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**COMPILED REPORTS ON THE APPLICABILITY
OF SELECTED CODES AND STANDARDS
TO ADVANCED REACTORS**

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Abstract

The following papers were prepared for the Office of Nuclear Regulatory Research of the U.S. Nuclear Regulatory Commission under contract DE-AC06-76RLO-1830 NRC FIN L2207. This project, *Applicability of Codes and Standards to Advance Reactors*, reviewed selected mechanical and electrical codes and standards to determine their applicability to the construction, qualification, and testing of advanced reactors and to develop recommendations as to where it might be useful and practical to revise them to suit the (design certification) needs of the NRC.

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**APPLICABILITY OF THE ASME CODES AND
STANDARDS TO ADVANCED REACTORS (ASME
Section XI)**

Truong V. Vo
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September 1993

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1.0 INTRODUCTION

This report provides an assessment of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" to determine the applicability of the code to the design of advanced light water reactors (ALWRs). Four ALWR designs are considered in this analysis: the General Electric Advanced Boiling Water Reactor (GE ABWR), GE Simplified Boiling Water Reactor (GE SBWR), Combustion Engineering System 80+ (CE 80+), and the Westinghouse AP600.

This evaluation identifies, to the extent possible, the potential need for modification or revisions to ASME B&PVC, Section XI, Division 1. A study was performed for Subsections IWA, "General Requirements", IWB, "Requirements for Class 1 Components of Light-Water Cooled Plants", IWC, "Requirements for Class 2 Components of Light-Water Cooled Plants", and IWD, "Requirements for Class 3 Components of Light-Water Cooled Plants" to identify potential changes. Potential changes to other Sections XI Subsections are given preliminary consideration.

Section 2 of this report describes the approach. Conclusions are provided in Section 3.

2.0. APPROACH

The assessment involves two basic steps. The first step identifies and summarizes the unique features or operating modes of the four advanced light water reactor designs being considered. The second step reviews current ASME Codes and Standards for applicability to the ALWRs and identifies potential changes needed to address the ALWRs unique design features. The evaluation is focused on the ASME, Section XI. A study for ASME Section XI, Subsections IWA, IWB, IWC, and IWD was performed. Additionally, tentative changes to other subsections are identified in Table 5. Details involved in performing these steps are described below.

2.1. Unique Features of the ALWRs

This subsection summarizes the unique features of the ALWRs. The major changes in ALWR designs involve two aspects: (1) the advanced reactors incorporate passive safety system concepts in their designs for preventing and responding to potential upset and accident conditions, and (2) the design life of the advanced reactor plants is increased to 60 years. Use of passive safety systems allows many systems and components classified as safety-related in current light water reactor designs to be downgraded to non-safety related in the advanced reactor designs. Failure mechanisms with causes related to the active features of systems may be insignificant for passive safety systems. Extending the design life from 40 years to 60 years increases the relevance of aging related failure mechanisms and the corresponding need for inspection and testing.

1. GE ABWR

- a. Added an Automatic Power Regulator to control reactor power in startup and operation by changing rod positions and reactor circulation flow.
- b. Added ability to flood the lower drywell with water from the suppression pool for response to severe accidents where core melt could cause vessel failure.
- c. Added rod scram accumulators that use high pressure nitrogen to pressurize water and force the control rods into the core on a reactor trip.
- d. Replaced the jet pumps with canned motor pumps internal to the reactor vessel for forced coolant circulation.
- e. Extends the design life of the plant to 60 years.

2. GE SBWR

- a. Eliminates large nozzle connections to the reactor vessel below the top of active fuel level based on enhanced natural

circulation capabilities that allows elimination of forced coolant circulation and coolant circulation piping loops external to the vessel, and deletion of the recirculation flow control system.

