APPLICATION OF CANDU STEAM GENERATOR RESEARCH EXPERIENCE IN THE FIELD OF SMR STEAM GENERATORS

APLICAREA EXPERIENȚEI DE CERCETARE AFERENTĂ GENERATORULUI DE ABUR CANDU ÎN DOMENIUL GENERATOARELOR DE ABUR PENTRU SMR

Dumitra LUCAN[1],[2]

Abstract: *The paper synthetically presents a CANDU Nuclear Power Plant and a NuScale reactor and the steam generators used for the two types of reactors. Also, the paper includes a description of the processes and structural materials for the CANDU steam generator and the identification of known processes or those for which intense research activity must be developed to fill the knowledge gaps for the NuScale steam generators. Finding the answers for these issues supposes inter and transdisciplinarity in engineering sciences and technologies. Only by working in research teams including chemists, physicists, engineers specializing in materials science, metallurgy, energy, electronics can be identified and put into practice the optimal solutions for the proper functioning of innovative reactors.*

Keywords: CANDU, NuScale, steam generator, Alloy 800, Alloy 690

Rezumat: *Lucrarea prezintă sintetic o centrală nucleară CANDU și un reactor NuScale precum și generatoarele de abur utilizate pentru cele două tipuri de reactoare. De asemenea, lucrarea include o descriere a proceselor și materialelor structurale pentru generatorul de abur CANDU și identificarea proceselor cunoscute sau a celor pentru care trebuie dezvoltată o activitate intensă de cercetare în vederea completării cunoștințelor aferente generatoarele de abur NuScale. Găsirea răspunsurilor la aceste probleme presupune inter și transdisciplinaritate în științele inginerești și tehnologie. Doar lucrând în echipe de cercetare formate din chimiști, fizicieni, ingineri specializați în știința materialelor, metalurgie, energetică, electronică pot fi identificate și puse în practică soluțiile optime pentru buna funcționare a reactoarelor de Generație nouă.*

Cuvinte cheie: CANDU, NuScale, generator de abur, Incoloy 800, Inconel 690

[1] Senior Researcher, Nuclear Materials and Corrosion Dept., Institute for Nuclear Research, Pitesti, Romania, e-mail: dumitra.lucan@nuclear.ro; dida.lucan@gmail.com.

[2] Corresponding member, Technical Sciences Academy of Romania ASTR, Bucharest, Romania.

1. Introduction

CANDU Nuclear Power Plant. The basic operation of the CANDU design is similar to other nuclear reactors. Fission reactions in the reactor core heat pressurized water in a primary cooling loop. A heat exchanger, also known as a steam generator, transfers the heat to a secondary cooling loop, which powers a steam turbine with an electric generator attached to it. The exhaust steam from the turbines is then cooled, condensed and returned as feedwater to the steam generator, Figure 1 [1]. Most commercial reactor designs use normal water as the moderator. Water absorbs some of the neutrons, enough that it is not possible to keep the reaction going in natural uranium. CANDU replaces this "light" water with heavy water. Heavy water's extra neutron decreases its ability to absorb excess neutrons, resulting in a better neutron economy. This allows CANDU to run on unenriched natural uranium. This was a major goal of the CANDU design; by operating on natural uranium the cost of enrichment is removed [1].

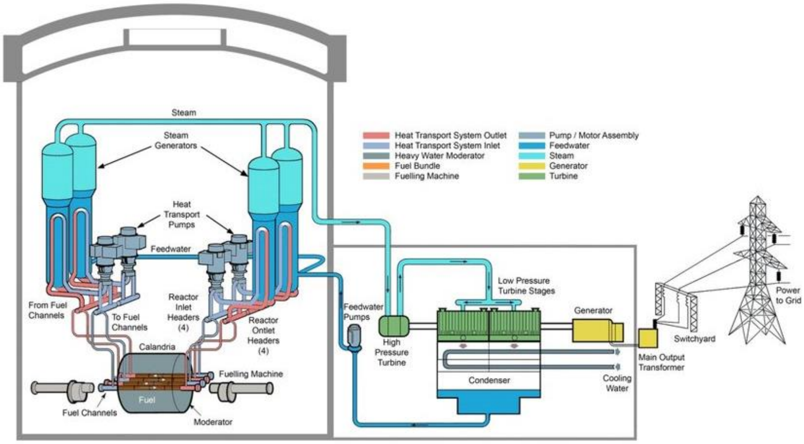

Figure 1. Schematic presentation of a CANDU Nuclear Power Plant

NuScale. The land-based water-cooled SMRs contains water-cooled SMR designs from various configurations of light water reactor (LWR) and heavy water reactor (HWR) technologies for on-land on-the-grid applications. These designs represent the mature technology considering most of the large power plants in operation today are of water-cooled reactors. There are twenty-five water-cooled SMR designs from 12 States that comprises integral-PWRs, compact-PWRs, loop-PWRs, BWRs, CANDU-type designs, and pool-type reactors for district heating.

Main design features for NuScale. According to the classification currently used by the IAEA, small reactors are the reactors with an equivalent electric power less than 300 MW, medium sized reactors are the reactors with an equivalent electric power between 300 and 700 MW [2].

The NuScale Power Module™ (NPM) is a small, light-water-cooled pressurized-water reactor (PWR). The NuScale plant is scalable and can be built to accommodate a varying number of NPMs to meet customer's energy demands. The 60 MW(e) NPM provides power in increments that can be scaled to 720 MW(e) gross in a single facility with twelve modules. A twelve-module configuration is the reference plant size for design and licensing activities. Each NPM is a self-contained module that operates independently of the other modules in a multi-module configuration. NuScale design is a modular reactor for electricity production and non-electrical process heat applications.

