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TERMS AND ABBREVIATIONS

AECL	Atomic Energy of Canada Limited
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CANDU	CANadian Deuterium Uranium®[a]
CE	Condensate Extraction
CHUG	CHECWORKS User Group
CODAP	Component Operational Experience, Degradation & Ageing Programme
COG	CANDU Owners Group
DN	Diameter Nominal
EFPY	Effective Full Power Years
EPRI	Electric Power Research Institute
FAC	Flow Accelerated Corrosion
HP	High Pressure
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
INPO	Institute of Nuclear Power Operations
KEPCO	Kansai Electric Power Company
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
NPS	Nominal Pipe Size
NRC	Nuclear Regulatory Commission
PWR	Pressurized Water Reactor
RT	Radiographic Testing
US	United States
UT	Ultrasonic Testing
VVER	Vodo-Vodyanoi Energetichesky Reaktor (Water-Water Power Reactor)

[a] CANDU is a registered trademark used by Candu Energy Inc. under licence from Atomic Energy of Canada Limited).

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1. INTRODUCTION

Flow accelerated corrosion (FAC) gained significant attention in the nuclear industry after fatalities were experienced following the rupture of the suction line to the main feedwater pump at the Surrey Unit 2 Pressurized Water Reactor (PWR) in the United States (US) in 1986 and the rupture of a high pressure condensate line at the Mihama Unit 3 PWR in Japan in 2004. The lines in the balance of plant piping systems at nuclear power plants (NPPs) are typically constructed from carbon or low alloy steel and are susceptible to this form of corrosion, which can result in significant loss of pipe wall thickness. The nuclear piping systems in the primary circuit of PWRs are typically fabricated from stainless steel and are therefore immune to FAC. However, CANDU nuclear piping systems, which carry heavy water in the primary circuit, are constructed of carbon steel; hence, under certain combinations of flow conditions and water chemistry, FAC degradation of CANDU nuclear piping is plausible.

In response to the concerns over the potential for FAC related degradation of pressure boundary systems and components, the US industry, through the Electric Power Research Institute (EPRI), undertook the development of a program to manage FAC. EPRI has issued a recommended practices document [1] and developed predictive modelling software (CHECWORKS) that have been implemented at US and Canadian NPPs to manage FAC related degradation. While the Canadian CANDU NPP operators are active participants in the EPRI programs, they form a minority group with NPPs that have the potential for FAC to impact nuclear piping systems, due to the aforementioned differences in materials and operating conditions relative to the balance of plant piping. CANDU NPP operators have established additional activities related to the management of FAC through the CANDU Owner's Group, but the majority of the recommended practices employed in FAC management are based upon PWR experience.

Given that the CANDU NPPs are entering periods of extended operation and initiatives are underway within the CANDU industry related to evaluations of large break loss of coolant accident probabilities and leak before break evaluations for large diameter piping systems, the present report provides a technical review of the application of the EPRI recommended practices for FAC management to assess their applicability to CANDU nuclear piping systems larger than 6 inches in diameter.

This report is broken down into the following sections;

Section 2 – Provides an overview of the FAC mechanism, highlighting how the mechanism may be impacted by the aforementioned differences between CANDU nuclear piping and balance of plant piping.

Section 3 – Provides a summary of major world wide experience with FAC failures in nuclear plants.

Section 4 – Identification and discussion of EPRI recommendations for an effective FAC program.

Section 5 – A comparison of factors that influence FAC between CANDU nuclear piping and balance of plant piping.

Section 7 – Discussion presented on the strengths and potential areas of improvement associated with the noted recommendations.

Section 8 – Provides a summary of the report.

2. FAC MECHANISM

Prior to reviewing the recommendations made by various organizations in implementing FAC management programs in Pressurized Water Reactors (PWR) and Boiling Water Reactors (BWR), it is useful to first consider the FAC mechanism. In particular, it is important to understand the fundamental prerequisites for FAC and considerations that may be unique to CANDU nuclear piping systems.

FAC is a mechanism in which magnetite, the normal protective oxide that forms on carbon steel surfaces exposed to water under 'reducing'[b] conditions, undergoes dissolution at higher than expected rates. To understand why the magnetite oxide may undergo rapid dissolution, it is useful to review the basic process of corrosion.

When a carbon steel surface is exposed to water, it is thermodynamically favourable for iron to give up its electrons through an oxidation half reaction forming either ferrous species (Fe^{2+}) and/or ferric species (Fe^{3+}).

$$Fe \rightarrow Fe^{2+} + 2e^-$$

$$Fe \rightarrow Fe^{3+} + 3e^-$$

The liberated electrons are then consumed on the same metal surface by reacting with species such as dissolved oxygen under alkaline conditions to form hydroxide molecules, or with water molecules to form molecular hydrogen, in the reduction half reaction.

$$O_2 + 2H_2O + 4e^- \rightarrow 4OH^-$$

$$2H_2O + 2e^- \rightarrow H_2 + 2OH^-$$

The oxidized iron forms an iron oxide. Under CANDU HTS conditions, the iron oxide takes the form of a magnetite, Fe_3O_4, which contains one ferrous ions (Fe^{2+}) and two ferric ion (Fe^{3+}). Under low-flow or stagnant conditions, the magnetite oxide grows thicker and more protective, thereby reducing further corrosion and passivating the surface. On carbon steel, the kinetics of the corrosion process typically follows a parabolic rate law, i.e., the metal loss and oxide thickness increases proportional to the square root of time.

[b] The term 'reducing' is relative, since carbon steel under high temperature aqueous conditions is sufficiently oxidizing to corrode. The term 'reducing conditions' describes the water chemistry environment in the primary circuit of CANDU and PWR plants that includes the addition of dissolved hydrogen (and elimination of dissolved oxygen) to reduce the electrochemical potential of circuit materials. These conditions differ from the oxidizing conditions that were originally implemented in early BWR plants (many BWRs currently operate with more reducing hydrogen water chemistry conditions).

The magnetite oxide that forms on the corroding carbon steel surface can dissolve in the water in which it is in contact. There are several factors than can influence this dissolution process;

- Magnetite solubility: The higher the iron solubility in the aqueous solution in contact with the carbon steel surface, the higher will be the magnetite dissolution rates from the corroding surface.
- Steel composition: Some transition metal oxides are significantly less soluble than iron oxides such as magnetite. Hence, small quantities of transition metals, such as chromium, present in the carbon steel can significantly reduce the magnetite dissolution rate.
- Hydrodynamics: The dissolution of iron from magnetite films coating the inside of carbon steel nuclear piping is often mass transport limited as the dissolved iron must diffuse across the laminar boundary layer that typically forms in the fluid at the inside surface of the pipe.

If the oxide dissolution rate and the accompanying metal oxidation rate are sustained at values that are significant on an engineering scale, then the process is referred to as flow accelerated corrosion. Under steady state water chemistry and hydrodynamic conditions, the rate of wall loss is typically linear with time.

Each of the aforementioned factors affecting magnetite dissolution is considered in more detail in the following subsections. In particular, the differences in these factors between the CANDU primary side HTS and the secondary side of PWRs and CANDU plants are highlighted.

