ARC-100 Sodium Fast Reactor Pressure Boundary Classification and Material Selection

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Abstract

Pressure boundary classification and material selection in CSA N285 was written for water cooled reactor technology operating at temperatures below where material creep would need to be considered. Pressure boundary classification is based primarily on pressure boundary system function and radionuclide content in the pressure retaining system. CSA N285 classification can be applied to non-water based technology on the basis of pressure boundary function and radionuclide content. For sodium fast reactor technologies, sodium activation (Na-24) can be compared to tritium limits found in CSA N285 by conversion of radionuclide concentration. Once converted, the pressure boundary system can be classified using rules found in CSA N285. After classification, material selection and design rules appropriate to the classification can be applied which account for high temperature service and unique high temperature effects.

1. Introduction

The ARC-100, being developed by ARC Clean Technology, is a 100 MWe sodium fast reactor (SFR). Sodium cooled fast neutron (un-moderated) reactors have been in operation around the world since the early stages of nuclear power development. However, no commercial SFR power plant has operated in Canada to date. An ARC-100, a 100 MWe commercial demonstration SFR, is planned to be constructed and in operation by 2030 at the existing Point Lepreau Nuclear Generating site in Southern New Brunswick.

The ARC-100 is a pool type sodium fast reactor. The “pool” refers to the main reactor vessel, as shown in Figure 1 (labelled “Reactor”). The reactor vessel contains liquid sodium which is the Primary Heat Transport fluid, as this sodium removes heat from the reactor core. The Primary Heat Transport System (PHTS) is completely contained within the reactor vessel, shown in yellow in Figure 1. PHTS sodium is circulated through the core by Electromagnetic (EM) pumps located in the reactor vessel, submersed in sodium.

Heat is removed from the PHTS by heat exchangers that reside in the reactor vessel submerged in the PHTS sodium. The heat exchangers are the transition from the PHTS to the Intermediate Heat Transport System (IHTS). Similar to the main Steam Generators in a CANDU plant, the IHTS heat exchangers transfer heat directly from the PHTS, but not for immediate steam generation. Different than a CANDU plant, there is a secondary system between PHTS and steam generation, which is the IHTS. This system, like the PHTS, uses liquid sodium as a heat transfer medium. Sodium has a high thermal conductivity, low melting temperature, and a relatively high boiling point making it ideal for use as a PHTS fluid and IHTS fluid. The IHTS operates at a higher pressure than the PHTS (~60 psig and ~0 psig respectively). If there were ever a break in the IHTS, sodium would leak into the PHTS avoiding a loss of coolant in the reactor vessel.

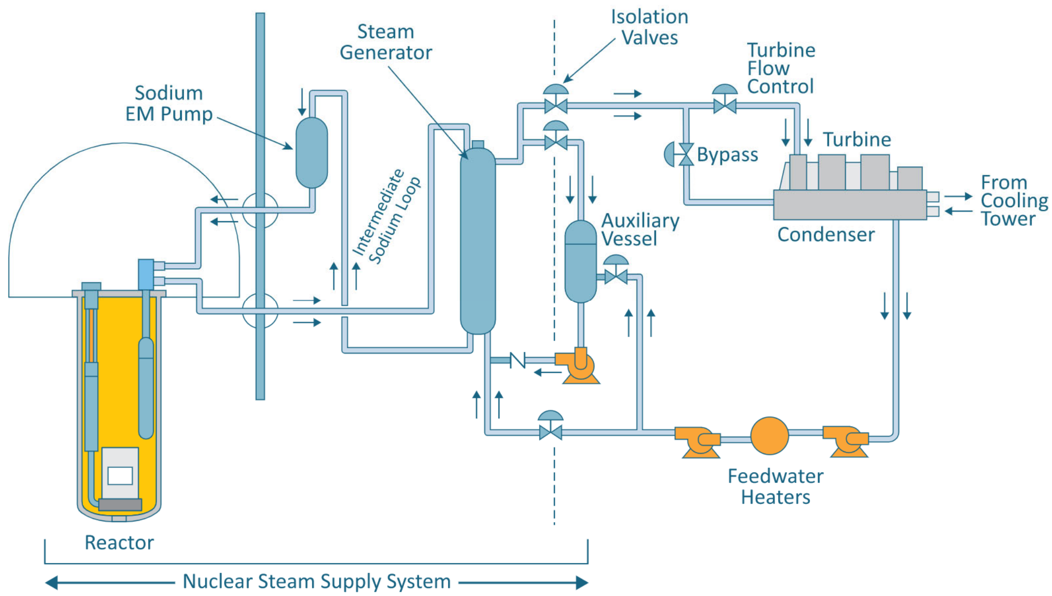

Figure 1   ARC-100 pool type SFR major systems.

