, 3 and 4 which have been in operation for over 13 years. Before commercializing the CANFLEX-NU fuel in the Wolsong units, there has been requested for an economic benefits analysis for the CANFLEX-NU fuel in the Wolsong units because the Uranium price at present is very different from that during the development phase of the CANFLEX-NU fuel.

The present paper shows the important characteristics of the CANFLEX NU fuel, such as the physics, thermal-hydraulics fuel performance and safety and the enhancement of reactor safety and operating performance are compared to those of the 37-element fuel bundle. Also, the results of the economic benefits analysis for the full core loading of the CANFLEX-NU fuel into Wolsong units in Korea are presented.

CNSC Approach to the Review of Nuclear Fuel Systems Design for New Nuclear Power Plants

H-C. Suk and M. Couture (Canadian Nuclear Safety Commission)

In Canada, over the past 2 years between 2007 and 2009 there has been renewed interest in the nuclear energy option, and in particular in Ontario where the province has been conducting a bidding process to choose a vendor. The Canadian Nuclear Safety Commission (CNSC) is undertaking a number of preparatory activities to ensure that it is ready to licence whichever technology is submitted by an applicant.

These preparatory activities have included the development of regulatory document RD-337 whose purpose is to set out the expectations of the Canadian Nuclear Safety Commission (CNSC) with respect to the design of new water-cooled nuclear power plants (NPPs). RD-337 represents the CNSC's adoption of the principles set forth by the International Atomic Energy Agency (IAEA) in NS-R-1, *Safety of Nuclear Power Plants: Design*, and the adaptation of those principles to align with Canadian expectations. RD-337 was approved by the Commission and issued in November 2008.

Other documents which are either completed or under development include the "GD-369 - Construction Licence Applications for Nuclear Power Plants: Guidelines" and "Staff Review Procedures (SRPs)". GD-369 is a document that describes the scope of the CNSC staff review and the information that is required to be submitted in an application for a licence to construct a nuclear power plant. SRPs are being written for all phases of the licensing process, including the environmental assessment, the licence to prepare site, and the licence to construct (SRPs for a licence to operate will be prepared in the future). The SRPs outline the principal technical criteria that will guide staff in determining the adequacy of the submissions. For the licence to construct, there are SRPs that address all aspects of the nuclear power plant design.

The focus of this paper is on the CNSC Approach to the Design Review of Nuclear Fuel Systems for New Nuclear Power Plants, and will be limited to the thermal, mechanical and material design of the fuel system. Although to the extent practicable, the documents RD-337, GD-369 and the SRG for fuel system designs is technology-neutral with respect to water-cooled reactors, this paper is focused on the review of the design of CANDU fuel bundles. Clauses of RD-337 which are relevant to CANDU fuel designs will be listed. Also discussed in this paper are the objectives of the review, the review criteria, and the information and data expected in the licensee's application.

Monday, 2010 October 18, 1:30 PM-4:00 PM
Technical Session #M1
Fuel Design

Technical Basis for ACR-1000® Fuel Acceptance Criteria

A. Sun and M. Tayal (AECL Sheridan Park)

The ACR fuel design acceptance criteria (FAC) provide assurance that the fuel is not damaged during normal operating conditions and anticipated operational occurrences to meet its design and regulatory requirements. The criteria cover three broad areas: criteria on thermal integrity; criteria on structural integrity; and criteria on dimensional compatibility with the fuel channel. These criteria are supported with solid and rich technical bases from relevant and applicable experience of fuel design and operation for CANDU and light water reactors around the world. This paper presents these technical bases. Specifically, this paper:

a) Discusses the completeness of the damage mechanisms and the associated acceptance criteria for application to ACR-1000 fuel;
b) Provides technical bases for Minimum Acceptable Margins that are embedded in each individual criterion for assessing ACR-1000 fuel performance.

CNSC Approach to the Review of Reactor Core Thermalhydraulic Design for New Nuclear Power Plants

Y. Guo, N. Hammouda, H. Tezel and M. Couture (Canada Nuclear Safety Commission)

The Canadian Nuclear Safety Commission (CNSC) has developed staff review procedures, providing the CNSC reviewer with guidance on acceptable ways of demonstrating that CNSC requirements and expectations are met by a new nuclear power plant design. This paper discusses the review procedure of reactor core thermalhydraulic design, which is part of the CNSC staff review procedures for a licence to construct. It provides a definition of the topical areas of review, a concept of thermalhydraulic safety limits, a description of the review criteria, and a list of the information and the data expected in the licensee's application. These contents are presented, while practical, in a manner that is technology neutral for water-cooled reactor designs including PWRs and CANDU reactors. Separate discussions are provided for the areas that are design specific. The intent of this paper is to inform applicants of the new review processes at the CNSC and to state the CNSC expectations on future construction application submissions.

Coolant Solubility of Burnable Neutron Absorbing Material: A Thermodynamic Treatment in Support of Advanced CANDU Reactor Fuel

A.S. Blackier[1], E.C. Corcoran[1], M.H. Kaye[2], B.J. Lewis[1] and W.T. Thompson[1]
([1] Royal Military College of Canada), ([2] University of Ontario Institute of Technology)

The Advanced CANDU Reactor (ACR) employs a newly-designed fuel bundle that contains a Burnable Neutron Absorbing (BNA) material in the central position. The BNA is composed of elements with high neutron absorption cross sections (Gd and Dy) dissolved in zirconia. If a sheath failure were to occur, there is concern that the possible leaching of these elements into the coolant could cause a reactivity re-distribution. To address this concern, the solubility of Gd and Dy over a range of pH and temperatures has been examined in the context of the possible existence of hydroxyl complex ions. Estimated thermodynamic properties are proposed that provide the means to compute the low Dy and Gd concentrations in the reactor coolant in the event of a BNA cladding breach.

Internal Gas Pressure and Circumferential Ridge Sheath Strains in CANFLEX-ACR Fuel

A. Gill and D. Evens (AECL Sheridan Park)

ACR-1000® fuel is designed to operate with an average exit burnup of up to 20 MWd/kgU. This average exit burnup is in excess of the burnup in current CANDU® designs. The increased burnup will result in higher fission product inventory. This paper reports the evaluation of fuel sheath strains and internal gas pressures during normal operation. The internal gas pressures and sheath strains are assessed using the ELESTRES computer code. Predicted strains and pressures have adequate margin to the relevant acceptance criteria.

Selection of Instruments Used for Vibration Measurement of Fuel Bundles in a Pressure Tube under CANDU Reactor Operating Conditions

P. Alavi, K. Chakraborty and D. Gossain (AECL, Sheridan Park)

Vibration characteristics of CANDU fuel bundle and fuel elements is a key parameter considered in the design of a fuel bundle. Out-reactor frequency and temperature sweep tests, under reactor operating conditions, are performed to verify vibration characteristics of CANDU fuel bundles. Several options have been considered in the selection of vibration instrumentation to perform out-reactor frequency and temperature sweep tests.

This paper compares the benefits and disadvantages of various vibration instruments and summarizes the rationale behind the selection of instruments used for vibration measurements over a range of temperature and pressure pulsation frequencies. The conclusions are presented from the bench tests performed, which confirm the use of the selected instruments.

Monday, 2010 October 18, 1:30 PM-4:00 PM
Technical Session #M2
Fuel Modelling and Computer Code Development I

Numerical Verification of Equilibrium Chemistry Software within Nuclear Fuel Performance Codes

M.H. Piro, B.J. Lewis, W.T. Thompson (Royal Military College of Canada),
S. Simunovic and T.M. Besmann (Oak Ridge National Laboratory)

A numerical tool is in an advanced state of development to compute the equilibrium compositions of phases and their proportions in multi-component systems of importance to the nuclear industry. The resulting software is being conceived for direct integration into large multi-physics fuel performance codes, particularly for providing transport source terms, material properties, and boundary conditions in heat and mass transport modules. Consequently, any numerical errors produced in equilibrium chemistry computations will be propagated in subsequent heat and mass transport calculations, thus falsely predicting nuclear fuel behaviour. The necessity for a reliable method to numerically verify chemical equilibrium computations is emphasized by the requirement to handle the very large number of elements necessary to capture the entire fission product inventory. A simple, reliable and comprehensive numerical verification method is presented which can be invoked by any equilibrium chemistry solver for quality assurance purposes.

Modelling CANDU Fuel Element and Bundle Behaviour for In- and Out-Reactor Performance of Intact and Defective Fuel

K. Shaheen, A.D. Quastel, J.S. Bell, B.J. Lewis, W.T. Thompson
and E.C. Corcoran (Royal Military College of Canada)

A proposed platform-based fuel performance code integrates treatments for intact fuel performance and defective fuel oxidation. The intact fuel performance code is verified against the ELESTRES and ELESIM industry-standard toolset for heat transport, fission gas diffusion, and deformation and interaction of the pellet and sheath. The oxidation model integrates equilibrium thermodynamics into oxygen transport equations and is validated against coulometric titration data from Chalk River Laboratories. Ongoing work aims to incorporate the intact fuel performance model into a bundle heat transport and deformation model, and to apply the oxidation to the design and analysis of an out-reactor instrumented fuel oxidation experiment.

