AECL EACL

Licensing Submission

Canadian Codes and Standards Used in CANDU Plants

ACR USA

108US-03621-LS-002
Revision 0

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2004 March
Mars 2004

CONTROLLED - Licensing
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Release and Revision History / Liste des documents et des révisions

0939B Rev. 13

Document Details / Détails sur le document

Title / Titre	Total no. of pages / N^bre total de pages
Canadian Codes and Standards Used in CANDU Plants	

CONTROLLED – Licensing / CONTRÔLÉ - Permis

Release and Revision History / Liste des documents et des révisions

Release / Document No./N°	Release / Document Date	Revision / Révision No./N°	Revision / Révision Date	Purpose of Release; Details of Rev./Amendement / Objet du document; détails des rév. ou des modif.	Prepared by / Rédigé par	Reviewed by / Examiné par	Approved by / Approuvé par
1		D1	2004-02-20	Issued for Review and Comment.	A. H. Stretch	G. Rousseau M. Elgohary M. Jankovic R. Jaitly/D. Johal E. Choy K. Hau J. Millard A. Josefowicz	
2		0	2004-03-02	Issued as "Approved for Use".	A. H. Stretch	G. Rousseau	V. Langman

DCS/RMS Input / Données SCD ou SGD

Rel. Proj. / Proj. conn.	Project / Projet	SI	Section	Serial / Série	Sheet / Feuille No. / N°	Sheet / Feuille Of / De	Unit No.(s) / Tranche n°
	108US	03621	LS	002	1	1	

TABLE OF CONTENTS

SECTION **PAGE**

TABLES

1. PURPOSE

AECL Technologies will be applying to the US NRC for the Design Certification of the Advanced CANDU® Reactor (ACR™*) design. This document provides information about the generic design of the ACR to the US NRC as part of the pre-application activities. This document outlines the Canadian nuclear codes and standards used in the design of CANDU plants.

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* ACR™ (Advanced CANDU Reactor™) is a registered trademark of AECL.

2. OBJECTIVES

This document provides information about the development of Canadian codes and standards (i.e. the CAN/CSA N Series of standards) for nuclear safety related systems, and their application to the design of CANDU plants.

The objective of this licensing submission is to enable discussions between the design team and the USNRC about the potential application of Canadian codes and standards in the ACR design (especially to the CANDU specific aspects of the ACR design), and the suitability of these standards for certification of the design in the US.

3. NUCLEAR STANDARDS APPLIED TO CANDU PLANTS

3.1 Development of Nuclear Standards

Standards in Canada are developed and maintained by the National Standards System, which includes over 7200 National Standards of Canada. A federal Crown corporation, the Standards Council of Canada (SCC), coordinates and manages the National Standards System. The Standards Council of Canada accredits organizations to develop, maintain, and revise standards. The main standards development organizations are the Canadian Standards Association (CSA), the Underwriters Laboratories of Canada (ULC), the Bureau de Normalisation du Quebec (BNQ), and the Canadian General Standards Board (CGSB).

In the 1970s, the CSA established the nuclear series (N Series) of standards applicable to the safety related systems in nuclear power plants (note that the term "safety related systems" as defined in these standards is broader than defined in US 10CFR50, including more than just accident mitigation systems). The Strategic Steering Committee on Nuclear Standards was established to oversee the development of standards for siting, design, manufacture, construction, commissioning, operation, and decommissioning of nuclear power generating plants, systems, and components. Other types of facilities, including radioactive waste management facilities and fuel cycle facilities also fall within the responsibility of this committee. This Committee is not responsible for requirements for the prevention of electrical shock or prevention of fire from electrical sources, which are included in the Canadian Electrical Code, which is addressed by another committee.

The Strategic Steering Committee provides leadership and guidance to Technical Committees, which are established to manage the development and maintenance work for a number of related standards. Each Technical Committee is required to have members from the following categories:

- Supplier/Fabricator/Contractor: those involved in the manufacture, construction, or installation of nuclear facilities or equipment.
- Owner/Operator/Producer: those involved in the operation and maintenance of a nuclear facility.
- Service Industry: those involved in the design, procurement, inspection, or decommissioning services for nuclear facilities.
- Government and/or Regulatory Authority: those who have a statutory responsibility for regulation of any aspect of nuclear facilities or materials.
- General Interest: this category includes professional and lay people employed by academic and scientific institutions, consumer groups, industry associations, testing laboratories, and safety associations.

The number of members in each of the above categories is controlled within a specified minimum and maximum to ensure that each group does not exercise an undue influence on the standard. Each member has one vote in the approval process for the standard. A standard is not issued until all negative votes on a final draft standard are satisfactorily resolved.