2.1 Magnetite Solubility

Magnetite is an ionic solid that can undergo reductive dissolution[c] in aqueous solutions. Its solubility depends upon;

- Temperature
- Water pH
- Electrochemical Potential

Depending upon the pH conditions, magnetite can display both normal solubility, which means the solubility increases with increasing temperature, and inverse solubility, which means the solubility decreases with increasing temperature. Hence, depending upon the pH conditions, an increase in temperature can either increase or decrease FAC rates.

The pH of the coolant in CANDU primary side systems is controlled using lithium hydroxide, which when dissolved in heavy water yields a solution of Li^+ and OD^-. Lithium hydroxide is a non-volatile, strong base that remains fully dissociated in the liquid phase under CANDU HTS conditions, including at elevated temperatures.

[c] During the dissolution process, it is generally thermodynamically more favourable for the one ferric ion to be reduced to a ferrous ion.

The pH of the water in secondary side systems is controlled using one or more volatile alkalizing agents. Volatile agents are used to ensure that they are transported, along with the steam, from the steam generator to the turbines, thereby protecting the carbon steel surfaces. These volatile alkalizing agents are typically amines such as ammonia or morpholine and are referred to as weak bases since their dissociation in water, and effectiveness as an alkalizing agent, decreases with increasing temperature. As a result of the varying volatilities and dissociation constants for these volatile alkalizing agents, the pH of secondary side systems, and the resulting FAC rate dependence, is more complex than in the CANDU primary side systems.

Magnetite solubility also depends upon the electrochemical potential. As noted earlier, magnetite oxide is comprised of both Fe^{2+} and Fe^{3+} ions. However, due to the lower solubility of Fe^{3+}, magnetite dissolution typically requires the reduction of Fe^{3+} to Fe^{2+}. When oxygen is present, this reduction reaction is decreased dramatically, effectively stopping FAC[d]. Hence, chemistry control in the balance of plant in nuclear plants is typically a compromise between

- ensuring the presence of sufficient concentrations of dissolved oxygen within much of the feedtrain to reduce FAC rates and
- ensuring relatively little dissolved oxygen entering the steam generator where it can contribute to the corrosion of the nickel alloy tubing.

In contrast, the CANDU HTS is operated with an overpressure of hydrogen, which means that during normal operation, there is virtually no dissolved oxygen in the water.

2.2 Steel Composition

As previously noted, the solubility of some transition metals is significantly less than that of iron. Hence the presence of even low concentrations of these transition metals in the carbon steel can significantly reduce the FAC rate by forming lower solubility metal oxides. The most important transition metal with respect to reducing FAC rates of carbon steel is chromium. Several sets of data examining the FAC rate dependence on the chromium concentration have been produced and these data were reviewed by EPRI in 2003 [2]. Generally, data produced by AECL under CANDU HTS primary side conditions indicated a greater reduction in FAC rate for a given chromium concentration than that reported by some of the other groups. However, it is worth noting that the AECL data was obtained following long term testing, involving exposures of up to 611 days, whereas the data produced by some of the other groups were obtained from short term testing after only a few 100’s of hours. Since the benefit of chromium is expected to increase with time, as the chromium oxides are preferentially retained on the surface, the larger reductions observed in the AECL data are not surprising.

After reviewing all data, EPRI concluded that all of the test data represented equally valid rate reductions and an updated correlation curve based on all of the data was recommended for incorporation into CHECWORKS. Furthermore, the effectiveness of

[d] Oxygen addition has been practised in fossil plants as a means of controlling FAC.

chromium in reducing FAC rates was considered equal for both single phase and two phase conditions.

2.3 Hydrodynamics

Hydrodynamics is important in assessing FAC susceptibility as it directly influences the transport of dissolved species, in equilibrium with the magnetite oxide, from the oxide-water interface to the bulk water. When water flows through a long straight pipe at high velocity, a laminar boundary layer is established at the pipe wall surface; the higher the velocity, the thinner the laminar boundary layer. The rate of magnetite dissolution is often limited by the mass transport, by diffusion, of ferrous ions, produced at the oxide-water interface, across the laminar boundary layer; the thicker the boundary layer, the lower the rate of oxide dissolution and the lower the FAC rate. Turbulence introduced into the water flow reduces the thickness of the laminar boundary layer adjacent to the corroding surface, and increases the mass transfer rates and associated magnetite dissolution rates.

Components such as flow elements, elbows, tees, reducers, etc., can all introduce turbulence into the flow and locally increase FAC rates [1], [4]. These components are equally found in the CANDU primary side nuclear piping systems and secondary side of CANDU and PWR plants. The geometry factors contained within software tools such as CHECWORKS are expected to equally apply to the same geometries in the CANDU nuclear piping systems.

3. WORLD WIDE NUCLEAR PLANT EXPERIENCE

The most extensive database of FAC related events is called Component Operational Experience, Degradation & Ageing Programme (CODAP) and is maintained by the Organisation for Economic Co-operation and Development. In the most recent reporting [4], the database at the time of the report contained 1,990 records involving wall thinning below the minimum allowable wall thickness, including events related to through-wall leaks and ruptures, over the period of 1970 to 2012. Based on the geographical breakdown of these events, 1655 of the 1990 events were reported for North American plants, suggesting a disproportionate representation considering that of the world's 450 operational and long-term shutdown reactors listed on the IAEA website, only 121 are located in the United States, Canada and Mexico. Hence, the reported events likely represent only a fraction of events that have occurred worldwide.

A detailed review of the 1,990 events in the CODAP database is beyond the scope of the present document. However, there are a number of observations from these reported data [4] worth noting;

- ~80% of the events are from plants operated in the US, which results in a strong US influenced perspective.
- In the case of large bore piping (defined as >DN100 or >4 NPS), there are
    - 1138 events of wall thinning,

  - 217 events involving through-wall leaks,
  - 50 events involving rupture
- In the case of small bore piping (defined as ≤DN100 or ≤4 NPS), there are
  - 265 events of wall thinning,
  - 296 events involving through-wall leaks,
  - 25 events involving rupture

The smaller number of wall thinning events for small bore piping but yet a larger number of through-wall leaks is likely the result of small bore piping not being as extensively addressed through in-service inspection. This situation has changed in recent years as the attention on small bore piping has increased.

- For the period of 1990 to 2012, there has been an average of 38 FAC events reported annually. For the period of 2009 to 2012, the annual average has been 14.

This reduction in FAC events in recent years appears to be a statistically significant trend and is likely a result of the increased effectiveness of the pipe wall thinning programs in managing FAC degradation.

3.1 Significant FAC Failures

EPRI has identified a number of significant FAC failures that have occurred at fossil and nuclear plants worldwide. A summary of these events, taken from Reference [1], is provided in Table 1. Some of the more significant failures, which have been documented in more detail in the literature, are described in the following subsections. The subsections are organized in reverse chronological order, starting with the more recent failures. The lessons learned from these failures have directly resulted in many of the recommendations discussed in Section 4.

Since the most recent revision of the EPRI guidelines [1] is based on operating experience available up to 2012, the CODAP on-line database [3] was searched for failures in piping ≥6 NPS attributed to the following mechanisms;

- FAC
- Erosion
- Erosion-Cavitation
- Erosion-Corrosion

No piping failures attributed to these mechanisms were contained in the database.