The basic configuration of a single NuScale reactor module is shown schematically in Figure 2 [3].

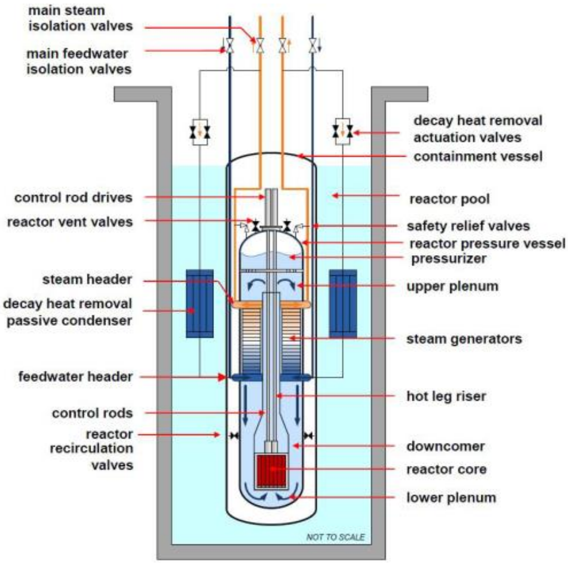

Figure 2. NuScale Module

The integrated nuclear reactor pressure vessel contains the nuclear core, a helical coil steam generator, and a pressurizer. It is approximately 20.0 m long by 2.8 m in diameter. The nuclear core consists of an array of reduced height LWR fuel assemblies and control rod clusters at standard enrichments.

The helical coil steam generator consists of two independent sets of tube bundles with separate feedwater inlet and steam outlet lines.

Feedwater is pumped into the tubes where it boils to generate superheated steam. A set of pressurizer heaters is located in the upper head of the vessel to provide pressure control. The entire Nuclear Steam Supply System (NSSS) is enclosed in a steel containment that is 24.6 m long by 4.6 m in diameter. There are five essential features of the NuScale plant, which in combination, distinguish it from the many other small nuclear plants being developed today. The first is its *compact size*.

The NSSS can be entirely prefabricated offsite and shipped by rail, truck or barge. At this size, it can be fabricated by mid-size domestic large plants. Second, the nuclear *core is cooled entirely by natural circulation*. Water is heated in the nuclear core to produce a low density fluid that travels upward through the hot leg riser. The helical coils wrapped around the outside of the riser provide a heat sink that cools the water, causing its density to increase. The density difference acting over an elevation difference results in a buoyancy force that drives the fluid flow around the loop. Natural circulation operation provides a significant advantage in that it eliminates pumps, pipes, and valves and hence the maintenance and potential failures associated with those components. It also reduces in-house plant loads. This added simplicity enhances overall plant safety. Third, *the design relies on well-established light water reactor technology*. The NuScale plant can be licensed within the existing LWR regulatory framework, drawing on a vast body of established R&D, proven codes and methods, and existing regulatory standards.

Fourth, the *NuScale design is supported by a one-third scale*, electrically heated integral test facility which operates at full pressure and temperature. Operation of an integral test facility for such a non-conventional design is highly useful in seeking regulatory approval from governing bodies such as the U.S. Nuclear Regulatory Commission (NRC). Customers will also want assurance from integral facility tests that the plant will perform as predicted. This prototype test facility will provide the necessary supporting data required for licensing.

Finally, the *NuScale design is the only SMR that utilizes a compact movable modular containment*, in contrast to a traditional cast in-place concrete design. Similar to the reactor vessel, the use of a compact containment that is prefabricated off site and shipped by rail, truck or barge, enhances the economics and deployment flexibility.

This unique set of features, specifically the synergy created by plant simplicity, reliance on existing light water technology, and the availability of an integral test facility, all combine to position the NuScale plant for early deployment.

The NuScale plant design adopts design simplification, proven light-water reactor technology, modular nuclear steam supply system, factory-fabricated power modules, and passive safety systems that allow for unlimited coping time after a design basis accident without power, operator action, or makeup water. The NPM is designed to operate efficiently at full-power conditions using natural circulation as mean of providing core coolant flow, eliminating the need for reactor coolant pumps [3-5]. A comparison of selected operating conditions for a PHWR, and the water cooled SMR NuScale is presented in Table 1 [6].

Table 1. **Selected data for typical PHWR and comparison with NuScale SMR concept**

	PHWR (CANDU 6)	**NuScale**
Country of origin	Canada	USA
Gross thermal power	2064 MWt	160 MWt
Gross electric power	735 MWe	48 MWe
Primary Side	**PHWR**	**NuScale**
Flow direction	horizontal	vert., upward
Reactor coolant	heavy water	light water
Circulation mode	forced	natural
Operating pressure	11.2 MPa	12.8 MPa
Core outlet temperature	310^0C	310^0C
Core inlet temperature	266^0C	258^0C
Total core flowrate	7700 kg/s	587 kg/s
Secondary Side	**PHWR**	**NuScale**
Number of SG	4	2
Feedwater flow direction	upward	upward
SG tubing arrangement	inverted U-tube	helical coil
Number of tubes per SG	3550	690
Tube OD	15.9 mm	15.9 mm
Tube material	Alloy 800	Alloy 690
Steam outlet pressure	4.6 MPa	3.5 MPa
Steam outlet temperature	260^0C	300^0C
Steam inlet temperature	187^0C	149^0C
Feedwater flow per SG	262 kg/s	33.6 kg/s

2. Steam generators for CANDU NPP and NuScale inovative reactor

CANDU Steam generator. This equipment is of the vertical recirculating type built by Babcock & Wilcox Canada Ltd. The CANDU 6 model contains 4 SGs per unit, Incoloy 800 tubing and SS-410 grid tube support and anti-vibrations bars (AVB), Figure 3.