EM pumps circulate IHTS sodium from the Reactor Vessel to the Steam Generator (SG), as shown in Figure 1. The IHTS transfers heat to Feedwater contained in helical tubes inside the SG. Steam generated from the SG powers a Turbine Generator typically found in a steam power plant.

Currently, Advanced Small Modular Reactor (aSMR) technology development is ahead of Canadian nuclear standards. There are no standards issued specific to metal cooled reactors. However, application of Canadian nuclear standards as written, though originally intended for water-cooled reactor technologies, is possible.

Allowances are made by the Canadian Nuclear Safety Commission (CNSC) for reactors other than water-cooled reactors. Technologies other than water cooled reactors, like the ARC-100, must meet the safety objectives, high-level safety concepts, and safety management requirements found in Reg Doc 2.5.2 [1].

The application of current water-cooled reactor based CSA nuclear standards to the ARC-100 SFR technology has been applied to meet the requirements of Reg Doc 2.5.2. Using the functional based requirements of the CSA nuclear standards, and by understanding the original intent of the nuclear standards, has made their direct application to a nuclear technology that is different than water-cooled technology possible.

2. Applicability of CSA N285 to ARC-100 systems

Pressure boundary component and system requirements in Canada come from the CSA N285 standard. There are multiple directives to follow the requirements of CSA N285 in New Brunswick: CNSC regulatory documents for reactor facility design (Reg Doc 2.5.2) [1], CNSC reactor operating licence conditions [2], and Provincial regulations [3].

CNSC Reg Doc 2.5.2 is the most appropriate of the three stated directives for deriving pressure boundary requirements for the ARC-100. Section 7.7 of Reg Doc 2.5.2 requires all pressure boundary Systems, Structures, and Components (SCC) to be "classified, designed, fabricated, erected, inspected, and tested in accordance with established standards" [1]. Section 7.7 of the Reg Doc provides guidance on how to satisfy the CNSC requirements, by stating: "This is achieved by using industry standards – such as CSA N285, General requirements for pressure-retaining systems and components in CANDU nuclear power plants and ASME Boiler and Pressure Vessel Code" [1].

2.1 CSA N285 pressure boundary classification of ARC-100 systems

The purpose of classifying pressure boundary systems is to ensure design rules and material selection (among other considerations) are commensurate with the function of the pressure boundary system, the risk to the public if the system were to fail, and the systems expected service conditions. Higher pressure boundary classes do not necessitate a safer system, but force the designer/licensee to apply appropriate consideration of the pressure boundary systems intended purpose. Much of the reason for the higher class design rules, material integrity, and higher testing requirements are to relieve cost and undue over-thickness of pressure retaining components. If the simplified design rules of lower pressure boundary classes were applied to systems that are designed to withstand extreme events (like the PHTS for example) pressure retaining component thicknesses would be un-practically thick to meet the higher factor of safety of the lower pressure boundary class design codes [4]. Pressure boundary Class 1 systems require more complex stress analysis rules, higher material integrity requirements, and a division of service level loading conditions paired with higher allowable stresses (not just the simple application of the highest simultaneous operating temperature and pressure conditions expected) which allow for thinner pressure retaining thickness of components and systems that can withstand extreme events.

2.1.1 Pressure boundary Class 1

The functional requirement for a Class 1 system classification is a system that contains "fluid that directly transports heat from nuclear fuel, and whose failure would cause a loss of coolant accident" [5]. This would make the ARC-100 reactor vessel a Class 1 system. The failure of the reactor vessel would cause a loss of liquid sodium which is the PHTS fluid that removes heat from the reactor core (to a point, as there are protections against this typical to SFRs, ie the guard vessel). The functional requirement of a Class 1 system from CSA N285 being fairly straight forward, regardless of the technology that the standard was originally written for. Like a CANDU plant, there is a PHTS purification system in the ARC-100. Similarly, this system also would qualify for Class 1 as it will contain PHTS sodium.

2.1.2 Pressure boundary Class 2

The functional requirement for a Class 2 system classification is "systems that penetrate the containment structure and form part of the containment boundary" [5]. This requirement being fairly straight forward is simply applied to the ARC-100 IHTS. The IHTS is one of few systems that will penetrate the concrete containment structure of the ARC-100 reactor building, forming part of the containment system. Therefore, the portion of the IHTS that is part of the containment system will be classified as CSA N285 Class 2. Once the piping of the IHTS breach containment, isolation valves are used as a classification break, per requirements of CSA N290, and the IHTS changes to CSA N285 Class 6. The Class 6 classification applies to the piping between the ARC-100 Steam Generator and containment isolation valves. The Class 6 classification applies to the Steam Generator.