All of the failures noted in Table 1 are associated with balance of plant piping and not nuclear piping, which is the focus to the current study. As a result, a second search was performed of the CODAP database [3]for nuclear piping with a size ≥6 NPS for failures that have been attributed to the above four FAC/erosion mechanisms. No failures have been reported. This is not surprising since higher alloy steels, which are immune to FAC, rather than carbon steel or low alloy steels, are typically used in the nuclear piping in PWRs and BWRs.

3.1.1 Mihama Unit 3

Mihama Unit 3 is a 826 MWe 3-loop Westinghouse-type PWR located in Mihama, Japan. The unit was connected to the grid in 1976. On August 9, 2004, a 22-inch pipe immediately downstream of an orifice in the condensate system ruptured resulting in the deaths of five workers and the injury of six additional workers. The nominal wall thickness of the pipe when the station went into service was 10 mm. The thinnest wall in the failed region of the pipe was 0.4 mm. The same pipe section in the 'B' loop was 1.8 mm [5].

The failed pipe run should have been included in the plant's pipe wall thinning management program following the introduction of a program guideline adopted by Japanese PWR operators in 1990. However, the pipe run appeared to have been inadvertently left out of the pipe wall management program and subsequent opportunities to correct errors and omissions from the inspection program were missed [6]. Further investigations revealed that the affected pipe run had been inspected at all other Japanese operated plants except those operated by KEPCO where 14 similar pipe portions had never been inspected. It was concluded that the quality assurance and maintenance management had failed to function properly in KEPCO and that eventually resulted in the pipe failure.

3.1.2 Iatan Unit 1

Iatan Unit 1 is a 651 MW cool fired fossil power plant located in Iatan, Missouri, US. The unit was connected to the grid in 1980. On May 9, 2007, a 4-inch superheater attemperator spray line ruptured resulting in the deaths of two workers and the injury of one additional worker [7]. The failure occurred in a portion of straight pipe, immediately downstream of the control valve that emits a fine spray of water into the superheated steam. The pipe wall thickness had been reduced significantly and FAC was identified as the likely mechanism responsible for the wall thickness reduction.

3.1.3 South Ukraine Unit 2

South Ukraine Unit 2 is a 1000 MWe VVER V-338 PWR located in Yuzhnoukrainsk, Ukraine. The unit was connected to the grid in 1985. On May 19, 2005, a 500 mm diameter pipe from a high-pressure heater ruptured [8]. On August 26, 2005, a 219 mm diameter pipe carrying condensate also ruptured. No other details of the failures are readily available.

3.1.4 Callaway Unit 1

Callaway Unit 1 is a 1275 MWe 4-loop Westinghouse type PWR located near Fulton, Missouri, US. The unit was connected to the grid in 1984. On August 11, 1999, a 6-inch pipe in the drain line between the moisture separator reheater drain tank and the feedwater heater ruptured while the unit was at full power [9]. There were no fatalities or injuries involved in the incident.

The failure occurred in a section of schedule 40 (nominal thickness of 7.1 mm) straight pipe immediately downstream of a 45° elbow. The failure was attributed to a reduced wall thickness due to FAC, although a contribution from water drop impingement was also noted. The 45° elbow was included in the wall thickness inspection program and had been previously inspected. However, although it is common practice to also include a section of the downstream piping from a targeted fitting as part of the inspection, it was not mandatory and was not performed in this case. This omission was driven by a perception in the industry at the time that the elbow would be the most susceptible component to FAC, and not the downstream straight piping.

3.1.5 Point Beach Unit 1

Point Beach Unit 1 is a 640 MWe 2-loop Westinghouse type PWR located near Two Creeks, Wisconsin, US. The unit was connected to the grid in 1970. On May 14, 1999, the shell side of the feedwater heater ruptured. There were no fatalities or injuries involved in the incident.

The extraction steam entering the top of the shell of the feedwater heater was known by the operators to contain residual moisture in spite of the presence of a preseparator tank immediately upstream. Although a stainless steel diffuser plate was located directly under the extraction steam inlet, the deflected steam impinged upon the carbon steel shell in the area where the rupture occurred. The 0.5-inch nominal wall thickness of the shell had been reduced to as little as 0.05 inch in the area of the rupture. Following the accident, an inspection of the second feedwater heater in the same unit revealed similar wall loss. However, inspection of the feedwater heaters in Unit 2 revealed no degradation (it is not clear what, if any, differences existed between the materials and operating conditions between the two units). The station indicated that the feedwater heaters were not included in any periodic inspection program even though similar failures had occurred in feedwater heaters at the Dresden Power Station in 1983, the Susquehanna Steam Electric Station in January 1999 and the Pilgrim Station, also in April 1999.

3.1.6 Fort Calhoun Unit 1

Fort Calhoun Unit 1 is a 512 MWe 2-Loop Combustion Engineering type PWR located near Fort Calhoun, Nebraska. On April 21, 1997, a 12-inch diameter sweep elbow in the fourth stage extraction steam piping ruptured [10]. There were no fatalities or injuries resulting from the incident.

The fourth stage extraction steam system delivers wet steam (92% quality) from the outlet of the high-pressure turbine to the feedwater heaters. The carbon steel piping had a nominal wall thickness of 9.5 mm and was known to be susceptible to wall thinning following the replacement in 1985 of a similar elbow, which was located upstream of the failed elbow, after it had developed a pinhole leak. The extraction steam system had been included in the station’s pipe wall thinning management program and had been modeled with CHECWORKS. The CHECWORKS model predicted greater wall loss in the long radius elbows, which had been replaced, than in the sweep elbows. However, it was subsequently realized that the line correction factors that had to be applied were outside of the acceptable range, which invalidated the predicted wall loss rates. Secondly, the wall loss rates were largely based on measurements from an elbow that had been replaced in 1985 and yet the elapsed time was calculated using the 14 years since commissioning instead of the 2 years since the elbow replacement, resulting in a significant underestimation of the wall loss rate.

3.1.7 Millstone Unit 2

Millstone Unit 2 is a 870 MWe 2-loop Combustion Engineering type PWR located near Waterford, Connecticut. On November 6, 1991, while the unit was operating at 100% full power, an 8-inch elbow located between the first stage moisture separator reheater drain tank and the feedwater heater in the B train ruptured [11]. The release of steam triggered portions of the turbine building fire protection deluge systems.

The wall thickness along the extrados of the elbow, where the failure occurred, which had a nominal wall thickness of 0.322 inch, had been reduced by 95%. The wall thickness in the same elbow in the A train had been reduced by 34%. The elbows had not been selected for wall thickness inspections. Although the station had a program in place for monitoring high energy piping since 1981, the selection of inspection locations was largely based on engineering judgement.

3.1.8 Millstone Unit 3

Millstone Unit 3 is a 1150 MWe 4-loop Westinghouse type PWR located near Waterford, Connecticut. On December 31, 1990, while the unit was operating at 86% full power, two 6-inch, schedule 40 pipes in the moister separator drain system ruptured [12]. The release of steam triggered portions of the turbine building fire protection deluge system, leading to the local flooding and loss of electric power, shutting down the plant process computer and disabling the instrument air supply in the containment building.

A through-wall leak had been previously observed in the line, just downstream of the level control value in the A train, and preparation was under way to repair the line. In isolating the line, a pressure transient was generated which caused the lines in both the A and B trains to rupture. After the failure, the wall thickness of the lines was found to have been reduced to 0.5 mm. The lines had been identified as being potentially susceptible to "erosion corrosion" and were intended to be analyzed. However, due to a communication error, the analysis was never performed.

