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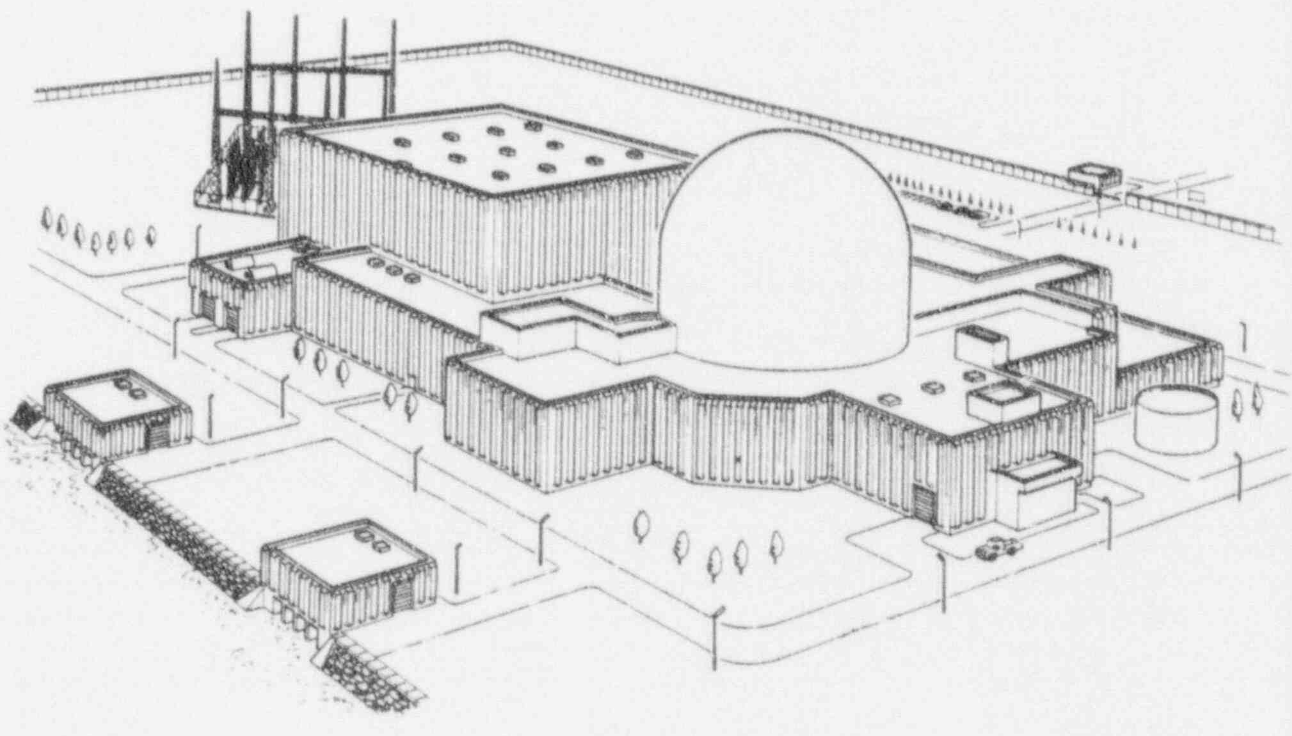
CANDU 3 and U.S. NRC Requirements

Equivalent Safety Issues: Containment Design

TTR-411



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Abstract

This document compares selected features of the CANDU 3 Containment System to U.S. NRC requirements. Four issues including: containment load combinations, containment structural design for accidents, containment isolation, and containment heat removal, are discussed.

These comparisons lead to the conclusion that, despite the design differences, the CANDU 3 design provides an adequate level of safety.

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CHAPTER 1

EQUIVALENT SAFETY ISSUES

INTRODUCTION

by

F. Ardc-hiri

1.1 INTRODUCTION

Licensing and safety analysis practice has evolved independently in Canada and the United States. The two countries' approaches to design, safety analysis and licensing have identical safety objectives and have been guided by national and international agreements and standards and the natural easy flow of information and knowledge across the international borders.

In both Canada and the United States the assessment of reactor design and safety of operation proceeds through the application of broadly similar processes with similar objectives. Utilities and designers operate within a set of federal government prescribed guidelines which have developed since the middle of this century. Analysis and operational practice are submitted by the utilities to federal government regulatory agencies for review. Utility submissions provide details of design and safety assessments which are then reviewed with respect to established practice prior to the granting of an operating license by the regulatory agency.

A comprehensive set of regulations and guidelines has been developed by the United States Nuclear Regulatory Commission (U.S. NRC). These naturally focus on the Light Water Reactors (LWR) which are prevalent in the U.S. According to 10 CFR 50, Appendix A (Introduction), modifications to these regulations may have to be developed by the U.S. NRC for other reactor types to assure an acceptable level of safety. Differences exist, however, between the CANDU (CANadian Deuterium Uranium) reactor design and the LWR. Some of the major ones include:

- The CANDU reactor utilizes natural uranium fuel which is contained in pressure tubes while the LWR design uses enriched uranium contained in a large pressure vessel;
- The CANDU reactor is cooled by heavy water which is separated from the heavy water moderator while the LWR is cooled and moderated by the same light water cooling system.

Applying the NRC requirements, which are influenced by the LWR design, to the CANDU 3 reactor design is not always straightforward. To demonstrate that the CANDU 3 meets NRC regulations requires that specific design differences be addressed.

Key differences have been identified and evaluated to show that:

a. CANDU 3 meets the U.S. NRC requirements;

b. CANDU 3 design and U.S. NRC requirements meet the same or similar ultimate safety objectives, i.e., equivalence of safety.

Some key features of the containment system for the CANDU 3 have been reviewed against the U.S. NRC requirements for LWR systems. This review and comparison has resulted in preparation of the "Equivalent Safety Issues" papers included in this document. It is shown that the CANDU 3 containment design affords the degree of safety required by the U.S. NRC licensing criteria.

The following issues are discussed:

- Containment Load Combinations

  The U.S. NRC and Canadian Standards apply two different approaches in determining the loads and load combinations used for containment analysis. This issue demonstrates that both of these approaches provide adequate safety.

- Containment Structural Design For Accidents

  The CANDU 3 structural design is shown to meet an adequate level of safety compared to the requirements of an LWR with large dry containment, for postulated accident scenarios.

- Containment Isolation

  U.S. NRC and CANDU 3 governing requirements, both specify valve configurations required to ensure prevention of radionuclide release to the environment. The CANDU 3 configurations are shown to provide a level of safety equivalent to that of the U.S. NRC requirements.

- Containment Heat Removal

  The U.S. NRC regulations specify a heat removal system designed with safety-grade requirements. The CANDU 3 containment heat removal system is not considered nuclear grade. However, it is shown that the CANDU 3 containment heat removal system is sufficiently qualified and reliable to provide an adequate level of safety comparable to that of the U.S. NRC requirements.

Therefore, despite differences between CANDU 3 and the LWR designs which underlie the U.S. NRC requirements, CANDU 3 containment features are shown to have adequate safety for each of these key issues.

Note that each of these issues is presented in a self-contained chapter with its own set of references, figures, and appendices.

CHAPTER 2

EQUIPMENT SAFETY ISSUES

CONTAINMENT LOAD COMBINATIONS

by

S.J. Barnes

R. Ricciuti

2.1 INTRODUCTION

The containment system is a safety feature in nuclear power plants, designed to limit the release of radiation to the environment. It must maintain its integrity under all anticipated load scenarios due to transients and postulated accidents. Strength requirements of the containment system are determined by establishing a series of load combinations the reactor must be designed to withstand. The series should reflect the most severe combinations of applicable loads.

The methods of determining the load factors and combinations have been developed over the years, based on well established engineering practices. The approaches used by the NRC, which endorses ASME Section III Division 2, and for CANDU differ, however. For this reason, a one-to-one comparison between the different loads and load combinations cannot be made. The intent of this report is to show that the two approaches have equivalent safety, and that the loads and load factors have been developed from the same basic assumptions.

2.2 NRC REQUIREMENTS AND GUIDANCE

The NRC document, the Code of Federal Regulations 10 CFR 50, section 50.34 (f) (3)(v), requires that containment be designed in accordance with ASME requirements (Reference 2.6-1). Article CC-3000 of ASME Section III Division 2 (Reference 2.6-2) outlines the design requirements for concrete containment structures. The approach used by ASME for the design of containment structures is the working stress approach. A description of this method is provided in Appendix 2A.

The loads and load factors which have been established by ASME are found in Reference 2.6-2. They have been divided into several categories. These categories are divided into service loads (CC-3221), which include normal, construction and test loads, and factored loads (CC-3222), which include abnormal, severe environmental and extreme environmental loads. Abnormal categories include the maximum loads which might occur under accident conditions, while environmental categories include loads from natural phenomena. The severe environmental categories include the effects of either an operating basis earthquake (OBE) or the local wind conditions, while the extreme environmental categories accommodate either a safe shutdown earthquake (SSE) or a design basis tornado. The load combinations in ASME (CC-3230) combine the load categories into three service and five factored load combination categories.

2.3 CANDU 3 APPROACH

The approach used by CSA-N287.3 (Reference 2.6-3) for the design of CANDU concrete containment is the limit state method. A description of this method is provided in Appendix 2A. The revised version of standard CSA CAN3-N287.3 is in draft form currently. The load combinations and factors from the draft are shown in Table 2.1.

These loads are divided into site-dependent loads, which include earthquake, climatic and ground water loads; and design-dependent loads, which include all foundation, structural and process related loads. The CANDU 3 generic design accounts for site-dependent loads by applying an envelope to all environmental factors. For example, the maximum wind conditions and the hottest and coldest sites found in applicable areas for CANDU 3 sites, are used for the load determinations. Earthquakes too are determined so as to cover all sites for which a CANDU 3 would be applicable. In all cases, design analysis with site specific parameters is done before building commences, for design compliance verification.

The load categories of CSA-N287.3 encompass all load categories of ASME, both in definition and concepts of evaluating their magnitude. As in ASME, the load combinations in CSA-N287.3 combine the load categories into service and abnormal/environmental categories. Here however, there are three service and three abnormal/environmental load combination categories. There are no severe environmental categories. The ASME severe environmental loads, such as Site Design Earthquake load (Q_{es}) and design wind load (W), have been combined in the service category load combinations. Also, there is no combination of abnormal peak LOCA pressure (P_a) and extreme environmental earthquake (Q_{ed}). However, the Q_{ed} is combined with the maximum pressure which is assumed to occur simultaneously with it, that is, a reduced pressure (P_r) associated with the postulated failure of systems that are not qualified for the extreme seismic loads.

2.4 EVALUATION

The design loads required by the NRC for LWR design, per ASME Section III – Division 2, and those used for the CANDU 3, per CAN3-N287.3 and AECL design requirements, are very similar both in their definition and concepts of evaluating their magnitude. Process loads not identified explicitly are implicit in other load categories. For example, the ASME load category G (relief valves and other high energy device actuation) is included in the CSA-N287.3 R_o and R load categories. As explained in Appendix 2A, the limit state method allows for the effects of creep and shrinkage and thus these loads are included in the CAN3-N287.3 load combinations.

The difference in load factors for service category load combinations, between ASME and CSA codes, is due to the use of working stress design method in ASME, and the limit state design method in CSA. However, as discussed in Appendix 2A, the application of load (increase) and material performance (reduction) factors used in the limit state method provides a safety margin no less than the margin provided by the working stress design method.

A severe environmental load definition does not exist in CSA codes. The severe environmental load category in ASME accounts for the effects of wind and an operating basis earthquake (OBE). However, the CSA includes the effects of wind under the normal service category and does not define an OBE. Instead, it defines a site design earthquake (SDE). (The definition of the earthquakes used by ASME and the CSA are given in Appendix 2B.) The SDE is a low intensity, more probable event which is also included in the normal service category. These two load combinations (OBE and wind versus SDE and wind) have comparable safety margins.

All systems and equipment in a CANDU 3 whose failure can result in a loss of coolant accident (LOCA) are qualified to a design basis earthquake (DBE) (which corresponds to the safe shutdown earthquake (SSE) in ASME), and so the combination of LOCA and DBE (i.e., P_a and Q_{ed}) is not considered in the design of the CANDU 3 containment structure. These systems and equipment are designed by applying ASME level 'C' stress limits. The CSA abnormal/environmental category does, however, include a slightly increased pressure in the seismic load combination. This is established based on the rupture of the non-seismically qualified (NSQ) systems and components. NRC requirements specify ASME level 'D' stress limit, which is two or more times as high as level 'C', for SSE load combinations, as per Section 3.9.3 of NUREG-800 (Standard Review Plan). Also, ASME considers the combination of a LOCA and an SSE, as reflected in its abnormal/environmental load category.

2.5 CONCLUSIONS

The two approaches to load and load combinations, working stress and limit state, used by the NRC and for the CANDU, respectively, have been shown to have adequate safety. The main differences in the load combinations are due to the differing seismic classifications which are used. These classifications have been shown to provide sufficient safety.

2.6 REFERENCES

2.6-1 Code of Federal Regulations, Chapter 10, Part 50, 10 CFR 50.34, (f) (3)(v).

2.6-2 ASME Code for Concrete Reactor Vessel and Containments, Division II, CC-3200.

2.6-3 Canadian Standards:

CSA CAN3-N287.3, "Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", Canadian Standards Association.

CAN3-S16.1-M89, "Limit State Design of Steel Structures", Canadian Standards Association.

CAN3-A23.3-M84, "Design of Concrete Structures for Buildings", Canadian Standards Association.

2.6-4 MacGregor, J.G., "Safety and Limit States Design for Reinforced Concrete", Canadian Journal of Civil Engineering, Vol. 3, 1976, pp 484-513.