Based on the above, it is noted that the Canadian nuclear facilities regulator, the Canadian Nuclear Safety Commission (CNSC), serves as a member of each Technical Committee. While a standard is acceptable to the CNSC when issued, the CNSC does have the right to impose additional requirements or different requirements, generally through the issue of regulatory documents and license conditions. In some cases, the regulator imposes a standard in regulatory documents or as a license condition, but otherwise the plant designer is free to use other design methods acceptable to the regulator to address requirements in the standards.

3.2 Current Canadian Nuclear Standards

Canadian nuclear standards are issued in several different series, each of which addresses a general technical topic, as outlined below. A consolidated list of these standards is shown in Table 1, along with revision dates. Revisions and reaffirmations occur on a regular planned basis, and more recent dates may apply – note that the re-affirmation dates are not shown. There are also some additional standards that have been completed, but are not yet issued – these are expected to be issued in the near future.

A summary of the content of some standards is given where these are considered to be more relevant to the design and of particular interest to the USNRC. In a few cases, the rationale behind a particular requirement is also noted.

3.2.1 N285 Series: Systems and Components

This series provides the design requirements and the application of those requirements to pressure boundary components. The standards in this series are listed below, with a short summary of their content.

- N285.0 General Requirements for Pressure–Retaining Systems and Components in CANDU Nuclear Power Plants

  This standard provides the rules and requirements for:

  a) The effective date of standards. Generally, the date of the standards applied would be the date of the construction license for a plant, with earlier or later dates subject to the approval of the regulator.
  b) The responsibilities of the owner, designer, fabricator and installer.
  c) Classification of pressure boundary systems:

     Class 1 is applied to systems that directly transport heat from nuclear fuel and whose failure would cause a loss of coolant accident.

     Class 2 is applied to piping systems and components that form part of the containment boundary (see also Class 4).

     Class 3 applies to all other systems that contain radioactive substances with a tritium concentration above 0.4 TBq/kg (10 Ci/kg) or equivalent (note: the rationale for this number is that it would cause a radiation dose of more than 100 mSv (10 rem) to a person standing next to a major failure of the pressure boundary, and considers that this type of failure would be an accident situation that falls outside of the normal occupational dose limits for plant staff).

     Class 4 applies to containment boundary components such as airlocks, seal plates, and electrical penetration assemblies that are not addressed in Class 2.

Class 6 is applied to all other safety related systems, including systems containing radioactive materials below the criteria established for Class 3.

d) Registration requirements (applicable to plants in Canada, where all pressure boundary systems and components are registered with the regulator).
e) Design requirements for systems and components. In general, subject to the other standards listed below, the following ASME requirements are applied to the design:

   Class 1 systems and components are designed to ASME Section III Class 1, Class 2 systems and components are designed to ASME Section III Class 2, Class 3 components are designed to ASME Section III Class 3, and Class 6 systems and components are designed to "non-nuclear" standards as listed in another CSA standard, CSA B51 "Boiler, Pressure Vessel, and Pressure Piping Code" (this standard calls up industry standards such as ANSI/ASME standards B31.1 and B31.3)

   Note that the revision date of the above standards would be that of the date of the construction license (per (a) above, unless agreed otherwise with the regulator.
f) Overpressure protection requirements
g) Design documentation for systems and components
h) Fabrication and installation requirements
i) Quality assurance requirements (i.e. application of the N286 series listed below)
j) Inspection, examination, and testing
k) Records, nameplates, and identification
l) Inservice inspection and testing

- N285.2: Requirements for Class 1C, 2C and 3C Pressure–Retaining Components and Supports in CANDU Nuclear Power Plants

  This standard provides requirements for pressure boundary components for which the requirements of ASME Section III are not applicable or need to be supplemented to perform their specific function (note that ASME Section III was developed for pressure vessel type reactors that are not refueled while at power and do not use pressure tubes as the reactor coolant boundary). The components addressed in this standard are:

  a) Fuel channel assemblies (pressure tubes, end fittings, closure plugs)
  b) Calandria assembly (calandria tube, rolled joint, lattice tube)
  c) Reactivity control units (control rod drive housing, liquid injection nozzles)
  d) Fuel handling equipment (elastomeric hose, fuel channel closure and fueling machine safety locks, fueling machine supports).

- N285.3: Requirements for Containment Systems Components in CANDU Nuclear Power Plants.

  This standard applies to various types of containment components, including airlocks and transfer chambers, seal plates, flexible bellows and seals, inflatable seals, and materials for these components.