3.1.9 Surry Nuclear Power Station Unit 2

Surry Nuclear Power Station Unit 2 is a 890 MWe 3-loop Westinghouse type PWR located near Surry, Virginia. On December 9, 1986, an 18-inch suction line to the main feedwater pump A on Unit 2 ruptured [13]. There were four staff fatalities and additional four staff injuries resulting from the incident.

The pipe failure was initiated when the main steam isolation valve on the steam generator failed closed, which caused a pressure surge that resulted in an elbow in the 18-inch feedwater pump suction line to fail. The elbow was found to have failed because of a reduced wall thickness attributed to "erosion-corrosion", which is more accurately described today as FAC. There had been few previously reported failures in large-bore piping systems containing high-purity water and the operator did not have an in-service inspection program, which was consistent with the general practice at the time.

3.1.10 Trojan Nuclear Power Plant Unit 1

Trojan Nuclear Power Plant is a 1080 MWe 4-loop Westinghouse type PWR located near Portland, Oregon. On March 9, 1985, a 14-inch heater drain pump discharge pipe ruptured [2]. There was one staff injury resulting from the incident.

The pipe ruptured at a location where the flow, passing through a globe valve, was directed at the pipe wall, resulting in a reduction in the wall thickness from its nominal value of 0.375 inch to 0.098 inch. The section of pipe had been installed in 1977 and was intended to be used only during startups to assist in maintaining heater drain tank levels; it was not designed for full flow, normal, full-power operation. However, due to other operational issues, the section of replaced piping was made the dominant flow path from the heater drain tank and routinely experienced full flow during normal full-power operation.

4. RECOMMENDATIONS FOR AN EFFECTIVE FLOW-ACCELERATED CORROSION PROGRAM

Following the failure of two moisture separator drain lines at Millstone Unit 3 in 1990, EPRI conducted a series of visits to nuclear power plants to determine how well FAC programs were being implemented. Although utilities were committed to implementing FAC programs, the success of the programs and their implementation was found to have varied significantly from station to station. Based on the findings from those visits, EPRI complied and published a series of recommendations for improving the quality and implementation of FAC programs. The resulting report has subsequently been updated on several occasions to incorporate lessons reported by the industry, especially through forums such as the CHECWORKS User Group (CHUG), and the recommendations contained therein are widely incorporated into the FAC programs at most of the US nuclear plants. Revision 4 of this report [1] is used as reference in the present report.

EPRI identified six key elements deemed necessary for an effective FAC program;

- Corporate commitment to ensure programs are adequately funded and prioritized and staff are properly trained and engaged in industry-wide forums sharing relevant operating experience.
- Analysis tools for assessing potentially susceptible locations, and filtering out low-risk locations, to produce a manageable inspection scope.
- Site and industry experience published through CHUG or through other industry organizations such as Institute of Nuclear Power Operations (INPO).
- Inspections to provide accurate estimates of current wall thicknesses and rates of wall loss. Generally, it is better to inspect a smaller number of components

thoroughly, mapping wall thickness of entire component, rather than reporting just the minimum wall thickness from a larger number of components.

- Training and engineering judgement to ensure those responsible for the FAC programs, and their alternates, are knowledge about FAC and the tools used to manage FAC, such as CHECWORKS, and are well integrated with system engineers and other station experts to facilitate good engineering judgement in the performance of their duties.
- Long term strategy to reduce FAC rates through water chemistry, process or material changes and prioritizing most at risk locations, especially as the plant ages.

These elements are expected to apply equally to an FAC program set up for the nuclear piping in the CANDU HTS. In the case of site and industry experience, a more CANDU focus may be required since organizations such as CHUG are dominated by balance of plant experience. It is possible that organizations such as COG could play an expanded role in tracking CANDU HTS specific experience.

EPRI has also highlighted the importance of procedures and documentation specifically developed to support the implementation of the FAC program. In the case of multi-unit stations, these procedures and documentation should be as generic as practical. These procedures and documents would include;

- Governing document showing corporate commitment to monitor and mitigate FAC.
- Procedures, which are updated as required, addressing all tasks within the FAC program.
- Documents noting all susceptibility analysis, inspection results including locations inspected and their selection basis, inspection findings, locations deferred, evaluation and disposition results.
- Documents noting all component and line replacements.

Documentation is an important element in the effectiveness of the FAC program as noted by the piping failures noted in Section 3.1. In many of those cases, failures occurred as a direct result of a breakdown of documentation, such as failure to properly track inspection locations or component replacements.

4.1 EPRI Recommendations for FAC Tasks

Management of FAC degradation can be broken down into the following steps;

- Using a Predictive Methodology
- Identifying Susceptible Systems and Equipment
- Performing FAC predictive analysis
- Selecting and Scheduling Components for Inspection
- Performing Inspections

- Evaluating Inspection Data
- Evaluating Worn Components
- Repairing and Replacing Components
- Determination of the Safety Factor

Hence, the recommendations provided by EPRI [1] are organized according to these steps, summaries for which are provided in the following subsections. It is worth stressing that these steps are focussed on the management of FAC rather than where and how FAC degradation can be predicted. Hence, even though there may be some differences in how FAC manifests itself in the CANDU nuclear piping systems and the secondary side of CANDU and PWR plants, the steps associated with the management of FAC largely apply equally to both. Key differences, where they exist, are noted in the respective subsection.

4.1.1 Using a Predictive Methodology

The predictive methodology should take into account

- The geometry, temperature, velocity, water chemistry and material composition of each component
- Consider the range of hydrodynamic conditions, including the ability to estimate flow and thermodynamic conditions in lines
- Consider the water chemistry in the various systems, in particular the alkalinity of the water (pH) and the concentration of dissolved oxygen, and including local water chemistry conditions that may deviate from those of the bulk system.
- The impact of changing plant operating states on the FAC rates throughout the plant.
- Use refined FAC rate predictions that take into account hydrodynamic, water chemistry and material information and that are validated based on plant data.
- Predict future wall thickness based on estimates of FAC rate using models that account for previous wall thickness measurements (for example, CHECWORKS Pass 2 type analysis).
- Review accuracy of predictive methodologies by comparing predictions with measured values.

It is worth noting that concerns were expressed many years ago about the potential impact of hydrazine on FAC. Hydrazine is routinely added to the feedtrain in the balance of plant to reduce the concentration of dissolved oxygen in the water before it enters the steam generator. It was speculated that high concentrations of hydrazine, which can shift the electrochemical potential to negative, more reducing, values, could result in higher than expected FAC rates and a factor for hydrazine was included in earlier versions of the CHECWORKS code. However, an EPRI study released in 2005 concluded that hydrazine did not influence FAC rates and the associated factor was removed in subsequent versions of the code [14], [15]. Hydrazine has not been widely used in the HTS of CANDU plants, with only limited use to reduce dissolved oxygen concentrations on startup, so should not be a concern in either case.

Other influences of water chemistry on FAC have been assessed at times and modifications made to the CHECWORKS code accordingly. In particular, the influence of dissolved oxygen, which was discussed in Section 2.1, was also investigated by EPRI and recommendations were made for changes to the CHECWORKS code to improve the accuracy of wall loss predictions [14]. As noted earlier, dissolved oxygen is generally not expected in the nuclear piping of the CANDU HTS during normal operation.