Table 2.1

Load Combinations and Load Factors

	D	S_c	L	P_a	P_r	P_{sm}	T_{sm}	T_o	T_a	Q_{es}	Q_{ed}	Q_w	Q_t	R_o	R_a	R	T_r
Service Category																	
Construction	1.25	1.25	1.5	–	–	–	–	1.25	–	–	–	–	–	–	–	–	–
Test	1.25	1.25	1.5	1.7	–	–	–	1.25	–	–	–	–	–	–	–	–	–
Normal	1.25	1.25	1.5	–	–	–	–	1.25	–	1.5	–	–	–	1.5	–	–	–
	1.25	1.25	1.5	–	–	–	–	1.25	–	–	–	1.5	–	1.5	–	–	–
Abnormal/Environmental Category																	
Abnormal	1.0	1.0	1.0	–	–	1.0	1.0	–	–	–	–	–	–	–	1.0	1.0	–
	1.0	1.0	1.0	1.5	–	–	–	–	1.0	–	–	–	–	–	1.0	1.0	
Extreme Environmental	1.0	1.0	1.0	–	–	–	–	1.0	–	–	1.0	–	–	1.0	–	–	–
	1.0	1.0	1.0	–	–	–	–	1.0	–	–	–	–	1.0	1.0	–	–	–
Abnormal/Extreme Environmental	1.0	1.0	1.0	–	1.0	–	–	–	–	–	1.0	–	–	1.0	–	–	1.0

Notes:

D = dead loads, represent the permanent mass of all structural elements and permanent equipment loads.

Q_{es} = forces generated by the site design earthquake.

Q_{ed} = forces generated by the design basis earthquake.

L = live loads, including roof loads, soil pressures, snow loads, hydrostatic loads (ground water and internal flooding) and temporary loads applied during construction, testing, operation and maintenance or replacement of major components.

P_a = design accident pressure caused by the pressure within containment due to a Loss of Coolant Accident (LOCA) or LOCA combined with Loss of Emergency Core Cooling (LOECC) or LOCA combined with Loss of Air Coolers (LOAC).

P_r = reduced design accident pressure due to the rupture of non-seismically qualified (NSQ) components.

P_{sm} = design accident pressure due to steam main break.

T_{sm} = design accident temperature due to steam main break.

T_r = reduced design accident temperature due to the rupture of NSQ components.

R = loads that may be generated by the rupture of a high-energy pipe during the severest postulated accident 'R'. This may be due to a LOCA or MSB.

'R' shall include the effects of:

(a) pipe reaction (R_r);
(b) jet impingement (R_j): and
(c) impact (R_m) of a ruptured pipe on the containment component or element under consideration.

R_a = pipe reactions from thermal expansion generated during an accident.

R_o = pipe reactions during normal operation, shutdown or start-up conditions

S_c = load effects due to shrinkage and creep including the effects of concrete mix compositions, cement content and type, relative humidity, age of concrete at load application, thickness of concrete elements, differential shrinkage and external restraint.

T_a = design accident temperature load caused by the temperature within the containment due to LOCA, LOCA + LOECC or LOCA + LOAC.

T_o = thermally induced loads that may occur during construction, test, or normal operating and shutdown conditions, and including the severest combination of internal and external ambient temperatures, producing the maximum effect.

Q_w = loads generated by the design wind specified for the site in accordance with the requirements of the National Building Code, and with a probability of being exceeded once in every 100 years.

Q_t = loads that may be generated by the design tornado specified for the site

Q_t shall include the effects of:

(a) tornado wind pressure (Q_{tp})
(b) differential pressure loads (Q_{td}) due to rapid atmospheric pressure change; and
(c) tornado-generated missile (Q_{tm}) impact.

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APPENDIX 2A
A COMPARISON OF DESIGN METHODS FOR CONCRETE CONTAINMENT STRUCTURES

2A.1 LIMIT STATES AND WORKING STRESS DESIGN METHODS

Two philosophies of structural design are prevalent and have long been used by structural engineers in the design of steel and reinforced concrete structures. The working stress method was the principal one used from the early 1900s until the early 1960s. In 1963 the American Concrete Institute Building Code (ACI 1963) pioneered the North American use of split load factors to account for overloads and understrength members. Since 1963 there has been a rapid transition to the Limit States method because it is considered to be a more conceptually realistic and rational approach in the establishment of structural safety.

The Limit States method is a probability based design code that requires structures to be proportioned such that the probability of failure occurring is low enough to be acceptable while at the same time specifying that the serviceability requirements of the structure are satisfied.

In contrast to the Limit States approach is the Working Stress method. Here the structural elements are proportioned such that the internal stresses resulting from service loads are within acceptable limits. Generally the stress levels are restricted to the elastic range of the structural materials.

A more complete discussion of the two design philosophies is given below.

2A.1.1 THE WORKING STRESS METHOD

In the working stress method a structural element is designed so that the stresses resulting from the action of service loads and computed by the mechanics of elastic members do not exceed some predesignated allowable values. Service load is the load, such as dead, live, wind, earthquake, which is assumed actually to occur when the structure is in service.

The allowable stresses are prescribed by a building code to provide a factor of safety against attainment of some upper limiting stress, such as the specified compressive strength f'_c for concrete and the minimum specified yield stress f_y for non-prestressed reinforcement steel.

The procedure used for defining structural safety differs between the working stress and limit states design method. In the working stress method of reinforced concrete design the factor of safety is assumed to be closely related to:

$$\text{Factor of Safety} = \text{Yield Strength of Steel} / \text{Allowable Steel Stress} \quad (1)$$

or

$$\text{Factor of Safety} = \text{Concrete Strength} / \text{Allowable Concrete Stress} \quad (2)$$

As an example, ASME Section 3 Division II limits the concrete stresses to $.45 f'_c$ for the case of membrane plus bending under service loads. The allowable tension stresses in the reinforcing bars is also limited under the working stresses method. ACI Appendix B.3 specifies 20000 psi for grades 40 and 50 steel, and 24000 psi for grade 60 and above. Therefore for members properly designed under the Working Stress method, the stresses computed under the action of service loads will be well within the elastic range, so that the linear (straight-line) variation between stress and strain is used.

The major shortcomings of this approach are discussed in detail in Reference 2.6-4. They are summarized briefly below:

1. It does not adequately take into consideration the variability of the applied loads and the variables that can affect the structural strength of a member.

   If the strength of a structural member and the loads applied to it were assumed to have a normal distribution, then the probability of failure would be represented by the intersection of the two normal distribution curves. For the case where the loading and strength are well controlled, in which case there would be a small dispersion of loads and strengths (small standard deviation in probability terminology), then there is relatively little probability of failure.

   However for the case of poor strength and load control, where the the mean strength and loads are the same for the well controlled case but the dispersion of the strengths and loads is the much greater (larger standard deviation), then the probability of failure would be much greater than the well controlled case. However this reduction in safety margins is not accurately represented by the working stress safety factors above.

   In the working stress method for reinforced concrete design the greater variability of concrete is taken into account by using a slightly higher safety factor in computing the allowable stresses in concrete. However the variability of the loading is not rationally accounted for.

2. The variations in loading that increase at different rates or have different signs is not rationally accounted for.

   The factor of safety in the working stress method assumes that all loading will increase at approximately the same rate. This can lead to serious errors in circumstances when a highly variable load such as wind or earthquake loading causes forces opposite in sign to those resulting from relatively constant loads such as dead load or prestressing loads. The resulting stresses from an overload may be opposite in sign to those at service load and the reinforcement provided for service load conditions may not be sufficient to prevent failure.

3. The ultimate load capacity of a member is not evaluated.

   The working stress design method assumes that the ratio between the service load and the ultimate capacity is the same as the ratio between allowable stresses and material strengths. This assumption is not as accurate as defining safety margins in terms of the ultimate load/applied load ratio.

4. There is no rational method of considering the consequences of failure or the type of failure.

2A.1.2 THE LIMIT STATES METHOD

Limit States design is a design method in which the performance of a structure is checked against limiting conditions at appropriate load levels. The limiting conditions to be checked in structural design are Ultimate limit states, Serviceability limit states and Damage limit states. Ultimate limit states are those states concerning safety, such as exceeding of load carrying capacity, overturning, sliding and fracture due to fatigue or other causes. Serviceability limit states are those states in which the behaviour of the structure is unsatisfactory, and include excessive deflection, excessive concrete cracking, excessive vibration and excessive permanent deformation. Damage limit states are related to non-structural damage.

Essentially the Limit States design method attempts to ensure that the maximum strength of a structure (or elements of a structure) is greater than the loads that will be imposed upon it, with a reasonable margin of safety. This is known as the "ultimate limit states criterion". In addition the designer attempts to ensure that the structure will fulfil its function satisfactorily when subjected to service loads. This is the "serviceability limit states criterion". The "damage limit states criterion" aims to reduce the probability of occurrence of non-structural damage in a structure such as premature or excessive cracking or spalling of the concrete or damage to non-structural elements.

Therefore, Limit States Design is a design process that involves :

1. Identification of all potential modes of failure (Limit States).
2. Determination of acceptable levels of safety against occurrence of each limit state.
3. Consideration by the designer of the significant limit states.

2A.1.2.1 Basic Ultimate Limit States Design Equation

The ultimate limit states criterion is related to a structural collapse of part or all of a structure. The ultimate limit states should have a very low probability of occurrence since it may lead to loss of life and major financial losses. The basic equation for checking the ultimate limit states condition is the following:

$$\phi R \geq \alpha_d D + \gamma \psi (\alpha_l L + \alpha_e E + \alpha_t T \ldots \text{ other loads}) \qquad (3)$$

Where

ϕ = Performance factor;

R = Nominal resistance of a structural element;

γ = Importance factor;

ψ = Load combination factor;

$\alpha_d, \alpha_l, \alpha_e, \alpha_t$ = Load Factors;

D, L, E, T = Specified loads for Dead load (D), Live load (L), Earthquake (E) and Temperature loading (T).

The number of load combinations and the applicable loads are specified in the appropriate design codes (e.g. CSA Standards CAN/CSA-N287.3-M91, CAN3-S16.1-M89, CAN3-A23.3-M84, Reference 2.6-3).

On the left side of equation 3, the performance factor ϕ is applied to the nominal member strength, or resistance, to take into account the fact that the actual strength of a member may be less then anticipated due to variability of material properties, dimensions and workmanship. The performance factor also may take into account the type of failure anticipated for the member and the uncertainty in the prediction of the member resistance.

The resistance, R of a structural member, connection or structure is the nominal strength based on specified material properties, nominal dimensions and equations describing the theoretical behaviour of the member, connection or structure.

Hence, the limit states requirement specifies that the factored resistance of a structural element ϕR, is the product of the resistance and the performance factor and must equal or exceed the effect of the factored loads.

The performance factor, γ, is a factor that takes into account the consequences of collapse as related to use and occupancy of the structure.

The load combination factor, ψ, is a factor that takes into account the reduced probability of a number of loads from different sources acting simultaneously.

The specified loads include Dead (D), Live (L), Temperature (T), Earthquake (E) etc. All design loads for Concrete Containment Structures for CANDU Nuclear Power Plants are prescribed by Canadian Standard CAN/CSA-N287.3-M91 (Reference 2.6-3)

The load factor, α, by which a specified load is multiplied to obtain a factored load takes into account the possibility that loads larger than those anticipated may act on the structure, the uncertainty involved in predicting the loads, and the approximation in the analysis of the effects of the loads on the structure. Different load factors, α_d, α_l, α_e, α_t etc. are assigned to the different load effects, thus recognizing the uncertainty of predicting dead load (accounted for by α_d) is less then the uncertainty of predicting live load (as measured by α_l).

The ultimate limit states design uses a maximum probability of failure method to achieve an appropriate safety margin in the structural design. In mathematical terms, the probability of structural failure is expressed by:

$$P_f = P\,[(R - U) < 0] \qquad (4)$$

where

P_f = Probability of Failure

R = Design Structural Resistance

U = Maximum Load

In equation 4, (R-U) represents the safety margin for a given structure. If R and U follow standard distributions (e.g. normal distribution) this probability can be calculated or obtained from tables provided the dispersion of data for R and U is known. Generally equation 4 is not used directly because of the work involved in evaluating the probability of failure for every structure and the need for statistical data on many aspects of loading and construction. However, equation 4 forms the basis of computing Load Factors (α), and Performance Factors (ϕ) specified in the limit states design based codes (Reference 2.6-3).

Thus the Limit States design philosophy is generally recognized to be a more rational approach for establishing structural safety. Furthermore, it is able to address the four major drawbacks that exist in the working stress definition for safety factors.

2A.1.2.2 Serviceability and Damage Limit States

Serviceability limit states is related to disruption of the functional use of the structure. Serviceability criterion include excessive deformations for normal service and undesirable vibrations.

Damage limit states can tolerate a higher probability of occurrence than the ultimate limit states since these criterion generally do not reduce the structural safety margins. Damage criterion is measured by: premature or excessive cracking; excessive deformations leading to non-structural damage or changes in the distribution of forces; permanent inelastic deformations.

APPENDIX 2B
EARTHQUAKE DEFINITIONS

2B.1 CANDU EARTHQUAKES

The Design Basis Earthquake (DBE)

The Design Basis Earthquake means an engineering representation of the potentially severe effects of earthquakes applicable to the site that have sufficiently low probability of being exceeded during the lifetime of the plant. The DBE effects on the site are described by the DBE Ground Response Spectra (GRS). Its effects within structures at the site are described by the Floor Response Spectra (FRS) which are developed for selected locations in each structure. All systems and components which are required to ensure the safe shutdown and cooldown of the reactor, are to be seismically qualified for a DBE.

Site Design Earthquake (SDE)

The Site Design Earthquake (SDE) means an engineering representation of the effects at the site of a set of possible earthquakes with an occurrence rate, based on historical records, not greater than 0.01 per year. The SDE effects on the site and within structures at the site are described by Ground Response Spectra and Floor Response Spectra. All systems and components which are required to ensure the onset of ECC is posible 24 hours after a LOCA, must be seismically qualified for a SDE.