There are few other pressure boundary systems that form part of the containment system. PHTS purification penetrates containment, however, because it contains PHTS and a failure could cause a loss of coolant accident (although small), this system carries the higher classification of Class 1. There are two passive decay heat removal systems that reject reactor decay heat to the atmosphere. To do so requires that portions of both decay heat removal systems penetrate the containment boundary, so these will also carry a CSA N285 Class 2 classification.

2.1.3 Pressure boundary Class 3

CSA N285 pressure boundary Class 3 has non-functional requirements for classification. Although the criterion for classification is different, the ultimate intent is the same as CSA N285 Class 1 and Class 2: protection of the public from damaging effects of radiation exposure. A pressure boundary Class 3 classification is for a system that is not required to be Class 1 or Class 2, and the system contains tritium in amounts exceeding 74 Gbq/kg [5].

A commentary on CSA N285 was issued in November of 2016 with the intention of providing "a context and explanation for the structure and content of CSA N285, and outlines the governing principles and requirements" of the standard [6]. In section A.2 of the commentary the distinctions between CSA N285 pressure boundary Class 1, Class 2, Class 3, and Class 6 are discussed. The tritium concentration that distinguishes between a Class 3 and Class 6 classification was originally based on a dose of 100 mSv from a major pressure boundary failure, not part of normal plant operation [6]. Since major pressure boundary failures are infrequent a normalized limit for normal operation was more practical. The current regulatory limit is an effective dose of 20 mSv/yr (100 mSv averaged over 5 years, no more than 50 mSv in one year) [6]. Therefore, a limit based on the "expected equilibrium tritium concentration" is used for classification of pressure boundary systems [5]. This shows the point being made is tritium is not the specific concern of a Class 3 classification. Instead, radiation dose limits from systems containing radionuclides affecting workers and respecting the yearly dose limits set by the regulation. G.G. Legg also wrote a commentary on CSA N285 issued to the CSA N285A Technical Committee in July of 1998 [7]. Similarly, he explains that the distinction of pressure boundary Class 3 classification was (at that time) "0.4 TBq/kg (0.01 Ci/g) for tritium, or an equivalent energy-weighted activity of other radionuclides" [7]. Relating equivalent energy-weighted activity of radionuclides present in systems of non-water cooled technologies is how non-water based technologies can classify systems to CSA N285 Class 3 that do not contain tritium.

Table 1 ARC-100 equivalent radionuclide concentration comparison.

Radionuclide	Coolant (Primary or Secondary)	Acceptable Concentration (GBq/kg)	Calculated Concentration (GBq/kg)	Acceptable (A)/Not Acceptable (NA)
H-3 (Tritium)	Primary	74	0.25	A
Na-22	Primary	0.29	0.20	A
Na-24	Primary	0.09	1333.72	NA
Na-24	Secondary	0.09	3.23E-06	A
I-131	Primary	0.64	33.56	NA
Cs-137	Primary	0.56	10.76	NA

Table 1 is a summary of calculated radionuclide concentrations for sodium in the PHTS and IHTS compared to the tritium limit published in CSA N285 for the purpose of pressure boundary classification (there is little to no tritium in ARC-100 systems). The PHTS is listed for comparison only, as it is classified by function as CSA N285 Class 1. The IHTS contains activated sodium (Na-24) as a portion of the system is in the reactor vessel and exposed to a neutron field. The sodium of the IHTS does not reach radionuclide concentrations that would qualify it for a Class 3 classification. Portions of the IHTS do form part of the containment system, but the sections that are not part of the containment system can be classified as CSA N285 pressure boundary Class 6 (non-nuclear).

3. Pressure Boundary Design and Material Selection for High Temperature Systems

Pressure boundary components and systems classified using CSA N285 are required to follow design rules, material selection, and fabrication requirements commensurate with the classification. These requirements are typically, if not always, delegated to the ASME Boiler and Pressure Vessel Code (BPVC). For water-cooled reactor technologies ASME BPVC Section III Division 1 is used by CSA N285 to provide requirements for design, material, and fabrication. However, Division 1 of ASME BPVC Section III limits temperatures for pressure boundary material. Figure 2 shows the temperature limits of materials accepted for use in Division 1 of ASME BPVC Section III. The PHTS and IHTS of the ARC-100 operate above the temperature limits of ASME BPVC Section III Division 1 (950°F and 890°F respectively).