The recommendations for using a predictive methodology also apply to CANDU nuclear piping. However, the pH in the balance of plant of PWRs and CANDU plants is achieved through the addition of volatile amines, as noted in Section 2.1, and the tools currently used to provide a predictive capability, such as CHECWORKS, have been developed for this particular type of water chemistry. There may be challenges associated with directly applying these tools for predicting FAC in the CANDU HTS nuclear piping in view of the differences in how pH is controlled.

4.1.2 Identifying susceptible systems and equipment

When identifying susceptible systems and equipment;

- Consider all carbon and low alloy steel piping systems containing flowing water or steam as being potentially susceptible to FAC
- Exclude systems or portions of systems deemed to have a low level of susceptibility through detailed evaluations based on
  - presence of higher alloy steels,
  - transport of superheated steam,
  - transport of water containing high levels of dissolved oxygen,
  - transport of single phase water at low temperatures,
  - low duty factors or no flow
- Any system or portions of systems excluded should be documented along with the basis for the exclusion.

Due to the significant impact that low concentrations of chromium can have on FAC rates, it has been recommended that stations determine the chromium content of their carbon and low alloy steels and include that information in their CHECWORKS model so that the software can estimate the lower rates using a model based on the available test data [2].

These recommendations all apply to CANDU nuclear piping, noting that

- most of the material in CANDU nuclear piping is carbon steel,
- none of the systems carry superheated steam, and
- none contain high levels of dissolved oxygen during normal operation.

The EPRI document [1] provides recommendations for excluding systems or portions of systems based on the following specific criteria;

- material fabricated from stainless steel or low alloy steel containing ≥1.25 wt% Cr
- piping carrying superheated steam
- piping carrying fluid containing high concentrations of dissolved oxygen (>1000 ppb)
- carrying single phase flow below 93°C.
- piping experiences no flow, operates <2% of the plant operating time, or operates with temperatures >93°C <2% of the plant operating time.

These conditions are also expected to apply to nuclear piping in the CANDU HTS.

However, the EPRI document [1] also provides recommendations for excluding systems or portions of systems based on the following criterion;

- piping fabricated from carbon steel containing ≥0.1 wt% Cr

It is worth noting that SA106 carbon steel feeder pipe supplied with the CANDU plants prior to Qinshan in the late 1990's contained ~0.02 to 0.10 wt% Cr and all these plants experience, or have experienced, FAC, albeit to varying degrees. In addition,

- testing performed under simulated CANDU HTS conditions indicated a ~50% reduction in FAC rate for material with 0.14 wt% Cr relative to material with 0.013 wt% Cr [17], and
- station data indicated a reduction in the FAC rate of 64±5% for feeder pipes containing 0.33 wt% relative to those containing 0.02 wt% [18].

Hence, although carbon steel piping under CANDU HTS conditions experiences a significant reduction in FAC rates due to the presence of small amounts of Cr, for example 0.1 wt%, the material continues to experience wall loss, albeit at lower rates. These rates of wall loss may be significant depending upon the available wall thickness margins available in the piping. It is therefore recommended to not categorically exclude carbon steel CANDU nuclear piping from FAC evaluation unless at least one of the other aforementioned exclusion requirements is met.

4.1.3 Performing FAC predictive analysis

Once the susceptible lines in the plant have been identified, the FAC wall loss should be predicted using an appropriate predictive methodology and knowledge of the operating conditions for each location.

- Lines should be calibrated when wall thickness data are available to correct predictions based solely on operating conditions (referred to Pass 2 analysis in CHECWORKS).
- Produce a new power level in the predictive model if changes are made to the plant heat balance diagram.

These recommendations apply to CANDU nuclear piping.

It is worth noting that in the case of outlet feeders and other carbon steel components, the concept of Effective Full Power Years (EFPY) has been extensively used to capture the impact of varying reactor power on FAC rates. Essentially this means that the wall loss expected from, say 90% full power operation for 1 calendar year is equivalent to the wall loss expected from 100% full power operation for 90% of a calendar year. Although the prediction of wall loss may become inaccurate at some power levels, it has proven to be a reasonable representation of operating time when evaluating FAC rates in HTS components (feeders, header nozzles) based upon the power variations typically experienced in CANDU plants.

4.1.4 Selecting and Scheduling Components for Inspection

Once FAC wall loss rates are predicted, inspection locations of the susceptible piping need to be identified based upon the considerations such as;

- Select lines based on model predictions, trending of previous measurements, operating experience, engineering judgement, and potential for an entrance effect[e].
- Locations should include both new as well as previously inspected locations. In addition, a proportionally larger number of non-calibrated versus calibrated lines should be included. Previously inspected locations should include locations with the highest trending wall loss rates and lowest remaining operating life. Locations downstream of FAC-resistant components should be included due to possible entrance effects.
- Susceptible-not-modeled lines (not modelled due to unknown or widely varying operating conditions) should be included.
- Locations selected based on plant experience/conditions (components operating downstream of equipment operating outside of design, such as valves, locations that are similar to those of known FAC susceptibility in parallel lines, etc.).
- Locations based on past operating experience.
- Inspections of susceptible equipment.
- Increase inspection scope when previous inspections reveal unexpected or inconsistent findings.

These recommendations all apply to CANDU nuclear piping.

4.1.5 Performing inspections

A number of recommendations are made with respect to the performance of inspections.

- UT is better for piping components, RT is better for irregular surfaces

[e] Entrance effect describes a phenomenon that is associated with a higher than expected rate of wall loss on a pipe susceptible to FAC when it is located downstream of a pipe that is not susceptible to FAC, such as stainless steel. The higher than expected rates of wall loss are localized immediately downstream of the weld transition.

- UT wall thickness grid measurements of a component are better than a single minimum reported from a scan of the component.
- Use recommended methods for establishing grids or using templates and for identifying locations within the grid
- Perform a baseline inspection of replacement components prior to service, including inspection of the locations on both sides of the replacement welds to account for counter-boring needed to facilitate fit-up, etc.
- Extend measurements beyond the exit of a component such as a bend or tee as localized wall loss can extend beyond the boundaries of the component.
- The selection of grid lines for a particular component should be compliant with that required by any software being used to model the wall thickness data.
- RT can be used to inspect wall thickness of large bore piping
- Visual inside inspection can be used to augment wall thickness inspections
- Valves cannot be inspected by UT; techniques such as visual and radiography can be used instead.
- Measuring alloy content of components and crediting it in predictive models can improve the accuracy of predictions
- Option available to exclude measured components with a Cr content greater than 0.10 wt% from future inspections if there is no significant wall loss.
- Review data and verify questionable values, consider potential impact of manufacturing variability in assessing wall loss.

These recommendations all apply to CANDU nuclear piping with one exception. With respect to excluding measured components with a Cr content greater than 0.10 wt%, see comment at the end of Section 4.1.2 with respect to Cr contents and FAC. If the component is inspected, and no significant wall loss is observed, then excluding component from future inspections may be appropriate.

4.1.6 Evaluate Inspection Data

The following recommendations apply to the evaluation of inspection data.

