1ST INTERNATIONAL CONFERENCE OF MATERIALS, CHEMISTRY AND FITNESS-FOR-SERVICE SOLUTIONS FOR NUCLEAR SYSTEMS

Challenges and Innovative Solutions in Fitness-for-Service

Canadian Nuclear Society ®

May 15 - 17, 2019 — Hilton Toronto/Markham Suites Conference Centre

Advance Technical Program

Thursday 16 May 2019 - Morning

8:30 — 10:00	**Opening/Keynote**	
	• General Chair's opening remarks, introduction of keynote speakers	M. Baghbanan
	• Challenges and Innovative Solutions in Fitness-For-Service	M. Knutson
	• The Role of Fitness-for-Service Methodologies as a Key Part of Nuclear Plant Asset/Life Cycle Management Programs	G. Newman
10:00 — 10:15	**break**	
10:15 — 12:15	**Session A1: Fuel Channels**	
	• Regulatory Perspective on Fitness-for-Service Requirements for the Pressure Tube to Calandria Tube Contact in CANDU Reactors	Sankar Laxman, Kostas Tsembelis and John Jin
	• Pressure Tube Flaw Characterization and Replication Challenges	Rebecca Wu, Mohammadreza Baghbanan and Steve Motomura
	• Delayed Hydride Cracking	Glenn McRae, Kit Coleman and Sean Hanlon
	• Evaluation of In-Service Pressure Tube Flaws	Christopher Manu, Ben Dobbie and Danny Mok
	• Use of Channel Specific Gap Measurements in Probabilistic Assessments of Pressure Tube to Calandria Tube Contact	Eric Nadeau, Peter Daoud, Adrian Baniak and Jenny Kong
	• Potential OPEX-Based Refinements to Fuel Channel Sag Models for Future Life Extensions	Paul Sedran
12:15 — 13:00	**lunch**	

Thursday 16 May 2019 - Afternoon

13:00 — 13:40	**Session A2: Fuel Channels**	
	• FFS of Rolled Joints	Glenn McRae, Kit Coleman and Scott Langille
	• Resistivity Change in CANDU® Pressure Tube Material due to Heat Treatment	Perryn Bennett, Matt Topping, Ross Underhill, Jordan Morelli, Mark Daymond and Thomas Krause
13:40 — 15:20	**Session B: Probabilistic Methods and Simulations**	
	• Regulatory Considerations for the Adoption of Probabilistic Assessment Methodologies for Pressure Boundary Component Aging Evaluations	Blair Carroll
	• Advanced Simulation Software to Assess How, Why, Where, and When Components Will Fail	James Carter
	• Development of Probabilistic Models for Integrated Life Cycle Management	Joseph Cluever, Paul Bruck, Thomas Esselman, Mark Woodby and Michael Taylor
	• Investigation of Model Uncertainties for Stability Evaluation of Circumferential Crack	Yifan Huang, Xinjian Duan
	• Probabilistic Reactor Core Assessment of Degradation Mechanisms Related to Pressure Tube Flaws	Ben Dobbie, Christopher Manu and Danny Mok
15:20 — 15:35	**break**	
15:35 — 17:15	**Session C: Balance-of-Plant**	
	• Maintenance Cooling System HX Shell Replacement without Affecting Recall Times	Stephen Fluit, Jeff Jarvis, Mark Bevan, Geoffrey Lawson and Tiberiu Preda
	• Vibration Monitoring Locations & Allowable Levels Along Small Bore Pipes	Usama Abdelsalam, Stephen Jeremia and Dk Vijay
	• Flow Accelerated Corrosion Evaluation of Main Steam Pipe	Morteza Toloui, Sung Tran Tran, Brain Seed and Ella Pakravan
	• Deaerator tank seam weld crack repair using Gas Tungsten Arc Welding (GTAW) process - Pickering NGS Unit 1	Ashraf Sadek
	• Fitness for Service Assessment of a Flange Joint with Missing bolt(s)	Vishash Sharma, Aman Usmani, Reza Adibiasl and Edwin Chen

Advance Technical Program

Friday 17 May 2019 - Morning

8:30 — 10:10 **Session D1: OPEX, FFS methodology, Materials factors, Other**

• Continued Fitness for Service of the Bruce B Unit 7 Calandria Relief Ducts	Andrew Brooks and Sahil Gupta
• Reactor Inlet Header Temperature Overview: External preheater fouling study	Ernest Lu
• NRU Return to Service Project – Fitness-for-Service	Michael Kozluk
• Comprehensive Study of Carbon Steel to Alloy 600 Dissimilar Metal Weld	Xinjian Duan, Ming Li, Andrew Glover and Dongmei Sun
• Fitness for Service Assessment in Local Thinned Areas of a Cylindrical Vessel	Mohammadreza Noban, Aman Usmani and Reza Adibiasl

10:10 — 10:25 **break**

10:25 — 11:25 **Session D2: OPEX, FFS methodology, Materials factors, Other**

• Concrete Performance at Elevated Temperature – A State of the Art Review	Hesham Mohammed, Medhat Elgohary and Ayman Saudy
• Origins of Feeder Fitness-for-Service Guidelines	Michael Kozluk
• Design Evaluation of SMRs for Utilization of Spent fuel	Rami Nessim

11:25 — 12:05 **Session E1: Steam Generators**

• Case study - Bruce Power reactor original steam generator divider plate degraded bolt	Ernest Lu and Justin Howatt
• Inspection of Newly Installed and In-service Plug Welds	David Burr, Nathan Bruns, Steve Fluit and Ken Sedman