The primary coolant enters SG at 310 – 315^{0}C and leaves at 270 - 280^{0}C on the cold leg side [5]. Composition for Incoloy 800 is the next: C<0.02, Ni=30-35, Cr=19-23, Mn<1.5, Si<1, Cu<0.75, Ti=0.15-0.6, Fe=Bal.

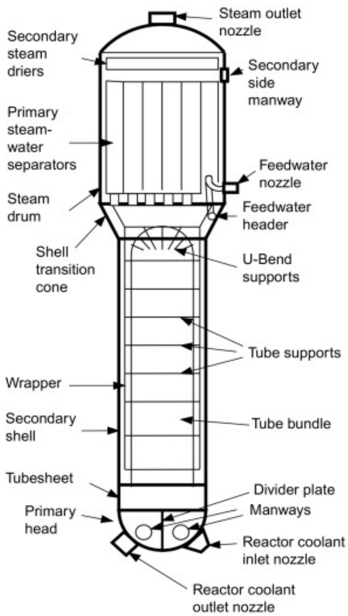

Figure 3. CANDU Steam Generator

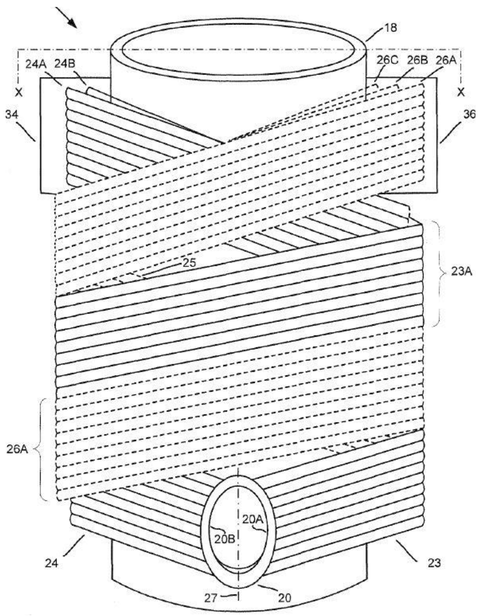

Figure 4. Helical Coil Heat Exchanger

NuScale Steam Generator. NuScale employs a once-through, vertical helical coil steam generator design. With this design, feedwater enters into a lower integrated tubesheet plenum (ITP), flows upward through the tube bundle where it is converted to steam that exits through the upper ITP. One significant difference from a standard once-through design is that the secondary fluid (feedwater/steam) flows through the inside diameter (ID) of the tubes and primary water flows on the outside diameter (OD) [7]. Steam generators that use a helical tube array have the advantage of a compact design with relatively high heat transfer efficiency.

The Romanian school was preoccupied with the sizing and operation of different types of heat exchangers, including the helical ones from the beginning of the researches in the field of chemical engineering, an example in this sense being the work of professor Emilian Bratu, [8].

Figure 4 is an illustration of flowpaths for the helical coil design provided for the purpose of general illustration only. In this illustration, the center element (18) is the riser. The riser is described as a structure that functions as a hot leg, directing core outlet flow upward through the center region of the vessel where the core flow is directed radially outward, then downward through the steam generator section. The lower integrated tubesheet/plenum (ITP) directs feedwater flow through into the tube bundles (at 20A & 20B). The tube bundles represented by items 23 & 24 carry fluid to the upper ITP. Current plans include two independent tube bundles. Feedwater 5-27 and steam flows may be independently controlled to the independent bundles to allow for continued operation in the event of a failure of a tube in one of the steam generator bundles.

NuScale indicates that tubes will be hydraulically expanded into the tubesheet, although the specific details of this design do not appear to be finalized. The tubesheet connection is likely to be significantly thinner than prior designs and would need either to be clad with an austenitic material such as Alloy 52 or constructed from an Alloy 690 plate or forging. In addition, evaluation remains in progress to determine if connecting welds can be seal welds only or if the welds must have a structural function. Regardless of the specific configuration, the tube to tubesheet design configuration will be different in some respects from traditional tube to tubesheet connections. If a low-alloy tubesheet is used, then the tubesheet must be clad on the tube side, since primary water containing boric acid flows on the OD of the tubes. If the tubesheets are fabricated from Alloy 690 plate or forgings, this will be a relatively new application of Alloy 690 in a LWR. An additional configuration detail of note is the need to install flow orifices at in the tube inlets for the purpose of precluding density wave flow oscillations within the tubes. These orifices must be removable to facilitate inspection from the inlet end of the tubes.

Notably, the risk of PWSCC within the steam generator should be somewhat reduced in comparison with large advanced PWRs. Operating temperatures associated with the NuScale design are lower; hot leg temperature for the NuScale design is approximately 310^0C, compared with hot leg temperatures at or above 315.5^0C for all Gen. II and advanced PWR designs. Additionally, the large radius of curvature of the helical coil reduces the potential for high fabrication-induced stresses, as can occur at recirculating steam generator U-bends. However, it has not yet been determined if the helical coil bends will be heat treated after forming. Consequently, the amount of cold strain in the thermally treated Alloy 690 tubes will need to be considered relative to existing data on PWSCC growth rates in cold worked material.