- Apply standardized evaluation methods and engineering judgement to mitigate the impact of weak or missing information related to initial wall thickness, presence of counterbores at pipe welds, missing or erroneous data, etc.
- Data should be properly reviewed, including in the context of data from upstream and downstream components.
- Estimate the total wall loss and wall loss rate using a method that is appropriate for the type of measurement data available and whether baseline data are available.

These recommendations all apply to CANDU nuclear piping.

4.1.7 Evaluating Worn Components

Once wall loss and wall loss rates have been estimated, the components must be evaluated against their required wall thickness.

- Use a conservative approach such as assuming the maximum wall loss rate for all locations on the component and applying a reasonable safety factor to account for uncertainties in the wall loss rate calculations. The safety factor is expected to decrease as more measurement data become available.
- Determine the minimum wall thickness requirement for each component using appropriate ASME Section III rules.
- Predict future wall thickness using an appropriately conservative wall loss rate taking into account any changes in operating conditions that could arise in the future relative to past operation. Wall thickness predictions based on two sets of in-service measurements collected from two different outages are preferred.
- Estimate the remaining service life using the estimated wall loss rate, current wall thickness, minimum acceptable wall thickness and appropriate safety factor.

These recommendations all apply to CANDU nuclear piping. It should be noted that CANDU nuclear piping is Class 1 whereas most of the piping in the balance of plant is Class 6. The requirements for evaluating wall thickness in Class 1 are more demanding than those for Class 6. In cases were local wall loss occurs in Class 2 or Class 3 piping, methods have been developed and are provided under code cases, such as ASME N-597, for evaluating local thinning. Wall losses in Class 1 nuclear piping could be dispositioned using ASME Section III rules. In the case of CANDU Class 1 feeder pipes, the CANDU industry developed fitness for service guidelines to disposition local thinning that could not otherwise be successfully dispositioned using ASME Section III rules. However, the methodologies used for Class 2 and Class 3 piping or for Class 1 feeder pipes, are currently not available for other ≥6 NPS Class 1 nuclear piping and could be a limitation in evaluating wall thicknesses found during in-service inspections.

4.1.8 Repairing and Replacing Components

In the event that a component is deemed to have inadequate wall thickness, repairing or replacing components is required.

- Option available for repairing may include interior or exterior weld build up, with the former generally being preferred.
- When replacing components, the selection of a more FAC resistance material is preferred to reduce future wall loss.
- Consideration should also be given following a repair or replacement to reduce the susceptibility to wall loss by modifying the operating conditions and/or improving the water chemistry where possible.

These recommendations all apply to CANDU nuclear piping with the following caveat. It should be noted that CANDU nuclear piping is Class 1 whereas most of the piping in the balance of plant is Class 6. Hence, the methods employed for repairing or replacing a component must be in compliance with Class 1 requirements.

4.1.9 Determination of the Safety Factor

The determination of a safety factor is required to address uncertainties associated with estimating past and future wall loss.

- It is recommended to use only one safety factor when determining fitness for continued service and the re-inspection interval. Developing multiple safety factors or applying a safety factor at different points in the process can impact the relative susceptibility ranking, diverting attention away from the most at risk locations. NSAC-202L-R4 provides guidance on developing a safety factor and should be considered.

This recommendation applies to CANDU nuclear piping.

4.1.10 Development of a Long-Term Strategy

Section 5 of NSAC-202L-R4 provides a discussion on the importance of developing a long-term strategy to manage wall loss in system components. The discussion on the selection of more FAC resistance materials for replacement components is applicable to CANDU nuclear piping. This has been the practice with the replacement of feeder pipes, including both single replacements and multiple replacements (refurbishment).

Section 5 of NSAC-202L-R4 also discusses potential changes to water chemistry to mitigate wall loss rates. Many of the options identified, such as the use of alternate amines for pH control, which provide more favourable partitioning through the steam cycle piping, or considerations of oxygen concentrations in BWR plants, do not apply to the CANDU HTS. Some pHa[f] optimization was carried out in the HTS of CANDU plants following the discovery of FAC in outlet feeders [19]. All CANDU plants revised their HTS pHa specifications so as to limit operation towards the low end of the original pHa operating range to take advantage of these lower FAC rates. Hence further reductions in FAC rates through further water chemistry optimization are unlikely.

4.2 Recommendations from Other Organizations

The OECD, and in particular its Nuclear Energy Agency (NEA), established a program entitled “Component Operational Experience, Degradation & Ageing Programme” (CODAP) to facilitate the collection and analysis of data related to the degradation and failure of piping systems in commercial nuclear power plants. A document was produced by CODAP looking specifically at FAC [4]. Although the CODAP report does not provide formal recommendations for an effective FAC program, it identifies four factors that contributed to

[f] Note that pHa is the apparent pH and is the pH reported for heavy water measured with a pH meter calibrated with light-water buffers.

program deficiencies leading to the occurrence of FAC related failures. An effective FAC program should therefore address these deficiencies. In particular;

- Through-Wall Leak Develops Prior to Next Scheduled Inspection
    - Failures have occurred prior to the next scheduled inspection either because the wall thinning rate had been underestimated or that the operating conditions changed, resulting in increased wall thinning rates. It is therefore important to use conservative rates when predicting future wall thicknesses, and to adequately account for the impact of changes in operating conditions on those rates.
- Failure Occurs in a Location Identified by Predictive Program as Susceptible
    - Failures have occurred at locations where predictive tools identified a susceptibility to FAC, but the susceptible location was not properly incorporated into the inspection program. This speaks to the need for adequate review and verification of inspection locations.
- Failure Occurs Due to FAC Software Model Input Errors
    - Failures have occurred at locations covered by predictive modelling because the inputs used in the predictive model were incorrect. This speaks to the need for adequate review and verification of input parameters.
- Less-Than-Adequate FAC Program Implementation
    - An example was cited (See Section 3.1.4 for details) where a failure had occurred in a section of straight pipe immediately downstream of an elbow identified as being susceptible because the inspection scope didn't include the downstream portion of straight pipe and was limited to just the elbow fitting. Hence, piping and components downstream of a susceptible location, such as an elbow/bend, should also be inspected.

These deficiencies could also occur in the management of CANDU nuclear piping. Hence they too need to be addressed.

Reports produced by the International Atomic Energy Agency (IAEA) have also highlighted the importance of programs such as Ageing Management Program to manage FAC at nuclear power plants using a process of

- periodic inspection, to monitor wall thicknesses, and
- corrective actions, such as pipe replacement, to address wall thickness deficiencies once they are identified [20].

The IAEA has highlighted the importance of predictive tools, such as COMSY, which is a software tool developed by AREVA similar to the CHECWORKS code developed by EPRI, for predicting areas of degradation and guiding inspection, as well as the replacement of susceptible components with more FAC resistant materials [21].

5. COMPARISON OF CANDU NUCLEAR PIPING AND BALANCE OF PLANT PIPING

It is important to note that there are some key differences between CANDU nuclear piping and balance of piping which may be significant in terms of how they are impacted by FAC. These differences are summarized in Table 2.

With respect to safety class, CANDU nuclear piping is Class 1 whereas balance of plant piping is Class 6. This difference is significant in terms of the rules applied to dispositioning wall thickness measurements (see discussion in Section 4.1.7) as well as when repairing and replacing components (see discussion in Section 4.1.8).