- N285.4: Periodic Inspection of CANDU Nuclear Power Plant Components.

  This standard establishes the minimum inspection requirements for pressure retaining components and their supports, other than containment boundary components that are addressed in N285.5 listed below. A revision of this standard, reflecting changes in the inspection of feeders, pressure tubes, and steam generator tubes to reflect recent experience, is due for issue in the near future.

  The standard establishes the criteria for selecting a suitable inspection sample that will provide sufficient evidence that a generic deterioration is not taking place that was not addressed in the design of the plant, while maintaining the radiation exposure of inspection staff at a reasonable level. An “Inspection Rationale” statement is provided as an introduction to this standard to explain the development of the criteria.

  This standard establishes the inspection criteria in the following manner:

  a) Systems are selected for the inspection program if they directly transport heat from the nuclear fuel or whose failure could result in a significant release of radioactive substances, if they are essential for the safe shutdown of the reactor and cooling of the nuclear fuel, and systems whose failure could jeopardize the integrity of these systems.

  b) The extent of the systems subject to inspection is established, based on isolation valves and consequence of failure.

  c) Acceptable inspection procedures and criteria are established

  d) An “inaugural” or “baseline” inspection sample is defined, as well as a smaller “periodic inspection” sample. The inspection areas defined in the “periodic inspection” sample are initially inspected during a five year period starting one year after first net power generation, and thereafter on a ten year cycle (except for heat transport system pressure tubes, which have a separate inspection schedule).

  e) The criteria for the selection of the inspection sample include:

     1) Size of failure, in terms of energy release rate compared with the largest energy release rate in an accident.

     2) Fatigue usage factor.

     3) Stress intensity compared with the allowable stress intensity.

     4) Extent of inspection, based on the type of component or inspection area.

     5) Reduction of the sample size based on the number of identical components and the number of identical reactor units.

  f) Additional specific periodic inspection criteria are provided for fuel channel pressure tubes, steam generator tubes, and fuel channel feeder pipes because their failures generally fall outside of the failure criteria established above, due to their small size, as outlined below (inaugural inspections are not addressed below):

     1) For single unit plants, five pressure tubes are selected for volumetric and dimensional inspection, and are inspected within a 3 year period, starting 4 years after first power generation, and thereafter on 6 year intervals. One pressure tube is selected from the designated lead unit for material surveillance (i.e. testing for

hydrogen concentration, facture toughness, etc) after 12 years of operation within a 2 year period, and thereafter on 3 year intervals.

2) For the first unit, 10 fuel channel feeder pipes are selected for inspection, and 7 feeders are selected from the second unit, with a further reduction for subsequent units.
3) For steam generator tubes, 10% of the tubes in one steam generator of the first unit are selected for inspection, at intervals not exceeding five years, and 2% of the tubes in one steam generator in the second unit (assuming identical design and construction).

- N285.5: Periodic Inspection of CANDU Nuclear Power Plant Containment Components

  This standard establishes the minimum inspection requirements for metallic containment boundary components.

  As for N285.4 described above, the standard establishes the criteria for selecting a suitable inspection sample that will provide sufficient evidence that a generic deterioration is not taking place that was not addressed in the design of the plant, while maintaining the radiation exposure of inspection staff at a reasonable level. The type and extent of inspection is based on the component characteristics, its loading during normal and accident conditions, and the most likely type of deterioration that may occur (loss of material, corrosion related cracking, discontinuities).

- N285.6: Material Standards for Reactor Components for CANDU Nuclear Power Plants.

  This standard consists of nine separate material standards for pressure boundary components for which suitable ASME or ASTM material standards do not exist. In some cases, ASME or ASTM material standards do exist, but need to be modified for this application as outlined in these standards.

  These include the following, for which the title is explanatory of the content:

  a) CAN/CSA N285.6.1-88 Seamless Zirconium Alloy Tubing for Fuel Channels
  b) CAN/CSA N285.6.2-88 Seamless Zirconium Alloy Tubing for Reactivity Control Rods
  c) CAN/CSA N285.6.3-88 Annealed Seamless Zirconium Alloy Tubing for Liquid Injection Shutdown System Nozzles
  d) CAN/CSA N285.6.4-88 Thin-Walled, Large Diameter Zirconium Alloy Tubing
  e) CAN/CSA N285.6.5-88 Zirconium alloy Wire for Fuel Channel Spacers
  f) CAN/CSA N285.6.6-88 Inspection Criteria for Zirconium Alloys
  g) CAN/CSA N285.6.7-88 Zirconium alloy Design Data
  h) CAN/CSA N285.6.8-88 Martensitic Stainless Steel for Fuel Channel End Fittings
  i) CAN/CSA N285.6.9-88 Materials for Supports for Pressure Retaining Items

3.2.2 N286 Series: Quality Assurance

This series provides the requirements for the management of quality assurance for all phases of the nuclear power plant. The quality assurance standards have been the subject of separate focus topic discussions with the USNRC, so are not addressed further in this document.