Table HAA-1130-1
Values of T_{max} for Various Classes of Permitted Materials

Materials	T_{max}, °F (°C)
Carbon steel	700 (370)
Low alloy steel	700 (370)
Martensitic stainless steel	700 (370)
Austenitic stainless steel	800 (425)
Nickel–chromium–iron	800 (425)
Nickel–copper	800 (425)

Figure 2 ASME BPVC Sec. III Div. 5 Table HAA-1130-1 [10]: limits for low temperature service rules.

ASME BPVC Section III has developed a code division specific to high temperature reactors: ASME BPVC Section III Division 5. Materials under Division 5 are being qualified for long operating life and with high temperature considerations. Figure 3 shows allowable stress values, S_{mt}, for 9Cr-1Mo-V steel considering long life and high temperature effects. Unlike ASME BPVC Section VIII Division 1 and the ASME B31 Codes which limit material allowable stress to 100,000 hrs for high temperature creep, ASME BPVC Section III Division 5 is qualifying material and providing design rules for creep and creep fatigue up to 500,000 hrs, as shown in Figure 3.

Figure HBB-I-14.3E
S_{mt} — 9Cr-1Mo-V

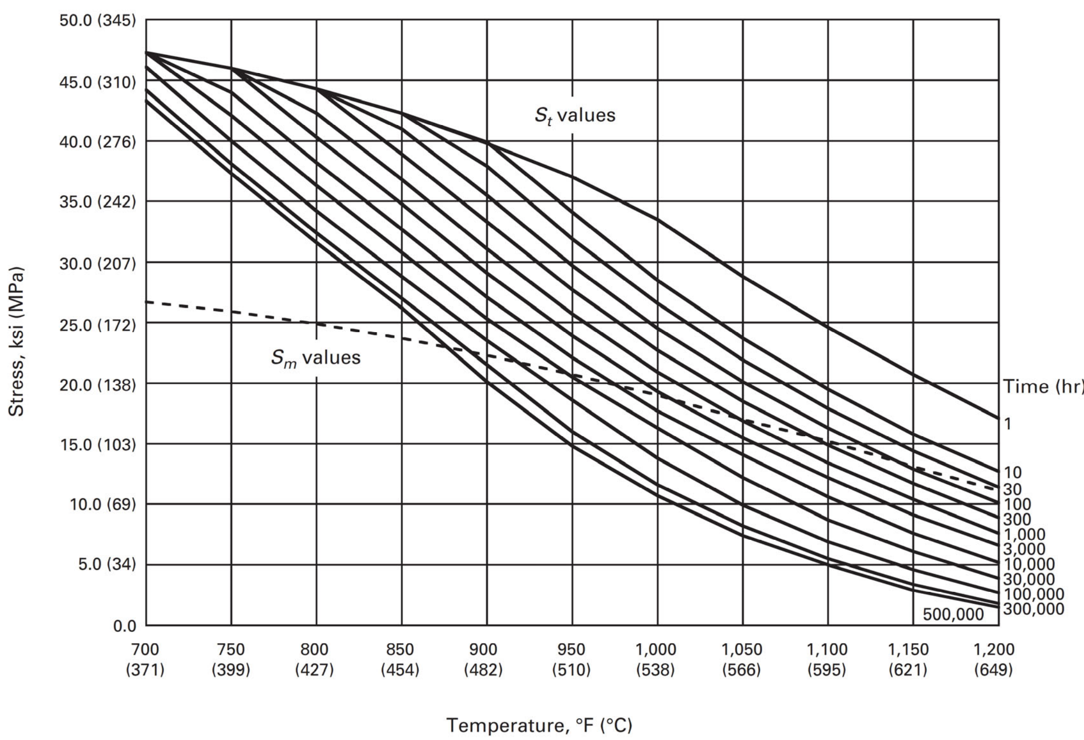

Figure 3 ASME BPVC Sec. III Div. 5 Table HBB-I-14.3E [10]: S_{mt} allowable general membrane stress with time at temperature consideration.

Water-cooled reactor technology typically does not operate at temperatures where creep and creep fatigue are a required design consideration. As such, ASME BPVC Section III Division 1 limits temperatures to where creep and creep fatigue are negligible.