12:05 — 12:50 **lunch**

Friday 17 May 2019 - Afternoon

12:50 — 14:10 **Session E2: Steam Generators**

• Evaluation of Wear Degradation for Steam Generator Tubing Rupture	Brent Capell
• Non-destructive Detection and Characterization of De-alloying Degradation in Steam Generator Tubing	Rosita Mousavi, Peter Kwan and Gordon Bruce
• Case Study of Steam Generator Foreign Material Assessment	Wolf Reinhardt, Rosita Mousavi, Ramtin Kahvaie-Zad and Yifan Huang
• Management of Ongoing Tube Support Plate Degradation	Nathan Bruns, David Burr, Dongmei Wang, Steve Fluit and Ken Sedman

14:10 — 14:20 **Closing**

Fuel Channels

Regulatory Perspective on Fitness-for-Service Requirements for the Pressure Tube to Calandria Tube Contact in CANDU Reactors

Sankar Laxman, Kostas Tsembelis and John Jin
Canadian Nuclear Safety Commission

Canadian licensees periodically inspect CANDU pressure tubes in accordance with clause 12 of CSA standard N285.4. The Zr-2.5% Nb pressure tubes carry nuclear fuel and are insulated from the cooler moderator by an outer Zircaloy 2 tube, called the calandria tube. The pressure tube is supported on four spacers, or “garter springs,” which help prevent contact between these two tubes. Contact between the hot pressure tube and colder calandria tube can lead to a cold spot in the pressure tube and subsequent formation of a brittle hydride blister at the cold spot, and eventual cracking of the blister during reactor operation.

Among several volumetric and dimensional inspection requirements, clause 12 of CSA N285.4 requires the determination of the pressure tube to calandria tube gap and stipulates that the pressure tubes are considered acceptable for operation only when “no pressure tube to calandria tube contact” is predicted to exist at the end of the next periodic inspection interval.

When the screening criteria of CSA N285.4 are not satisfied, N285.8 provides additional requirements to disposition. Additionally, N285.8 requires a risk assessment for the entire core performed either deterministically or probabilistically.

In this paper, CNSC staff provides the regulatory perspective for meeting CSA N285.4 and N285.8 requirements regarding the pressure tube to calandria tube contact.

Pressure Tube Flaw Characterization and Replication Challenges

Rebecca Wu, Mohammadreza Baghbanan and Steve Motomura
Ontario Power Generation

Integrity assessment of service induced flaws and artifacts detected on the inside surface of the CANDU pressure tubes is required prior to unit restart. Non-destructive examination (NDE) methods are used to characterize and size these flaws. For limiting flaws, additional inspection information is often required to demonstrate fitness-for-service and satisfy the acceptance criteria described in CSA N285.8.

Replication, which is obtaining a negative impression of the pressure tube surface at the flaw location, is used to obtain detailed flaw geometry information and/or assist in flaw characterization to confirm or validate NDE data. Depending on the quality of the replica, flaw geometry information such as length, width, depth, orientation angle (with respect to the pressure tube longitudinal axis), and flaw tip root radius may be obtained. This presentation will review replica evaluation for different flaw types in pressure tubes and will present some OPEX on challenges in obtaining a good quality replica from these flaws to support fitness-for-service evaluation and unit restart.

Delayed Hydride Cracking

Glenn McRae[1], Kit Coleman[2] and Sean Hanlon[2]
[1]Carleton University, [2]Canadian Nuclear Labs

The FFS of the Annulus Gas System depends on demonstrating sufficient agility to detect heavy water leaks before pressure tube breaks. Heavy water can leak when hydrogen isotopes are directed to flaws under tension in the body of pressure tubes to form brittle hydrides that can crack through wall in a process called delayed hydride cracking (DHC). In the 1970s and 1980s, DHC was responsible for several pressure tube failures in Canadian CANDU reactors. The most recent DHC failure was possibly last year in India. Leak-before-break is assured when crack rates are sufficiently slow to be detectable by observation of moisture in the annulus gas. Engineering models are unable to predict DHC rates reliably. Recently, a new formalism based on thermodynamics has been used to predict rates of laboratory experiments to unprecedented accuracy and precision. This formalism should be applicable to reactor conditions.

This presentation will review the basis for the new formalism, show the results of laboratory confirmation, and provide a roadmap for how to develop an equation for FFS.

Evaluation of In-Service Pressure Tube Flaws

Christopher Manu, Ben Dobbie and Danny Mok
Kinectrics Inc.

Pressure tubes are periodically inspected volumetrically for the presence of flaws as per the requirements of Canadian nuclear standard CSA N285.4 “Periodic Inspection of CANDU Nuclear Power Plant Components”. Clause 5 of CSA N285.8 “Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors”, provides the evaluation requirements for inspected flaws that exceeded the N285.4 disposition limit. Computer code PTFAP (Pressure Tube Flaw Assessment Program) was created to perform pressure tube flaw evaluations consistent with the requirements of Clause 5 of CSA N285.8. Pressure tube flaws are volumetric in nature and can be assessed as such or can be conservatively assessed as planar. In the volumetric assessment the flaw is evaluated for protection against crack initiation via constant load delayed hydride cracking (DHC), hydrided region overload, fatigue and also against plastic collapse. In the planar assessment the flaw is evaluated for protection against fracture initiation and plastic collapse after crack growth via DHC and fatigue are postulated.

Depending on the unit and the number of pressure tubes inspected, the number of flaws requiring assessment and disposition during an outage may range from about 20 to 500. Most such flaws, and thus the pressure tubes in which they reside, can be demonstrated to be fit for service using computer code PTFAP in a time efficient manner to support outage schedules. Occasionally, more difficult flaws are found during inspection which require additional analysis refinements.