- N286.0 Overall Quality Assurance Program Requirements for Nuclear Power Plants
- N286.1 Procurement Quality Assurance for Nuclear Power Plants
- N286.2 Design Quality Assurance for Nuclear Power Plants

  This standard, in addition to ISO 9000, is used as the basis for the development of the Quality Assurance Program applicable to the ACR design.
- N286.3 Construction Quality Assurance for Nuclear Power Plants
- N286.4 Commissioning Quality Assurance for Nuclear Power Plants.
- N286.5 Operations Quality Assurance for Nuclear Power Plants
- N286.6 Decommissioning Quality Assurance for Nuclear Power Plants
- N286.7 Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants

3.2.3 N287 Series: Concrete Containment Structures

This series provides the requirements for the concrete containment structure (i.e. the reactor building for the ACR design). Their titles indicate the content and application of these standards, so they are not further addressed in this document.

- N287.1 General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
- N287.2 Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
- N287.3 Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
- N287.4 Construction, Fabrication and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
- N287.5 Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
- N287.6 Pre–Operational Proof and Leakage Rate Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
- N287.7 In–Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.

3.2.4 N288 Series: Environmental Radiation Protection

These standards provide the methodology for calculation of release limits and dose during normal operation and after accident conditions in Canada, specifications for air cleaning assemblies that filter radioactive particulates, and requirements for radiological monitoring. The titles indicate the content and application of these standards, so they are not discussed further in this document.

- N288.1 Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities
- N288.2 Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors
- N288.3.2 High Efficiency Air–Cleaning Assemblies for Normal Operation of Nuclear Facilities
- N288.4 Guidelines for Radiological Monitoring of the Environment

3.2.5 N289 Series: Seismic Design

These standards provide the methodology for developing the seismic response spectra and the requirements for seismic qualification of equipment. The titles indicate the content and application of these standards, so they are not discussed further in this document.

- N289.1 General Requirements for Seismic Qualification of CANDU Nuclear Power Plants
- N289.2 Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants
- N289.3 Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants
- N289.4 Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants
- N289.5 Seismic Instrumentation Requirements for CANDU Nuclear Power Plants

3.2.6 N290 Series: Control Systems, Safety Systems, and Instrumentation

These standards provide the requirements for support power systems (includes electrical supply systems and compressed air supply), instrumentation and control of shutdown and regulating systems, and requirements for post-accident monitoring. The titles indicate the content and application of these standards, so they are not discussed further in this document.

- N290.1 Requirements for the Shutdown Systems of CANDU Nuclear Power Plants
- N290.4 Requirements for the Reactor Regulating Systems of CANDU Nuclear Power Plants
- N290.5 Requirements for the Support Power Systems of CANDU Nuclear Power Plants.
- N290.6 Requirements for Monitoring and Display of CANDU Nuclear Power Plant Status in the Event of an Accident

3.2.7 N292 Series: Waste Management

This standard provides the requirements for dry storage of CANDU fuel. The details of this standard are not discussed further in this document.

- N292.2 Dry Storage of Irradiated CANDU Fuel

3.2.8 N293 Series: Fire Protection

This standard outlines the requirements for a Fire Protection Program in CANDU plants, and contains requirements for fire prevention in design, fire detection and suppression, separation between groups of safety related systems to ensure that required safety functions can be performed, evaluation of nuclear safety (fire hazard assessment), and fire protection requirements during plant operation. The standard references and applies standards of the National Fire Protection Association (NFPA) for detection and extinguishing systems (as is the practice in building and fire codes in Canada), as well as standards of the Underwriters Laboratories of Canada for establishing the fire rating of structures and components.

It is noted that this standard is currently undergoing a major revision to reflect recent advancements in fire protection, including the application of probabilistic risk assessment and risk informed requirements.

- N293 Fire Protection for CANDU Nuclear Power Plants

4. INDUSTRY STANDARDS APPLIED TO ACR DESIGN

CANDU plants apply many North American industry standards during the design and procurement of equipment and components, including standards from ANSI, ASME, ISA, IEEE, TEMA, etc. On currently operating plants, over 500 of these standards have been applied to the detailed design, procurement, and construction of the plant. These standards are applied by designers, based on the function and requirements of the equipment or structure, through technical specifications of various types. These standards are not listed in this document.