With respect to water chemistry pH control (see discussion in Section 2.1), lithium hydroxide, which is a strong base, is used in CANDU nuclear piping whereas volatile amines, which are weak bases, are used in the balance of plant piping. The use of a strong base may be a limitation when using software, such as CHECWORKS, which were originally developed for balance of plant piping, for predicting FAC rates in CANDU nuclear piping.

The flow in CANDU nuclear piping is either single phase flow or two phase flow with up to ~4% steam quality. In the balance of plant, the flows include the full range of conditions from single phase water flow to dry steam. Hence, the FAC expected in CANDU nuclear piping is expected to be similar to the FAC observed in CANDU outlet feeders, which is characterized by a scalloped surface. Under two-phase conditions, as experienced in the balance of plant piping, a “tiger striping” appearance is often observed [22]. This tiger stripping is not expected in CANDU nuclear piping.

The CANDU nuclear piping is fabricated from carbon steel while a much wider range of materials from carbon steel to low alloy steel to stainless steel is found in the balance of plant. Hence the phenomenon of an entrance effect, as discussed in Section 4.1.4) is less likely to occur in CANDU nuclear piping.

6. COMPARISON OF FAC MANAGEMENT PRACTICES WORLDWIDE

Many of the recommended EPRI practices discussed in Section 4.1 have been captured in EPRI’s software package CHECWORKS. Although CHECWORKS is the dominant software package used in US nuclear plants, there are other packages used at nuclear plants elsewhere in the world. Some of the other more widely used packages are discussed in this section.

- BRT-CICERO

  The software package BRT-CICERO was developed by EDF in the 1990’s and is now mandatory for all 58 nuclear power plants in France.

- COSMY-FAC

  Condition Oriented ageing and plant life Monitoring System (COSMY) is a software code developed by Areva and was based, in part, on two earlier codes developed by Siemens;

    - WATHEC, which provided prediction of FAC in view of material and operational factors and was based on extensive laboratory test data, and
    - DASY, which provided capabilities for recording, managing, evaluating and documenting wall thickness inspection data.

- RAMEK

  RAMEK is a software code developed in Russia in 2003 to assist in the management of pipe wall thinning in Russian nuclear power plants. The code provides predictions for local wall thickness based on the local impact of geometry and flow conditions, as well as alloy composition.

Interest has been expressed by nuclear industry stakeholders as to the accuracy of these various software codes for predicting wall loss due to FAC. The IAEA initiated a Coordinated Research Project (CRP) and several meetings have been held since 2013 involving the 17 participating countries with the objective of benchmarking CHECWORKS and the three codes noted in this section. According to the IAEA website for this initiative, the results generated from the running of the four codes on references cases are currently being assessed. Upon completion, the results will be made available in an IAEA Nuclear Energy Series publication.

7. STRENGTHS AND POTENTIAL AREAS FOR IMPROVEMENT

Many of the recommendations identified by EPRI [1] and other organizations, such as the NEA and the IAEA [4], [20], [21], that highlight key elements of an FAC management program for the piping in the balance of plant of PWRs and CANDUs and the piping in BWRs, also apply to CANDU nuclear piping. The effectiveness of the overall program relies upon all of the elements addressed by these recommendations and a weakness in any one area can result in plant failures, as highlighted by the OPEX events noted in Section 3. With that in mind, it is difficult to single out individual strengths. However, it is useful to specifically note the following three program areas as providing significant benefits to the overall program success.

- Once susceptible systems or sub-systems are identified, predictive methodologies are valuable for predicting degradation in the various locations. This information can be used to focus the inspections on the most at risk locations. In cases where details of the operating conditions are known, FAC rates can be estimated using software such as CHECWORKS (referred to as a Pass 1 analysis in CHECWORKS). These rate estimates can be improved if wall thickness inspection data are available (referred to as Pass 2 analysis in CHECWORKS). Review and verification of the model inputs are important to reduce the likelihood of erroneous predictions.
- The accuracies of FAC rate predictions can be improved significantly when wall thickness inspection data are added to the predictive model. In CHECWORKS software, this is referred to as a Pass 2 analysis. For this reason, it is important to update the predictive models as inspection data become available. This information

can be used to assess the accuracies of the previous predictions and identify potential areas for improvement in the models, for example, where operating conditions may not have been adequately defined.

- Accurately maintained system or subsystem documentation provides inputs required for the predictive models and to understand FAC susceptibility. This includes information on system design, component fabrication (including any non-conformances), component configuration, component alloy content, service and repair histories, etc. Establishing and maintaining accurate records requires a commitment to quality assurances and effective work place review and verification procedures.

As noted earlier, success relies on all elements of the FAC program and failure to implement any of the recommendations identified by EPRI and others [1], [4], [20], [21], become areas for improvement. The mitigating measures could include;

- increased corporate commitment for the FAC program,
- increased resourcing for the FAC program
- improved communications between FAC program owner and engineering staff, including clarifying roles and responsibilities
- increased training to FAC program staff to ensure proper use of predictive methodologies, inspection technologies, etc.
- improved awareness and incorporation of operating experience from other units, in the case of multi-unit stations, and other utilities, especially those sharing similar system designs
- improved understanding of system design and configuration, including abnormal valve configuration and operation, abnormal process conditions, changes in reactor power, changes in operating chemistry conditions, etc.

In addition to these potential general areas for improvement, the following points, summarized in Table 3, are noted specifically in the context of managing FAC in CANDU nuclear piping;

- As noted in Section 5, predictive methodologies, such as CHECWORKS, have largely been developed for BWR and PWR/CANDU secondary side water chemistries. The water chemistry of the CANDU HTS differs significantly from these water chemistries and may not fully be captured by tools such as CHECWORKS. Hence, additional considerations may be required to account for HTS water chemistry in the predictive models for CANDU nuclear piping systems.
- As part of the process to identify susceptible systems and equipment, the EPRI recommendations include a provision to exclude carbon steel components containing ≥0.1 wt% Cr. However, based on operating experience from the CANDU HTS, and in particular from outlet feeder pipes, the FAC rate of carbon steel containing 0.1 wt% can be significant, depending upon the wall thickness margins

available in the component, especially as the operating life of these components is extended. Hence, it would be prudent not to exclude components based solely on this criterion.

- As noted in Section 5, wall losses from in-service inspections could be dispositioned using the conservative methodologies provided through ASME Section III rules. However, as noted in Section 4.1.7, more elaborate methodologies are available in code cases, such as ASME N-597, for dispositioning local thinning in Class 2 and Class 3 or in the COG Fitness for Service Guidelines for dispositioning local thinning in Class 1 feeder piping. The development of similar methodologies for ≥6 NPS nuclear piping could improve the ability to disposition local wall thinning found during in-service inspections.

It is worth noting that although the present study exclusively considered the impact of FAC on CANDU nuclear piping, erosion mechanisms such as cavitation or liquid drop impingement, may also contribute to a loss of pressure boundary. Historically, erosion mechanisms were not explicitly addressed within prediction methodologies such as CHECWORKS. However, with release of CHECWORKS Revision 4.0 in 2013, erosion mechanisms were formally incorporated for the first time, although the predicted wall losses may not be as refined as for wall loss by FAC (for example, only a go/no-go result is provided for flashing erosion). The inclusion of erosion mechanisms in these tools should make for a more complete “wall thinning” program.