Table HBB-U-1
Recommended Restrictions

Element	Type 304	Type 316
(a) Chemical Composition [Note (1)]		
Carbon	0.04–0.06	0.04–0.06
Nitrogen	0.04–0.07	0.04–0.07
Silicon	0.6	0.6
Manganese	1.0–2.0	1.0–2.0
Nickel	8.00–10.00	11.00–12.5
Chromium	18.5–20.00	17.00–18.00
Molybdenum	0.2	2.5–3.0
Sulfur	0.02	< 0.02
Phosphorus	0.045	< 0.03
Niobium	0.02 [Note (2)]	...
Aluminum	0.05	0.05
Antimony	0.02	0.02
Boron	...	0.003 [Note (3)]
Lead	0.003	0.003
Selenium	0.015	0.015
Tin	0.015	0.015
Vanadium	0.05	0.05
Zinc	0.01	0.01
(b) Grain Size (ASTM)	3–6	3–6
(c) Melt Practice	AOD or AOD/ESR	AOD or AOD/ESR
(d) Suggested upper long-term use limit for improved performance:		
Temperature, °F (°C)	1,100 (595)	1,100 (595)

Figure 4 ASME BPVC Sec. III Div. 5 Table HBB-U-1[10]: chemistry recommendations for high temperature service.

ASME BPVC Section III Division 5 also recommends chemistry control of materials to enhance high temperature performance, as seen in Figure 4. High temperature embrittlement of 300 series austenitic stainless steel occurs at temperature above 500°F (significant embrittlement at 885°F) with amounts of ferrite present in the material [8]. It is the authors opinion that similar chemistry control (like that in Figure 4), within limits of ASME BPVC Section II material chemistry specifications, can reduce the amount ferrite present in the material. This would lessen the embrittlement phenomena at high temperatures.

Ductility is an important characteristic for pressure boundary materials. With low system design pressures the limiting stresses in pressure boundary systems are likely to be caused by thermal or dynamic loading (seismic/earthquake), which will be a shift in design philosophy from high pressure stresses being the primary driver for wall thickness requirements. This being the case, maintaining pressure boundary material ductility, in high temperature operation, to avoid brittle fracture during a dynamic loading event, such as an earthquake, is important.

ARC-100 systems operating at temperatures above the limits of ASME BPVC Section III Division 1 are the PHTS and IHTS. These systems require the design rules and qualified material found in the newer Division 5 code. Adoption of Division 5 in Canada has not yet occurred. The U.S. Nuclear Regulatory Commission (NRC) has only recently accepted ASME BPVC Section III Division 5 [9]. The “Class system” found in the Subsections of ASME BPVC Section III Division 1 is also present in the new Division 5. This allows it to pair well CSA N285 as written today and applied to SFR technology.

4. Conclusion

The current Canadian regulatory framework and requirements paired with existing Canadian nuclear standards are, in the authors opinion, applicable to SFR technology. While the original intention and context of the regulations and standards were for water-cooled reactor technology (CANDU), the same objectives can be met when applied to other technologies. The functional requirements of CSA N285 along with ASME BPVC Section III Division 5 (High Temperature Reactors) can successfully meet the “safety objectives, high-level safety concepts, and safety management requirements” of the regulations in Canada [1].

5. References

[1] CNSC, "REGDOC-2.5.2 Design of Reactor Facilities: Nuclear Power Plants", 2014, pp. 1, 46-47.

[2] CNSC, "Licence Conditions Handbook LCH-PR-17.00/2022-R000", 2017, pp. 63.

[3] NB Gov., "New Brunswick Regulation 84-174, Boiler and Pressure Vessel Act", 2011, Section 401 (3) (a).

[4] Reedy Engineering, Inc., "Development of ASME Section III", 2022.

[5] CSA Group, "N285.0-12 General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants/ Material Standards for Reactor Components for CANDU Nuclear Power Plants Update No. 2", 2014, pp. 12, 65.

[6] CSA Group, "Commentary on CSA N285.0-12 General requirements for pressure-retaining systems and components in CANDU nuclear power plants", 2016, pp. 8, 48.

[7] G.G. Legg, "Pressure-Retaining Systems in CANDU Nuclear Power Plants Explanation of Requirements", 1998, sec. 2.4.3.

[8] ASME, "ASME Boiler and Pressure Vessel Code Section II Materials Part D Properties", 2010, pp. 865.

[9] US NRC, "ACCEPTABILITY OF ASME CODE, SECTION III, DIVISION 5, "HIGH TEMPERATURE REACTORS", 2023.

[10] ASME, "ASME Boiler and Pressure Vessel Code Section III Division 5 High Temperature Reactors", 2021, pp. 2, 285, 112.