- b. Reactor vessel uses forged rings that results in 50% reduction of vessel welds.
- c. Uses rod scram accumulators that use high pressure nitrogen to pressurize water and force the control rods into the core on a reactor trip which eliminates the standby liquid control pumps.
- d. Main steam is non-reheat cycle and main feedwater uses adjustable speed electric motor driven pumps.
- e. Reactor water cleanup system also performs shutdown cooling and pool cooling functions using sealless pumps.
- f. Process instrument and control systems use multiplexed solid state fiber optics where practical and the process computer performs control functions.
- g. Primary containment is a cylindrical reinforced concrete design with a suppression similar to the Mark III BWR-6 design with passive containment cooling provided by the suppression pool and the gravity driven isolation condenser.
- h. The standby gas treatment system is deleted.
- i. Neutron monitoring systems use a startup range monitor with auto insert and withdrawal during power maneuvering, eliminates the intermediate range monitor, incorporates an automatic fixed incore probe that replaces the traversing incore probe, and adds a power range monitor.
- j. Uses an Automatic Power Regulator to control reactor power in startup, power operation, and reactor shutdown by appropriate changes to rod positions.
- k. Eliminates need for AC power for Engineered Safety Features (ESF) actuation and uses vital DC power with 72 hour capacity and four divisions of controls and instrumentation power during loss of AC power and LOCA. Non-safety grade diesel generator is provided and can be used for DC power.
- l. Uses automatic depressurization relief through four safety/relief valves with six squib valves for depressurization which allows gravity driven cooling system to flood containment through squib valves to an elevation one meter above the top of active fuel for indefinite post LOCA cooling. The high pressure core spray system, low

pressure core spray system, residual heat removal system, and standby service water system are deleted.

- m. Extends the design life of the plant to 60 years.

3. CE 80+

- a. Incorporates the in-containment refueling water storage tank concept which serves as the initial source of water for ECCS and as a quench tank for releases from the pressurizer safety valves and the safety depressurization system, and eliminates the recirculation actuation signal.
- b. Added two additional pumps which are dedicated to the shutdown cooling mode of operation. This eliminates the need for using the low pressure safety injection pumps for this purpose.
- c. Uses four high pressure safety injection pumps for ECCS. This eliminates two low pressure safety injection pumps.
- d. Uses the containment spray pumps to perform containment spray and shutdown cooling.
- e. Reclassifies the chemical and volume control system as non-safety based on safe shutdown functions being performed by the safety injection system.
- f. Extends the design life of the plant to 60 years.

4. AP 600

- a. Uses canned motor pumps for reactor coolant pumps.
- b. Uses an automatic depressurization system to depressurize the reactor coolant system during some accidents to allow passive water injection into the reactor core.
- c. Incorporates the in-containment refueling water storage tank as a large volume of water stored above the reactor core available for passive response to LOCAs upon RCS depressurization.
- d. Eliminates the ECCS function of the residual heat removal system and retains its shutdown and refueling cooling functions.
- e. Adds two core makeup tanks inside containment which float at RCS pressure and provide a quick passive source of water during accidents upon depressurization. This allows the safety function of the chemical and volume control system to be eliminated.

- f. Auxiliary feedwater pumps are eliminated based on use of the passive residual heat removal system to route primary water through the in-containment refueling water storage tank during some accidents to cool the primary water.
- g. Uses a large steel containment shell with passive external cooling from air and water which transfers heat to the atmosphere around the plant as the ultimate heat sink and results in reactor coolant being condensed on the inside containment shell and recycled for cooling the reactor core.
- h. Uses safety grade DC power to actuate and reposition some valves for passive systems response which allows downgrading AC power systems to non-safety grade including the standby diesel generators.
- i. Extends the design life of the plant to 60 years.

2.2. Review of the ASME and Analysis Approach

1. Overview of ASME, Section XI, Division 1, "Rules for Inspection and Testing of Light-Water Cooled Plants" review for ALWRs:
 - a. A preliminary review of the 1992 Edition of ASME, Section XI, Division 1 was performed. Subsections that are applicable to the safety related mechanical systems and components affected by the unique ALWR features were considered. Generally this limited the scope of the review to the reactor coolant system pressure boundary, containment system pressure boundary, and systems and components important to plant safety.
 - b. The ASME portion of the review does not include quality assurance, fire protection, fuel handling, waste handling and treatment, electrical, and instrumentation and control systems.
2. Analysis approach and assumptions for identification of potential impacts of unique ALWR features on the ASME Code:
 - a. The analysis is performed to identify potential impacts on the ASME Code to address unique features of the ALWR designs. The approach includes:
 - 1) Identification of the unique features for each ALWR design considered as presented in section 2.1 above.
 - 2) Development of categories for potential revisions to the ASME Code based on the nature of the identified ALWR unique features and ASME Code structure and content.