2B.2 NRC EARTHQUAKES

The Safe Shutdown Earthquake (SSE)

The SSE is that earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology and seismology and specific characteristics of local subsurface material. It is that earthquake which provides the maximum vibratory ground motion for which certain structures, systems and components are designed to remain functional. These structures, systems and components are those necessary to assure: (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safety shutdown condition, or (iii) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to prescribed guidelines.

The Operating Basis Earthquake (OBE)

The OBE is that earthquake which, considering regional and local geology and seismology and specific characteristics of local subsurface material, could reasonably be expected to affect the plant site during the operating life of the plant; it is that earthquake which produces the vibratory ground motion for which those features of the nuclear power plant necessary for continued operation without undue risk to the health and safety of the public are designed to remain functional. It is also required that the maximum vibratory ground acceleration of the OBE be assumed at least as one-half the maximum vibratory ground acceleration of the SSE.

CHAPTER 3

EQUIVALENT SAFETY ISSUES

CONTAINMENT DESIGN FOR LOCA AND STEAM LINE BREAK ACCIDENTS

by

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3.1 INTRODUCTION

The safety of CANDU reactors against uncontrolled releases of radioactivity to the public is achieved by a multi-tiered defense. The containment system provides the last defense against activity releases to the environment after various accidents. The accidents might be abnormal events associated with various large pipe breaks, or natural, external, environmental events, such as earthquakes, floods, tornadoes, etc.

In the U.S. large dry containment structures are designed, constructed, inspected and tested to satisfy the U.S. NRC regulations (Reference 3.6-1), the standards of the American Concrete Institute and the ASME Boiler and Pressure Vessel Code (Reference 3.6-2), and various regulatory guides and guidance documents, such as References 3.6-3, 3.6-4 and 3.6-5.

The CANDU 3 containment building design, in essence, is similar to a Pressurized Water Reactor (PWR) dry containment design. The containment system for a CANDU reactor is considered as a Special Safety System to prevent releases of radioactive material to the environment, or limit those releases within the permissible dose limits. The CANDU containment design conforms with the applicable Canadian requirements by the Atomic Energy Control Board (Reference 3.6-6) and the standards of Canadian Standards Association in the CSA CAN3-N287 series (Reference 3.6-7).

This document aims to demonstrate that the engineered features and the evaluations of the CANDU 3 containment structure satisfy the U.S. NRC containment design requirements for Primary Heat Transport System (PHTS) breaks and secondary coolant circuit steam line breaks.

After a short outline of the pertinent NRC and Canadian design requirements, the structural design of the CANDU 3 containment will be outlined. Then the effects of accident peak pressures on containment integrity and on the rates of out-leakage of radioactivity will be discussed, to show how the CANDU 3 design provides an equivalent level of safety to the U.S. NRC requirements for concrete containments.

3.2 NRC REQUIREMENTS AND GUIDANCE

The United States Code of Federal Regulations, 10 CFR, Part 50, Appendix A (Reference 3.6-1), mandates the U.S. Nuclear Regulatory Commission requirements for containments as follows:

- Criterion 16 – Containment Design:

  "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions arise."

- Criterion 38 – Containment Heat Removal:

  "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

- Criterion 50 – Containment Design Basis:

  "The reactor containment structure, including access openings, penetrations, and the containment heat removal system .... can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident." The margin should allow for the energy in the steam generators, the effects of metal-water reactions and other chemical reactions, but not the total failure of emergency core cooling.

The U.S. NRC safety analysis requirements are mandates for "applicants" or "licensees" to identify all conceivable accident initiating events, and evaluate the worst credible loss-of-coolant and associated accidents within the containment and the related risks to the environment. In a PWR such events are any Loss-of-Coolant Accident (LOCA) in the primary coolant circuit, including steam generator tube rupture accidents, any secondary side Steam Line Break (SLB), etc.

The NRC recommended evaluations identify a possibly worst combination of pressure, thermal, mechanical and radioactivity loads on the containment. For Light Water Reactors (LWRs) the containment Design Pressure has to take into account the higher of the LOCA and SLB induced pressures. Specifically, NUREG-0800 (Reference 3.6-3) prescribes that the containment design pressure must envelope all high-energy pipe breaks inside containment. Section 6.2.1.1.A of NUREG-0800 in the Standard Review Plan states the containment structure must be capable of withstanding, without loss of function the temperature and pressure conditions in the containment due to a spectrum (including break size and location) of postulated loss-of-coolant accidents (i.e., reactor coolant system pipe breaks) and secondary system steam and feedwater line breaks. Section 6.2.1.4 states that the SLB accident should be analyzed for a spectrum of pipe break sizes and various plant conditions from hot standby to 102% of full power. Only the 102% power condition need be analyzed provided the feedwater flows and fluid inventory are greatest at full power.

The containment structure Design Specifications are written by the owner or the owner's designee, reflecting the peak loads of the above evaluations.

The U.S. NRC recommendations for deterministic analyses of the containment are outlined in NUREG-0800 (Reference 3.6-3). The U.S. NRC Regulatory Guide 1.136 (Reference 3.6-5) endorses the requirements for design, fabrication, erection and testing of concrete containment structures for nuclear reactors outlined in Section III, Division 2 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Reference 3.6-2). This ASME Code is a U.S. national standard.

ASME Code Section NCA-2140 specifies the Design Basis. The pressures, temperatures and mechanical loads (e.g., from internal missiles or whipping pipes) to which the containment structure is subjected due to plant design, operating and test conditions are identified in a Design Specification by the owner or the owner's designee (e.g., an architect-engineer). The Section states that the "Design Pressure shall not be less than the maximum difference in pressure between the inside and outside of the item, ..., which exists under the most severe loadings for which the Level A Service Limits are applicable. The Design Pressure shall include allowances for pressure surges, control system error, and system configuration effects such as static pressure heads." Similarly, the Section also specifies the Design Temperature and the Design Mechanical Loads.

The ASME Code sets out the design requirements for concrete containments with regard to the structural concrete pressure resisting shell, the metallic liner, and the penetration liners extending the containment liner through the shell concrete. The metallic liner is not used as a strength element for the containment shell, but its interaction with the concrete containment is used for determining the evaluation of strains and stresses in the liner.

The containment load combinations are specified by NRC's NUREG-0800 (Reference 3.6-3).

The ASME Code considers as service loads any loads encountered during construction and in the normal operation of a nuclear power plant. Included in such loads are any anticipated transient or test loads during normal and emergency startup and shutdown of the nuclear steam supply, safety and auxiliary systems. Also included in this category are those severe environmental loads which may be anticipated during the life of the facility.

The ASME design is based on "factored load" requirements, where the "factored loads include loads encountered infrequently, such as severe environmental, extreme environmental, and abnormal loads." The service, factored, severe, extreme and abnormal load categories and their combinations are given in ASME Code, Sect.III, Div.2, CC-3000.

The allowable stresses and strains in steel and concrete are defined in CC-3000. Limitations on maximum concrete temperatures are also given.

The ASME sizing of the concrete and of the reinforcing bars is according to a "working stress" method, which specifies load combinations with prescribed load factors, but has no factoring for the material attributes.

The detail design methods for the containment, including foundation, liner, penetrations, transitions between steel and concrete, crack control in concrete, etc., and also, the requirements for fabrication and construction, construction testing and examination and structural integrity test are also set out in the ASME Code.

The U.S. Code of Federal Regulations, 10 CFR, Part 50, Appendix A requires an essentially leak-tight containment structure after postulated accidents to prevent uncontrolled activity release.

3.3 CANDU 3 APPROACH

A CANDU Containment System consists of a number of components and subsystems:

- containment building structure;
- process line penetrations (steam, water, air, electrical, etc.);
- fuel transfer ports for new and irradiated fuel;
- an equipment hatch and airlocks for operator access;
- process line isolation valves (one or two per line, and automatic on lines potentially open to the containment atmosphere);
- hydrogen/deuterium control (dispersal and recombination);
- containment atmosphere cooling, e.g., by local air coolers;
- gross containment leakage monitoring;
- active waste handling.

The CANDU 3 containment building is the principal component of the containment system. The containment building is a steel-lined, reinforced concrete structure and has been designed with several aims, besides normal housing and weather shelter purposes for a reactor:

a. provide radiation shielding to the environment outside the containment;

b. serve as a pressure barrier and envelope that can retain – or limit the release of – radioactive hazardous effluents from credible primary loss-of-coolant accidents (LOCA);

c. serve as a pressure barrier that can resist the ignition of combustible gases (e.g., hydrogen), that might be present as a result of some loss-of-coolant accidents in the containment, without deterioration of the pressure envelope;

d. maintain the containment structural integrity in the event of a severe secondary side break inside the containment, e.g., a double-ended discharge from a Steam Line Break (SLB);

e. maintain the containment structural integrity in the event of a Design Basis Earthquake (DBE);

f. protect the reactor and its process, control and safety systems from damage by external causes, such as a flood and a tornado.

The containment design is based on established safety considerations in agreement with the AECB R-7 requirements (Reference 3.6-6). The containment safety studies, are based on identifying initiating events, and probabilistically and deterministically evaluating the worst credible accidents in LOCA and SLB accidents. The evaluations extend to the highest peak pressures, thermal loads, mechanical loads (e.g., internal missiles and whipping pipes), and radioactivity concentrations in the containment atmosphere. Stress evaluations are then performed for assuring structural integrity and essential leak-tightness of the containment

envelope. It is assumed that before the most severe pressurizing accident the containment is intact and in its normal operating mode, the access doors and equipment hatch are closed and the components of the containment system perform within specifications except that some of the air coolers may not be available.

After a LOCA or a SLB accident the triggering of the containment isolation is based on detection of increased levels of containment pressure or radioactivity, and the isolation is automatic. A primary LOCA event may initiate isolation by either pressure increase or activity release, while a secondary side steam line break (SLB), practically without radioactivity, will likely cause isolation by high pressure only. The CANDU 3 containment response to an SLB is also analyzed by assuming some pre-existing leaks in some boiler tubes, in which case a minimal, acceptable amount of radioactivity may be released from the containment.

The CANDU containment structure design and construction conforms with the applicable Canadian Standards in the N287 series (Reference 3.6-7) and in the N285 series (Reference 3.6-8). Key requirements set by these standards are given here to highlight some of the relevant design aspects:

CAN3-N287.1-M82 classifies a concrete containment structure and its embedded parts as "Class containment". CAN/CSA-N285.0-M91 classifies the components of a containment system, with a concrete containment structure and its appurtenances capable of closure in the event of an accident, as a Class 4 system. If the Design Pressure is larger than 5 psig (35 kPa(g)), the system has to be designed, fabricated, installed, inspected and registered (including the necessary documentation) by the appropriate ministry department having jurisdiction for the site.

CAN3-N287.1-M82 presents the following design requirements:

a. A containment Design Pressure that is higher than the calculated peak value of overpressure in design basis accidents, which are selected from postulated single failure and dual failure accidents of the PHTS, e.g., loss-of-coolant accident (LOCA) for single failure accidents, and LOCA with coincident unavailability of the Emergency Core Cooling System (LOCA + LOECC), for dual failure accidents.

b. Pipe reaction force values for the steam and feedwater system main piping and other parts of the heat transport and steam systems that require anchorage in the event of rupture;

c. Temperature values and, where applicable, their predicted duration for:

   1. Normal operating conditions;

   2. Safe shutdown conditions; and

   3. All accident conditions (this includes the effects of higher temperatures caused by superheated steam as well);

d. Site-dependent loads obtained from the National Building Code of Canada, and other prescribed sources;

e. A leakage rate acceptance value approved by the licensing authority, to which a factor is applied in order to determine the maximum allowable leakage rate.

To be consistent with the AECB R-7 regulatory requirements (Reference 3.6-6) the CANDU 3 design identifies:

a. test acceptance leak rates, and

b. maximum allowable leak rates.

For practical purposes the test acceptance leak rate level is taken as the design leak rate.

CSA Standard CAN3-N287.3 prescribes the load categories, load combinations and the permissible levels of stress and strain for the containment structure. These are further discussed in Chapter 2.

The physical design of the CANDU 3 containment is in conformance with the outlined approach. The CANDU 3 steel-lined reinforced concrete containment structure, with upright cylindrical perimeter wall, flat base slab and torispheroidal dome (Figure 3-1) is very similar to a number of U.S. PWR containments.

The sizing of the CANDU 3 containment structure was initiated by equipment layout considerations and sizing of the free air volume 1,380,000 cu.ft (39,100 cu.m) on pressure peak considerations. The sizing of the concrete wall thickness was based on shielding and structural requirements. The choice of a steel-lined, reinforced concrete structure was based on constructability, maintainability, repairability, life expectancy, lack of cable inspection, proven design, etc.

The containment cylinder has an outside diameter of 134.5 ft (41 m), a height of 180.7 ft (55.1 m), and a uniform wall thickness of 3.94 ft (1.2 m), except at two locations adjacent to internal subcompartments, where extra shielding is needed. The containment perimeter walls are above grade level and are subject to atmospheric pressure.

The Canadian Standards require the containment to withstand the worst LOCA Design Basis Accident (DBA). The DBA does not require a combination of primary and secondary accidents with seismic events. In fact, the CANDU design is set on a DBA pressure based on the worst primary circuit accident, but makes certain that the worse pipe break accident in the secondary circuit, which may create a minimal radioactive load, is still safe for containment structure integrity and acceptable for activity release levels.The main design aspects of CANDU 3 containment for primary LOCA and secondary SLB accidents are discussed in the next Section.