Use of Channel Specific Gap Measurements in Probabilistic Assessments of Pressure Tube to Calandria Tube Contact

Eric Nadeau, Peter Daoud, Adrian Baniak and Jenny Kong
SNC-Lavalin Nuclear

Since the advent of the surface-riding eddy current probe that measures the gap between the pressure tube (PT) and the calandria tube (CT), gap measurements have become the critical input to derive the key distributions used in all probabilistic assessments of PT/CT contact and blister susceptibility. A benchmarking methodology is used to calibrate (or tune) the distributions of PT creep factor and PT end-slopes for a reactor by comparing it's pooled PT/CT gap measurements against the corresponding probabilistic predictions using the Monte Carlo method. Probabilistic assessments provide a list of channels that could be at risk if some specific combinations of PT creep and end-slope would occur in some of those channels. Gap measurements are regularly used to determine which channels from the probabilistic list are truly at risk from those that are not. A method was needed to feedback the information from channel specific PT/CT gap measurements into the probabilistic assessment of the reactor core. Hence, a new benchmark methodology has been developed to determine the channel specific PT creep factor and end-slope values that are compatible with the PT/CT gap measurements. The channel specific uncertainty of PT creep factor and end-slope is significantly less than that for the entire core, as it does not include the tube-to-tube variability. It is mostly dominated by the impact of measurement uncertainty, which is taken into account with this methodology. The use of channel specific information allows for optimal management of the reactor core maintenance.

Potential OPEX-Based Refinements to Fuel Channel Sag Models for Future Life Extensions

Paul Sedran
RESD Inc.

Of the predictive models for fuel channel deformation, the most critical for fitness-for-service evaluations is the sag model, implemented in the well-known CDEPTH code. The CDEPTH deformation model, based on current technical understanding and empirical data, have been successfully used in the CANDU industry for over forty years to assess the acceptability of fuel channel sag, including evaluations of post-SLAR spacer locations, of critical importance to the industry.

The model has been deliberately designed to produce conservative results and have been periodically modified to account for various new OPEX, such as inspection results. Over the years, several OPEX items related to the fuel channel deformation have come to the fore, as follows:

a. CANDU 6 Calandria Tube (CT) spring back measurements for the original Fuel Channel (FC) F06 in Point Lepreau, indicating a greater lateral stiffness than the current predicted value for the standard FC beam model
b. CANDU 6 CT in-service ovality measurements from periodic ultrasonic inspections which indicate a deviation from the circular CT shape assumed in the standard fuel channel model.

This paper presents the OPEX information in question and presents a case to correlate the spring back measurement results to various fuel channel structural details, such as CT ovality and sag, and FC elastic modulus, which may be used to justify modifications to the standard model.

It is expected that the deformation models , modified as proposed , will generate less conservative elongation and sag predictions than those from the current models, which could prove useful in future fuel channel life extension campaigns.

FFS of Rolled Joints

Glenn McRae[1], Kit Coleman[2] and Scott Langille[1]
[1]Carleton University, [2]Canadian Nuclear Labs

The rolled joint is a region in CANDU fuel channels where hydrogen isotopes accumulate in-service to the highest levels. Circumferential variation of concentrations in the rolled joints are higher than in the body of the pressure tube. Models of hydrogen diffusion by Fick's Law do not provide satisfactory predictions of hydrogen concentrations in the rolled joint. These models include the effect of concentration gradients, but do not include how stress gradients cause hydrogen to move, except in ad hoc ways that are not consistent with best-practice statistics and thermodynamic laws. This presentation will introduce an equation for hydrogen flux that includes the effects of both gradients of concentration and stress, and a formalism that is thermodynamically sound. An example showing predictions of hydrogen moving in zirconium will be shown. This presentation will also show the results of a novel method used to reduce hydrogen concentrations in simulated rolled-joint laboratory experiments.

Resistivity Change in CANDU® Pressure Tube Material due to Heat Treatment

Perryn Bennett[1], Matt Topping[2], Ross Underhill[1], Jordan Morelli[2], Mark Daymond[2] and Thomas Krause[1]
[1]Royal Military College, [2]Queen's University

For eddy current based inspections, material electrical resistivity is an important parameter. For example, material resistivity values for the pressure tube (PT) and calandria tube (CT) are required to ensure accurate measurement of the PT-CT gap in CANDU® reactor fuel channels. To test the hypothesis of whether in-reactor conditions, such as higher temperatures could change a material's resistivity, this study examined the effect of heat treatment on the resistivity of Zr2.5%Nb. At 400 °C and 450 °C, under anaerobic furnace conditions, sectioned PT samples were held for varying periods of time in order to partially decompose beta-Zr and produce varying fractions of omega-Zr. The resistivity of the heat treated PT samples was measured using the four-point method and changes in resistivity with time at temperature were recorded. The magnitude of the resistivity was observed to decrease by up to 10% with time in the furnace. Reduction of resistivity with heat treatment was associated with changes in the microstructure. Examination by transmission electron microscopy (TEM) showed an increase in the volume fraction of hcp omega-phase, and associated bridging between higher conductivity alpha-Zr grains, which as a consequence would result in an overall decrease of resistivity. These results have implications for the uncertainty of PT to CT gap measurement, where temperature variation arises along the channel and also between 6 and 12 o'clock at a given axial position.