- 3) Identification of the individual articles, subarticles, and subsubarticles that might be affected to address the identified ALWR unique features.
 - 4) Presenting a brief description of the rationale for the potential revisions identified.
- b. The criteria used in performing this analysis include:
- 1) ALWR unique features that interface directly with the reactor coolant pressure boundary and involve a new concept not specifically addressed in the ASME Code create a candidate potential impact.
 - 2) ALWR unique features that interface directly with the containment pressure boundary and involve a new concept not specifically addressed in the ASME Code create a candidate potential impact.
 - 3) ALWR unique features which involve downgrade of previously safety related systems based upon reliance on passive safety systems create a candidate potential impact.
- c. Assumptions and limitations for this analysis include:
- 1) ALWR unique features were considered conceptually, and subjective judgement was used conservatively to identify potential impacts to the ASME Code. When detailed design specifications for the unique ALWR features are available and the need for some identified potential revisions to the ASME Code may need to be reconsidered.
 - 2) ALWR unique features that involve extended design lives and design concepts which depart from the characteristics of active safety systems may raise degradation or failure mechanisms which were not previously relevant or significant. As such failure mechanisms become known, there may be a need for additional changes to the inspection and testing requirements or acceptance criteria to appropriately address these degradation and failure mechanisms.
3. Applications were performed for Subsections IWA, IWB, IWC, and IWD. Results of the applications are summarized in Tables 1, 2, 3, and 4. Potential impacts to other subsections were also identified but were not developed further (Table 5). These impacts will be addressed at a later date.
- a. The analysis process was performed for ASME Code Section XI, Division 1, Subsection IWA, "General Requirements". This

review resulted in identification of potential impacts to two articles, subarticles, or subsubarticles as follows:

- 1) Repair and Replacement, IWA-4120, "Exemptions".
 - 2) Section IWA-9000, "Glossary".
- b. The analysis process was performed for ASME Code Section XI, Division 1, Subsection IWB, "Requirements for Class 1 Components of Light-Water Cooled Power Plants". This review resulted in identification of potential impacts to one article, subarticle, or subsubarticle as follows:
- 1) Examination and Inspection, IWB-2500, "Examination and Pressure Test Requirements", and Table 2500-1, "Examination Categories".
- c. The analysis process was performed for ASME Code Section XI, Division 1, Subsection IWC, "Requirements for Class 2 Components of Light-Water Cooled Power Plants". This review resulted in identification of potential impacts to three articles, subarticles, or subsubarticles as follows:
- 1) Components Exempt From Examination, IWC-1221, "Components Within RHR, ECC, CHR Systems or Portions of Systems".
 - 2) Components Exempt From Examination, IWC-1222, "Components Within Systems or Portions of Systems Other than RHR, ECC, and CHR Systems".
 - 3) Components Exempt From Examination, IWC-1223, "Integral Attachments".
- d. The analysis process was performed for ASME Code Section XI, Division 1, Subsection IWD, "Requirements for Class 3 Components of Light-Water Cooled Power Plants". This review resulted in identification of potential impacts to one article, subarticle, or subsubarticle as follows:
- 1) Components Subject To Examination, IWD-1220, "Components Exempt From Examination".
- e. The assessment performed for the Section XI Subsection beyond IWD resulted in identification of several tentative potential impacts which are identified in Table 5.

3.0. SUMMARY AND CONCLUSIONS

The preliminary results show that the existing provisions of the ASME Code are generally applicable to the unique design features of the ALWR designs considered. For the ASME Sections selected for study (IWA, IWB, IWC, and IWD, Section XI), the results, as shown in Tables 1, 2, 3, and 4 indicated the need for only limited changes in ASME Section XI. Based on this preliminary information, as shown in Table 5, it is indicated that changes in O&M Code may be anticipated. The impacts on ASME Code are primarily due to one or more of the following categories:

- 1) code terminology (IWA-9000, Section XI; O&M Code)
- 2) inservice inspection/testing program (IWA-2000, IWB-2000, IWC-1220, and IWD-1200, Section XI; O&M-6, O&M-8, O&M Code)
- 3) design life.