Given the relatively large radius of the coils, this is not expected to be a concern.

The major concern of alloy 690 SCC in secondary water due to concentrated impurities (lead in particular) should also be reduced since line-contact crevices are eliminated on the tube ID. There have been some issues in advanced gas-cooled reactor steam generators where secondary water is boiled into steam and superheated on the tube ID with a tendency for fouling and concentration of impurities in the boiling crisis zone. It is important to note that the NuScale steam generator is expected to operate quite differently. More detailed analysis is recommended for the SCC potential in the alloy 690 tubing as the steam generator design evolves further.

Composition for alloy 690 is the next: Ni = 58.0, Cr = 27-31, Fe = 7-11, C = 0.05, Si < 0.5, Mn < 0.5, S < 0.015, Cu < 0.5, [9].

Materials of construction for the tube supports have not yet been finalized, but will likely be Type 405 stainless steel. Selection of materials for this application introduces a significant new application of ferritic stainless steel as a passive, long-lived structure within the primary system. Type 405 stainless steel has been used extensively in secondary-side tube support applications, but not in the primary system. Although the application does not raise any significant degradation concerns, some investigation to ensure no vulnerabilities exist seems reasonable.

To summarize the steam generator evaluation, no specific materials performance concerns were identified. However, the NuScale steam generator design differs significantly from the recirculating or straight tube once-through designs used for PHWRs. As a result, research focused on mitigating SCC risk in large advanced PHWRs will not be relevant to the NuScale design. There may also be unique considerations associated with the tube-to-tubesheet connections. Additional materials R&D with a focus on the NuScale design specifics would be needed to demonstrate long-term performance [10; 11].

3. PHWRs SGs Degradation Mechanisms and the possibility of appearing at SGs of SMRs

Over the past 40 years or so, the degradation process of highest concern has been corrosion, especially under-deposits, fouling, and mechanical degradation, including vibration-fretting wear. In the early years of the US nuclear power generation industry, relatively low capacity factors were largely a consequence of material degradation issues. The economic success of SMRs will depend upon high capacity factors being achieved right from the start.

The degradation mechanisms in light-water SMR SGs are expected to be similar to those for pressurized water reactors where those of the latter are illustrated in Figure 5 (specifically for the recirculating type nuclear SGs). Many of the numerous degradation mechanisms were not anticipated in the original design. Hence, it could be expected that other degradation modes will be identified during the design, construction, commissioning and operation phases of water-cooled SMRs. The following will briefly outline these degradation processes, how they have been addressed, and will summarize how the SMR SG designs under consideration will be impacted by them [6].

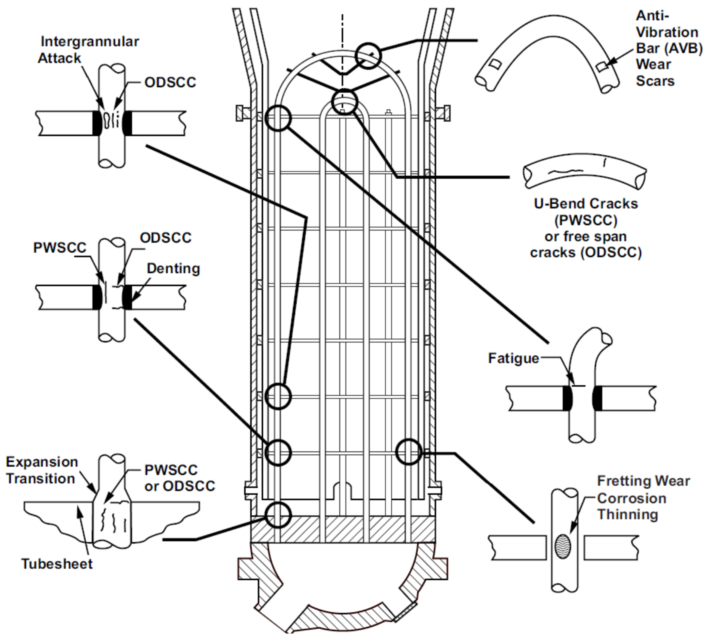

Figure 5. Types and locations of tubing degradation that have occurred in recirculating steam generators (*ODSCC - Outside Diameter Stress Corrosion Cracking, PWSCC - Primary Water Stress Corrosion Cracking)*

More advanced SG designs with tubing fabricated from either Alloy 800 (nuclear grade) or Alloy 690 (thermally treated) have experienced relatively little degradation. A brief summary of the higher-risk degradation mechanisms that have the potential to impact SMR SGs is provided in the next sub-sections.

3.1. Stress-Corrosion Cracking

This type of corrosion was more frequently identified on the U-shaped upper region of Incoloy-800 tubes, but cracks have been noticed in other areas, too. The cracks that appeared in the U-bend region has been generally initiated from the inside of the tube. The examination of such tubes shown that these cracks initiated on the side of the primary agent are of intergranular nature, oriented along the longitudinal axis of the tube. Given the available information on SMR SGs, and in particular that for the NuScale design which specifies Alloy 690 tubing, SCC is not expected to be a concern for an anticipated design life of 40 years. However, there is little or no information available on SCC susceptibility for a SG with a helical design and with unknown levels of residual stress or cold work, or with secondary side chemistry on the inside of the tube where fouling and deposition could be a significant concern.