Probabilistic Methods and Simulations

Regulatory Considerations for the Adoption of Probabilistic Assessment Methodologies for Pressure Boundary Component Aging Evaluations

Blair Carroll
Canadian Nuclear Safety Commission

Traditionally, deterministic methods have been used to evaluate the effects of aging to establish the remaining useful life of pressure boundary components for Canadian CANDUTM nuclear power plants (NPPs). The deterministic methods have been effective in maintaining the safe operation of NPPs, and they incorporate safety factors and bounding values for input parameters to address uncertainties associated with the actual operation of components. Considerable effort is underway within the CANDUTM industry to gain regulatory acceptance of probabilistic methodologies to evaluate the effects of aging on NPP pressure boundary components. Probabilistic methodologies, such as probabilistic fracture mechanics (PFM), use research and operational experience to better characterize the in-service performance of pressure boundary components and more accurately predict remaining useful life

Canadian Nuclear Safety Commission (CNSC) staff recognizes that there are potential benefits to using probabilistic methodologies; however, their adoption is often not a simple process. The pressure boundary codes, standards and assessment methodologies that currently form the licensing basis for NPPs are based primarily on deterministic approaches. When considering the acceptability of probabilistic assessment methodologies, CNSC staff must ensure that acceptable design and safety margins are maintained.

This technical presentation provides an overview of the considerations of CNSC technical staff tasked with reviewing the application of probabilistic aging evaluations to pressure boundary components.

Advanced Simulation Software to Assess How, Why, Where, and When Components Will Fail

James Carter
Vextec Inc., JECA

An advanced simulation system and software tool has successfully supported diagnostics; prognostics; forensic failure analysis; design, operation, and maintenance optimization; life-extension decisions; and other applications. This paper addresses the “Virtual Life Management®” (VLM®) process, which employs the “VPS-MICRO®” software to accurately account for a metal’s reaction to the imparted stress by creating a 3D, time and condition-based simulation to determine how, why, where, and when a part, assembly, or system will fail or how it has failed due to fatigue and fracture. The technology goes beyond Finite Element Analysis (FEA), expensive physical testing, and low-confidence historical material failure analysis by incorporating the material grain structure into the FEA model and performing Monte Carlo probabilistic techniques to address non-homogeneity of the structure. Operating conditions, such as in-service loading, cycling, environment, lubrication, etc. are also incorporated into the model.

Successful virtual life simulations have been employed in the airline, manufacturing, medical device, automotive, military, and energy industries. However, the process can be applied to all industries,

including nuclear power. Components in any industry, can be and have been addressed including large gear boxes, bearings, shafts, turbine blades, fan blades, pressure vessels, rollers, and piping/components. The virtual simulation can be performed for different metals and alloys. Costly and time-consuming physical testing can be reduced or eliminated with improved predictability. High confidence results can be used to eliminate critical `equipment downtime.

This paper and presentation will explain the VLM process, the VPS-MICRO software functionality, and offer technical and economic case studies of real-life applications.

Development of Probabilistic Models for Integrated Life Cycle Management

Joseph Cluever[1], Paul Bruck[1], Thomas Esselman[1], Mark Woodby[2] and Michael Taylor[2]
[1]LPI, Inc., [2]EPRI

The Electric Power Research Institute (EPRI) has developed the Integrated Life Cycle Management (ILCM) computer code to provide a standard methodology to support long-term asset management decisions. The initial release of ILCM provided likelihood of failure (LoF) curves for various systems, structures, and components (SSCs) found throughout nuclear power stations. The probabilistic models used to calculate these curves were based on a blend of physics-of-failure algorithms and empirical data. Since the initial release, ILCM has been augmented with an expert elicitation tool that calculates a likelihood of replacement (LoR) for a generic SSC and an optimization tool for value-based replacement scheduling called Component Optimization Analysis Tools (COATs). The LoR calculator uses Bayesian methods to convert an expert's estimate about the reliability of an SSC into a set of Weibull parameters that best describe the likelihood of component replacement due to failure or obsolescence. LoF and/or LoR curves can be used in the optimization COATs tool. COATs uses genetic algorithms to search through all of the possible replacement schedules in order to determine which schedule provides the lowest expected life cycle cost. This value-based optimization considers not only the reliability curves of the components and the associated costs for an in-service failure but also the timing of spare purchases and end-of-license considerations.

Investigation of Model Uncertainties for Stability Evaluation of Circumferential Crack

Yifan Huang, Xinjian Duan
Candu Energy Inc.

Quantification of model uncertainties is probably the most difficult part in the application of Probabilistic Fracture Mechanics (PFM), especially for the phenomenological models, which are typically developed by regression from the experimental data. These phenomenological models could have a variety of forms, involves different parameters and have various levels of embedded conservatisms.

Candu Energy Inc (CEI) has actively participated multiple international round robin activities related to PFM and Leak-before-break (LBB). One of the key observations is related to the stability evaluation of through-wall circumferential crack. North America PFM practitioners usually use J-integral based Elastic Plastic Fracture Mechanics (EPFM), limit load solution or a combination of these methods as the crack stability model. On the other hand, the Failure Assessment Diagram (FAD) is much more frequently used

by European nuclear community. To reconcile the results, there is a need to investigate the difference in evaluating crack instability using FAD, EPFM, and limit load solution such as net-section collapse (NSC).

PRAISE-CANDU Version 2.0 has been developed by CEI for various fitness for service and safety applications. As part of the validation effort of the P-C 2.0 stability model, a case study using NSC model in CEI's LBB Guideline and various FAD procedures in R6 and ASME FFS-1 were compared for a through-wall circumferential crack in feeder pipe dissimilar metal welds. The analytical assessments are also compared with the component test results. The variations among the stability models were quantified.

Probabilistic Reactor Core Assessment of Degradation Mechanisms Related to Pressure Tube Flaws

Ben Dobbie, Christopher Manu and Danny Mok
Kinectrics Inc.