Articles, subarticles, and subsubarticles within the ASME Codes and Standards may need to be revised in order to completely address the unique features of the ALWRs. The code may be revised to address the design changes from active to passive safety systems and potential for different failure mechanisms, and the increase in plant design life (60 years). The ASME Codes and Standards may also be revised to address inservice inspection and testing requirements for new components (i.e., canned motor pump for the RCS, AP 600 design). For instance, the potential revisions to inservice testing requirements for IWP, "Pump Testing" and IWV, "Valve Testing" are addressed in O&M-6 and O&M-8, respectively.

This preliminary analysis was based on available information regarding ALWR designs. Detailed design information on certified ALWR designs, when available and reviewed may reveal additional information that could alter the identified potential impacts to the ASME Code. Future work will identify and develop potential impacts on the remaining sections of ASME B&PVC Section XI and O&M and may also consider other ALWR designs. This will include further review of ASME Section XI Subsections IWP, IWV, and the O&M Codes. The rationale for the potential impacts to codes and standards would also be described.

4.0. REFERENCES

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1992 Edition.

Table 1

**ALWR IMPACTS ON ASME B&PV
CODE SECTION XI, SUBSECTION IWA**

TABLE 1 ALMR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ^{**} REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used as reactor coolant pumps.	2. a,b 3. a,b,c 4. a,b	1. Under Repair and Replacement, IWA-4120, "Exemptions". 2. Subsection IWA-9000, "Glossary".	1. Change may be desired to state if replacement of canned motor reactor coolant pumps (that have no shaft seals) are exempted. 2. Change may be desired to define the canned motor reactor coolant pump concept.
2. Automatic RCS depressurization system added to allow for passive system water injection into the core.	1. a,d 2. a,b 3. b,c	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	1. a,d 2. a,b 3. a,b,c 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of IRWST and associated passive response under reactor coolant system depressurization.
4. Residual heat removal system does not provide ECCS function but is used for shutdown and refueling cooling.	1. b,c 2. a,b 3. b,c 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of residual heat removal system that does not provide an ECCS function and is therefore non-safety related.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL** REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	1. a,d 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ^{**} REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
6. Chemical and volume control system including charging pumps and primary coolant make-up pumps downgraded to non-safety based on CMT addition along with other changes.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	1. a,d 2. a,b 3. b,c 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define a passive residual heat removal system that together with the IRWST does provide a cooling function for some accidents which allows elimination of auxiliary feedwater pumps from the design.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL* REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
8. Removed the auxiliary feedwater pumps based on passive RHR function.	1. b,c 2. a,b 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of pressurized water reactor without the need for an auxiliary feedwater system.
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	1. a,b,c, d 2. a,b 3. b,c 4. a	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the passive design concept for the containment heat removal function and associated containment design features.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ⁺ REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	1. b,c 2. a,b 3. b,c	N/A	N/A
11. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of 60 year plant design life.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
GE ABWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt severe accidents.	1. a,d 2. a,b 3. a,b,c 4. b	N/A	N/A
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	2. a,b 3. a,b,c 4. a,b	1. Under Repair and Replacement, IWA-4120, "Exemptions". 2. Subsection IWA-9000, "Glossary".	1. Change may be desired to state if replacement of canned motor reactor coolant pumps (that have no shaft seals) are exempted. 2. Change may be desired to define the canned motor reactor coolant pump concept.
3. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of 60 year plant design life.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA

CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and safety depressurization system.	1. a,d 2. a,b 3. a,b,c 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of IRWST and associated passive response under reactor coolant system depressurization.
2. Eliminated the recirculation actuation signal based on adding the IRWST.	1. a,b,c,d 2. a,b 4. b	N/A	N/A
3. Two pumps dedicated to shutdown cooling operation.	1. a,b,c,d 2. a,b 4. b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALMR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL* REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	1. b 2. a,b 3. b,c 4. b	N/A	N/A
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	1. b 2. a,b 3. b,c 4. b	N/A	N/A
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ^{**} REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
7. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of 60 year plant design life.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALMR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IMA			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IMA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	2. a,b 4. a,b	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	3. a 4. a,b	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	1. a,b,c, d 2. a,b 4. a,b	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	2. a,b 3. a,b 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALMR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IMA			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ^{**} REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
5. Eliminates reactor core isolation cooling system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
6. Adds decay heat removal system to perform reactor water cleanup function, shutdown cooling, pool cooling and pool cleanup functions using sealless pumps and hollow fiber filters.	1. b 2. a,b 3. b,c 4. a,b	N/A	N/A
7. Deleted standby gas treatment system.	1. b,c 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL* REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	1. a,d 2. a,b 3. b,c	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the passive design concept for the containment heat removal function and associated containment design features.
9. Vital AC power is provided by two DC busses with 72 hour battery capacity.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
11. High pressure core spray eliminated.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL* REVISIONS**	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A
13. Eliminated low pressure core spray.	1. b,c, 2. a,b 3. b,c 4. a,b	N/A	N/A
14. Eliminated residual heat removal emergency core cooling system function.	1. b,c 2. a,b 3. b,c 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of residual heat removal system that does not provide an ECCS function and is therefore non-safety related.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWA			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ^{**} REVISIONS	IWA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	1. a,d 2. a,b 3. b,c 4. b	N/A	N/A
16. Uses an isolation condenser containment design.	1. a,b,c, d 3. b,c 4. b	N/A	N/A
17. Eliminated standby service water system.	1. b 4. a,b	N/A	N/A
18. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	1. Subsection IWA-9000, "Glossary".	1. Change may be desired to define the concept of 60 year plant design life.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION XI, POTENTIAL REVISION CATEGORIES

- 1. Safety Classification**
 - a. passive safety systems and components
 - b. eliminated active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
- 2. Code Terminology**
 - a. concepts
 - b. definitions
- 3. Design Life**
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile
- 4. Ongoing Code Work**
 - a. code committees
 - b. risk-based approaches

Table 2

**ALWR IMPACTS ON ASME B&PV
CODE SECTION XI, SUBSECTION IWB**

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used as reactor coolant pumps.	2. a,b 3. a,b,c 4. a,b	1. Under Examination and Inspection, IWB-2500, "Examination and Pressure Test Requirements", and Table IWB-2500-1, "Examination Categories" for B-K-1, B-L-1, and B-L-2.	1. Change may be desired to categorize canned motor reactor coolant pumps and describe the applicable examination requirements, methods, acceptance criteria, and frequency.
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	1. a,d 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL ^{*,**} REVISIONS	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	1. a,d 2. a,b 3. a,b,c 4. a,b	N/A	N/A
4. Residual heat removal system does not provide and ECCS function but is used for shutdown and refueling cooling.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	1. a,d 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
6. Chemical and volume control system including charging pumps and primary coolant make-up pumps downgraded to non-safety based on CMT addition along with other changes.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A
7. Passive residual heat removal system cools primary water through the IRWT during some accidents.	1. a,d 2. a,b 3. b,c 4. a,b	N/A	N/A
8. Eliminated auxiliary feedwater pumps based on passive RHR function.	1. b,c 2. a,b 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
11. Plant design life extended to 60 years.	3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
GE ABWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	1. a,d 2. a,b 3. a,b,c 4. a,b	N/A	N/A
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	2. a,b 3. a,b,c 4. a,b	1. Under Examination and Inspection, IWB-2500, "Examination and Pressure Test Requirements", and Table IWB-2500-1, "Examination Categories" for B-K-1, B-L-1, and B-L-2.	1. Change may be desired to categorize canned motor reactor coolant pumps and describe the applicable examination requirements, methods, acceptance criteria, and frequency.
3. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and safety depressurization system.	1. a,d 2. a,b 3. a,b,c 4. a,b	N/A	N/A
2. Eliminated the recirculation actuation signal based on adding the IRWST.	1. a,b,c, d 2. a,b 4. a,b	N/A	N/A
3. Dedicated two pumps to shutdown cooling operation.	1. a,b,c, d 2. a,b 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
CE SYSTEMS 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	1. b 2. a,b 3. b,c 4. a,b	N/A	N/A
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	1. b 2. a,b 3. b,c 4. a,b	N/A	N/A
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A
7. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	2. a,b 4. a,b	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	2. a,b 3. a 4. a,b	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	1. a,b,c,d 2. a,b 4. a,b	1. Under Examination and Inspection, IWB-2500, "Examination and Pressure Test Requirements", and Table IWB-2500-1, "Examination Categories" for B-N-1, B-N-2, and B-N-3.	1. Change may be desired to reflect the change in reactor vessel internals design features that result in increased natural circulation such that the need for forced recirculation is eliminated and describe the potential change to the applicable examination requirements, methods, acceptance criteria, and frequency.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
4. Uses adjustable speed electric motor driven feedwater pumps.	2. a,b 3. a,b 4. a,b	N/A	N/A
5. Eliminates reactor core isolation cooling system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	1. b 2. a,b 3. b,c 4. a,b	N/A	N/A
7. Deleted standby gas treatment system.	1. b,c 2. a,b,c 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS*	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	1. a,d 2. a,b 3. b,c 4. a,b	N/A	N/A
9. Vital AC power is provided by two DC busses with 72 hour capacity.	1. b,c 2. a,b 3. b,c 4. a,b 4. c	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
11. High pressure core spray deleted.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS*	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	N/A	N/A
13. Deleted low pressure core spray.	1. b,c, 2. a,b 3. b,c 4. a,b	N/A	N/A
14. Eliminated residual heat removal emergency core cooling system function.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWB			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWB SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	1. a,d 2. a,b 3. b,c 4. a,b	N/A	N/A
16. Uses an isolation condenser containment design.	1. a,b,c,d 2. a,b 3. b,c 4. a,b	N/A	N/A
17. Eliminated standby service water system.	1. b 2. a,b 4. a,b	N/A	N/A
18. Plant design life extended to 60 years.	2. a,b 3. a,b,c,d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION XI, POTENTIAL REVISION CATEGORIES