3.4 EVALUATION

The choice of the containment Design Basis Accident (DBA) pressure is based on peak pressure predictions of a spectrum of postulated significant pipe break accidents. CANDU reactors as well as U.S. PWRs, have indirect cycle heat transport systems from the reactor to the steam turbines. In either reactor type, pipe breaks may occur in the primary and secondary coolant circuits. In a PWR both circuits use light water (H_2O). In CANDU reactors only the secondary circuit coolant is light water, and the primary circuit coolant is heavy water (D_2O).

For CANDU 3 the maximum pressure of 26 psig (180 kPag) is obtained from the peak pressure of a 100% pump suction header break primary LOCA coincident with a loss-of-emergency core cooling (LOECC) according to Figure 3-2. A 10% margin was added in this case, and therefore a DBA pressure of 29 psig (200 kPag) is used for primary LOCA events in CANDU 3 containment. This is the containment DBA pressure used for abnormal design condition. This LOCA + LOECC transient also releases the largest amount of radioactivity from the primary circuit into the containment. Also, due to the loss of the emergency core cooling, the heat load on the containment would be very high, and the containment heat removal systems would be burdened heavily and the containment wall would experience the highest temperature for LOCAs. In SLB accidents the discharged steam (which becomes superheated when the secondary coolant level drops below the top end of the heat exchanger U-tubes) will cause higher containment wall temperatures than the primary LOCAs. These high wall temperatures are within CSA Standard CAN3-N287.1-M82 permitted limits and taken care of in the civil engineering stress calculations.

The radioactivity releases after this severe primary LOCA, and the radiation dosages and risks to the population are also evaluated and are within allowable limits.

With a double-ended break of a main steam line the peak pressure in the containment may reach nearly 58 psig (400 kPag) at half hour after the break, in the most severe case, if none of the air coolers are credited (Figure 3-3). At that time operator intervention can reduce the pressure in the building by remotely opening some of the Main Steam Safety Valves (MSSVs). Figure 3-3 shows the effect on pressure of opening all eight MSSVs. Even the opening of one MSSV would depressurize the building but at a slower rate. Based on this extremely conservative SLB accident, a design pressure of 61 psig (420 kPag) is used to calculate the containment stresses after a steam line break. With half of the air coolers credited, the peak pressure reaches only about 43 psig (300 kPag) in about the first minute, and the pressure slowly decreases afterwards (Figure 3-4). The depressurization rate of the containment can be increased at half an hour after the break by opening some MSSVs. The containment atmosphere temperature is predicted to be in the range of about 266 to 284°F (130-140°C) in the first half hour, without any or with half of the air coolers, and the atmospheric temperature decreases afterwards.

In the CANDU 3, after a secondary coolant circuit SLB accident, there is no significant release of radioactivity into the containment. In the CANDU 3 reactor any fuel failure during normal operation is constantly monitored and rapidly identified, and the failed fuel is replaced by new fuel using the on-power fueling machine. Leakage of the heavy water primary coolant in a steam generator into the light water secondary coolant, via cracking of a boiler tube, is also carefully monitored and safeguarded. If the leakage rate exceeds the level at which a failed boiler tube could be identified, the reactor is shut down and the failed tube is plugged. This operating requirement also evolved from an economic penalty associated with heavy water losses from the primary coolant circuit. The fueling machine is supplied with environmentally qualified backup cooling for the SLB case, so there will also be no releases from the fuel handling system. Thus in a CANDU 3 SLB accident there is practically no release of radioactivity to the environment and no risk to the public. The CANDU 3 containment building was designed to remain structurally sound after a SLB accident, to prevent damage to the contained reactor system.

The CANDU 3 containment building is designed for numerous combinations of loads. The extent and size of the rebars is calculated according to the limit state design method required by the CSA N287 series of standards (Reference 3.6-7). The load combinations (described in Chapter 2) include the applicable live and dead loads, thermal loads and abnormal loads with load factors allowing for load uncertainties. The thermal effects include not only differential temperatures between inside and outside of a concrete wall, but also the actual peak temperatures of the concrete walls.

According to CSA/CAN3-N287.3-M91 the primary LOCA-based Design Pressure is of concern in the Service Load Category pressure test load combination (with a load factor of 1.7) and in the Abnormal/ Environmental Load Category abnormal load combination (with a load factor of 1.5). The containment building is also designed for a load combination which includes the peak pressure after a secondary steam line break (with a load factor of 1.0), in combination with other loads. Thus for both the primary LOCA-based and the secondary SLB-based accidents the CANDU 3 containment structure performs with safety and structural integrity. The occurrence of a LOCA or of a SLB inside containment together with a DBE is excluded based on adequate seismic qualification and design margins of components and structures.

The design for the pressure test load combination in the Service Load Category is performed for the test pressure of 33 psig (230 kPag), which is 15% above the Design Pressure, resulting in small elastic response only. This test provides a proof of verification for the methods of the structural design of the containment building, and also provides a measure of the leak-tightness.

The low carbon (mild) steel liner is not credited in the evaluation of stresses of the containment structure, but the stresses and strains of the liner are checked for all containment load combinations. In addition, the transient LOCA thermal loads, which are seen by the liner only, are also considered in its design. The CANDU 3 liner is about 1/4-inch thick (6 to 8 mm), and is secured to the walls with studs. The liner serves as the barrier against releases of radioactivity through the containment walls, preventing the out-leakage of the containment post-LOCA atmosphere or liquid effluents from a pressurized containment via any porosity and micro-cracks in the concrete. Finally, the liner is readily reparable. The CANDU 3 steel liner satisfies the U.S. design, construction and inspection requirements.

The overall leakage rate of the containment is determined during commissioning by the initial leakage rate tests at 15% above the design basis accident pressure and later on at intervals required by maintenance. The test acceptance leakage rate is 0.5% of the containment building free volume/day during the commissioning test. In long term operation the permissible leak rate for a containment may be increased before repair to the steel liner becomes necessary. The acceptability of a higher limit of leakage rate is determined from evaluations of doses and risks to the public, after credible accidents with a proposed increased leakage rate. The leakage rate is tested periodically, and if the test acceptance leakage specifications are exceeded the containment liner is repaired and other leakage preventing measures are taken. While the plant is operating the building air pressure, temperature, etc. are monitored to provide timely indication of any gross breach of containment.

In a CANDU 3 reactor containment the containment volume to reactor power ratio is large, and this feature helps to prevent large, fast pressure build-up. The carbon steel liner also serves to condense steam on the containment wall. An initial pressure increase after a primary LOCA can be turned over by condensation on building walls and internals, and in the longer term the building pressure is brought down by air coolers. After a SLB the building pressure can be brought down by the air coolers or if air coolers are not available by opening some Main Steam Safety Valves (MSSV).

3.5 CONCLUSIONS

The containment structure is designed to survive the depressurization of the primary coolant system without structural damage and without unacceptable release of radioactivity to the environment. Most of the design reflects established CANDU practices and operational experience accumulated over many years of service with single unit CANDU reactors which utilized several aspects of the CANDU 3 containment design. The containment design is in agreement with . e Atomic Energy Control Board (AECB) regulatory policy document R-7 of 1991 which sets forth the nuclear reactor containment requirements, and is in agreement with the applicable codes and standards in Canada, in particular the CSA Standards for CANDU containment structure.

The CANDU 3 containment structure and its liner are designed for overpressure due to both primary LOCAs and secondary side steam line breaks satisfying the requirements of the applicable Codes and Standards. The containment integrity and its leak-tightness are assured after all probable primary and secondary coolant circuit break accidents. A coincident LOCA + LOECC is a DBA for the containment; similarly a SLB is a DBA. A CANDU 3 containment, designed with a load factor of 1.0 for the steam line break case, does not result in structural damage or uncontrolled release of radioactivity.

Therefore, these requirements provide an adequate level of safety.

3.6 REFERENCES

3.6-1 U.S. Nuclear Regulatory Commission, "General Design Criteria for Nuclear Power Plants", Code of Federal Regulations, Title 10 – Energy, Chapter I, Appendix A, published by (USA) Office of the Federal Register, National Archives and Records Administration, Washington, D.C., revision of 1991 January 1.

3.6-2 ASME Boiler and Pressure Vessel Committee (Subcommittee on Nuclear Power) and ACI-ASME Joint Technical Committee, 1989 ASME Boiler and Pressure Vessel Code, American National Standard, ACI Standard 359-89, Section III, Rules for Construction of Nuclear Power Plant Components, Division 2, Code for Concrete Reactor Vessels and Containment, 1991 January 1 edition.

3.6-3 U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", LWR Edition, NUREG-0800, three volumes, 1987 June.

3.6-4 U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiolo... ...ences of a Loss of Coolant Accident for Pressurized Water Reactors", U... ...egulatory Guide, RG-1.4, Revision 2, 1974 June.

3.6-5 U.S. ... Commission, "Materials, Construction, and Testing of Concrete C... NRC Regulatory Guide, RG-1.136, Revision 2, 1981 June.

3.6-6 Atom... Board (Canada), "Requirements for Containment Systems for CAND... ...wer Plants", AECB Regulatory Policy Statement, Regulatory Document R-7, Ottawa, Ontario, Canada, 1991 February 21.

3.6-7 CSA N287 series Standards:

- Canadian Standards Association, "General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN3-N287.1-M82, 26 pp., Toronto, Ontario, Canada, 1982 May.
- Canadian Standards Association, "Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN/CSA-N287.2-M91, 47 pp., Toronto, Ontario, Canada, Edition of 1991 December.
- Canadian Standards Association, "Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN3-N287.3-M91, Toronto, Ontario, Canada, Draft revision, 1991 June.
- Canadian Standards Association, "Construction, Fabrication, and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN3-N2... M83, Toronto, Ontario, Canada, 1983.
- Canadian Standards Association, "Testing and Examination Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN3-N287.5-M81, Toronto, Ontario, Canada, 1981.

- Canadian Standards Association, "Pre-Operational Proof and Leakage Rate Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN3-N287.6-M80, 13 pp., Toronto, Ontario, Canada, 2nd edition, 1980 April.
- Canadian Standards Association, "In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants", National Standard of Canada, CAN3-N287.7-M80, 13 pp., Toronto, Ontario, Canada, 2nd edition, 1980 April.

3.6-8 CSA N285 series standards:

- Canadian Standards Association, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants", National Standard of Canada, CAN/CSA-N285.0-M91, 85 pp., Toronto, Ontario, Canada, Edition of 1991 November.
- Canadian Standards Association, "Requirements for Containment System Components in CANDU Nuclear Power Plants", National Standard of Canada, CAN/CSA-285.3-88, 15 pp., Edition of 1988 February.