Clause 7 of Canadian nuclear standard CSA N285.8 "Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors", provides the evaluation requirements for pressure tube integrity assessment of an entire reactor core. The computer code SCEPTR (Stochastic Core Evaluation of Pressure Tube Rupture) was developed to perform probabilistic pressure tube flaw evaluations and pressure tube leak-before-break (LBB) analysis using the procedures and models defined in CSA N285.8 and is intended to satisfy Clause 7.3.2 (Assessment of degradation mechanisms related to flaws) and Clause 7.4 (Evaluation of leak-before-break). The Monte Carlo simulation method with random sampling generates input data from the start of reactor operation to a user selected end of evaluation period which is subsequently evaluated for crack initiation via constant load delayed hydride cracking (DHC), hydrided region overload and fatigue. If crack initiation is predicted then evaluation of flaw growth to through wall penetration is performed, followed by LBB sequence of events assessment of any through wall flaws. This process is repeated for the desired number of simulated reactor cores (typically 100,000) to ensure numerical convergence of results. The information for the reactor cores is then post processed and the pressure tube rupture frequency is determined.

Balance-of-Plant

Maintenance Cooling System HX Shell Replacement without Affecting Recall Times

Stephen Fluit[1], Jeff Jarvis1, Mark Bevan[2], Geoffrey Lawson[2] and Tiberiu Preda[2]
[1]BWXT Canada Ltd., [2]Bruce Power

The Maintenance Cooling System Heat Exchangers (MCS HX) in Bruce A U3 and U4 are Class 3 (ASME Sec. III, ND) heat exchangers that have been in service since 1978/1979. The shell side of the HXs have experienced significant degradation from localized pitting caused by Microbiologically Influenced Corrosion (MIC). Engineering analysis of the degradation predicted perforation of the MCS HX shells prior to the planned replacements in 2021 / 2022.

A traditional solution would be to replace the shell, however the MCS HXs are required to be available for service as an emergency heat sink while the unit is operating (with a recall window of 10 hrs), and must also be available when the unit is shut down. Furthermore, the MCS HXs are located in a room with limited clearance and a small man door for access. These constraints preclude a traditional shell replacement.

BWXT, in conjunction with Bruce Power, designed, registered, mocked-up, and implemented a unique repair strategy that resulted in a complete encapsulation of the MCS HX shells. New shell segments were maneuvered into the MCS HX room and welded in-situ, allowing the MCS HX to be available for service at all times. The encapsulation repair limited the amount of welding on the existing pressure boundary to the nozzles, penetrations and supports. This minimized the risk of burn-through, which would compromise the integrity of the existing shell during installation. The Unit 3 and 4 MCS HX encapsulation repairs were performed with the units at power, and without affecting availability for recall. The MCS HX shells are now fit for service until the vessels are replaced during the Asset Management replacement window.

Vibration Monitoring Locations & Allowable Levels Along Small Bore Pipes

Usama Abdelsalam[1], Stephen Jeremia[2] and Dk Vijay[2]
Independent Contractor, [2]Kinectrics Inc.

The vibration velocity is widely used as a limiting criterion to guard against fatigue damage in piping systems. The velocity criterion limits the measured vibration velocity at a specific location to a calculated allowable level. Ideally, locations of maximum vibration velocity are sought. However, in the absence of a well-defined forcing function that causes the actual piping vibration, the location of the maximum vibration velocity along the piping system is not known a priori. This paper presents a finite element analysis performed to identify appropriate locations for placing vibration monitoring equipment along a small bore piping layout. Generic excitations over a wide frequency range are used to excite the piping system and the corresponding velocity responses at several locations are determined. The main criterion for ranking these locations is twofold; ability to experience reasonable vibration levels in all three orthogonal coordinates, and being accessible for the installation of the instrumentation. Two types of excitations are considered; harmonic excitation at each individual pipe bend and base excitation at the two terminal anchor points in each individual orthogonal coordinate. Estimates of the allowable vibration

velocities at the identified locations are discussed. This paper provides a systematic approach to determine proper locations for installing vibration monitoring instrumentation. In addition, it provides a conservative estimate of the allowable velocity at each individual location.

Flow Accelerated Corrosion Evaluation of Main Steam Pipe

Morteza Toloui, Sung Tran Tran, Brain Seed and Ella Pakravan
Ontario Power Generation

Flow accelerated corrosion (FAC) is a major concern in nuclear and thermal power generating plants as it can cause rapid thinning of pipes by the dissolution of the passive oxide layer formed on the pipe surface. A large number of components made from carbon steel in the primary and secondary side are prone to wall thinning (loss of metal) caused by FAC. Pipelines afflicted by FAC due to exposure to either single phase water or two phase wet steam in the secondary side, have significant potential for afflicting sudden ruptures. Wall loss caused by FAC is the known cause of multiple, significant pipe failures across the nuclear industry and poses a significant risk to high-energy lines carrying water or steam.

During a planned outage of unit 8 at Pickering Nuclear Generating station, a routine FAC inspection on one of the Main Steam header outlet piping discovered a band of low wall region adjacent to a weld connection. As the result, the organization decided to replace the piping and fitting upstream and downstream of the weld. This provided an opportunity to conduct destructive examination on the Main Steam piping internal diameter (ID). The purpose of the examination was to confirm FAC by identifying signs or indications of FAC. Signature patterns of FAC, minor localized single phase flow-accelerated corrosion semi horseshoe pit features, were reported. Ultrasonic wall thickness measurements, together with the examination of surface features observed on the inner surfaces of the pipe were used for extent of condition assessment. Careful examinations revealed that the thin walled section adjacent to the weld was due to inadequate welding preparation at the time of installation. Extent of condition assessment results are expected to help with Fitness-for-Service (FFS) assessments of future inspection results of components in similar locations, avoiding unnecessary future repairs.