3.2 .Flow Accelerated Corrosion

Flow accelerated corrosion (FAC) was a problem in early SG designs that used carbon steel support plates and flow tubes. This problem was resolved by switching to stainless steel (type 410S) for the supports. For SMRs with natural circulation, and hence low flow rates and the use of FAC-resistant materials, this will likely be of little concern.

3.3. Crevice Corrosion

The crevices formed at the intersections of the tubes and various supports are potential sites for the accumulation of activation products, even in the absence of flow-boiling on the outside surface of the steam generator tubes. Corrosion products or impurities that have been deposited in the core (or simply passed through) may become activated and disperse throughout the primary flow circuit. Deposition of activation products at tube and support intersections could lead to the formation of radiation “hot spots”. The effect of the tube support structure on fluid flow and the deposition of corrosion products and impurities should be assessed.

3.4. Fouling

On the primary side of the various water-cooled SMR designs, the tube and support materials include stainless steel and zirconium alloys, which have relatively low corrosion rates and low corrosion product release rates. For NuScale are established regarding the control of water chemistry to minimize solid deposits on the reactor core and steam generator tubes, but specific details were lacking on how this control would be achieved.

Based upon previous experience with fouling in the primary circuits of PWRs and PHWRs, a number of points are to be noted regarding water purification in the primary circuit of an SMR.

The flow rate of the reactor coolant purification system needs to be high enough to ensure that activation products are removed from the reactor coolant at a higher rate than that of deposition on out-of-core surfaces. In other words, the purification half-life should be shorter than that for out-of-core deposition in order to minimize the growth rate of radiation fields away from the core.

Fouling deposits will tend to increase the hydraulic roughness of surfaces, and this will tend to reduce the flow rate in a natural circulation system. The effect of increasing hydraulic roughness on thermalhydraulic conditions in the reactor coolant system needs to be assessed. In particular, an assessment should be made on how increasing hydraulic roughness affects the heat fluxes and corresponding deposition rates of corrosion products on heat transfer surfaces in the reactor core and steam generator. This might be an issue for SMR designs that rely on natural circulation for the primary coolant. If build-up of magnetite on the inside diameter of the tube walls occurs, this will likely be a concern in terms of inspectability of the tubes with existing probes. The primary source of magnetite is usually carbon steel, which is typically used in the feedwater system of PHWRs. Using low alloy steels or stainless steels in the feedwater system would nearly eliminate magnetite creation, but material cost would increase significantly.

From the research and operation experience held now and based upon an impurity concentration of iron oxide of 1 part per billion and a feedwater flow rate of 67.2 kg/s, a deposit accumulation of 2.1 kg per year could be expected in the NuScale plant SG. Using the same method of estimation and the same impurity concentration, a PHWR (CANDU 6) would see 33 kg of accumulated deposits per year. From chemical cleaning data for two CANDU 6 plants reported in the specialty literature, it can be observed that the iron oxide transport rate calculated from cleaning at one plant was 70 kg/year (based on chemical cleaning of the tube bundle, not including deposits removed by sludge lancing the tubesheet) while the calculated transport rate for another plant was 47 kg/year.

Assuming that the deposits consist primarily of iron oxide, accumulating with a density of approximately half of the nominal density of 5140 kg/m^3, then the volume of deposit accumulation would be 1650 cm^3. The estimated total ID surface area of the NuScale SG tubes is 1400 m^2 per module, and since the fuel cycle is 2 years long, this volume of deposits would accumulate to a thickness of 1.2 μm in that time span (assuming uniform distribution over the entire ID surface of the tubes). This level of accumulation will likely not prevent ID inspection of the tubes using typical inspection probes, unless the accumulation was significantly non-uniform on the ID. With non-uniform fouling deposits, the peak value will be higher, possibly by more than a factor of two. In regions of the tube bundle where boiling occurs and the steam quality is between 50% and 70%, the deposit thickness could be significantly greater than the average. Also, this estimation does not account for the contribution from crud bursts that tend to happen during reactor startup or as a consequence of operating transients. Hence, the average deposit loading based only on feedwater iron oxide during operation at power likely represents a lower bound.

If the materials and chemistry control strategies for the NuScale secondary system, including the feedwater piping and condensate system, appear to be consistent with those of the existing PHWR reactor designs, with the exception that the secondary water in the SMR steam generator itself is not in contact with a carbon steel pressure boundary.

The PHWR steam generators are a once-through design with the boiling coolant on the inside of the tubes. This design separates the boiling coolant from the tube-support structure, thus reducing the potential for deposit formation and hideout of impurities compared to the steam generators in current pressurized water-cooled reactors which have boiling secondary coolant on the shell side. However, there are two aspects of the steam generator design discussed below for which deposit formation could be an issue.

Firstly, for PHWR SG, flow restrictors are installed just before the inlet tubesheet (i.e., secondary side flow), and these are intended to mitigate the effect of pressure drop oscillations in the tube-side flow and thereby ensure secondary-side flow stability. These could be sites of significant deposition of corrosion products. Those structures create an abrupt reduction in flow cross-sectional area at the entrance and cause boundary-layer separation that lead to precipitation and particulate fouling at the entrance to the broached hole. A similar phenomenon could take place at the inlet region of the flow restrictors of the SMR steam generators.