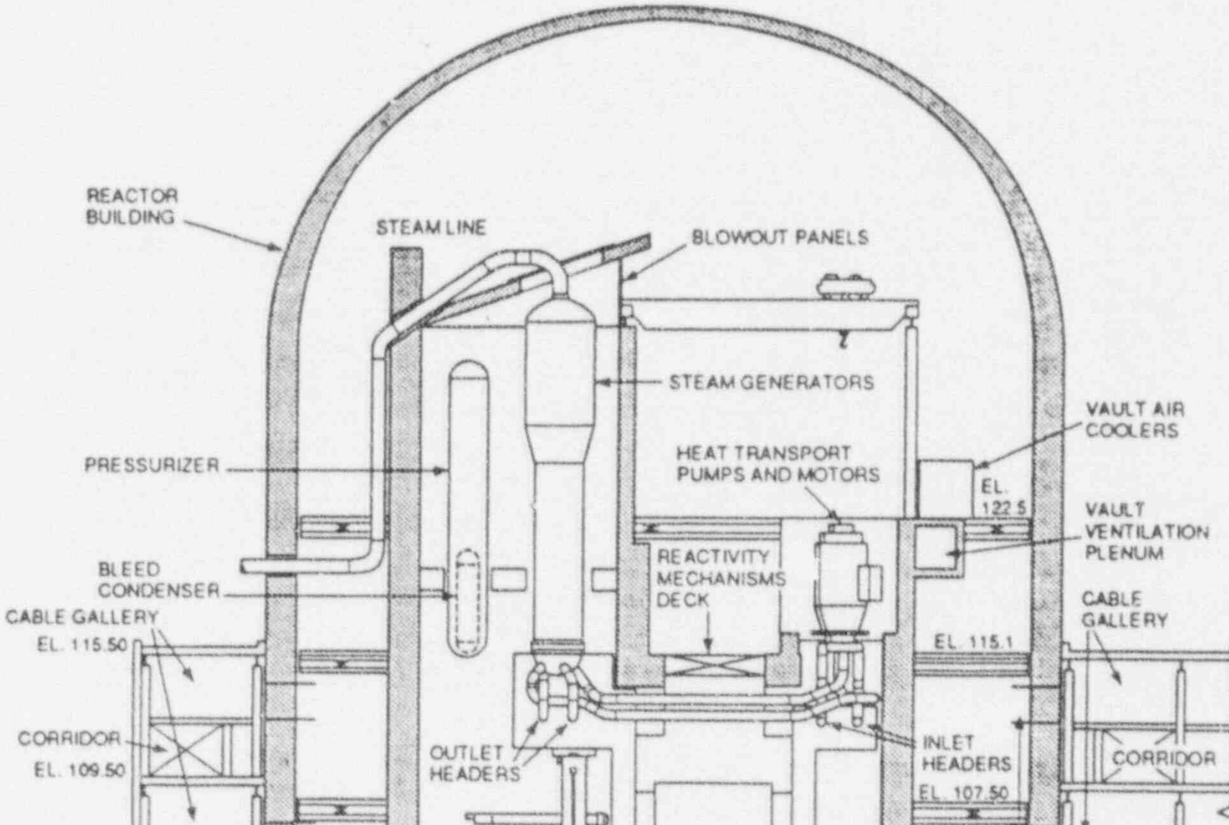

Figure 3-1

CANDU REACTOR BUILDING-SECTION

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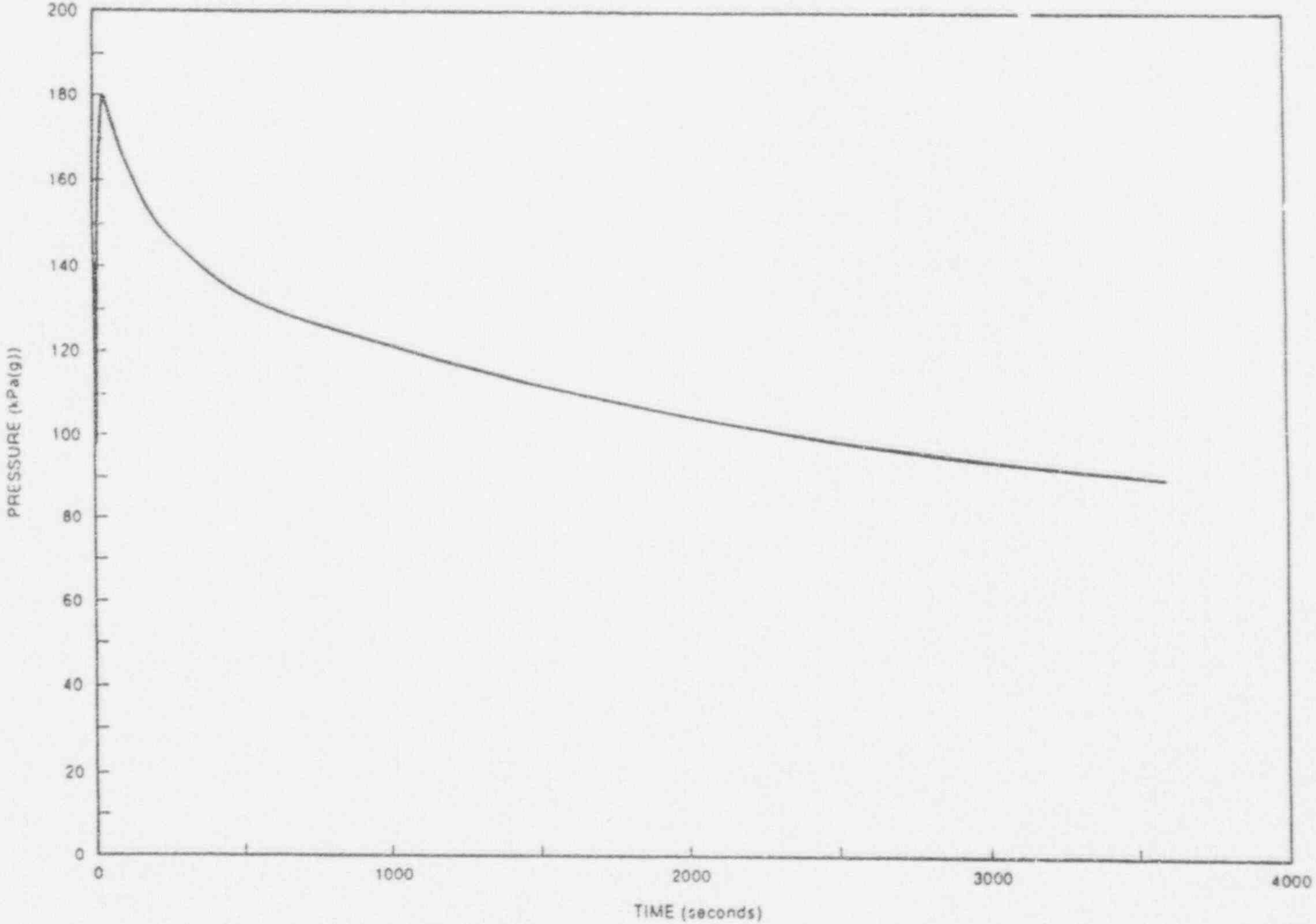

Figure 3-2

CONTAINMENT PRESSURE AFTER 100 PERCENT PUMP SUCTION PIPE BREAK COINCIDENT WITH LOSS OF EMERGENCY CORE COOLING SYSTEM

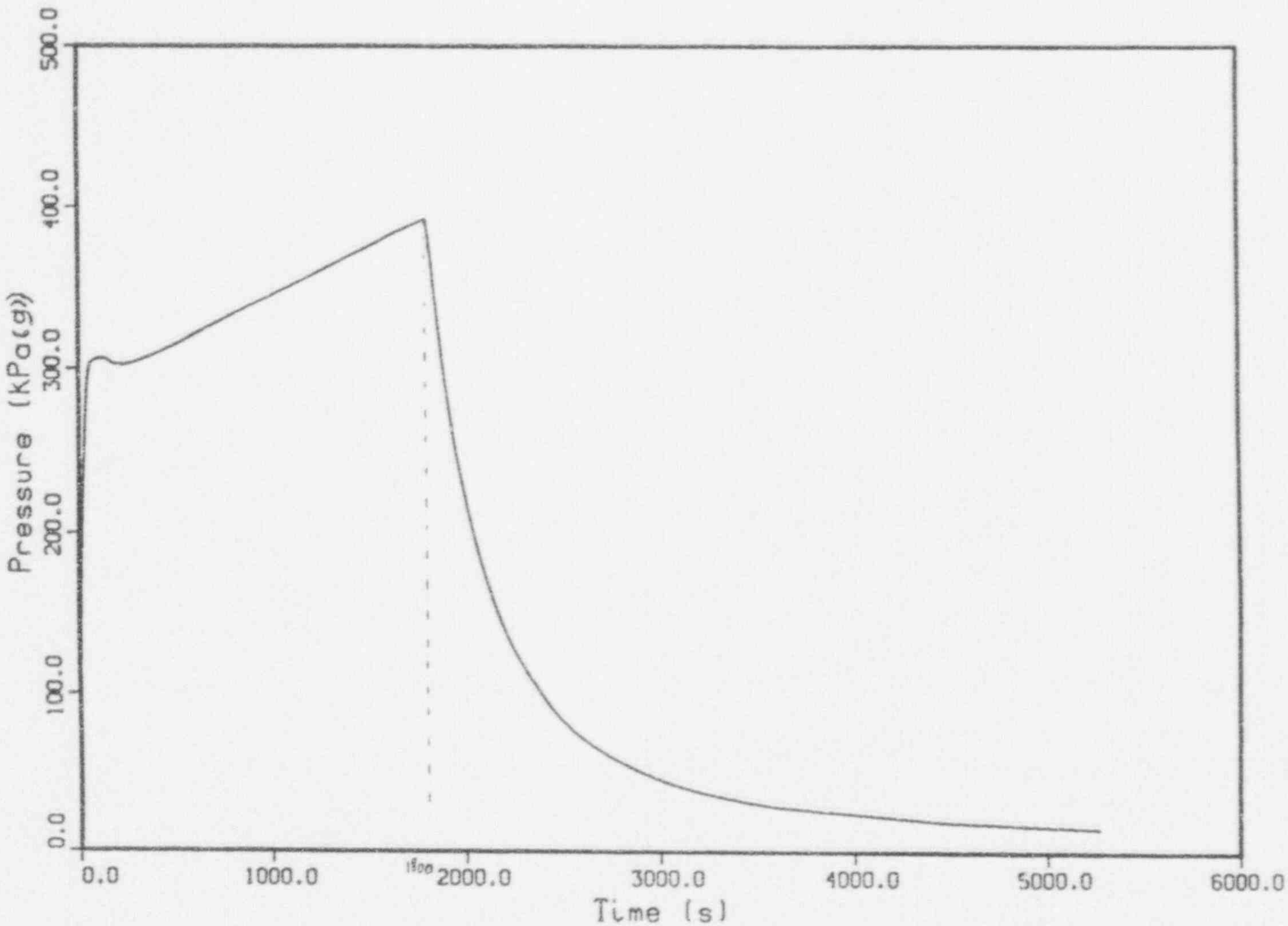

Figure 3-3

CONTAINMENT PRESSURE AFTER 100% STEAM LINE BREAK WITH NO AIR COOLERS OPERATING AND ALL MSSV-S OPENED AT HALF AN HOUR

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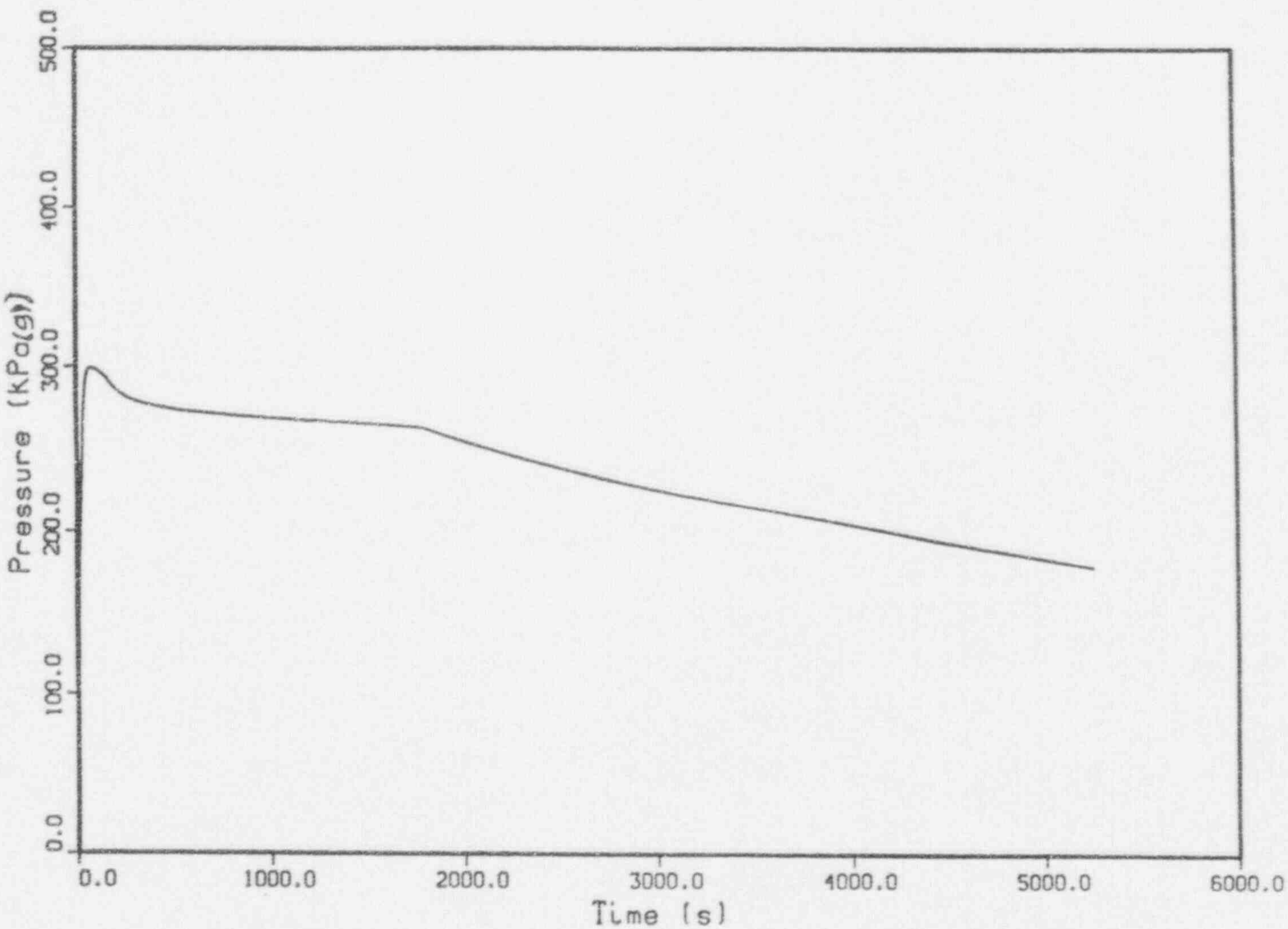

Figure 3-4

CONTAINMENT PRESSURE AFTER 100% STEAM LINE BREAK WITH HALF OF THE AIR COOLERS OPERATING

CHAPTER 4

EQUIVALENT SAFETY ISSUES

CONTAINMENT ISOLATION SYSTEM

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4.1 INTRODUCTION

The CANDU 3 containment isolation system is a subsystem of the containment system which is identified as a special safety system in CANDU design concept.

The regulatory requirements for the design of the CANDU 3 containment isolation system are described in AECB Regulatory Document R-7 "Requirements for Containment System for CANDU Nuclear Power Plants" (Reference 4.6-1). The CANDU 3 containment system design defines an envelope around the reactor core, the heat transport system, the moderator system, and auxiliary systems, and forms a barrier to limit the release of radioactive material to outside environment during normal reactor operation and accident conditions including LOCA.

This report documents an evaluation of the CANDU 3 containment isolation system design based on U.S. Nuclear Regulatory Commission regulations and guidance. The applicable U.S. licensing requirements and guidance for Light Water Reactor (LWR) plants for the containment isolation system are outlined and the CANDU 3 approach for the containment isolation system design is provided. Similarity and differences between U.S. design and CANDU 3 design and their impact on certification of the CANDU 3 with U.S. Nuclear Regulatory Commission are identified and evaluated. The intent of this chapter is to show that, in spite of differences, the CANDU 3 containment isolation system has a level of safety equivalent to the requirements of the U.S. NRC.

4.2 NRC REQUIREMENTS AND GUIDANCE

The design objective of the containment isolation system for Light Water Reactor (LWR) plants is to prevent the escape of radionuclides through the containment boundary following accidents. Major U.S. NRC requirements on containment isolation are attached in Appendix 4A for convenient reference.