Deaerator tank seam weld crack repair using Gas Tungsten Arc Welding (GTAW) process - Pickering NGS Unit 1

Ashraf Sadek
Ontario Power Generation

Linear indications were discovered on Pickering Unit 1 Deaerator weld W6 during P1711 outage inspections. The nine linear indications observed were at the 6 O' Clock position. Light surface polishing of the indication area was carried out and confirmed the presence of cracks in the shell adjacent to the Heat Affected Zone (HAZ). Additional inspections were performed to determine the extent of condition, and identify crack depth. Wet fluorescent magnetic particle (MT), visual examination (VT) and Phased Array Ultrasonic Testing (PAUT) inspections confirmed the indications were confined to the discovery area and no through wall cracks were observed from the vessel exterior. The repair strategy was to grind the crack indications and build back using GTAW process. The remaining ligament on three of the nine cracks was found too close to the qualified welding procedure's minimum allowed wall thickness. The three deep cracks were ground through wall and the location was built by welding both from the tank interior and exterior.

Fitness for Service Assessment of a Flange Joint with Missing bolt(s)

Vishash Sharma, Aman Usmani, Reza Adibiasl and Edwin Chen
Kinectrics Inc.

Flanged joints are a commonly used component in the nuclear power plants. Occasionally, during in-service inspections of an operating plant such joints may be found deficient such as having one or more missing closing bolt(s). The assessment is required to determine if the joint will be fit for service 'as-is' until the next planned outage when a permanent fix can be planned and installed. This paper provides methodology to perform fitness for service assessment using an example. Conventional and Finite Element techniques are used to illustrate the proposed methodology. The impact on leak tightness, bolt preload and gasket seating loading is discussed.

OPEX, FFS methodology, Materials factors, Other

Continued Fitness for Service of the Bruce B Unit 7 Calandria Relief Ducts

Andrew Brooks[1] and Sahil Gupta[2]
[1]Bruce Power, [2]Kinectrics Inc.

This paper presents an overview of the specialized inspection tooling and the Fitness-for-Service (FFS) work completed on the 18 inch Calandria Relief Ducts (CRDs) at the Bruce Nuclear Generating Station B (BNGS) Unit 7. The Unit 7 CRDs are constructed from 304L stainless steel which is exposed to low levels of chlorides. It is believed that metal was impregnated with iron contamination during original construction, which makes it susceptible to degradation from chloride induced transgranular stress corrosion cracking (TGSCC). A first of a kind specialized pipe crawler robotic tooling system was developed, that used combination of Ultrasonic (UT), and Eddy Current probes capable of performing high precision volumetric inspections from the inside diameter (ID) of the CRD. In addition, a new simplified enhanced CRD visual inspection tooling system was also deployed at BNGS Unit 7 during the February 2019 outage campaign. Results from the volumetric examinations and detailed visual inspections were subsequently used to characterize and size the indications for length, depth and orientation angles. The inspection data along with finite Element Analysis (FEA) stress modelling was used to complete Fitness for Service (FFS) analysis for CRD. FFS analysis was completed using the fracture mechanics techniques described in ASME SEC XI and IWA-3300 flaw characterization guidance [R-1] adapted for CRD specific application.

Reactor Inlet Header Temperature Overview: External preheater fouling study

Ernest Lu
Bruce Power

Reactor Inlet Header Temperatures (RIHT) are rising in the Bruce Units and in CANDU Units in general. Its increase is mainly due to the ongoing accumulation of deposits on Inside Diameter (ID) of SG and Preheater (PH) tubes. This paper explains the challenges in the RIHT monitoring process and provides an update from recent studies.

Fouling is the deposition and accumulation of unwanted materials such as magnetite on the internal or external surfaces of heat exchanger tubing. It impedes tube inspection and degrades heat exchanger thermal performance which causes rise of Reactor Inlet Header Temperature (RIHT), one of the key safety parameters monitored in CANDU type reactors. This paper shares a few studies Bruce Power has completed over the past years on external preheater to understand the extent of degradation for the purposes of trend analysis and mitigation.

NRU Return to Service Project – Fitness-for-Service

Michael Kozluk
CANTECH Associates Ltd.

The National Research Universal reactor (NRU) safely shutdown on 2009 May 14th due to a loss of Class 4 power. During the inspections conducted on May 15th, to enable re-start of the reactor, a leak was discovered in the inner heavy water reactor vessel. The reactor was kept in the shutdown state and inspections were undertaken to determine the location of the leak and an in-depth assessment of the condition of the reactor vessel was initiated. The assessment revealed that the wall of the inner vessel above the bottom circumferential weld had several local areas of 'loss-of-section' due to corrosion. A thorough inspection of the bottom 320 mm of the inner vessel was carried out and the stresses in the thinned area of the vessel with and without considering corrosion allowance were evaluated. This paper describes the NRU reactor, the extent of observed degradation, the preliminary fitness-for-service assessment performed as input into establishing the repair plan for the inner heavy water reactor vessel, and the subsequent fitness-for-service assessment performed to support the return to service and ongoing 28 day medical isotope production operating cycle in 2010 August.