Secondly, a notable feature of the OTSG designs in PHWRs is that essentially all of the corrosion products carried into the steam generator with the secondary coolant are deposited on the OD surface of the steam generator tubes in a single pass. Examination of the deposit distribution on tubes removed from OTSGs shows that the deposits are heaviest between steam qualities of approximately 50% and 70% [6]. The data from pulled SG tubes are consistent with experimental results from AECL's fouling program which found that the rate of particle deposition under flow-boiling conditions increases significantly with increasing steam quality for steam qualities greater than about 30%. Thus, there is the potential for the formation of heavy tube deposit on at least some portion of the tube bundle in the once-through steam generators of the SMR. These heavy deposits can become sites for the concentration of impurities and the development of an aggressive chemistry, thus increasing the risk of localized corrosion.

Because of the risks of enhanced deposit formation discussed above, the water chemistry guidelines for PWRs with once-through steam generators are likely a better guide for the SMR steam generators than those used for recirculating nuclear steam generators. In addition, consideration should be given to including a feedwater bypass loop in the design of the feedwater and condensate system. Such a loop is used during startup following a maintenance outage to filter suspended corrosion products that have been released to the feedwater by a 'crud burst' before the water is fed forward to the steam generator. If left unfiltered, a single crud burst can result in the transport of the same quantity of corrosion product to the steam generator as is transported during a full year under normal operating conditions.

The fouling concerns just discussed can lead to deposit formation, and the formation of crevices between tubing and support structures, or within the tubing itself. Crevice corrosion is a problem in PHWR SGs because of tube deposits and sludge piles on the secondary side, and the main concern for localized corrosion under deposits has been SCC. However, wastage, pitting and intergranular attack have all been observed on SG tubing under deposits when chemistry control has been off-specification. For SMR SGs, Alloy 690TT is susceptible to pitting under deposits in the presence of chloride or sulphide contamination and with an oxidizing chemistry. The proposed helical tube SGs with the secondary side water on the tube ID may be susceptible to this problem unless deposits can be minimized and/or routinely removed.

Fouling and crevice formation on the tube OD (i.e., the primary side) should not be an issue since the tubing is positioned such that deep sludge piles are likely avoided and the primary side chemistry is not expected to be corrosive.

3.5. Foreign Material Incursion

Foreign material damage has been an infrequent but expensive degradation mechanism experienced by PHWRs because this type of degradation is unanticipated, often requires an unplanned shutdown and can be difficult to diagnose and correct. For a helical tube SG, the primary flow rate may be too low to suspend any foreign objects (or loose parts) and hence this type of damage is not expected to occur. However, there is a concern that loose parts could be capable of significant damage or blockage if this occurs on the tube secondary side (i.e., tube-side).

3.6. Vibration and Fretting Wear

Mechanical degradations that may alter the PHWR SG tubing can be divided into: vibrations wear (fretting) and fatigue wear. These degradations belong to the category of localized attack. The strength that determines them is produced by tubes vibration, induced by flow circulation. This time, corrosion appears as an additional factor that accelerates mechanical degradation of the tubes; it acts synergistically. The effect of the synergetic action of the two factors varies from the erosion of passive films on the materials surface to the accelerating effects of certain aggressive environments on the quality of the metal. In the case of cracks initiated on defects (for example in regions where local thinning of tubing walls took place) a transgranular attack was identified on the tubes outer surface. The mechanism of these cracks includes the fatigue fretting corrosion in the presence of corrosive species in the environment.

In a helical tube SG, fretting wear of the tubes and supports will be highly dependent upon tube vibration frequencies, support conditions and the overall vibration damping. Damping in liquid flow is typically lower than in two-phase flow, except at very high void fraction.

A first order estimation of the potential for FIV damage was made for the outer-most tube in a helical tube bundle, using dimensions commensurate with the NuScale SG design, and also for an outer U-bend tube of a CANDU 6 SG. This was done on the basis of comparing estimates on the reduced gap flow velocity that each tube would see in cross flow. Reduced velocity is a dimensionless scaling parameter that provides a means of comparing the potential for flow excitation effects to occur for two similar structures, where the nature of the two flows may be different (i.e., two-phase steam-water versus single-phase liquid water).

In the case of a helical tube bundle subjected to liquid cross-flow, vortex shedding induced by the primary cross-flow of liquid water may become a concern since it is difficult to predict accurately. Tube-side boiling flow may induce excitation forces on the curved tubes in a helical coil steam generator, either due to momentum fluctuations of the vapour-liquid mixture, or due to the forces arising from the acceleration component of pressure drop that occurs when the higher density liquid evaporates into lower density vapour.

3.7 Denting

Denting of the tubes at the tube sheets was a problem in early SGs of PHWRs because of corrosion of carbon steel tube sheets and expansion of the corrosion products in the crevices between the tube OD and the lands of the support plate drilled flow holes. This issue was resolved by changing support plate materials to stainless steels and changing support plate designs from drilled holes to broached-hole designs or lattice bars. For the NuScale and other water-cooled SMRs steam generators, the design of the tube supports and the material selection should minimize this mechanism as a concern. The key feature to notice is the evolution of the various degradation mechanisms. Denting, for instance, was a significant issue for early SG designs, caused by the build-up of corrosion on the carbon steel support plates that constricted the tubes that passed through holes with relatively tight clearance. This issue was resolved in newer model SGs by changing the support plate design to use either broached holes or lattice bar arrangements, and by use of stainless steels for the support plates. As the importance of various degradation mechanisms changed with time, this led to changes to design, material, and operational parameters. Currently, stress corrosion cracking (SCC) on the secondary side of tubes (i.e., shell side) is of significant concern, because it is difficult to detect until it has progressed through approximately 40% of the wall thickness. Upon detection, a tube with this depth of degradation is usually removed from service. Crack progression to the point of through-wall penetration can result in tube rupture and leakage, and such an occurrence has a significant impact on SG and plant operation, [11].