U.S. NRC Standard Review Plan (SRP) 6.2.4, "Containment Isolation System" (Reference 4.6-2), provides the guidance and standards for the containment isolation. The following NRC Regulations and Regulatory Guides are applicable to the Containment Isolation System:

- 10 CFR Part 50.34(f)(2)(xiv), "Containment Isolation" (Reference 4.6-3, Appendix 4A.1),
- 10 CFR Part 50, Appendix A (Reference 4.6-7 Appendix 4A.2),
  - "General Design Criterion 54 - Piping Systems Penetrating Containment"
  - "General Design Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment"
  - "General Design Criterion 56 - Primary Containment Isolation"
  - "General Design Criterion 57 - Closed System Isolation Valves".
- Regulatory Guide 1.26, "Quality Group Classifications and Standards for water-, steam-, and radioactive-waste containing components of Nuclear Power Plants" (Reference 4.6-5).
- Regulatory Guide 1.29, "Seismic Design Classification" (Reference 4.6-6).
- Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment" (Reference 4.6-4).

10 CFR Part 50.34(f)(2)(xiv) (See Appendix 4A.1) requires containment isolation using two valves in series. The containment isolation system should include automatic closing on a high radiation signal for all systems that provide a path to the environment. It should also utilize a containment setpoint pressure for initiating containment isolation, as low as is compatible with normal operation.

GDC 54, "Piping System Penetrating Containment" (See Appendix 4A.2) is concerned mainly with leak detection, isolation and containment capabilities of the piping systems penetrating primary reactor containment.

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment" (See Appendix 4A.2) deals with the isolation valves required for each line that is part of the reactor coolant pressure boundary and that penetrates the containment.

GDC 56, "Primary Containment Isolation" (See Appendix 4A.2) deals with the isolation valves for lines that connect directly to the containment atmosphere and penetrate the primary reactor containment. For these lines, two containment isolation valves in series, one inside and one outside containment, should be used.

GDC 57, "Closed System Isolation Valves" (See Appendix 4A.2) is concerned with isolation valves for each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere.

The above NRC General Design Criteria require that each containment penetration be provided with a redundant barrier so that in the event that a single failure is postulated and one barrier does not perform as intended, the containment integrity is maintained. NRC Standard Review Plan 6.2.4 (Reference 4.6-3) and Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems" (Reference 4.6-8) provide acceptable alternative arrangements to the explicit arrangements given in GDC-54, 55, 56, and 57.

Containment isolation provisions for lines in engineered safety feature systems normally consist of two isolation valves in series. As per SRP 6.2.4 Section II.6.e (Reference 4.6-3), a single isolation valve will be acceptable if it can be shown that the system reliability is greater with only one isolation valve in the line, the system is closed outside containment, and a single active failure can be accommodated with only one isolation valve in the line. The closed system outside containment should be protected from missiles, designed to seismic Category 1 standards, classified Safety Class 2, and should have a design temperature and pressure rating at least equal to that for the containment. The closed system outside containment should be leak tested. For this type of valve arrangement the valve is located outside the containment, and the piping between the containment and valve should be enclosed in a leak tight or controlled leakage housing. In lieu of a housing the piping and valve design should conform to the requirements of SRP Section 3.6.2.

The use of a closed system inside containment as one of the isolation barriers is acceptable if the design of the closed system does not communicate with either the reactor coolant system or the containment atmosphere and the system is designed as Class 2, Seismic Category 1, capable to withstand the containment design temperature and external pressure from containment acceptance test, the system is protected against missiles and pipe whip, and is designed to withstand LOCA transient and environment.

As per SRP 6.2.4, Section II.6.h, the systems penetrating the containment are to be classified as either essential or non-essential to accident management. Essential systems may include remote-manual containment isolation valves, but provisions should be made to detect possible leakage from the lines outside containment. It is also required that non-essential systems be automatically isolated by the containment isolation signal.

According to SRP 6.2.4, the containment isolation system is required:

- to be designed as seismic category 1 and classified as Quality Group B in accordance with the Regulatory Guides 1.29 (Reference 4.6-6) and 1.26 (Reference 4.6-5) respectively, unless the service function dictates that Quality Group A be applied,
- to be designed to provide the valve operator to be in the "safe" position following loss of power; the valve should also be provided with position indication in the control room,
- to be designed to meet the requirements of Regulatory Guide 1.11 (Reference 4.6-4) for the instrument lines penetrating the containment; this basically requires that a single isolation valve capable of automatic operation, located outside the containment be employed. In addition, instrument lines that are closed both inside and outside containment, are designed to withstand the pressure and temperature conditions following a LOCA, and are designed to withstand dynamic effects, are acceptable without isolation valves.

4.3 CANDU 3 APPROACH

The design objective of the CANDU 3 containment isolation system is to prevent the escape of radionuclides following accidents which may lead to release of radioactivity from the reactor.

The containment isolation system is required to seal the reactor building containment envelope automatically when high radioactivity and/or abnormal pressure rise occurs in the reactor building. The CANDU 3 containment envelope is shown in Figure 4-1. To detect any leakage, the CANDU 3 containment is equipped with the Gross Leakage Monitoring (GLM) system. Its function is to provide the station operator with information that will clearly and reliably indicate any significant breach of containment.

The CANDU 3 containment isolation system is required to comply with AECB Regulatory Document R-7 (Reference 4.6-1). Excerpts are attached as Appendix 4B. Piping systems penetrating containment shall be provided with isolation devices having redundancy, reliability, and performance capability which reflect their importance to safety. The automatic isolation devices are designed to fail in the safe position. Manual isolation valves are locked closed or continuously monitored to show that they are in the closed position.

All containment isolation actions required in the first 15 minutes of an accident are initiated automatically. The CANDU 3 design logic of initiating the containment isolation is shown in Figure 4-2. To prevent inadvertent valve opening after containment isolation, the system is locked in by control logic until all alarms are cleared and the system reset.

The status of the containment isolation valves is monitored continuously and all necessary automatic actions are initiated from the main control room or the secondary control area.

The valves and piping that form the containment isolation system are seismically qualified to the DBE.

In the CANDU 3 design, the following types of piping systems penetrating containment are provided with isolation as specified below:

a. <u>Systems Connected to the Containment Atmosphere</u>

Systems connected to the containment atmosphere are provided with two isolation barriers, which should meet the following requirements:

1. Two automatic Class 2 isolation valves, one of which may be a check valve on the inside, for lines that may be open to the containment atmosphere.
2. Two closed Class 2 isolation valves, for lines normally closed to the containment atmosphere.
3. A single closed Class 2 isolation valve for lines of 50-mm (2-inch) diameter or less, normally closed to the containment atmosphere. This is in addition to a closed system outside containment (the closed system is considered as one of the isolation barriers).

Normally, the two isolation valves and their controls are located outside the containment in order to facilitate their maintenance and to avoid exposing them to the severe environmental conditions that exist inside the containment.

The line from and including the isolation valve closest to the containment, up to and including the outer isolation valve is part of the containment envelope and is constructed to the requirements of ASME Code Section III Class 2.

b. HTS Auxiliary Systems Penetrating Containment

1. Lines that are connected to the Heat Transport System (HTS) and penetrate the containment structure are provided with two Class 1 isolation valves in series. The normal arrangement is one inside and one outside containment. Two valves in series inside or outside containment structure to provide equivalent barrier are acceptable.

2. For lines connected to the HTS that are open during normal plant operation, one of the two isolating valves shall be either an automatically closing (e.g., check valve), or a powered isolation valve operable from the control room. Note that the provision of two check valves in series is not considered as an acceptable barrier. Manual (i.e., not powered) isolating valves must not be used inside the containment structure. Check valves used as an isolation barrier must be located inside the containment structure.

3. For small lines 25-mm (one-inch) diameter or less, a single closed Class 1 isolation valve inside containment is used, provided the line is connected outside containment to a closed system.

The line from the HTS up to and including the outer isolation valve is constructed to the requirements of Class 1 per CSA Standard CAN/CSA-N285.0-M91 (Reference 4.6-9) which essentially implies ASME Code Section III, Subsection NB. If the outer isolation valve is inside the containment, then the line from the outer isolation valve to containment penetration is constructed at least to the requirements of Class 2 Code.

c. All Other Systems Penetrating Containment

1. For closed systems inside or outside containment which form part of the containment envelope, meet the requirements of CSA Standard CAN/CSA-N285.0-M91 (Reference 4.6-9) Class 2 (this implies ASME Section III Class 2) and can be continuously monitored for leaks, no further isolation is used.

2. All other closed systems penetrating containment, which meet the requirements of CSA standard CAN/CSA-N285.0-M91 (Reference 4.6-9) Class 2 but can not be continuously monitored for leaks, are provided with a single isolating valve on each line penetrating containment, located outside containment as close as practicable to the containment structure.

3. Those closed systems inside containment, which have a design pressure at or greater than 72.5 psig (0.5 MPa(g)), and are continuously operated at or above containment design pressure at all points in the system, and which can be monitored for leaks, are provided with a single manual isolation valve located outside the reactor building on each line penetrating containment. These systems are constructed to the requirements of CSA standard CAN/CSA-N285.0-M91 Class 6 (Reference 4.6-9), which implies non-nuclear standard.

d. Instrument Lines

1. For small bore ductile tubing (3/4-in (19 mm) or less), crimping of the tube is a means of providing containment isolation when the tubing designated for crimping is in an accessible area and is readily identifiable.
2. All other instrument lines penetrating the containment are provided with isolation valves as described above suitable for each appropriate application.

4.4 EVALUATION

The information in Sections 4.2 and 4.3 show that both CANDU 3 design and U.S. design prevent the escape of radionuclides that may result from postulated accidents. Both designs require that the containment isolation function be maintained assuming any single active failure in the containment isolation provisions.

The practice for LWRs, sanctioned by the NRC in SRP 6.2.4, is to have the containment isolation system design incorporate all requirements for safety-grade systems, including redundancy of isolation devices, nuclear code (ASME Code, Section III, Class 2) requirements, seismic qualification, harsh environment qualification, leak detection, and containment isolation capabilities having redundant and reliable performance capabilities. In accordance with 10 CFR Part 50, NRC requires that, for lines penetrating the containment which are either part of the reactor coolant pressure boundary (RCPB) or connect directly to the containment atmosphere, two containment isolation valves in series, one inside and one outside the containment, be used. For lines that are neither part of RCPB nor connected directly to the containment atmosphere, one isolation valve outside the containment is required. The locked closed isolation valve and the automatic isolation valve are the acceptable type of isolation devices; a simple check valve is not normally an acceptable automatic isolation valve located outside the containment.

CANDU 3 design employs two automatic isolation valves in series for lines normally open to the containment atmosphere, and two closed isolation valves in series for lines normally closed to the containment atmosphere. Normally the two isolation valves are located outside the containment in order to facilitate their maintenance and to avoid exposing them to the severe environmental conditions that exist inside the containment. U.S. NRC regulations and guidance, however, require that two valves in series, one inside and the other outside the containment be employed. For certain applications, if it is not practical to locate a valve inside containment then as per SRP 6.2.4 Section II.6.d, both valves may be located outside the containment. For systems connected to the HTS and penetrating containment, CANDU 3 design normally requires one valve inside and the other valve outside the containment structure and hence complies with the U.S. NRC requirements. Similar to U.S. NRC regulations, the CANDU 3 design does not accept a check valve as an isolation valve outside the containment. A check valve is only acceptable if it is located inside the containment and only for lines connected to the HTS.

CANDU employs a single closed isolation valve for lines of 2-inch diameter or less that are connected to the containment atmosphere when the line is normally closed to the containment atmosphere and connected to a closed system outside containment. This arrangement can accommodate a single active failure and the possibility of passive failure is very low because the isolation valve is as close to the containment as possible. Therefore, the requirements of GDC 56 (Reference 4.6-7) are met.

For small lines of 1-inch diameter or less, a single closed Class 1 isolation valve inside containment provides adequate barrier, provided the line is connected outside containment to a closed system. This arrangement can accommodate a single active or passive failure. Therefore, safety objectives similar to the ones required by the U.S. NRC are met.

In the CANDU 3 design, certain closed systems inside containment, which have design pressure at or greater than 72.5 psig (0.5 MPa(g)) and operating pressure at or greater than containment design pressure, and which can be monitored for leaks (such as the group 1 Recirculated Cooling Water system) have a single manual isolation valve located outside the reactor building on each line penetrating containment. These systems are constructed to Class 6 (non-nuclear) requirements and meet an equivalent level of safety compared to the U.S. NRC requirements, because:

- there is negligible radioactivity in the systems, and these can be continuously monitored for leaks; the monitoring of leaks by different methods provides adequate reliability and assurance that systems will function as intended;
- a single active or passive failure is accommodated;
- these systems do not communicate with either the reactor coolant system or the containment atmosphere;
- the systems design pressure and temperature ratings are more than that for the containment.

In the CANDU 3 design, instrument lines penetrating the containment meet the guidelines of Regulatory Guide 1.11 and Standard Review Plan 6.2.4.

The CANDU 3 containment isolation system has performance capabilities which reflect the importance to safety of isolating the various types of piping systems penetrating containment. The containment isolation system is designed as a safety system, it is seismically and environmentally qualified, the valves are periodically tested to ensure operability and to check that the valve leakage is within acceptable limits. This is in compliance with GDC 54 (Reference 4.6-7).

CANDU 3 design does not classify the systems penetrating the containment into essential and non-essential systems, as in SRP 6.2.4, Section II.6.h (Reference 4.6-3). However, equivalence of safety is met by ensuring that all systems, which penetrate the containment and are connected to the containment atmosphere by a normally open line, are automatically isolated on the containment isolation signal. The systems that are needed to mitigate accidents are individually identified and are therefore not automatically isolated by the containment isolation signal.