Comprehensive Study of Carbon Steel to Alloy 600 Dissimilar Metal Weld

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In CANDU nuclear power plants, dissimilar metal welds (DMWs) between the Alloy 600 flow element and the SA-106 Grade B carbon pipe with Alloy 82 or Alloy 182 filler material are present in some of the feeders. Light Water Reactor (LWR) operating experience and research suggest that such dissimilar metal welds on outlet feeders may be susceptible to primary water stress corrosion cracking (PWSCC) especially at the welding start/stop locations. The Initiation and growth of PWSCC in a DMW are driven primarily by welding residual stresses (WRS). Large scale comprehensive R&D programs have been funded by Bruce Power, Ontario Power Generation (OPG) and Candu Energy during the last decade to thoroughly study the factors that affect PWSCC initiation, growth and stability. The following topics are to be covered in the present paper:

- A series of DMW samples between Alloy 600 pipe and SA-106 carbon pipe are fabricated using different welding processes, joint design and welding techniques. Insights into the effects of these different welding variables on mechanical properties (tensile properties and hardness of weld materials and heat affected zone), metallurgical properties (macro and microstructure examination) and chemistry (root pass alloying dilution etc.) are provided.
- WRS measurements using X-ray diffraction, neutron diffraction and Contour methods are compared. • WRS modeling using finite element method has been developed and validated against the WRS measurements.
- Fracture mechanics tests are performed to validate the analytical models for predicting the stability and crack growth of postulated large through-wall crack.
- A tiered composite risk-informed LBB case has been developed.

The knowledge gained from these 10-year R&D programs has been applied to assess the emergent technical issues encountered in the execution of Bruce Power's Major Component Replacement (MCR)

and OPG's The Re-tube and Feeder Replacement (RFR), and many other safety significant operation issues.

Fitness for Service Assessment in Local Thinned Areas of a Cylindrical Vessel

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The industrial components such as pipe, vessels and heat exchanger shells are usually subject to wall loss degradation due to corrosion and pitting during operation of power plants. Fitness for service assessment is required to determine the integrity and safety of component for continuous operation. This is based on information from periodic in-service inspections, surveillance and assessment activities throughout the period of service life. Finite element analysis as an advanced technique may be necessary to evaluate more precisely acceptability of localized stresses due to wall thinning at Locally Thinned Areas (LTAs). The projected thickness data at the end of the evaluation period based on current measured inspection data grid can be directly mapped onto the finite element model using shell or solid elements, as permitted by API579-1/ASME Section III. The study in this paper provides review of the available linear and non-linear methodologies for fitness for service assessment of LTAs. As an example, methods of evaluation are demonstrated for a typical cylindrical vessel shell under internal pressure using both solid and shell elements. First the evaluation is based on a linear stress analysis with acceptability determined using stress categorization and comparison with code limits. The validity of the results based on variable thickness shell elements is established by comparing the results using solid elements. If the linear elastic method is found inadequate, nonlinear limit load analysis is performed. The acceptability of localized thinned region against blow off is demonstrated using strain-based criterion.

Concrete Performance at Elevated Temperature – A State of the Art Review

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The characteristics and behaviour of concrete in existing nuclear stations under various environmental conditions continue to receive attention to retain its protective qualities for new and existing nuclear stations and facilities. Currently, upper temperature limits of 65°C over large areas and 100°C over local areas are specified in the Canadian Standards Association (CSA) Standard N287.3 for the design of concrete containment when exposed to heat for prolonged durations. However during normal operation, nuclear concrete structures may experience elevated temperature excursions at some locations. Several studies were performed to examine the effects of high temperature on concrete. In some recent studies, it has been indicated that temperatures beyond these limit have little effect on concrete properties and durability. The literature consists of results of experimental programs, and studies based on numerical simulations. The experimental and numerical simulation studies addressed the changes of physical and chemical parameters of concrete under elevated temperature and their effect on its mechanical properties.

This paper provides a state of the art summary review from the literature and performed assessments and provides a convenient reference that can assist evaluation in various scenarios.

Origins of Feeder Fitness-for-Service Guidelines

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The CANDU® Owners Group (COG) Feeder Integrity Joint Project (FIJP) was initiated to consolidate and oversee the industry's research and development (R&D) programs necessary to manage service-induced degradation originally observed in the outlet feeders at the Point Lepreau Nuclear Generating Station (PLGS). 1996 The COG Feeder Thinning Joint Project was initiated because of wall thinning detected in extrados of PLGS outlet feeder bends. 1997 The COG Feeder Cracking Joint Project was initiated because of the through-wall axial crack in PLGS outlet feeder bend of fuel channel S08. 2001 The through-wall axial crack in PLGS outlet feeder bend of fuel channel K16 lead the industry to merge the two feeder joint projects into the COG/FIJP.

The COG/FIJP was active for ten years (2001 to 2010). The principal areas of R&D were:

- repair and replacement
- non-destructive examination
- cracking
- chemistry
- industry data integration
- fitness-for-service

Over time, some of the R&D work areas were transferred to the COG Chemistry, Materials & Components (CM&C) research program. This paper describes genesis and evolution of the feeder fitness-for-service guidelines (FFSG) that was originally developed under the COG/FIJP. In 2007 the feeder FFSG was transferred to the COG/CM&C program and was issued for three-year trial use. Currently, revision 3 of the feeder FFSG is cited in the CNSC Licence Conditions Handbook for the operating Canadian plants.

Design Evaluation of SMRs for Utilization of Spent fuel

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This report presents the findings of the project undertaken to explore potential design changes to three small modular reactors (SMR) concepts so that the SMRs can use spent fuel from pressurized water reactors (PWRs) and/or CANDU reactors. Countries with existing nuclear power plants are now focused on nuclear waste management concerns. Thus, SMR technologies that use spent fuel are an attractive option. The following SMR designs are explored in the project: the ARC-100 sodium-cooled reactor, the SMR-LLC/SMR-160 pressurized water-cooled reactor, and the Integral Molten Salt Reactor (IMSR). The possible design changes and their implications have been discussed. A methodology is developed to quantify the difficulty of implementing these design changes and then compare the modification potentials of the three SMRs to use spent fuel.