Review of High Temperature Thermochemical Properties and Application in Phase-Field Modelling of Incipient Melting in Defective Fuel

M.J. Welland, B.J. Lewis and W.T. Thompson (Royal Military College of Canada)

In a defective fuel element, fuel oxidation may occur, which may affect the thermal performance of the fuel element thereby increasing the potential for centreline melting. A review of UO_{2+x} material properties is presented, and methods of extrapolation to melting conditions are, where necessary, recommended based on mechanistic considerations and self-consistency among the parameters. Properties are implemented in a phase-field model derived through the theory of irreversible processes, to simulate coupled heat and mass transport in the presence of a dynamic, non-congruent phase change. Simulation results are presented for centreline melting in operational, defective nuclear fuel.

* Note: Author M.J. Welland is currently at the EC-JRC, Institute for Transuranium Elements, Karlsruhe, Germany.

Framework for Sheath Hydriding Model for Defective Nuclear Fuel

G. Bruni (Oxford University, UK), B.J. Lewis and W.T. Thompson (Royal Military College of Canada)

Models have been developed to describe secondary hydriding phenomena. Thermodynamic models for the Zr-H and Zr-O-H systems provide terminal solubility of dissolved hydrogen and oxygen in the α-Zr phase as a function of temperature and partial pressures. A finite-element kinetic model of the Sawatzky and Vogt experiment reports the redistribution of hydrogen in a Zircaloy-2 sample placed in a temperature gradient; this confirms the hydrogen diffusion coefficient. A remedy to suppress secondary hydriding by maintaining passivity of the zirconium will also be presented. The concept involves slightly oxidizing the surface of the fuel pellets (UO_{2+x}) immediately prior to loading the pellets into the fuel sheath. Interactions with the hydrogen and the CANLUB provide the means of oxygen transfer from the pellet surface to the inner surface of the fuel sheath in the event that bare zirconium has been exposed as a result of expansion during a power ramp.

A Method to Model Hydrogen Precipitation

L. Lai, Z. Xu and Q. Jiang (AECL Sheridan Park)

Hydrogen precipitation or hydriding of Zircaloy component in CANDU® fuel is a complex problem and difficult to model. Due to the nature of the hydriding process, conventional methods cannot solve the problem and a special method is required. This paper describes a method to model hydriding, including theory, solution method and illustrative examples.

The total hydrogen concentration in a solid solution (i.e., Zircaloy) includes dissolved hydrogen (freely flowing mass) concentration C and hydride (precipitate mass) concentration. The diffusion differential equation only deals with the dissolved hydrogen.

In order to use the diffusion differential equation that can only deal with the dissolved hydrogen, the hydriding is treated as a sink or source of hydrogen corresponding to the local temperature. In two phase region, since the solid solution at every point is assumed to be in equilibrium with the terminal solid solubility (TSS) at the corresponding temperature, there is a hydriding rate existing that under this hydriding rate the dissolved hydrogen concentration is equal to TSS (Terminal Solid Solubility). This precipitation rate can be defined as positive for sinks and negative for sources in the diffusion differential equation. This hydriding rate can be used in a finite-element computer program to define the hydrogen sink or source rate.

Monday, 2010 October 18, 1:30 PM-4:00 PM
Technical Session #M3
Fuel Performance, Reliability and Operating Experience I

Wet Sipping System at Wolsong-1

J.Y. Park, J.C. Shin, Y.C. Kim, C.H. Park, T.Y. Choi, C.J. Park (Korea Nuclear Fuel) and A.M. Manzer (AECL Sheridan Park)

After many years of operation, the on-power failed fuel detection and location systems along with alarm area gamma monitors at Wolsong-1 have successfully demonstrated that most, if not all, defective and suspect fuel bundles can be located before discharge to the fuel bay. Today, discharged bundles are now being transferred from the fuel bay to the AECL designed Modular Air-Cooled Storage (MACSTOR) canister facilities. Since these canisters are licensed for storing intact fuel bundles only, a procedure was needed at Wolsong-1 to separate any suspect or defective bundles that do not release fission products in detectable quantities. Therefore, KNF designed and built a wet sipper to enclose an irradiated bundle inside a sealed container at the bottom of the fuel bay. Various techniques were then used to enhance the release of water soluble fission products from defective fuel elements before circulating water samples from the immediate vicinity of an irradiated fuel bundle to an inspection station located at the top of the fuel bay. Any water samples with elevated levels of gamma activity were direct indications of a fuel cladding breach. The presence of defective fuel elements were then verified by visual inspection. The system performance test was performed in the Wolsong-1 nuclear power plant on March 2009.This paper describes the results of the wet sipping tests.

Ultrasonic Inspection of CANDU Fuel Bundles for Water Ingress in Compromised Fuel Element Sheaths

W.H. Huang, T.W. Krause and B.J. Lewis (Royal Military College of Canada)

In the rare occurrence of a defective fuel, the primary coolant can enter into the element and flash to steam in the fuel-to-sheath gap, where the steam can subsequently react with the fuel matrix. Visual techniques are normally used for the post-irradiation inspection of discharged CANDU fuel bundles to help identify fuel failures. In this work, a more sensitive method based on underwater angled-beam inspections demonstrates that the introduction of water into the fuel elements acts as a couplant for sound waves, thereby providing for a clear demarcation of the fuel pellets within the element in observed scans. This study demonstrates that the inspection of the outer-ring (i.e., higher-powered) elements in a fuel bundle is possible.

Progress in Developing an On-Line Fuel-Failure Monitoring Tool for CANDU Reactors

S. Livingstone (AECL Chalk River Laboratories), B.J. Lewis (Royal Military College of Canada), M. Ip (Bruce Power), F.C. Iglesias and A. Fitchett (CANDESCO)

This paper describes the continued development of an on-line defected fuel diagnostic tool for CANDU reactors. One of the key capabilities of this tool is the ability to estimate the power and number of defects in the core based on the Gaseous Fission Product Monitoring System (GFP), and grab sample data. To perform this analysis, a clear understanding of the empirical diffusion coefficient *D*' [s^{-1}] is required. This paper examines two existing models for *D*' and presents a new model based on ^{133}Xe release data from commercial reactor experience. The new model is successfully applied to commercial data to demonstrate a novel technique for extracting defected fuel element power from GFP data during a reactor power change. The on-line defected fuel diagnostic tool is in a developmental stage, and this paper reports the latest enhancements.

A General Model for Predicting Coolant Activity Behaviour for Fuel-Failure Monitoring Analysis

A. El-Jaby, B.J. Lewis, W.T. Thompson (Royal Military College of Canada), F.C. Iglesias (CANDESCO) and M. Ip (Bruce Power)*

A mathematical treatment has been developed to predict the release of volatile fission products from operating defective nuclear fuel elements. The fission product activity in both the fuel-to-sheath gap and primary heat transport system as a function of time can be predicted during all reactor operating conditions, including: startup, steady-state, shutdown, and bundle-shifting manoeuvres. In addition, an improved ability to predict the coolant activity of the ^{135}Xe isotope in commercial reactors is discussed. A method is also proposed to estimate both the burnup and the amount of tramp uranium deposits in-core. The model has been implemented as a stand-alone code written in the C++ computer programming language using a finite-difference variable-mesh numerical scheme for the mass transport equations in the UO_2 fuel grain. The model has been validated against in-reactor experiments conducted with defective fuel elements containing natural and artificial failures at the Chalk River Laboratories. Lastly, the model has been benchmarked against a defective fuel occurrence in a commercial reactor.

* Note: Author A. El-Jaby is currently at the Canadian Nuclear Safety Commission.

In-Reactor Degradation of Fuel and Cladding in Fuel Pins Operated with Weld Defects

P. Mishra, V.P. Jathar, J.L. Singh, D.N. Sah, P.K. Shah, J.S. Dubey and S. Anantharaman (Bhabha Atomic Research Centre)

A PHWR fuel pin having an incomplete fusion defect in the end plug weld operated in the reactor in the failed condition for a period of 710 days and accumulated a burnup of 4400 MWd/tU. Detailed non destructive and destructive PIE was carried out on the fuel pin to understand the nature and extent of degradation of fuel and cladding in this fuel pin. PIE included visual examination, ultrasonic testing, metallography of a large number of transverse sections of the fuel pin, beta gamma autoradiography, microhardness measurement of the cladding, ring tension test on the cladding.

Tuesday, 2010 October 19, 8:30 AM-10:00 AM
Plenary Session #2
Special Interest Topics

Short, Medium and Long Term Consequences of Inadequate Defect Fuel Management

J.G. Roberts (Cantech Associates Ltd.), R. Nashiem, M. McQueen and G. Ma (Bruce Power)

Defect fuel pencils result in short, medium and long term consequences to the environment within and external to the nuclear power station. The paper will describe these consequences and specify the Defect Fuel Management Practices required to avoid these consequences.

Some Aspects on Security and Safety in a Potential Transport of a CANDU Spent Nuclear Fuel Bundle, in Romania

G. Vieru (Institute for Nuclear Research)

Each Member States (MS) is responsible for the security and safety of radioactive material during transport, since radioactive material is most vulnerable during transport. The paper presents some aspects on security and safety related to the potential transport of a CANDU Spent Nuclear Fuel (SNF) bundle from NPP CANDU Cernavoda to INR Pitesti. The possible environmental impact and radiological consequences following a potential event during transportation is analyzed, since the protection of the people and the environment is the essential goal to be achieved. Some testing for the package to be used for transportation will be also given.