It is important to consider the differences that exist between CANDU reactors and the LWRs as they relate to containment isolation:

- Pressure tube rupture in CANDU could lead to "in-core LOCA". In this type of accident, the reactor coolant would be mixed with the moderator, which could lead to a containment isolation problem if the moderator leaves the containment boundary. But the moderator system and its auxiliaries are within containment, except for two lines connected to the D_2O Supply System and the Upgrading System. Both the D_2O Supply System and the Upgrading System are isolated from the containment by two pneumatically operated valves in series. This meets an equivalent level of safety compared to the U.S. NRC requirements.

- In contrast to LWRs, CANDU reactors are refueled on-power (Reference 4.6-10). Three fueling-related ports pass through the CANDU 3 containment boundary: a new fuel port, an irradiated fuel port, and an ancillary port/rehearsal facility which is used for a variety of fueling and fuel channel maintenance operations. During normal operation, the ancillary port is closed to containment and connects externally only to a closed circulating loop.

  The main components of the new fuel transfer mechanism include a totally enclosed magazine, a new fuel port, and an adapter which connects the magazine to the new fuel port. Double containment valves are located at the front and rear of the magazine to allow loading and unloading of the magazine during reactor operation.

  The irradiated fuel transfer mechanism also includes a magazine which is enclosed within the irradiated fuel lock with double containment valves at each end. This magazine contains light water at a level just below the inlet port and bundles rotate into this water during transfer. On completion of loading the magazine, the containment valves at the fueling machine end of the lock are closed, and the lock is filled with irradiated fuel storage bay water.

  Therefore, during on-power refueling, CANDU 3 containment isolation is maintained and meets an equivalent level of safety compared to the U.S. NRC requirements.

4.5 CONCLUSION

The design of the CANDU 3 containment isolation system was reviewed against the U.S. regulations and guidance.

Based on the evaluation of the isolation system, it is demonstrated that in spite of some differences in containment isolation arrangements, CANDU 3 design satisfies the safety objectives of the U.S. NRC requirements. Each CANDU 3 containment penetration is provided with redundant barriers so that in the event that a single failure is postulated and one barrier does not perform as intended, the containment integrity is maintained.

Therefore, the CANDU 3 containment isolation system design provides an equivalent level of safety to that of the U.S. NRC requirements and guidance.

4.6 REFERENCES

4.6-1 AECB Regulatory Document R-7 "Requirements for Containment Systems for CANDU Nuclear Power Plants", Appendix 1.

4.6-2 U.S. NRC, Standard Review Plan, Section 6.2.4, Containment Isolation System.

4.6-3 U.S. NRC, 10 CFR 50, Section 50.34(f)(2)(xiv), "Containment Isolation Using 2 Valves in Series".

4.6-4 U.S. NRC, Regulatory Guide 1.11 (Safety Guide 11), "Instrument Lines Penetrating Primary Reactor Containment".

4.6-5 U.S. NRC, Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plant".

4.6-6 U.S. NRC, Regulatory Guide 1.29, "Seismic Design Classification".

4.6-7 U.S. NRC, 10 CFR 50, Appendix A, General Design Criteria

GDC 54 – Piping Systems Penetrating Containment

GDC 55 – Reactor Coolant Pressure Boundary Penetrating Containment

GDC 56 – Primary Containment Isolation

GDC 57 – Closed System Isolation Valves

4.6-8 U.S. NRC, Regulatory Guide 1.141, "Containment Isolation Provisions for Fluid Systems".

4.6-9 CSA Standard CAN/CSA-N285.0-M91, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants".

4.6-10 R.K. Nakagawa, "The Technology of CANDU On-Power Fueling", TTR-305, 1991 January.

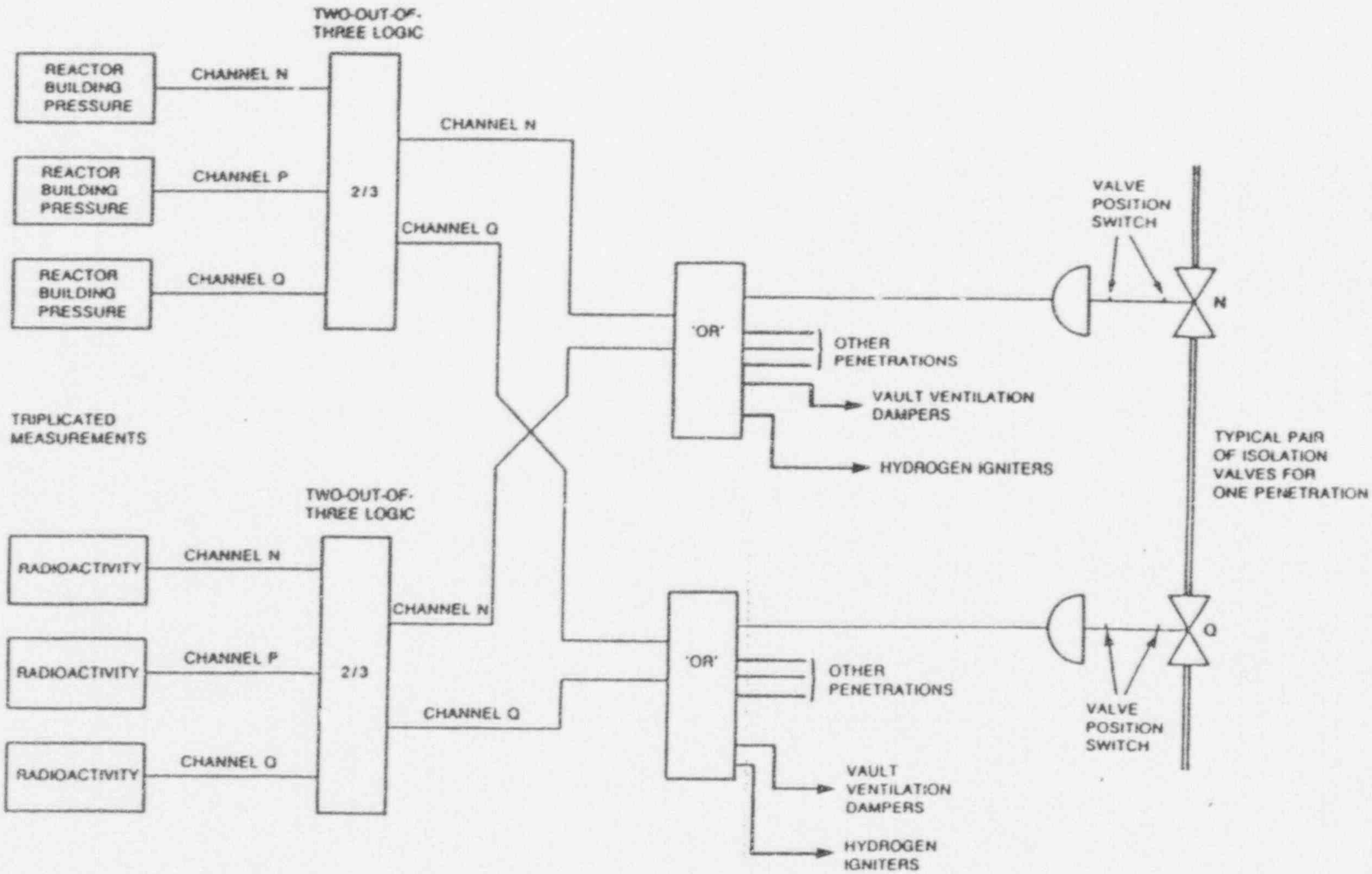

Figure 4-2: Block Diagram of Containment Initiation Control Loop

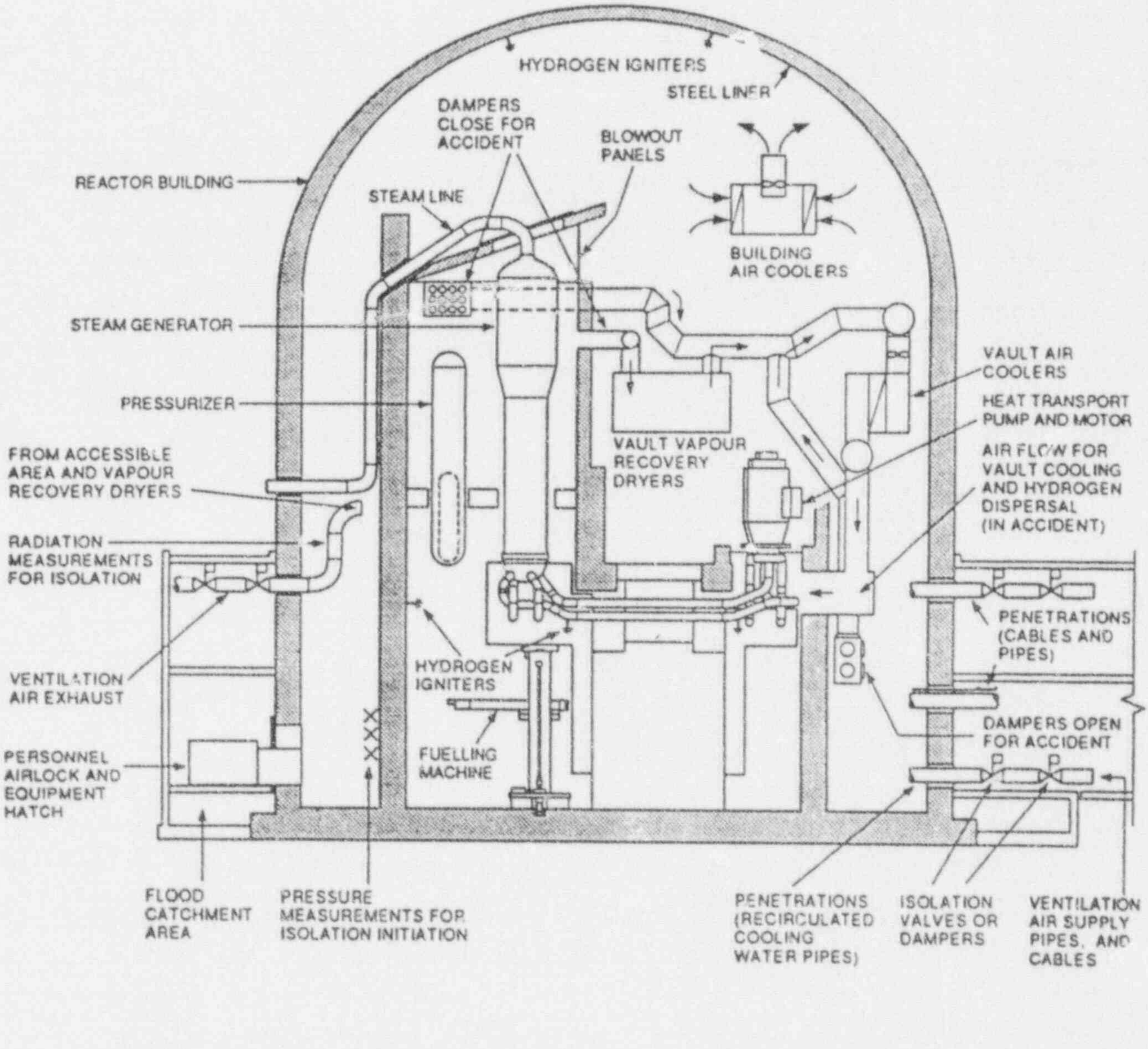

Figure 4-1: CANDU 3 Containment System

APPENDIX 4A
U.S. NRC REQUIREMENTS

4A.1 10 CFR PART 50.34 (f)(2)(xiv)

Provide containment isolation systems that:

(A) *Ensure all non-essential systems are isolated automatically by the containment isolation system,*

(B) *For each non-essential penetration (except instrument lines) have two isolation barriers in series,*

(C) *Do not result in reopening of the containment isolation valves on resetting of the isolation signal,*

(D) *Utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation,*

(E) *Include automatic closing on a high radiation signal for all systems that provide a path to the environs.*

4A.2 10 CFR PART 50 APPENDIX A: GENERAL DESIGN CRITERIA

Criterion 54 – Piping systems penetrating containment. Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping system shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Criterion 55 – Reactor coolant pressure boundary penetrating containment. Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) *One locked closed isolation valve inside and one locked closed isolation valve outside containment; or*

(2) *One automatic isolation valve inside and one locked closed isolation valve outside containment; or*

(3) *One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or*

(4) *One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.*

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.1

Criterion 56 – Primary containment isolation. Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Criterion 57 – Closed system isolation valves. Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

APPENDIX 4B
CANADIAN REQUIREMENTS

4B.1 EXCERPT FROM REGULATORY DOCUMENT R-7 "REQUIREMENTS FOR CONTAINMENT SYSTEM FOR CANDU NUCLEAR POWER PLANTS"

APPENDIX A:

REQUIREMENTS FOR METAL EXTENSIONS OF THE CONTAINMENT ENVELOPE

1. CODE REQUIREMENTS

Systems or portions of systems which form part of the containment envelope shall be constructed to the requirements of the ASME Boiler and Pressure Vessel Code, *Section III, Division 1, Subsection NC (Class 2 components) or Subsection NE (Class MC components) except for:*

a. *those systems whose process requirements are Class 1 or 2 in accordance with CAN3-N285.0;*

b. *those closed systems inside the containment structure which have a design pressure greater than 0.5 MPa(g) and are continuously operated at or above the positive design pressure of the containment at all points in the system, and which can be monitored for leaks. Such systems may be constructed to the process systems requirements, but they shall be constructed to not less than the non-nuclear requirements of CSA B51.*

Closed systems inside the containment structure which do not meet the requirements in paragraphs (a) and (b) may be built to the requirements of Class 3 if it can be shown to the satisfaction of the AECB that, due to smallness of size or other factors, the proposed design provides an adequate barrier.