4. R&D on the corrosion behaviour of the CANDU steam generator materials

To approach the R&D activity in the field of SMR, one can use the knowledge and expertise held by researchers regarding large reactors CANDU PHWR. Research is needed to understand, characterize, and manage service induced degradation of steam generator components. Technology development needs to be focused on timely detection, mitigation, and prevention of significant long-term effects of aging such as stress corrosion cracking and other types of degradation.

A research and development program to address the degradation of CANDU steam generator structures involving laboratory tests was developed in the Nuclear Materials and Corrosion Department from Institute for Nuclear Research Pitesti.

The main research areas and the principal experimental results of this program are given in the following:

Assessment of materials behaviour and current candidate for the construction of the steam generator in normal and abnormal operation conditions:

- Identification of corrosive degradation processes of the structural materials depending on the material conditions, environment conditions and the test parameters;
- Establishment of a correlation between the operating conditions and degradation mechanisms of the main structural materials from the primary and secondary circuits of the SG and mathematical modelling of involved processes;
- Development of mathematical models intended to forecast the intensity of the corrosive degradation during certain operating conditions;
- Extension of the long-term safe operation of the steam generator (Long Term Operation - LTO).

Water chemistry and specific processes for the transport, deposit and removal of the corrosion products and impurities in steam generator:

- Decrease of the structural materials corrosion by improving water chemistry or by applying chemical treatments to the surface;
- Implementation of the experimental results obtained in laboratory, of the operating parameters recorded during the equipment operation, as well as of the information and parameters from specialty literature or operation of similar facilities in other countries, into the database to assess the corrosion

behavior of structural materials and the formation and transport of corrosion products inside the steam generator circuits.

Operation, non-destructive testing, maintenance, repairing and modernization of the steam generator:

- Experimental program dedicated to new technologies for plugging the steam generator tubes in order to restore structural integrity.

Providing technical support for Cernavoda NPP in the field of steam generator:

- Provision of expertise in order to identify the types of degradation occurring due to corrosion of structural components during operation and proposals for decreasing them;
- Characterization of deposits on the steam generator tubesheet and establishment of their influence on the corrosion behaviour of the tubing material and of the heat transfer efficiency.

The experimental results obtained [12 - 23] and the expertise gained during the development of this research activity can be used to assess the types of degradations for NuScale structural materials.

The objectives of the program concerning the NuScale steam generator can be the follows: identifies the relevant ageing degradation mechanisms and discusses the issues associated with continued safe and economic operation of this key equipment, summarizes the results of current research and development activities and identifies the research needs in the future.

5. Conclusions

Based on PHWR experience with SG degradation and the materials, chemistry, design and operation/maintenance improvements that have been achieved, corrosion degradation of the SMR SG tubing is not expected to significantly affect SMR SG life up to a typical PWR design life of 40 years.

Nevertheless there is some uncertainty with this conclusion given the "inside-out" nature of the SMR SG design where secondary side water, typically the source of corrosion issues in PHWRs, is on the ID side of the steam generator tubes. There is limited data on whether this design difference will influence the degradation mechanisms.

It is anticipated that fouling on the tube ID may be a significant concern. This fouling can impact flow, provide sites for corrosion, reduce heat transfer and interfere with inspection and will require analysis and an R&D effort to resolve its significance to SMR SG operation. This concern highlights the need for excellent on-line chemistry control and monitoring to ensure low risk of fouling or corrosion degradation. There also exists the potential for flow-excitation of the tubes in the helical type steam generator, where the outermost tubes with the longest spans between supports are likely the most at risk.

The experimental results obtained for the CANDU steam generator consist in the correlation between the nature of materials used for the construction of various components of the steam generator, the chemical characteristics of the circulating environment and the way in which certain of their abnormalities can lead to the occurrence of different types of corrosion.

The paper states that the knowledge obtained based on experience gained in the field of structural materials degradation by research activity from large pressurized reactors can be applied to small modular reactors.

R E F E R E N C E S

[1]. *I. Pioro, R. B. Duffey, P. L. Kirillov, “*Current Status and Future Developments in Nuclear Power Industry of the World”, ASME J. of Nuclear Rad. Sci., Vol. **5**, Iss.2, 2019.

[2]. **** “Status of Small Reactor Designs without Onsite Refuelling”, IAEA TECDOC-1536, IAEA, Vienna, 2007.

[3]. *D. Sandusky, W. Lunceford, “*Assessment of Materials Issues for Light-Water Small Modular Reactors”, Pacific Northwest National Laboratory, PNNL-22290, 2013.

[4]. *J. N. Reyes, “*NuScale Plant Safety in Response to Extreme Events”, Nucl Tech, **128**, 2012, pp. 153-163.

[5]. *J. Riznic, “*Steam Generators for Nuclear Power Plants”, Science Direct, 2017.

[6]. *P. Feenstra1, R. Tapping, C. Turner, D. Horn, M. J. Pettigrew and M. Yetisir*, “Postulated Degradation Mechanisms of Steam Generator Tubing in Water-Cooled Small Modular Reactors”, 40th Annual Conference of the Canadian Nuclear Society and 45th Annual CNS/CNA Student Conference Virtual Conference, June 6 –9, 2021.