Overview of the UNENE/COG/NSERC Industrial Research Chair in Nuclear Fuel at the Royal Military College of Canada

B.J. Lewis, W.T. Thompson and E.C. Corcoran (Royal Military College of Canada)

An Industrial Research Chair (IRC) in Nuclear Fuel at the Royal Military College of Canada (RMC) was established in September 2007 for five years in partnership with the CANDU Owners Group (COG), University Network of Excellence in Nuclear Engineering (UNENE), Natural Sciences and Engineering Research Council (NSERC) and Department of National Defence (DND) to promote university research to better understand nuclear fuel performance during normal and reactor accident conditions, including the behaviour of advanced and next generation fuel designs. This paper specifically highlights research projects presently underway which connect the specialties of researchers at the university to commercial requirements.

Tuesday, 2010 October 19, 10:20 AM-12:00 PM
Technical Session #T1
Advanced Fuel Cycles

Post-Irradiation Examination of MOX Fuel with Varying Pu Homogeneity

N. Harrison, T. Janathasing and F.C. Dimayuga (AECL Chalk River Laboratories)

The Atomic Energy of Canada Limited (AECL) Pu-containing mixed-oxide (MOX) fuel program includes fuel fabrication development, irradiation testing, post-irradiation examination (PIE), reactor physics and fuel-management studies. The BDL-446 experiment investigates one particular fabrication parameter, plutonium homogeneity, and its effect on fuel performance. Three different distributions of Pu in the UO_2 matrix were tested: pure PuO_2 particles within the matrix, regions of master mix particles (containing an intermediate Pu concentration in UO_2) within the matrix, and a homogeneous, solid solution of $(U+Pu)O_2$. The fuel pellets had a Pu content of 1.35 wt.% Pu in total heavy elements (HE), mixed with depleted uranium powder. The irradiation test began at outer element linear power ratings of ~52 kW/m, declining to a final linear power rating of ~21 kW/m at burnups of ~500 MWh/kgHE (21 MWd/kgHE). This paper discusses the PIE results of various fuel types. The PIE results showed that the performance of all three types of MOX fuel is typical of UO_2 fuel irradiated under similar conditions of power and burnup. There is a difference in the fission gas release among the different fuel types, with the MOX fuel containing regions of pure Pu appearing to have higher gas release relative to the other two.

Post-Irradiation Examination of CANDU Fuel Bundles Fuelled with $(Th,Pu)O_2$

M. Karam, F.C. Dimayuga and J. Montin (AECL Chalk River Laboratories)

AECL has extensive experience with thoria-based fuel irradiations as part of an ongoing R&D program on thorium within the Advanced Fuel Cycles Program. The BDL-422 experiment was one component of the thorium program that involved the fabrication and irradiation testing of six Bruce-type bundles fuelled with $(Th,Pu)O_2$ pellets. The fuel was manufactured in the Recycle Fuel Fabrication Laboratories (RFFL) at Chalk River allowing AECL to gain valuable experience in fabrication and handling of thoria fuel. The fuel pellets contained 86.05 wt. % Th and 1.53 wt. % Pu in $(Th, Pu)O_2$. The objectives of the BDL-422 experiment were to demonstrate the ability of 37-element geometry $(Th, Pu)O_2$ fuel bundles to operate to high burnups up to 1000 MWh/kgHE (42 MWd/kgHE), and to examine the $(Th, Pu)O_2$ fuel performance. This paper describes the post-irradiation examination (PIE) results of BDL-422 fuel bundles irradiated to burnups up to 856 MWh/kgHE (36 MWd/kgHE), with power ratings ranging from 52 to 67 kW/m. PIE results for the high burnup bundles (>1000 MWh/kgHE) are being analyzed and will be reported at a later date. The $(Th, Pu)O_2$ fuel performance characteristics were superior to UO_2 fuel irradiated under similar conditions. Minimal grain growth was observed and was accompanied by benign fission gas release and sheath strain. Other fuel performance parameters, such as sheath oxidation and hydrogen distribution, are also discussed.

Safety Assessment on NUE Fuel Bundles for Demonstration Irradiation in a CANDU Reactor

H.Z. Fan, Z. Li, N. Barkman, C. Cottrell and A. Ranger (AECL Sheridan Park)

The CANDU reactors have a unique fuel cycle capability that enables them to utilize alternative fuel options. One option currently being developed is Natural Uranium Equivalent (NUE) fuel. This fuel is not fabricated from natural uranium but from a mixture of depleted uranium and recycled uranium that follows the fundamental specifications for natural uranium (NU) fuel and mimics its performance. Recycling uranium from reprocessing facilities reintroduces the uranium into the fuel cycle and will improve the utilization rate of NU resources; ultimately improving the sustainability of fuel resources. The concept of using NUE is currently being implemented as a demonstration in the Qinshan CANDU units in China. The NUE fuel has been manufactured in China and utilized in a two-channel demonstration irradiation in one of the Qinshan CANDU units. A safety assessment has been performed which has concluded that there are no safety concerns or adverse implications of replacing NU fuel with NUE fuel for the two-channel demonstration irradiation. This paper is an overview of this safety assessment.

Post-Irradiation Examination of DUPIC Fuel

M. Karam, N. Harrison and J. Montin (AECL Chalk River Laboratories)

The DUPIC (**D**irect **U**se of spent **P**WR fuel **I**n **C**ANDU reactors) program is designed to demonstrate the feasibility of fabricating DUPIC fuel and irradiating it under CANDU operating conditions. The scope of the Canadian part in the program included the fabrication of three DUPIC fuel elements, irradiation testing in the NRU research reactor and assessment of DUPIC fuel performance relative to CANDU UO_2 fuel. AECL successfully fabricated three 37-element geometry DUPIC elements. The irradiation testing of the three DUPIC elements in the NRU reactor in a demountable element bundle commenced in 1999 and was completed in 2003. The three DUPIC elements were irradiated to burnups ranging from 251 MWh/kgHE (10 MWd/kgHE) to 517 MWh/kgHE (22 MWd/kgHE), with maximum powers ranging from 49 kW/m to 51 kW/m. Post-irradiation examination (PIE) of the elements was conducted in the hot cells at Chalk River Laboratories (CRL) after their discharge from the NRU loops. DUPIC fuel exhibited comparable sheath strain and fission-gas release (FGR) to that observed in similarly operated UO_2 fuel. The pellet microstructural behaviour of DUPIC fuel (e.g., grain growth) is consistent with UO_2 fuel irradiated to higher burnup than that of the DUPIC fuel. PIE results for the first DUPIC element were previously reported at the 7th CANDU Fuel Conference. This paper describes sheath strain and microstructure results of the second and third DUPIC elements. The FGR and other post-irradiation examination (PIE) results of the DUPIC fuel will be reported at a later date.

Tuesday, 2010 October 19, 10:20 AM-12:00 PM
Technical Session #T2
Fuel Modelling and Computer Code Development II

Iodine Stress Corrosion Cracking and Possible Mitigation Strategy for CANDU Fuel

M.R. Kleczek, K. Shaheen, W.T. Thompson, B.J. Lewis (Royal Military College of Canada)
and F.C. Iglesias (CANDESCO)

Iodine induced stress corrosion cracking, I-SCC, is a recognized factor in the operation of nuclear reactors requiring the implementation of mitigation measures. I-SCC is believed to depend on certain factors such as iodine concentration, oxide layer type and thickness on the fuel sheath, irradiation history, metallurgical parameters related to the sheath like texture and microstructure, and the mechanical properties of zirconium alloys. This work develops a fully mechanistic kinetic model accounting for the iodine chemistry and thermodynamics in the fuel-to-sheath gap. The governing transport equations are solved with a finite-element technique using the COMSOL Multiphysics® commercial software platform. This study also proposes potential remedies for I-SCC.

Measurements of Elastic Modulus in Zr Alloys for CANDU Applications

Z.L. Pan, N. Wang and Z. He (AECL Chalk River Laboratories)

Measurements of elastic modulus as a function of temperature in the range of 20 to 400°C were carried out on specimens of Zr-2.5Nb, Zircaloy-4, Zircaloy-2 and Excel Zr alloy using an ultrasonic resonance technique. The specimens were machined from CANDU pressure tubes, a calandria tube and the fuel bundle material, etc. Effects of neutron irradiation and crystalline texture on elastic modulus were investigated. The results show that elastic modulus of the Zr alloys:

1) Decreases with the increase of temperature;
2) Depends strongly on the crystalline texture of the Zr alloys;
3) Increases slightly with neutron irradiation.

An Improved UO_2 Thermal Conductivity Model in the ELESTRES Computer Code

G.G. Chassie, M. Tochaie and Z. Xu (AECL Sheridan Park)

This paper describes the improved UO_2 thermal conductivity model for use in the ELESTRES (ELEment Simulation and sTRESses) computer code.

The ELESTRES computer code models the thermal, mechanical and microstructural behaviour of a 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 for fuel element design and assessment. It is also used to provide initial conditions for evaluating fuel behaviour during high temperature transients.