Table 1
Codes and Standards (National Standards of Canada)

Document Number	Title
CAN/CSA–N285.0–95	General Requirements for Pressure–Retaining Systems and Components in CANDU Nuclear Power Plants
CAN/CSA–N285.2–99	Requirements for Class 1C, 2C and 3C Pressure–Retaining Components and Supports in CANDU Nuclear Power Plants
CAN/CSA–N285.3–88	Requirements for Containment Systems Components in CANDU Nuclear Power Plants.
CAN/CSA–N285.4–94	Periodic Inspection of CANDU Nuclear Power Plant Components.
CAN/CSA–N285.5–M90	Periodic Inspection of CANDU Nuclear Power Plant Containment Components.
CAN/CSA–N285.6 Series 88	Material Standards for Reactor Components for CANDU Nuclear Power Plants.
CSA N286.0.1-92	Commentary on the Principles for Quality Assurance Programs of CSA N286 Series Standards.
CAN/CSA–N286.0–92	Overall Quality Assurance Program Requirements for Nuclear Power Plants.
CAN3–N286.1–00	Procurement Quality Assurance for Nuclear Power Plants
CSA N286.2–00	Design Quality Assurance for Nuclear Power Plants
CSA N286.3–99	Construction Quality Assurance for Nuclear Power Plants
CAN/CSA–N286.4–M86 (R2000)	Commissioning Quality Assurance for Nuclear Power Plants.
CSA N286.5–95	Operations Quality Assurance for Nuclear Power Plants.
CSA N286.6-98	Decommissioning Quality assurance for Nuclear Power Plants
CSA N286.7–99	Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants
CSA N287.1–93	General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
CAN/CSA–N287.2–M91 (R1998)	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
N287.3–93	Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
CAN/CSA–N287.4–92	Construction, Fabrication and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
CSA N287.5–93	Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
CSA N287.6–94	Pre–Operational Proof and Leakage Rate Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
CAN/CSA N287.7-96	In–Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.
CAN/CSA–N288.1–M87	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities
CAN/CSA–N288.2–M91	Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors

Document Number	Title
CAN3–N288.3.2–M85	High Efficiency Air–Cleaning Assemblies for Normal Operation of Nuclear Facilities
CAN/CSA–N288.4–M90	Guidelines for Radiological Monitoring of the Environment
CAN3–N289.1–80	General Requirements for Seismic Qualification of CANDU Nuclear Power Plants
CAN3–N289.2–M81	Ground Motion Determination for Seismic Qualification of CANDU Nuclear Power Plants
CAN3–N289.3–M81	Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants
CAN3–N289.4–M86	Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants
CAN/CSA–N289.5–M91	Seismic Instrumentation Requirements for CANDU Nuclear Power Plants
CAN3–N290.1–80	Requirements for the Shutdown Systems of CANDU Nuclear Power Plants
CAN3–N290.4–M82	Requirements for the Reactor Regulating Systems of CANDU Nuclear Power Plants
CAN/CSA–N290.5–M90	Requirements for the Support Power Systems of CANDU Nuclear Power Plants.
CAN3–N290.6–M82	Requirements for Monitoring and Display of CANDU Nuclear Power Plant Status in the Event of an Accident
CSA–N292.2-96	Dry Storage of Irradiated CANDU Fuel
CSA N293–95	Fire Protection for CANDU Nuclear Power Plants
CAN/CSA B51-M1991	Boiler, Pressure Vessel, and Pressure Piping Code
CAN3-Z299.0-86	Guide for Selecting and Implementing the CAN3-Z299-85 Quality Assurance Program
CAN3–Z299.1–85	Quality Assurance Program - Category 1
CAN3–Z299.2–85	Quality Control Program - Category 2
CAN3–Z299.3–85	Quality Verification Program - Category 3
CAN3–Z299.4–85	Quality Assurance Program - Category 4

Notes:

1. The Standards may call up other international or industry standards that are applicable. Standards prepared by ASME and ANSI for pressure vessels and piping are applied in this manner.
2. Compliance with the codes and standards will be only to those requirements applicable to the design of the plant.
3. The (Ryear) indicates the reaffirmation date for each standard – in some cases, a more recent reaffirmation may have occurred than is indicated above.
4. The CAN3-Z299 series above (end of table) is due to be withdrawn in the near future, replaced by the ISO 9000 series for quality assurance.