2. ISOLATION

Piping systems shall be provided with isolation devices having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating the various types of piping systems penetrating containment. Where isolation in a piping system is provided by valves, provisions shall be made to test the valve operability periodically, to check that the valve leakage is within acceptable limits and to allow maintenance of the valve without causing a breach of the containment envelope. In order for a manual isolation valve to be considered closed, it shall be either locked closed or continuously monitored to show that the valve is in the closed position.

The various types of piping systems penetrating containment shall be provided with the following isolation unless it can be shown that, for a specific type of line, other isolation provisions would be acceptable.

2.1 PRIMARY HEAT TRANSPORT AUXILIARY SYSTEMS PENETRATING CONTAINMENT

Each line that is connected to the primary heat transport system pressure boundary and that penetrates the containment structure shall be provided with two isolation valves in series. The valves shall normally be arranged with one inside and one outside the containment structure. If it can be shown that two valves inside the containment structure or two valves outside the containment structure can provide an equivalent barrier in certain applications, then this may also be an acceptable arrangement.

A check valve may be used as one of the isolation barriers but it shall be located inside the containment structure. Two check valves in series are not considered an acceptable barrier.

Where the valves provide isolation of the heat transport system during normal operation of the station, then both valves shall normally be in the closed position.

Systems directly connected to the heat transport system and which may be open during normal operation of the station shall also be provided with the same isolation as the normally closed system except that manual isolating valves inside the containment structure shall not be used. At least one of the two isolation valves shall be either an automatic isolation valve (for instance, a check valve) or a powered isolation valve operable from the control room.

For small lines of 25 mm in nominal diameter or less, a single closed isolation valve inside containment may be used provide the line is connected to a closed system outside containment.

The line up to an including the second isolation valve, or the first valve in the case of small lines 25 mm in nominal diameter or less shall be constructed to the requirements of Class 1 in accordance with CAN3-N285.0

2.2 SYSTEMS CONNECTED TO CONTAINMENT ATMOSPHERE

Each line that connects directly to the containment atmosphere, that penetrates the containment structure, and that is not part of a closed system, shall be provided with two isolation barriers as follows:

a. *two automatic isolation valves in series for those lines which may be open to the containment atmosphere;*

b. *two closed isolation valves in series for those lines that are normally closed to the containment atmosphere;*

c. *one closed isolation valve for lines of 50 mm in nominal diameter or less, which are normally closed to the containment atmosphere and connected to an easily defined closed system outside containment.*

The line up to and including the second valve, or the first valve in the case of paragraph (c), shall be part of the containment envelope and shall be constructed to the requirements of ASME Code, *(Section III, Class 2).*

2.3 CLOSED SYSTEMS

Closed systems inside or outside the containment structure which form part of the containment envelope and which meet the requirements of Class 2 and can be continuously monitored for leaks need no further isolation. All other closed systems shall be provided with a single isolation valve on each line penetrating containment. The valves shall be located outside containment as close as practicable to the containment structure. Valves required for process purposes may be used as the isolation valves for these closed loops.

2.4 SMALL LINES

For ductile piping of small bore, crimping of the pipe is a possible means of providing an isolation barrier instead of a valve. For this to be acceptable, the details of its application shall be submitted for approval in each case of its proposed used. In particular, the method of crimping, the location of the part to be crimped and the method of identifying the failed line shall be shown to be satisfactory. In the case of primary heat transport system instrument lines, the following extra conditions are required:

a. *space must be available for crimping the tubes where they penetrate through the containment structure;*

b. *the quality of the lines is to be as good as the rest of the primary heat transport system;*

c. *the relevant release limits must be shown not to be exceeded during the period in which the reactor is shut down consequent to the failure, and the crimping is executed; and*

d. *any outflow form the breaks can be filtered before release to the atmosphere to control the escape of fission products.*

CHAPTER 5

EQUIVALENT SAFETY ISSUES

CONTAINMENT HEAT REMOVAL

by

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5.1 INTRODUCTION

The containment is a barrier to contain activity releases which could be generated as a consequence of an accident sequence. Related design objectives of the systems which support the containment function are for containment to remain intact and to be able to reduce the pressure and temperature in containment following a pipe break. The containment heat removal system is provided to remove heat generated during normal plant operation and to remove excess heat and steam which would appear in the event of a Loss of Coolant Accident (LOCA) or secondary side pipe failure. Both U.S. and Canadian design practices require that containment heat removal capability be met for all plant conditions including those involving a LOCA.

In light water reactors (LWRs), the types of systems provided to remove heat from the containment include fan cooler systems, spray systems, and residual heat removal systems. The U.S. NRC has developed specific requirements and guidance for the design of these systems.

In the CANDU 3 reactor, the containment heat removal capability is provided by local air coolers designed and sized to handle both the normal operating heat loads and heat loads under accident conditions. The low pressure, recirculation portion of the emergency core cooling (ECC) system also removes decay heat from the core and reduces energy transfer to the containment atmosphere under post-accident conditions.

This report deals with the evaluation of containment heat removal system which is required for containment heat removal under post-accident conditions. Major differences between U.S. NRC requirements and the CANDU 3 design for containment heat removal system are identified and evaluated. Despite these differences, it is demonstrated that the CANDU 3 design provides an adequate level of safety compared to the U.S. NRC requirements.

5.2 NRC REQUIREMENTS AND GUIDANCE

Light water reactor (LWR) plants include a containment heat removal system. The types of systems provided to remove heat from the containment include fan cooler systems, spray systems, and residual heat removal systems. These systems remove heat from the containment atmosphere and the containment sump water, or the water in the containment wet well.

The fundamental requirement for containment heat removal is specified in the Code of Federal Regulations (Reference 5.6-1). Additional guidance is provided via the Standard Review Plan (SRP) (Reference 5.6-2) and Regulatory Guides (References 5.6-3 & 5.6-4).

General Design Criterion (GDC) 38 (Reference 5.6-1 and Appendix 5A) provides requirements for the reduction and stabilization of containment pressure and temperature following a loss of coolant accident. It considers containment heat removal to be a safety function and consequently, following SRP 6.2.2 (Reference 5.6-2), requires suitable redundancy of components and features, capability assuming a single failure with loss of either on site or offsite power, leak detection and isolation capability, and suitable seismic qualification.

Regulatory Guide 1.26 (Reference 5.6-3) requires the containment heat removal system to be classified as a Quality Group B system which is to be ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", Class 2.

Regulatory Guide 1.29 (Reference 5.6-4) requires the system to be seismically qualified to withstand the effects of the SSE and remain functional.

5.3 CANDU 3 APPROACH

The containment and decay heat removal function in CANDU 3 is performed by the Reactor Building Cooling System, and Emergency Core Cooling System (ECC) recirculation pumps and heat exchangers with associated piping, valves, and support services.

The reactor building cooling system consists of eight main local air coolers and other coolers in individual rooms such as the moderator room, shutdown cooling room, and fuelling machine maintenance lock. Each cooling unit comprises air coolers, dampers and fans, as well as the appropriate piping and instrumentation.

The vault and steam generator enclosure atmosphere is separated from the accessible areas by closing off the top of the steam generator enclosure and by using blowout panels. Four local air coolers are dedicated to the vault cooling circuit, of which three are normally operating and one is on standby. Air is drawn from the top of the steam generator enclosure by local air coolers and is directed via ducting across the reactor building into the inlet vault.

The other local air cooler units are located strategically in the accessible areas of the reactor building. Other smaller local air coolers are placed in various individual rooms. The local air coolers are supplied with recirculated cooling water.

The main containment local air coolers and flow isolation dampers are accessible during normal reactor operation and do not pose difficulty for operational checks and inspection. The fans in the eight main local air coolers are supplied with Group 1 Class III power.

Instrumentation and control for the reactor building cooling system is provided to control operation of the building air coolers, the vault cooling units, the respective isolating dampers, and to provide temperature measurements and annunciation of alarm conditions. The temperature is measured in both the inaccessible and accessible areas.

The eight main local air coolers remain functional during a LOCA or a steam main failure. Following either of the above events, the system continues to remove heat from the reactor building atmosphere and helps to reduce the temperature and pressure of the building within 24 hours following the event.

The local air coolers are not seismically qualified. The supports and associated duct work are qualified to Design Basis Earthquake (DBE) category A to prevent local air coolers from toppling over, thus damaging seismically qualified components.

The reactor building cooling system is designed to Class 6 requirements of the CSA standard N285.0, (Reference 5.6-5) which basically implies application of non-nuclear standards for the system pressure boundary.

Under post accident conditions, during long-term ECC operation, the seismically (DBE) qualified ECC recirculation pumps circulate a mixture of light and heavy water from the containment ECC sump via the seismically (DBE) qualified heat exchangers back in to the core and remove heat from the containment sump water and thus prevent decay heat from being added to the containment atmosphere. The cooling water to the heat exchanger is supplied by the recirculated cooling water system, and is backed by the Group 2 raw service water system, which is seismically qualified.

It should be noted that Canadian licensing requirements call for analysis of the consequences of postulated process system failures coincident with failure of a mitigating safety system (a so-called "dual failure"). For CANDU designs with a containment dousing system, one of these analyses is for loss of coolant coincident with a loss of dousing. Since the CANDU 3 design does not incorporate a dousing system, an analysis of a loss of coolant coincident with complete failure of the containment cooling system is required for CANDU 3. Therefore, the design is required to meet the Canadian "dual failure" radiological dose limits without crediting containment cooling.

5.4 EVALUATION

The outlines in Sections 5.2 and 5.3 show a difference between the U.S. NRC guidance and CANDU 3 design with regard to the containment heat removal system.

The guidance for LWRs, given by the NRC in SRP 6.2.2, (Reference 5.6-2) is to have a containment heat removal system whose design incorporates all requirements for safety-grade systems, including redundancy of equipment consistent with single failure criterion, nuclear code (ASME Code, Section III) requirements, seismic qualification and leak detection, isolation and containment capabilities.

CANDU 3 design employs the reactor building cooling system consisting of local air coolers (air/water) with appropriate dampers and ducting. This system is a non-nuclear and non-seismically qualified system. The system is able to reduce the building pressure following a LOCA to around atmospheric pressure in 24 hours without requiring venting to the external atmosphere. Sufficient redundancy is provided in the system, assuming a failure of a single active component coincident with loss of offsite power. The heat removal capacity of the reactor building cooling system is sufficient for a LOCA or main steam failure.

The local air coolers in the reactor building cooling system of CANDU 3 are designed such that each cooler has individual cooling water pipes and isolation valves. These pipes and valves are located in the accessible area and satisfy the isolation requirement as defined in GDC 38 (Reference 5.6-1). The local air coolers are designed with adequate draining provision to drain the large amount of condensate. Any water leakage from the local air coolers is collected by the active drainage system, the moisture detectors in the active drainage system detect leakage from the local air coolers.

The reactor building cooling system in CANDU 3 is designed to provide containment heat removal following a LOCA or main steam failure. Because the high energy piping inside the reactor building is seismically qualified, the reactor building air coolers are not required for Design Basis Earthquake (DBE). The reactor building air coolers are not required for Site Design Earthquake (SDE) either, because an SDE is postulated (i.e., considered credible) only 24 hours after a LOCA, at which time the reactor building air coolers have completed their mission following a LOCA. Thus, Regulatory Guide 1.29 (Reference 5.6-4) is not applicable to the CANDU 3 containment heat removal system.

The non-nuclear classification of the reactor building cooling system in CANDU 3 design is acceptable since an adequate level of safety is deemed to be demonstrated by the following:

- The system is appropriately qualified for the events that it must mitigate.
- PSA analysis demonstrates a system unreliability (considering both active and passive failures) sufficiently low as to make the failure of the system a negligible risk contributor.
- The system is continuously monitored for leakage.

In addition, the design is required to meet Canadian "dual failure" dose limits following a loss of coolant with loss of containment cooling.

Thus, the CANDU 3 design with regard to the containment heat removal system provides an adequate level of safety.

5.5 CONCLUSIONS

The design of the CANDU 3 reactor building cooling system for the containment heat removal function has been evaluated based on the U.S. requirements and regulatory guidance. The leak detection, isolation and containment capabilities are incorporated in the design of the CANDU 3 reactor building cooling system. Similar to LWR design, the reactor building cooling system in CANDU 3 is able to accomplish:

a. the containment heat removal function with redundancy of components and

b. the containment heat removal function assuming a single active component failure with either a loss of onsite or a loss of offsite power.

The reactor building cooling system in CANDU 3 is not seismically qualified, because the system is not required to function after a DBE or for more than 24 hours following a LOCA or main steam line break. The containment design is also required to meet Canadian "dual failure" dose limits following a LOCA and complete loss of containment cooling.

Based on the evaluation of CANDU 3 containment heat removal capability, it is demonstrated that, in spite of significant differences in design methods, the CANDU 3 design, with regard to the containment heat removal capability, provides an adequate level of safety.

5.6 REFERENCES

5.6-1 Title 10, Code of Federal Regulations, Part 50, "Licensing Production and Utilization Facilities", Appendix A, "General Design Criteria for Nuclear Power Plants", Criterion 38.

5.6-2 U.S. NRC, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", LWR Edition, NUREG 0800, Chapter 6, Section 6.2.2.

5.6-3 U.S. NRC, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive- Waste-Containing Components of Nuclear Power Plants", U.S. Nuclear Regulatory Commission, Regulatory Guide 1.26, Rev. 3, 1976 February.

5.6-4 U.S. NRC, "Seismic Design Classification", Regulatory Guide 1.29, Rev. 3, 1978 September.

5.6-5 Canadian Standard, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants", CAN/CSA-N285.0-M92, Canadian Standards Association.

APPENDIX 5A

SELECTED U.S. REQUIREMENTS REGARDING CONTAINMENT HEAT REMOVAL

10 CFR50, GENERAL DESIGN CRITERIA (Reference 5-6.1)

Criterion 38-Containment heat removal. A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.