The thermal conductivity of UO_2 fuel is one of the key parameters that affect ELESTRES calculations. The existing ELESTRES thermal conductivity model has been assessed and improved based on a large amount of thermal conductivity data from measurements of irradiated and un-irradiated UO_2 fuel with different densities. The UO_2 thermal conductivity data cover 90% to 99% theoretical density of UO_2, temperature up to 3027 K, and burnup up to 1224 MWh/kg U. The improved thermal conductivity model, which is recommended for a full implementation in the ELESTRES computer code, has reduced the ELESTRES code prediction biases of temperature, fission gas release, and fuel sheath strains when compared with the available experimental data. This improved thermal conductivity model has also been checked with a test version of ELESTRES over the full ranges of fuel temperature, fuel burnup, and fuel density expected in CANDU fuel.

Structural and Mechanical Properties of Urania Oxidized to U_3O_8

B. Szpunar, J.A. Szpunar (University of Saskatchewan) and D. Oh (AECL Sheridan Park)

Oxidation of UO_2 fuel pellets in air environments to oxides with higher oxygen content (U_4O_9 or U_3O_8) leads to enhancement of fission product release from UO_2 grain in severe accident scenarios. The enhanced fission product release is most often related to a re-organization of the crystal lattice. As a first step in investigating the microstructural changes following UO_2 oxidation to U_3O_8, an *ab initio* quantum mechanical calculation for UO_2 and U_3O_8 lattices and elastic properties has been performed. This paper presents a preliminary calculation of structural parameters, elastic constants and changes of dimension of UO_2 fuel pellets.

Tuesday, 2010 October 19, 10:20 AM-12:00 PM
Technical Session #T3
Fuel Fabrication I

PHWR Fuel Fabrication with Imported Uranium – Procedures & Processes

R.V.R.L.V. Rao, Sheela, A. Rameswara Rao, G.V.S. Hemantha Rao and R.N. Jayaraj (Nuclear Fuel Complex)

Following the 123 agreement and subsequent agreements with IAEA & NSG, Government of India has entered into bilateral agreements with different countries for nuclear trade. Department of Atomic Energy (DAE), Government of India, has entered into contract with few countries for supply of uranium material for use in the safeguarded PHWRs.

Nuclear Fuel Complex (NFC), an industrial unit of DAE, established in the early seventies, is engaged in the production of Nuclear Fuel and Zircaloy items required for Nuclear Power Reactors operating in the country. NFC has placed one of its fuel fabrication facilities (NFC, Block-A, INE-) under safeguards.

DAE has opted to procure uranium material in the form of ore concentrate and fuel pellets. Uranium ore concentrate was procured as per the ASTM specifications. Since no international standards are available for PHWR fuel pellets, Specifications have to be finalized based on the present fabrication and operating experience. The process steps have to be modified and fine tuned for handling the imported uranium material especially for ore concentrate.

Different transportation methods are to be employed for transportation of uranium material to the facility. Cost of the uranium material imported and the recoveries at various stages of fuel fabrication have impact on the fuel pricing and in turn the unit energy costs. Similarly the operating procedures have to be modified for safeguards inspections by IAEA.

NFC has successfully manufactured and supplied fuel bundles for the three 220 MWe safeguarded PHWRs. The paper describes various issues encountered while manufacturing fuel bundles with different types of nuclear material.

Use of Servo Controlled Weld Head for End Closure Welding

S.K. Pathak, D.S. Setty, A. Rameswara Rao, G.V.S. Hemantha Rao and R.N. Jayaraj (Nuclear Fuel Complex)

In the PHWR fuel fabrication line resistance welding processes are used for joining various zirconium based alloy components to fuel tube of similar material. The quality requirement of these welding processes is very stringent and has to meet all the product requirements. At present these welding processes are being carried out by using standard resistance welding machines.

In the resistance welding process in addition to current and time, force is one of the critical and important parameter, which influences the weld quality. At present advanced feedback type fast response medium frequency weld controllers are being used. This has upslope/down slope, constant and repetitive weld pattern selection features makes this critical welding process more reliable. Compared to weld controllers, squeeze force application devices are limited and normally standard high response pneumatic cylinders are used in the welding process. With this type of devices the force is constant during welding process and cannot be varied during welding process as per the material deformation characteristics. Similarly due to non-availability of feedback systems in the squeeze force application systems restricts the accuracy and quality of the welding process.

In the present paper the influence of squeeze force pattern on the weld quality using advanced feedback type servo based force control system was studied. Different squeeze forces were used during pre and post weld heat periods along with constant force and compared with the weld quality.

Fabrication of Simulated Intermediate-Burnup ACR Fuel in the RFFL

D. Woods, G. Cota-Sanchez and F.C. Dimayuga (AECL Chalk River Laboratories)

Due to the differences in reactor design between the Advanced CANDU Reactor (ACR) and a standard CANDU reactor, a program of reactor physics validation measurements is being conducted in the ZED-2 reactor to support the use of the reactor physics computer code tool set for application in ACR. These measurements include a series of experiments using MOX fuel that simulated intermediate-burnup ACR fuel. The Recycle Fuel Fabrication Laboratories (RFFL) at the Chalk River Laboratories, a facility designed to produce experimental quantities of MOX fuel for reactor physics and irradiation tests, conducted a fabrication campaign to manufacture this MOX fuel.

The objective of the RFFL fabrication campaign was to produce 41 MOX fuel bundles with the ACR geometry, which is a modified 43-element CANFLEX design. The ACR fuel bundle consists of 42 11.5 mm diameter elements in the outer rings and a 20 mm diameter centre element. Forty of these bundles were assembly welded and one was a demountable bundle that allows special elements to be installed and removed for fine structure experiments.

Based on a study done to determine the composition of the simulated intermediate-burnup ACR fuel, the MOX fuel bundles contained different fuel compositions (i.e., different Pu and Dy contents and different ^{235}U enrichments) for each ring of elements. The fabrication process used, from the starting fuel powders to the finished elements and bundles, will be presented, including qualification results and fabrication data.

Impact of Chemical Quality of Source Material on Sintering Recovery

E. Venkatesam, T.K. Sinha and D. Pramanik (Nuclear Fuel Complex)

Nuclear Fuel Complex (NFC), a constituent unit of Department of Atomic Energy (DAE), manufactures and supplies Pressurised Heavy Water Reactor (PHWR) fuel bundles for all the power reactors in India. The raw material is supplied by Uranium Corporation of India Limited (UCIL) in the form of Magnesium-Di-Uranate (MDU). The chemical quality of MDU plays a major role in the performance of various process steps in the Uranium di-Oxide (UO_2) powder and pellets.

MDU being received hitherto was from only one mill and the conversion process steps were quite well established for obtaining sinterable grade UO_2 powder, which has resulted in very good sintered density pellets with good integrity. With the opening of new mines and setting up of new mill the chemical characteristics of the ore and MDU thus produced have different characteristics.

These differences have called for re-setting of many process parameters in the powder conversion and pellet production plants to ensure consistent quality and recoveries. The paper gives in detail the corrective measures attempted, observations made and the conclusions drawn.

Tuesday, 2010 October 19, 1:30 PM-3:10 PM
Technical Session #T4
Spent Fuel Management

CANDU Fuel Rod Behaviour During Dry Storage

A.C. Marino (Comisión Nacional de Energía Atómica)

The BaCo code was applied to simulate the behavior for a PHWR fuel under storage conditions showing a strong dependence on the original design of the fuel and the irradiation history. In particular, the result of the statistical analysis of BaCo indicates that the integrity of the fuel is determined by the manufacture tolerances and the solicitations during the NPP irradiation. The main conclusion of the present study is that the fuel temperature of the device should be carefully controlled in order to ensure safe storage conditions.

Spent Fuel Storage Bay, Water Temperature Response under Full Reactor Discharge

A. Gidi (SNC-Lavalin Nuclear) and A. Al-zubaidi (McMaster University)

A Transient Heat Transfer Lumped Capacitance Model (LCM) was used to predict the bay water temperature of the Spent Fuel Storage Bay at New Brunswick Power, Point Lepreau Nuclear Generating Station. The bay water and the cooling system were modelled and the hourly temperature of the water during full discharge of the reactor was calculated. Results were analyzed and compared with previous results from a study using a Quasi-Steady modelling technique. Full reactor core fuel discharge periods of thirty five, thirty eight and forty days were used. The LCM proved to be notoriously more efficient than the Quasi-Steady approach in terms of computational efforts.

Delayed Hydride Cracking Properties of the Endplate Resistance Welds of CANDU Fuel Bundles

G.K. Shek, B.S. Wasiluk (Kinectrics Inc.), J. Freire-Canosa (Nuclear Waste Management Organization) and T. Lampman (AMEC Nuclear Safety Solutions)

In order to assess the susceptibility of CANDU fuel bundles endplate resistance welds to Delayed Hydride Cracking (DHC) during long term dry storage, the threshold stress intensity factor (KIH) and crack velocity of DHC in endplate welds of three unirradiated fuel bundles were determined. The three bundles tested covered the 28-element and 37-element designs and two Canadian manufacturers. The range of KIH values and DHC velocities obtained from the endplate welds of the three bundles are consistent with previous results obtained from a 37-element bundle produced by one of the manufacturers.