Finally, the present study is focused on recommendations related to managing FAC degradation in nuclear piping. However, other forms of degradation, with a potential synergistic relationship, may be present. For example, environmental fatigue often occurs at the same locations as FAC and local wall thinning may increase the susceptibility to fatigue due to stress concentration. Similarly stress concentration may also enhance the susceptibility of nuclear piping to the cracking that has been observed in some CANDU outlet feeders. The concerns associated with these degradation mechanisms increase as the age of the piping is increased through life extension. Such considerations are not explicitly addressed by the EPRI recommendations.

8. SUMMARY

In response to the life extension of CANDU plants and the introduction of new methodologies for managing large break loss of coolant accidents, there is increased interest in the management of FAC related degradation in the CANDU nuclear piping systems. To this end, a review has been performed of the recommendations made by EPRI, as well as other nuclear industry organizations, on the implementation of programs to manage FAC in BWR and the balance of plant systems of PWRs and CANDU plants, with a focus on the applicability of these recommendations for the management of FAC in CANDU nuclear piping.

In general, the majority of recommendations considered in the review are programmatic in nature and would also apply to the management of FAC in CANDU nuclear piping. The recommendations cover all aspects of an FAC program including;

- Identifying susceptible systems and equipment using selection criteria and ranking locations for inspection using prediction methodologies such as CHECWORKS
- Performing inspections and evaluating the data collected using prediction methodologies such as CHECWORKS to estimate remaining life
- Repairing or replacing components when they are deemed to have reached their end of life
- To implement all aspects of the program under an appropriately conservative framework that avoids pressure boundary failures while still realizing the useful operating life of the components.

In the context of CANDU nuclear piping, three areas are identified for improvement to ensure the success of the program.

- Additional considerations given to the uniqueness of the water chemistry in the CANDU HTS, which may not be readily accounted for in the prediction methodologies used in codes such as CHECWORKS, which were developed for BWR and the secondary side chemistries in PWRs and CANDU plants.
- Not excluding carbon steel components from evaluation based solely on the criterion of containing ≥0.1 wt% Cr. Such material is known to continue to experience FAC under CANDU HTS conditions, albeit at reduced rates compared to material with less Cr. The wall losses experienced over an extended operating life could still be significant pending the wall thickness margins available in the component.
- Specialized methodologies, which have been developed to evaluate wall thickness in Class 2 and Class 3 piping or Class 1 CANDU feeder piping, are not available for Class 1 CANDU nuclear piping. The development of such specialized methodologies specifically for CANDU nuclear piping could improve the ability to disposition local wall thinning found during in-service inspections.

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Table 1 Significant FAC Events

Plant	NRC/INPO Reference	Type	Date	Single-Phase?	System	Comments
Oconee	IN-82-22	PWR	6/82	No	Extraction	Large hole in elbow in HP extraction line.
Navajo	——	Fossil	11/82	Yes	Feedwater	Similar conditions to Surry
Surry	IN 86-106, Bull. 87-01 IN 88-17 INPO SOER 87-03	PWR	12/86	Yes	Condensate	Four fatalities
Trojan	IN 87-36 IN 88-17 INPO OE2109	PWR	6/87	Yes	Feedwater	Major damage found; however, no failure. IN 87-36 contains incorrect information (repeated in IN 88-17) about thinning in straight sections.
Arkansas Nuclear One	IN 89-53	PWR	4/89	No	Extraction	Accident occurred in the CE unit.
Santa Maria de Garona, Spain	INPO OE3690	BWR	12/89	Yes	Feedwater	Fist size blowout. Low oxygen chemistry.
Loviisa, Finland	IN 91-18	PWR	5/90	Yes	Feedwater	
Millstone 3	IN 91-18	PWR	12/90	Yes	Separator Drain	
Millstone 2	IN 91-18 Supp. 1 INPO OE4923	PWR	11/91	Yes	Reheater Drain	
Sequoyah	INPO OE 5847	PWR	3/93	No	Extraction	
Sequoyah	IN 95-11 SER 6-95	PWR	11/94	Yes	Condensate	“Unknown” flow element
Pleasant Prairie Power Plant	——	Fossil	2/95	Yes	Feedwater	Two fatalities.
Millstone 2	INPO OE 7420 SER 21-95	PWR	8/95	Yes	Heater Drain	
Fort Calhoun	IN 97-84 INPO SEN 164	PWR	4/97	No	Extraction	
Point Beach 1	IN 99-19	PWR	5/99	No	Feedwater Heater Shell	

Plant	NRC/INPO Reference	Type	Date	Single-Phase?	System	Comments
Callaway	NRC Event Notification 36015 INPO SEN 203 INPO OE10171	PWR	8/99	No	Reheater Drain	
H. A. Wagner 3 Power Plant	——	Fossil	7/02		Feedwater Heater Drain	
Mihama 3, Japan	INPO OE19368 INPO OE18895	PWR	8/04	Yes	Feedwater	Five fatalities.
Edwards Power Plant	——	Fossil	3/05	Yes	Feedwater	
South Ukraine 2	WANO MER MOW 05-019	VVER	7/05		Feedwater Heater Drain	
South Ukraine 2	WANO MER MOW 05-021	VVER	8/05		Reheater Drain	
Iatan		Fossil	5/07	Yes	Attemperator spray	Two fatalities.

Table 2 Comparison of FAC Considerations for CANDU Nuclear Piping and Balance of Plant Piping

Factor	CANDU Nuclear Piping	Balance of Plant Piping
Safety Class	Class 1 (no specifically developed local wall thinning evaluation methodologies)	Class 6 (less stringent requirements and guidance from methodologies developed in code cases such as ASME N-597)
Water Chemistry pH/pHa control	LiOD (strong base) (not modelled by prediction software such as CHECWORKS)	Volatile Amines (weak bases) (modelled by prediction software such as CHECWORKS).
Flow	Range from single-phase water flow to two-phase flow with up to ~4% steam quality.	Full range of flow conditions from single phase water flow to dry steam.
Materials	Carbon Steel	Carbon steel, low alloy steels, stainless steels

Table 3 Potential Areas for Improvement to the EPRI Recommendations

Deficiency/Area for Improvement	Comments
Modelling of FAC in CANDU nuclear piping based on CANDU HTS specific water chemistry	Current software tools, such as CHECWORKS, were developed to model FAC under BWR and balance of plant conditions in PWRs and CANDU plants. Due to the differences in how pH is controlled in the CANDU HTS, the use of such tools for predicting FAC rates in CANDU nuclear piping may require additional considerations.
Excluding carbon steel components from inspections based on a content of ≥0.1 wt% Cr.	Although carbon steel containing 0.1 wt% Cr experiences lower FAC rates than material without Cr, the reduced FAC rates could still be significant depending upon the wall thickness margins available. It would be prudent to therefore not exclude piping based only the Cr content of the pipe.
Evaluating wall loss in Class 1 nuclear piping	Balance of plant piping, for which the EPRI recommendations were largely developed, is Class 6 and pipe wall thickness requirements are less demanding and methodologies are available for guidance from code cases such as ASME N-597. Similar methodologies have not been developed for ≥6 NPS Class 1 piping and may limit utilities ability to disposition findings of local thinning (methodologies have been developed for CANDU feeders).