Steam Generators

Case study - Bruce Power reactor original steam generator divider plate degraded bolt

Ernest Lu and Justin Howatt
Bruce Power

Bruce Power station B original steam generator design uses a bolted joint design to connect the divider plate segments. Over time, some bolt was found degraded likely as a result of flow accelerated corrosion (FAC). Managing this kind of degradation mechanism is challenged by the high dose environment within the steam generator primary bowl. This paper provides a summary of the problem identification, fitness for service assessment on divider plate structural integrity and effects on downstream equipment, and mitigating actions taken by Bruce Power.

Inspection of Newly Installed and In-service Plug Welds

David Burr[1], Nathan Bruns[1], Steve Fluit[1] and Ken Sedman[2]
[1]*BWXT Canada Ltd.*, [2]*Bruce Power*

Tubes in CANDU steam generators (SGs) and Class 1 heat exchangers with active degradation are often removed from service by installing welded plugs. Much advancement has been made in automating tube plugging, with the goal of producing more consistent and better quality welds. However, a small number of cases in recent SG inspection campaigns at Bruce Power have demonstrated that additional advancements are possible. Recently, a small number of automatic tube plugs have shown evidence of potential, minor leakage after a short operating interval. These plug welds successfully passed the required visual inspection when installed. It is possible that this potential leakage was caused by in-service degradation. Other suggested causes of the failure include weld defects due to contamination, lack of fusion between the weld and parent metal, and porosity or other subsurface defects in the welds. The main methods for plug weld verification in the field are visual inspection (VI) and dye penetrant (PT) inspection, which are surface inspections. Development of volumetric inspections is ongoing which may allow for increased confidence of safety and reliability in the operation of nuclear power plants.

Evaluation of Wear Degradation for Steam Generator Tubing Rupture

Brent Capell
EPRI

Steam generator (SG) tubing represents the largest surface area between the primary and secondary circuits for pressurized water reactors. To maximize heat transfer, SG tubing is relatively thin (less than 1.2 mm in thickness) which means that degradation of the SG tubing can rupture leading to a significant primary to secondary leakage event. Historically, SG tube leaks were related to degradation due to stress corrosion cracking (SCC) and evaluations have been performed to correlate low level leakage during operation to the probability of burst (POB) for various plant conditions. However, in the US the most recent leakage events have instead been associated with wear degradation and not SCC. Wear occurs

due to foreign objects, interaction of the tubing with support structures, and even tube to tube contact. This presentation is an evaluation of the POB due to flaw shapes associated with tube wear, especially as a function of operational primary to secondary leakage amounts.

Non-destructive Detection and Characterization of De-alloying Degradation in Steam Generator Tubing

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A new degradation mechanism has been observed in Monel 400 Steam Generator tubing material. Monel tubing material is a nickel-copper alloy (63Ni-28Cu-2½Fe) with the ASME material designation SB-163/N04400. The degradation is associated with denickelification of Monel, where the Nickel alloying element leaches out, leaving behind a porous copper matrix. Detection, sizing and characterization of the steam generator tubing degradation by non-destructive examination (NDE) techniques is one of the fundamental requirements in establishing a fitness for service criteria. In this investigation, eddy current, bobbin probe and X-probe as well as Tiny Rotating Ultrasonic probe (TRUSTIE™) were used for detection and characterization of the flaws. The results were compared against the results of destructive metallurgical examination of the flaws, to assess the capabilities and limitations of each technique.

Case Study of Steam Generator Foreign Material Assessment

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While procedures to avoid introducing foreign materials (FMs) to the secondary side of steam generators are implemented and enforced by operators, the discovery of FMs in the secondary side of SGs is inevitable. Retrieval of the FMs is the best practice when faced with the discovery of FMs in the secondary side of steam generators, however foreign object search and retrieval (FOSAR) is not always feasible and must be weighed against the potential to introduce other foreign material with retrieval attempts. Considering that debris fretting is the number one cause of steam generator tube primary to secondary leakage, it is important to evaluate FOSAR options as well as study the potential for tube damage and assess fitness for service if retrieval is not feasible or unsuccessful and the FM must remain in place.

This paper presents a case study of a loose piece of FM that was introduced to the secondary side of a steam generator during routine inspection and maintenance activities. An evaluation of FOSAR options and the fitness for service evaluation to assess the potential for tube damage are discussed.

Management of Ongoing Tube Support Plate Degradation

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[1]BWXT Canada Ltd., [2]Bruce Power

Significant TSP degradation was first observed in the Bruce Power Unit 8 steam generators in 2003. While initial predictions suggested the steam generators would only be operable for a few years, the

degradation has been successfully managed in the fifteen years since. Techniques for managing the degradation involve installation of additional supports, monitoring and prediction of TSP degradation rates through analysis of primary side Eddy Current (ET) inspection data using automated techniques inter-tube visual inspections from the secondary side, and targeted plugging of regions at risk of losing support. A unique method is employed to provide consistent measurements of remaining ligament from still frames of the visual inspection videos that can be compared inspection to inspection. Due to repeated visual inspection over the past fifteen years a more accurate degradation rate has been determined, allowing more accurate future predictions, both to monitor the condition and for plugging decisions. Tube plugging decisions are made based on a combination of the analyzed ET data and visual inspection results, with visual inspection results used to validate the ET and to overwrite the ET results in critical regions. Employing these techniques has allowed for continued safe operation of Bruce Unit 8 and increased confidence that the SGs will be able to continue operation until the scheduled refurbishment outage.