FEAT 4.1: Modeling of Sheath Oxidation and Heat Flow in CANDU Fuel Elements

Q. Jiang, L. Lai, Z. Xu and U.K. Paul (AECL Sheridan Park)

This paper describes recent developments in the AECL-developed computer program, FEAT (Finite Element Analysis for Temperature), which is used to assess the thermal integrity of CANDU fuel elements. The FEAT code is used to calculate temperatures in the fuel pellet and in the Zircaloy sheath of a CANDU fuel element under normal operating conditions (NOC), as well as the temperature peaking due to end flux peaking during a transient such as a postulated loss of coolant accident (LOCA). For normal operation of high burnup fuel, the Zircaloy oxidation effect on fuel temperatures needs to be considered due to the long residence time in the reactor. The oxide layer on the coolant side of the fuel sheath has a lower thermal conductivity than that of Zircaloy. Therefore, the heat flow from the fuel element to coolant will be reduced resulting in increased fuel pellet and sheath temperatures.

To ensure that the FEAT code is suitable for application in analysis of advanced fuels such as the ACR-1000 fuel, a number of model developments and code improvements were conducted based on the existing version FEAT 4.0, including the modeling of sheath oxidation and its effect on heat flow in the fuel element, time-dependence of end flux peaking during the postulated LOCA (Loss of coolant accident) conditions, pre-processing and post-processing of analysis data.

This paper describes the theories for the models, as well as other improvements, and verification and validation of the new FEAT version (i.e. FEAT 4.1).

Tuesday, 2010 October 19, 3:30 PM-5:10 PM
Technical Session #T5
Fuel Fabrication II

Advances in the Manufacture of Clad Tubes and Components for PHWR Fuel Bundle

N. Saibaba, S.K. Jha, B. Chandrasekhar, S. Tonpe and R.N. Jayaraj (Nuclear Fuel Complex)

Fuel bundles for Pressurized Heavy Water Reactors (PHWRs) consists of Uranium di-oxide pellets encapsulated into thin wall Zircaloy clad tubes. Other components such as end caps, bearing pads and spacer pads are the integral elements of the fuel bundle. As the fuel assembly is subjected to severe operating conditions of high temperature and pressure in addition to continual irradiation exposure, all the components are manufactured conforming to stringent specifications with respect to chemical composition, mechanical & metallurgical properties and dimensional tolerances. The integrity of each component is ensured by NDE at different stages of manufacture.

The manufacturing route for fuel tubes and components comprise of a combination of thermo-mechanical processing and each process step has marked effect on the final properties. The fuel tubes are manufactured by processing the extruded blanks in four stage cold pilgering with intermediate annealing and final stress relieving operation. The bar material is produced by hot extrusion followed by multi-pass swaging and intermediate annealing. Spacer pads and bearing pads are manufactured by blanking and coining of Zircaloy sheet which is made by a combination of hot and cold rolling operations. Due to the small size and stringent dimensional requirements of these appendages, selection of production route and optimization of process parameters are important.

This paper discusses about various measures taken for improving the recoveries and mechanical and corrosion properties of the tube, sheet and bar materials being manufactured at Nuclear Fuel Complex, Hyderabad.

For the production of clad tubes, modifications at extrusion stage to reduce the wall thickness variation, introduction of ultrasonic testing of extruded blanks, optimization of cold working and heat treatment parameters at various stages of production etc. were done.

The finished bar material is subjected to 100% Ultrasonic and eddy current testing to ensure defect free material. A number of eddy current indications, above the standard, were observed in the final stage of the bar material, which was resulting in low recovery. An in-depth study was done with various heat treatment cycles and different process parameters. After a step by step analysis of microstructure and mechanical properties at each process stage, it was found that these indications were due to the residual stresses on the periphery of the material caused at multi roll straightening of bars.

Material recovery in the manufacture of spacer pads through conventional sheet route was observed to be poor. A better method was established for the production of spacer pads through wire route resulting in improved recovery and higher productivity.

Resistance Welding Equipment Manufacturing Capability for Exports

V.S. Sastry, Y.S. Raju, A.K. Somani, D.S. Setty, A. Rameswara Rao, G.V.S. Hemantha Rao and R.N. Jayaraj (Nuclear Fuel Complex)

Indian Pressurised Heavy Water Reactor (PHWR) fuel bundle is fully welded and is unique in its design. Appendage welding, end closure welding, and end plate welding is carried out using resistance welding technique. Out of many joining processes available, resistance-welding process is reliable, environment friendly and best suitable for mass production applications.

Nuclear Fuel Complex (NFC), an industrial unit is established in Hyderabad, under the aegis of the Dept of Atomic Energy to manufacture fuel for Pressurised Heavy Water Reactors. From inception, NFC has given importance for self-reliance and indigenization with respect to manufacturing process and equipment.

Sintering furnaces, centreless grinders, appendage-welding machines, end-closure welding equipment and end-plate welding equipments, which were initially imported, are either indigenized or designed and manufactured in house. NFC has designed, manufactured a new appendage-welding machine for manufacturing 37 element fuel bundles. Recently NFC has bagged an order from IAEA through international bidding for design, manufacture, supply, erection and commissioning of end-closure welding equipment.

The paper gives in detail the salient features of this welding equipment.

Automation in Inspection of PHWR Fuel Elements & Bundles at Nuclear Fuel Complex

M.N.V. Viswanath, B. Kamalesh Kumar, K.S. Subramanian, A. Lakshminarayna and A.V. Ramana Rao (Nuclear Fuel Complex)

Nuclear Fuel Complex (NFC), Hyderabad a constituent unit of Department of Atomic Energy, India produces fuel for all Indian nuclear power reactors. In order to meet the growing needs for the nuclear fuel, NFC is involved in massive expansion of the production facilities. This calls for enhanced throughput at various inspection stages. NFC is embarking on automation as a potential tool in this direction. The paper deals with automated material handling and inspection that are being introduced at various stages like fuel pellet inspection, Ultrasonic testing for end closure weld evaluation, optical character recognition (OCR) for fuel bundle identification, Helium back filling, air washing and Helium leak testing operations.

Beryllium Brazing Considerations in CANDU Fuel Bundle Manufacture

J. Harmsen (Cameco)

Appendages of CANDU fuel bundle elements are currently joined to zircaloy sheaths by vacuum beryllium brazing. Ongoing environmental and workplace concerns about beryllium combined with the continuous efforts by Cameco Fuel Manufacturing in its improvement process, initiated this study to find a substitute for pure beryllium.

The presentation will review the necessary functionality of brazing alloy components and short list a series of alloys with the potential to duplicate the performance of pure beryllium. Modifications to current manufacturing processes based on in-plant testing will be discussed in relation to the use of these alloys. The presentation will conclude with a summary of the progress to date and further testing expected to be necessary.

Tuesday, 2010 October 19, 1:30 PM-5:10 PM
Technical Session #T6
Fuel Modelling and Computer Code Development III

Longitudinal Ridging Tests for ACR-1000 Fuel Sheathing

M. Bates, S. Abbas, K. Chakraborty, X. Wang and Z. Xu (AECL Sheridan Park)

CANDU fuel uses thin wall sheathing to reduce neutron absorption, enhance uranium utilization and provide excellent heat transfer between the sheathing and the UO2 pellets. A large diametral clearance between the fuel pellets and the sheath can result in severe longitudinal ridging under reactor operating conditions.

The Advanced CANDU Reactor (ACR-1000) fuel will also adopt thin wall sheathing. Since the ACR-1000 reference fuel element dimensions and the primary heat transport operating parameters are different compared to the CANDU 6, out-reactor tests were performed for ACR-1000 fuel covering the applicable range of parameters that have an impact on longitudinal ridge formation.

This paper presents the range of parameters tested and the test method developed, which are used in predicting critical collapse pressures for longitudinal ridge formation of fuel elements with dimensions spanning both of the ACR-1000 fuel element types. The results from this testing and from empirical correlations, and the mechanistic model used to predict ridging behaviour are discussed.

LONGER: A Computer Program for Longitudinal Ridging and Axial Collapse Assessment of CANDU Fuel

U.K. Paul, Z. Xu, S. Xu, X. Wang and K. Chakraborty (AECL Sheridan Park)

CANDU fuel element sheath is designed to be thin and flexible for the benefit of enhanced heat transfer from the pellet to the coolant through the sheath. The flexibility of the sheath may allow the formation of longitudinal ridges on the sheath or collapse of the sheath into an axial gap under certain conditions. For both cases of deformations, the sheath may experience significant strains, and may result in sheath failure. To ensure the sheath mechanical integrity, the fuel element design needs to be assessed to preclude the conditions for longitudinal ridging and sheath collapse into the axial gap. The AECL developed LONGER computer program is used in fuel design analysis for such purpose.

The LONGER code contains a number of models derived based on measurements (empirical models) and based on analytical equations, to predict the following parameters related to the deformations of CANDU nuclear fuel element sheaths.

For longitudinal ridging:

- The critical diametral clearance for sheath longitudinal ridging;
- The critical pressure for longitudinal ridging of the sheath.

For axial collapse:

- The critical pressure for instantaneous sheath collapse into an axial gap.

For circumferential collapse:

- The critical pressure for elastic collapse of the sheath;
- The effective circumferential collapse pressure of the sheath by taking into account the axial and radial loads and the ovality of the sheath.

The LONGER code has been qualified in accordance with the CSA standard N286.7 99 compliant AECL Software Quality Assurance (SQA) program. This paper describes the features and capabilities of the LONGER code that are used in CANDU fuel design analysis.