[7]. *D. K. Williams, D. P. Fasset*, “Helical Coil Steam Generator”, United States Patent US no. 9188328B2, Nov. 17, 2015.

[8]. *Em. Bratu, “*Calculation of Spiral Heat Exchangers”, Politechnic Institute Bucharest, Communications Session, 1958.

[9]. *americanspecialmetals.com/inconelalloy690.html*

[10]. *D. Jianling, X. Yu, F. Jiyang, “*Study on steam generator helical tube integrity assessment of HTR”, 20th International Conference on Structural Mechanics in Reactor Technology, Espoo, Finland, August 9-14, 2009.

[11]. *N. V. Hoffer, P. Sabharwall, N. A. Anderson, “*Modeling a Helical-coil Steam Generator in RELAP5- 3D for the Next Generation Nuclear Plant”, Idaho National Laboratory, INL/EXT-10-1962, 2011.

[12]. *D. Lucan, “*Primary Water Chemistry and Corrosion Management in a CANDU NPP”, Chapter 5 in Advances in Chemistry Research, Volume 69, James C. Taylor (Editor), Nova Science Publishing, New York, Series: Advances in Chemistry Research, 2021, pp.189-217, ISBN: 978-1-68507-014-4.

[13]. *D. Lucan, S. C. Valeca, , G. Jinescu, "*Plant Life Extension Programs - Useful Resources for Preservation and Amplification of Nuclear Power Plants Efficiency", Journal of Engineering Sciences and Innovation, Volume **5**, Issue 4/2020, pp. 329 - 338, ISSN 2606-6699.

[14]. *D. Lucan, S. C. Valeca, G. Jinescu, M. Mihalache, "*Romania's Contribution to the Development of the Knowledge Society in the Field of Nuclear Energy Research", Journal of Engineering Sciences and Innovation, Volume **4**, Issue 4/ 2019, pp. 399-410, ISSN 2606-6699.

[15]. *D. Lucan, G. Jinescu, A. Dinu, M. Stanciulescu, M. Mihalache, "*CANDU Steam Generator Secondary Side Fouling Characteristics, Journal of Engineering Sciences and Innovation", Volume **4**, Issue 1/2019, pp. 31 - 42, ISSN 2606-6699.

[16]. *D. Lucan, "*Factori care contribuie la degradarea materialelor structurale ale generatorului de abur (Factors that contribute to the degradation of the structural materials of the steam generator), Editura SITECH, Craiova, 2018, ISBN 978-606-11-6615-2.

[17]. *D. Lucan, G. Jinescu, "*Corrosion Analysis and Diagnosis - Useful Tools for Plant Life Management and Long Term Operation Programmes in Nuclear Power Plant", Journal of Engineering Sciences and Innovation, Volume **2**, Issue 3/2017, pp. 80-92, ISSN 2606-6699.

[18]. *D. Lucan, "*CANDU Steam Generator Corrosion Due to the Impurities of Concentration and Deposition", Capitol in Steam Generators: Design, Types and Applications (Book), Editor: Clayson Schellenberg, Nova Science Publishing, New York, Series: Energy Science, Engineering and Technology, 2017, pp.197-226, ISBN: 978-1-53612-510-8.

[19]. *D. Lucan, L. Velciu, G .Jinescu, "*The Synergistic Impurities Effect on the CANDU Steam Generator Corrosion", Revista de Chimie, **vol.67**, no.4, pp. 603-608, 2016, ISSN 0034-7752.

[20]. *D. Lucan, "*Corrosion of the Secondary Side Steam Generator in the Presence of Impurities", Chapter 9 in Nuclear Power Plants (Book), Editors: George Petridis and Dimitrios Nicolau, Nova Science Publishing, New York, Series: Nuclear Materials and Disaster Research Physics, 2012, pp. 197-226, ISBN: 978-1-61470-150-7.

[21]. *D. Lucan, "*Diminuarea coroziunii prin optimizarea chimiei apei în echipamentele de transfer termic" (Reducing corrosion by optimizing water chemistry in heat transfer equipment), Editura Universității din Pitești, 2012, ISBN 978-606-560-303-5.

[22]. *D. Lucan, "*Behavior of the Steam Generator Tubing in Water with Different pH Values", Nuclear Engineering and Design, 241 **(4)**, pp.1172-1176, 2011, ISSN: 0029-5493.

[23]. *D. Lucan, "*Procese specifice generatorului de abur" (Processes specific to the steam generator), Editura PARALELA 45, Pitești, 2009, ISBN 978-973-47-0813-0, 2011, pp.3-20, ISBN 978-953-307-303-3.

Acknowledgements

The experimental results from the published papers cited were obtained within the "Steam Generator" R&D Program initiated and developed by the author since the year 2000.

Author' biography

Dumitra LUCAN holds a Ph.D. in Chemical Engineering and is a Senior Scientific Researcher Grade 1 at Institute for Nuclear Research Pitesti, Romania. She is a corresponding member of the Technical Sciences Academy of Romania.

Her research topics include: corrosion behavior of the CANDU Steam Generator structural materials depending on material parameters, corrosive environment and testing conditions, data processing and interpretation. Corrosion analysis and diagnosis performed can be used as tools for Plant Life Management and Long Term Operation programmes in NPP including SMRs. She also studied the chemistry of liquid lead and its interaction with structural materials for Generation IV Lead Fast Reactor.

Email: dumitra.lucan@nuclear.ro