FEAST 3.1: Finite Element Modelling of Sheath Deformation Such as Longitudinal Ridging and Collapse into Axial Gap

X. Wang, Z. Xu, Y-S. Kim, L. Lai, G. Cheng and S. Xu (AECL Sheridan Park)

During normal operation, the collapsible CANDU fuel sheath deforms, especially, it may deform into longitudinal ridges or collapse instantaneously into the axial gaps between the end pellet and endcap or between two neighbouring pellets. These phenomena occur under certain conditions, such as the coolant pressure exceeding critical pressures for longitudinal ridging or axial collapse. Both longitudinal ridging and axial collapse phenomena result from plastic instability in the sheath under coolant pressure. Longitudinal ridging features one or multiple lobes or "ridges" (outward from the sheath surface) formed along the sheath in the longitudinal direction. Axial collapse features a "valley" around the sheath circumference. Both phenomena can lead to sheath overstrain, which in turn potentially leads to sheath failure. The LONGER code, which contains empirical correlations, has been used to predict the critical pressures for these two sheath deformation phenomena.

To study the behaviour of fuels outside of the application ranges of the LONGER empirical correlations, a mechanistic model is needed. FEAST (Finite Element Analysis for Stresses) is an AECL computer code used to assess the structural integrity of the CANDU fuel element. The FEAST code has recently been developed (to Version 3.1) to model processes occurring during longitudinal ridge formation and instantaneous collapse into the axial gap. The new models include those for geometric non-linearity (large deformation, large material rotation), non-linear stress-strain curve for plastic deformation, Zr-4 sheath creep law, and variable Young's Modulus etc.

This paper describes the mechanistic model (FEAST 3.1) development for analyses of longitudinal ridging and instantaneous collapse into axial gap, and the comparison with the results from empirical correlations in LONGER.

ELESTRES 2.1 Computer Code for High Burnup CANDU Fuel Performance Analysis

G.G. Chassie, Q. Jiang, H. Liu, M. Tochaie and Z. Xu (AECL Sheridan Park)

The ELESTRES (ELEment Simulation and sTRESses) computer code models the thermal, mechanical and micro structural behaviours 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. It is also used to provide initial conditions for evaluating fuel behaviour during high temperature transients.

ELESTRES 2.1 was developed for high burnup fuel application, based on an industry standard tool version of the code, through the implementation or modification to code models such as fission gas release, fuel pellet densification, flux depression (radial power distribution in the fuel pellet), fuel pellet thermal conductivity, fuel sheath creep, fuel sheath yield strength, fuel sheath oxidation, two dimensional heat transfer between the fuel pellet and the fuel sheath; and an automatic finite element meshing capability to handle various fuel pellet shapes.

The ELESTRES 2.1 code design and development was planned, implemented, verified, validated, and documented in accordance with the AECL 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 2.1 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 2.1 is capable of modelling important fuel phenomena and the code can be used in the design assessment and the verification of high burnup fuels.

Development of a Rigidity Enhancement Factor (REF) Correlation for Use in AECL Fuel Codes

S. Xu, Z. Xu and Q. Jiang (AECL Sheridan Park)

A fuel element can be simulated in static and vibration analysis as a composite beam, which consists of fuel sheath and fuel pellets. Flexural rigidity of the fuel element is determined based on contributions from the sheath and pellets. The sheath contributes 100% of its flexural rigidity to the composite beam, while the contribution from the pellet stack depends on the interaction between the pellets and the sheath.

This interaction is represented by the rigidity enhancement factor (REF), which is a function of the interfacial pressure between pellet and sheath. When REF is known, the fuel element flexural rigidity can be determined.

The fuel element rigidity can be calculated from the frequency of natural lateral vibration of the fuel element. Both natural vibration frequencies and input powers were recorded in AECL experiment. The interfacial pressure between the sheath and pellets can be calculated. A correlation between the interfacial pressure and the REF is derived using these measured data.

This paper describes the derivation of the REF based on the measured data.

Computation of Actinide Pourbaix Diagrams at 298 K and 550 K, (U,Np,Pu,Am,Cm – H_2O)

M.H. Piro, G.M.F. Bruni, B.J. Lewis, W.T. Thompson (Royal Military College of Canada), F.C. Iglesias (CANDESCO), M.A. Guoping, R. Nashiem (Bruce Power) and J.G. Roberts (Cantech Associates Ltd.)

Pourbaix diagrams for the actinides ranging from uranium to curium have been developed from the enthalpy of formation, entropy and heat capacity (or estimates thereof) for the phases and aqueous species. Emphasis has been given to the region between hydrogen and oxygen gas saturation. Additional thermodynamic computations illustrate how the data may be employed in better understanding the chemical behaviour of fuel debris circulating in CANDU reactor coolant.

2009 Creep Correlations of Zircaloy for ACR-1000 FUEL

A. Sun, M. Tayal, P. Reid and S. Abbas (AECL, Sheridan Park)

The experimental creep data of Zircaloy-4 fuel cladding obtained in the late 1970's has been the main input data for developing creep correlations in CANDU fuel design. However, these data were measured for those creep strains with short time durations and under a single neutron flux rate. The ACR-1000 fuel bundle has been designed to operate at high burnup, high coolant temperature and different flux rate(s).
This paper presents new creep correlations developed for ACR fuel in 2009. It includes more recent world-wide data on the creep rates of Zircaloy-4. The additional data covers longer creep duration under various neutron flux rates and different temperatures and stress levels. The creep data and correlations are categorised as two types of materials: stress-relieved annealed (SRA) and recrystallised annealed (RXA). A time hardening and a strain hardening correlation are formulated respectively in this paper. The new correlations provide small statistical error, and agree well in overall predictions with the measured creep data for Zircaloy-4 sheaths.

BEAM 1.7: Development for Modelling Fuel Element and Bundle Buckling Strength

G. Cheng, S. Xu, Z. Xu and U.K. Paul (AECL Sheridan Park)

This paper describes BEAM, an AECL developed computer program, used to assess mechanical integrity of CANDU fuel bundles. The BEAM code has been developed to satisfy the need for buckling strength analysis of fuel bundles. Buckling refers to the phenomenon that compressive axial load is large enough, and a small lateral load can cause large lateral deflections. The buckling strength refers to the critical compressive axial load at which lateral instability is reached. The buckling strength analysis has practical significance for the design of fuel bundles, where the buckling strength of a fuel element/bundle is assessed so that the conditions leading to bundle jamming in the pressure tube are excluded. This paper presents the development and qualification of the BEAM code, with emphasis on the theoretical background and code implementation of the newly developed fuel element/bundle buckling strength model.

Tuesday, 2010 October 19, 1:30 PM-5:10 PM
Technical Session #T7
Fuel Safety and Operational Margin Improvement

Fuel Behaviour and Fission-Product Release in the Power Pulse 1 Experiment

R.S. Dickson, A.I. Belov, M.D. Gauthier, R.T. Peplinskie and C.A. Buchanan (AECL Chalk River Laboratories)

The Power Pulse 1 experiment was intended to determine fuel behaviour and fission-product releases under conditions approaching and above the upper limit of fuel behaviour in Loss-of-Coolant Accident (LOCA) power pulses (formation of molten material). Eleven tests were performed on sections cut from two CANDU power reactor fuel elements: one in typical discharged condition, the other after operation in defected condition. The samples were pre-heated in a slow flow of helium before applying direct electric heating; some samples were subjected to power pulses, while others were not. Formation of molten material and columnar grains, limited loss of grain cohesion, and release of ^{85}Kr were observed in these tests.

Fuel Safety Analysis Following Feeder Break Accident for Refurbished Wolsong 1

J. Jung and J.H. Park (Korea Atomic Energy Research Institute)

The objective of the fuel analysis for the postulated accident was to estimate the quantity and timing of a fission product release from fuels when a postulated single channel accident occurs in CANDU 6 reactors. In this study, a fuel safety analysis for the refurbished Wolsong 1 was carried out by using the latest IST (Industrial Standard Toolset) fuel code. The relevant accident scenario focused in this study was a feeder stagnation break accident. The amount of fission product inventory and its distribution during the normal operating conditions were calculated by using the latest ELESTRES-IST code. For a calculation of transient fission product release following the feeder stagnation break, it was assumed that all fuel sheaths in the channel were failed and the entire gap inventory was released instantaneously at the beginning of the accident. The additional releases from the grain boundary and in-grain bound inventories were estimated by applying the Gehl's release model.

Methodology for Fission Product Release Calculations During an ACR-1000 End-Fitting Failure Event

W. Zhu, F.L. Huang and Z. Bilanovic (AECL Sheridan Park)

The ACR-1000 reactor enhances and retains the proven features of the CANDU design such as the concept of the horizontal fuel channel core. At each end of a fuel channel, there is an end-fitting incorporating a feeder connection through which pressurized coolant enters and leaves the fuel channel, where 12 fuel bundles are inserted. The safety analysis cases include postulated end-fitting failure events to assess the fission product releases from all fuel bundles which would be ejected out of the channel and oxidized in the airstream environment under decay power. This paper presents the methodology used in assessing the fuel behaviour and the fission product releases during a postulated end-fitting failure in an ACR-1000 reactor. After the end-fitting failure, the 12 fuel bundles are ejected out of the channel and drop onto the fuelling machine vault floor. The fuel bundles are likely heavily damaged by impact and would break into small clusters of elements or fragments. To calculate the fission product releases from an individual fragment, the transient fuel temperature is numerically solved by differential heat equations; the air oxidation model is chosen for the event accordingly; and the fission product inventory and releases are estimated by computer codes ORIGEN-S, CATHENA, ELESTRES and SOURCE-IST. Finally, the total fission product releases from all fragments into containment are calculated. This methodology has been developed for ACR-1000 safety analysis, which is also applicable to CANDU. With the new methodology, the transient releases from up to 150 fission products can be estimated as detail as in fragment. In this paper, a sample calculation is also provided to show the application of the methodology in ACR-1000 safety analysis for end-fitting failure.

ACR-1000 End-Temperature Peaking Analysis under Postulated Accident Conditions

F. Gao, Q.M. Lei and Z. Bilanovic (AECL Sheridan Park)

This paper presents a novel and systematic approach to conduct end-temperature peaking analysis under accident conditions for an ACR-1000 reactor, using a two-dimensional (radial and axial) finite-element computer code FEAT. In the past, end-flux peaking effects were overly conservatively assessed by including power increase in the fuel end region without accounting for heat transfer enhancement due to flow disturbance at the bundle end region, especially at the down-stream of a bundle junction. The current analysis determines the end-flux-peaking induced increase in fuel sheath and fuel centreline temperatures while accounting for all relevant key phenomena such as end-flux peaking and heat transfer characteristics including the effects of flow/thermal boundary layer redeveloping at the bundle end region. Using this method significantly reduces the fuel sheath temperature increase caused by end-flux peaking in comparison with the conservative analysis. The postulated accident events considered in this analysis include large break loss-of-coolant accident (LOCA), small break LOCA, and pressure tube rupture within an intact calandria tube. The determined temperature increases relative to the case without end-flux peaking are required to be quantitatively included in detailed safety analyses for postulated accidents.

CATHENA Modelling of ACR-1000 Fuel Handling Events

M. Maltchevski, H.Z. Fan and F. Huang (AECL Sheridan Park)

The ACR-1000 Fuel Handling (FH) and Storage System provides on-power fuelling capability based on a proven CANDU technology. It includes all aspects of fuel storage and handling, from the arrival of new fuel to the storage of irradiated fuel. The fuelling machine (FM) water system provides the cooling water to the fuelling machines during the normal operation. The FH emergency water system is a process circuit that is separate from the FM water system and the spent fuel transfer process system. The FH emergency water system operates at low pressures and is seismically and environmentally qualified.

The paper presents a CATHENA model developed for the safety analysis of ACR-1000 FH events with the loss of cooling function in the fuelling machine water system.

Prediction of Power-Ramp Defects in CANDU Fuel

P. Gillespie, S. Wadsworth (AMEC Nuclear Safety Solutions) and T. Daniels (Ontario Power Generation)

Power ramps result in fuel pellet expansion and can lead to fuel sheath failures by fission product induced stress corrosion cracking (SCC). Historically, empirical models fit to experimental test data were used to predict the onset of power-ramp failures in CANDU fuel. In 1988, a power-ramped fuel defect event at PNGS-1 led to the refinement of these empirical models. This defect event has recently been re-analyzed and the empirical model updated. The empirical model is supported by a physically based model which can be used to extrapolate to fuel conditions (density, burnup) outside of the 1988 data set.

Validation Summary for REDOU Code

D. Oh[1], E. Zariffeh[1], R. Dickson[2] and S. Girgis[1] ([1]AECL Sheridan Park) ([2]AECL, Chalk River)

REDOU (REleases Due to Oxidation of UO2) 2.0 is an AECL computer code for modelling the iodine release during the air oxidation of UO_2 fuel. REDOU 2.0 calculates the fractional release of grain-matrix iodine from within a UO_2 fragment that is exposed to air due to fuel damage during a postulated end-fitting failure accident. REDOU 2.0 has been extensively verified, validated and fully documented in accordance with the Canadian software quality assurance requirements. A brief description of the code and a summary of the validation results are given in this paper.

Wednesday, 2010 October 20, 8:30 AM-11:45 AM
Technical Session #W1
Fuel Modelling and Computer Code Development IV

BOW 4.0: Modeling Deformations of Fuel String with Channel Ageing Effects

S. Xu, Z. Xu, G. Cheng, L. Lai (AECL Sheridan Park), S. Yu and D. Wen (Ryerson University)

CANDU fuel string deformation can be affected by phenomena related to the ageing of the fuel channels, namely pressure tube (PT) diametral expansion, and PT sagging due to creep. The PT diametral expansion increases the fuel bundle clearance with the PT wall. The PT sagging causes fuel bundles to parallelogram in a fuel channel. The fuel channel ageing effects are not fully considered in the bundle version of the BOW code (3.0).

To account for such ageing effect, the bundle version BOW 3.0 has been recently extended to the channel version (BOW 4.0), which has many new models and capabilities, for example:

- Modelling of pressure tube sagging and its effect on fuel bundle deformation (parallelogramming);
- Modelling of fuel string deformation (multiple bundles, bundle shift, misalignment);
- Endplate creep model;
- General creep laws for fuel sheath materials;
- Model for sheath oxide effect on the temperature distribution in the fuel element;
- Graphical output (post-processor).

BOW 4.0 models the in-reactor fuel string deformation under normal operating conditions (NOC). This paper describes the theory and models for the new capabilities of the BOW code.

Deformation of Unirradiated CANDU Fuel Elements under Bending Loads

J. Freire-Canosa (Nuclear Waste Management Organization), A. Popescu and T. Lampman (AMEC Nuclear Safety Solutions)

Nuclear fuel rods offer a simple multi-body structure with an interesting and complex mechanical behaviour. Fuel pellets within the Zircaloy cladding are free to move for low cladding deformations. The complexity of such interactions complicates the successful modelling of fuel pencils. Mechanical tests using non-axial loads were performed on typical unirradiated CANDU fuel elements by applying a load at the fuel element mid-point and measuring the resulting element deformation along the fuel element axis. An ANSYS finite element model of the fuel elements containing cladding, endcaps, and individual pellets was developed and used to predict the single fuel element deformation observed in the tests. The model provides excellent predictions of the experimentally measured fuel element deformation using known geometrical dimensions and mechanical properties for CANDU fuel.

Numerical Model for Thermal and Mechanical Behaviour of a CANDU 37-Element Bundle

L. Jiang, K. MacKay (Martec Limited) and R. Gibb (Canadian Nuclear Safety Commission)

Prediction of transient fuel bundle deformations is important for assessing the integrity of fuel and the surrounding structural components under different operating conditions including accidents. For numerical simulation of the interactions between fuel bundle and pressure tube, a reliable numerical bundle model is required to predict thermal and mechanical behaviour of the fuel bundle assembly under different thermal loading conditions. To ensure realistic representations of the bundle behaviour, this model must include all of the important thermal and mechanical features of the fuel bundle, such as temperature-dependent material properties, thermal viscoplastic deformation in sheath, fuel-to-sheath interactions, endplate constraints and contacts between fuel elements. In this paper, we present a finite element based numerical model for predicting macroscopic transient thermal-mechanical behaviour of a complete 37-element CANDU nuclear fuel bundle under accident conditions and demonstrate its potential for being used to investigate fuel bundle to pressure tube interaction in future nuclear safety analyses. This bundle model has been validated against available experimental and numerical solutions and applied to various simulations involving steady-state and transient loading conditions.

Modelling of Transient Dynamic Bundle Deformation Using Time Integration Scheme

S. Xu, Y-S. Kim and Z. Xu (AECL, Sheridan Park)

The BOW code has been examined whether its modeling capability can be extended to the simulation of interactions (i.e., fretting) between neighbouring fuel elements in a fuel bundle and between the fuel bundle and the pressure tube in a fuel channel. The current BOW code is specialized in simulating the static problems, such as the deflection of each element and interactions between neighbouring elements in a fuel bundle, and interactions between neighbouring bundles and between a bundle and the pressure tube in a fuel channel. The Wilson θ time integration scheme has been implemented in the BOW code, for the extension of its capability to modelling dynamic contact problems.

As part of verification to ensure that the modification in the code functions exactly as designed, the dynamic-modelling capability of the BOW code has been applied to simple support beam cases subjected to a uniform step load at the middle of the beam. The calculation results confirmed that the modified BOW code, where the contact algorithm is implemented in the step-by-step integration manner using the Wilson θ time integration scheme, can solve the dynamic problem with unconditional convergence.

This paper describes the theory and models for the new capabilities of the BOW code.

Modeling of Fuel Bundle Deformation at High Temperatures: Requiremens, Models and Steps for Consideration

S. Xu, Z. Xu, H. Fan (AECL Sheridan Park) and T. Nitheanandan (AECL Chalk River Laboratories)

To model thermal mechanical bundle deformation behaviour under high temperature conditions, several factors need to be considered. These are the sources of loads, deformation mechanisms, interactions within bundle components, bundle and pressure tube (PT) interaction, and boundary constraints on the fuel bundles under in reactor conditions.

This paper describes the modelling of the following three processes:

- Bundle slumping due to high temperature creep sag of individual elements and endplates,
- Differential element expansion and fuel element bowing, and
- Bundle distortion under axial loads.

To model these processes, a number of key mechanisms for bundle deformation must be considered which include:

- Interaction of fuel elements in a bundle with their neighbours,
- Endplate deformation,
- Fuel elements lateral deformation under various loads and mechanisms,
- Interaction within a fuel element,
- Material property change at high temperatures,
- Transient response of a bundle, and
- Bundle configuration change.

This paper summarises the new models needed for the mechanistic modelling of the key mechanisms mentioned above and provides an example to show how an endplate plasticity model is developed with results.

CFD Simulations of the Single-phase and Two-phase Coolant Flow of Water inside the Original and Modified CANDU 37-Element Bundles

F. Abbasian, G.I. Hadaller and R.A. Fortman (Stern Laboratories Inc)

Single-phase (inlet temperature of 180° C, outlet pressure of 9 MPa, total power of 2 MW and flow rate of 13.5 Kg/s), and two-phase (inlet temperature of 265° C, outlet pressure of 10 MPa, total power of 7.126 MW and flow rate of 19 Kg/s) water flows inside a CANDU thirty seven element fuel string are simulated using a Computational Fluid Dynamics (CFD) code with parallel processing and results are presented in this paper.

The analyses have been performed for the original and modified (11.5 mm center element diameter) designs with skewed cosine axial heat flux distribution and 5.1% diametral creep of the pressure tube. The CFD results are in good agreement with the expected temperature and velocity cross-sectional distributions. The effect of the pressure tube creep on the flow bypass is detected, and the CHF improvement in the core region of the modified design is confirmed. The two-phase flow model reasonably predicted the void distribution, and the role of interfacial drag on increasing the pressure drop. In all CFD models, the appendages were shown to enhance the production of cross flows and their corresponding flow mixing and asymmetry.

FEED 1.6: Modelling of Hydrogen Diffusion and Precipitation in Fuel Bundle Zircaloy Components

L. Lai, Z. Xu, Q. Jiang, G. Cheng and S. Xu (AECL Sheridan Park)

An as-fabricated Zircaloy component in a CANDU fuel bundle has certain amount of hydrogen. In addition, the Zircaloy component pickups hydrogen during operation, where sheath oxidation occurs on the water side. Hydrogen content in the Zircaloy component will change due to the diffusion under gradients of concentration and temperature. A hydrostatic stress gradient may also have some effect on hydrogen diffusion. When the local concentration of hydrogen exceeds the terminal solid solubility (TSS), hydrides will start to form (i.e. hydride precipitation). Because hydrides have a negative effect on material properties (e.g. lower ductility), the hydrogen content in Zircaloy sheath needs to be limited to ensure that the sheath strength is not affected. The FEED (Finite Element Estimate for Diffusion) code was developed to predict the local hydrogen concentration and formation of hydride.

The FEED code has the following capabilities:

- Model transient Hydrogen/Deuterium (H/D) diffusion in Zircaloy components (e.g. fuel sheath, endcap and endcap weld);
- Model H/D pickup in Zircaloy sheath;
- Account for the effect of gradients of concentration, temperature and stress;
- Model transient hydride precipitation and re-dissolutions.

This paper describes the FEED 1.6 code, including theory, models, and some validation examples.

Wednesday, 2010 October 20, 8:30 AM-11:45 AM
Technical Session #W2
Fuel Performance, Reliability and Operating Experience II

CNSC Fuel Oversight Programme

A. El-Jaby, K. Conlon, W. Grant and M. Couture (Canadian Nuclear Safety Commission)

Nuclear power plant licensees are required to submit an *Annual Fuel Performance Report* to the CNSC pursuant to *Regulatory Standard S-99*, clause *6.4.10 – Report on the fuel monitoring and inspection programme*. This paper summarises how the information in the annual *Report on the fuel monitoring and inspection programme* is used by CNSC staff to provide an assessment of fuel performance and licensing compliance, the results of which are used as input into the annual *CNSC Staff Integrated Safety Assessment of Canadian Nuclear Power Plants*. Lastly, possible changes and improvements aimed at simultaneously enhancing and streamlining the current reporting template are discussed.

Pickering NGS B – Unit 6 Strategy to Remove Fuel Defects during Operation at High Power

M. Dobrean and E. Sorin (Pickering NGS, Ontario Power Generation)

Unit 6 was operating at high power (100% FP) with all the adjuster rods (AAs) in core when Chemistry sample results of HTS I-131 activity increased from 1.65 µCi/kg (base line) to 5.4 µCi/kg. Sustained elevated activity levels and subsequently numerous significant spikes have lead to the conclusion that the Unit 6 core contains defected fuel bundles. 34 channels have been identified by Reactor Physics, as possible locations containing defective fuel bundles.

Reactor Physics has developed a set of criteria, to be used in prioritizing the sequence of channel re-fuelling and defect removal. The refuelling strategy was successfully implemented in February 2009, all the suspect channels were twice fuelled and any potential defected fuel bundle was discharged by end of summer 2009.

Fuel Defect Investigation at CERNAVODA NGS in Romania

E. Suk, A.M. Manzer (AECL Sheridan Park), P.J. Valliant, J. Pyoli (AECL Chalk River Laboratories), S. Holostencu, N. Baraiaru, D. Stanila and A. Gisnac (SNN)

Cernavoda units experienced fuel failures among bundles domestically supplied in the recent years. Unit 1 has operated with an excellent fuel performance with zero defects in the recent years until 2008 when a couple of defective bundles were discharged. Those two defects discharged in 2008 were single element failures due to hydriding and the root cause was reported unknown. Unit 2 experienced defect excursion from the start of commercial operation (November 2007) until 2008. As part of root cause investigation, visual examination confirmed 55 defective bundles from Unit 2. The investigation concluded that the root cause appeared to be an overriding defect mechanism (to be confirmed through post-irradiation examination) that involves two independent mechanisms: debris and manufacturing flaws (suspected incomplete sheath-to-endcap welds). This paper summarizes the results of the fuel inspections, the root cause analyses, and corrective actions taken and recommended.

Assessing the Impact of Increased Transport Time between Failed Fuel and Delayed Neutron Detectors

A.Fitchett, F. Iglesias (CANDESCO), B. Lewis and K. Shaheen (Royal Military College of Canada)

A typical DN monitoring system in use in CANDU reactors is discussed and a mathematical model is developed to simulate the transport from a fuel defect location to the DN detectors. This model will be used to determine the impact of an increased transport time to the DN monitors due to pipe fouling and to quantify the contribution from tramp uranium to the total background activity in the PHTS. The model may be applied to CANDU-6 designs with minor modifications.

Examination of Irradiated PHWR Fuel Pins Subjected to Isothermal Heating at 700-900°C inside the Hot Cells

D.N. Sah, Prerna Mishra, P.M. Satheesh, Sunil Kumar and S. Anantharaman (Bhabha Atomic Research Centre)

High temperature deformation behaviour of irradiated zircaloy-2 cladding has been studied by carrying out isothermal heating of irradiated PHWR fuel pins, having burnup up to 15000MWd/tU at 700°C -900°C inside hot-cells. Post-test examination included visual examination, leak testing, dimension measurement on the tested fuel pins and optical and SEM examination of cladding sections taken from the ballooned and failed fuel pins. The specific aspects investigated in these experiments included cladding tube deformation and ballooning-failure as function of cladding temperature, internal gas pressure and burn up of the fuel pin and microstructural aspects of high temperature creep deformation and mode of cladding failure.

Preliminary Assessment of Noble Gas Bundle Tagging Using a Partial Krypton Backfill

B. Pollack, B.J. Lewis and D. Kelly (Royal Military College of Canada)

Current limitations of CANDU reactors to reliably locate defective fuel bundles have sparked interest into new identification techniques. Noble gas tagging, which would involve the addition of specific combinations of Kr and Xe isotopes to the fuel-to-sheath gap during manufacturing, has the potential to offer a means of locating failed-fuel bundles. The released tag with a given isotopic signature could be measured in the primary heat transport system by mass spectrometry. This technique would allow on-power failure location. Moreover, the technique could be of particular interest for demonstration irradiations with new fuel bundle designs. This report outlines preliminary considerations towards a suitable tag isotope choice and discusses the impact on the thermal performance of a fuel element. The detection limit of two mass spectrometer systems was determined through measurements of prepared krypton samples with aqueous concentrations in the range of 10^{-12} to 10^{-9} [mol_{Kr}/mol_{H2O}].

Thermal Aspects of Alternative Fuels for Use in Supercritical Water-Cooled Nuclear Reactors

L. Grande, W. Peiman, B. Villamere, A. Rodriguez-Prado, S. Mikhael, L. Allison and I. Pioro (University of Ontario Institute of Technology)

Supercritical Water-cooled nuclear Reactors (SCWRs) are a Generation IV concept currently being developed. This reactor type uses light water coolant above its critical point. This paper presents a thermalhydraulic analysis on a single channel within Pressure Tube (PT) – type SCWR with a single reheat cycle. The Axial Heat Flux Profiles (AHFPs) applied are uniform and non-uniform with a variety of alternative fuels (mixed oxide, thorium dioxide, uranium dicarbide, uranium nitride and uranium carbide). The results depict bulk-fluid, outer sheath and fuel centreline temperature profiles along with the Heat Transfer Coefficient (HTC) profile against sheath and fuel centreline temperature limits.

Wednesday, 2010 October 20, 11:45 AM-12:15 PM
Plenary Session #3, Room – Oakes South
CANDU Fuel Cycle Options
Chair: S. Palleck (AECL)

Unique Fuel Cycle Capabilities of CANDU

C. Cottrell, R. Bodner and S. Kuran (AECL Sheridan Park)

Abstract - not available.