1. **Safety Classification**
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
2. **Code Terminology**
 - a. concepts
 - b. definitions
3. **Design Life**
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile
4. **Ongoing Code Work**
 - a. code committees
 - b. risk-based approaches

Table 3

**ALWR IMPACTS ON ASME B&PV
CODE SECTION XI, SUBSECTION IWC**

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used for reactor coolant pumps.	2. a,b 3. a,b,c 4. a,b	N/A	N/A
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	1. a,d 2. a,b 3. b,c	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.</p>	<p>1. a,d 2. a,b 3. a,b,c 4. a,b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p> <p>3. IWC-1223, "Integral Attachments".</p>	<p>1. Change may be desirable to specify applicable exemption conditions for IRWST and associated passive flow path for RHR, ECC, and CHR functions, (i.e., add IRWST to footnote number 3).</p> <p>2. Change may be desirable to specify applicable exemption conditions for IRWST and associated passive flow paths for other than RHR, ECC, and CHR functions.</p> <p>3. Change may be desirable to specify applicable exemption conditions for supports and restraints for the IRWST and associated passive flow paths.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>4. Residual heat removal system does not provide and ECCS function but is used for shutdown and refueling cooling.</p>	<p>1. b,c 2. a,b 3. b,c 4. a,b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p>	<p>1. Change may be desirable to distinguish applicable exemptions for non-safety-related RHR system functions from the safety-related RHR system functions.</p> <p>2. Change may be desirable to address non-safety-related RHR systems.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	1. a,d 2. a,b 3. b,c 4. a,b	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)". 2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems". 3. IWC-1223, "Integral Attachments".	1. Change may be desirable to specify applicable exemption conditions for core make-up tanks (CMT) and associated passive flow path for RHR, ECC, and CHR functions (i.e., add CMT to footnote number 3). 2. Change may be desirable to specify applicable exemption conditions for CMT and associated passive flow paths for other than RHR, ECC, and CHR functions. 3. Change may be desirable to specify applicable exemption conditions for supports and restraints for the CMT and associated passive flow paths.

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>6. Chemical and volume control system including charging pumps and primary coolant make-up pumps downgraded to non-safety based on CMT addition along with other changes.</p>	<p>1. a,b,c, d 2. a,b 3. b,c 4. a,b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p>	<p>1. Change may be desirable to distinguish applicable exemptions for non-safety-related CVCS (Charging Pumps)/ECC system functions from the safety-related CVCS (Charging Pumps)/ECC system functions.</p> <p>2. Change may be desirable to address non-safety-related CVCS (Charging Pumps)/ECC systems.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	1. a,d 2. a,b 3. b,c 4. a,b	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)". 2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".	1. Change may be desirable to distinguish applicable exemptions for RHR flow paths that provide or support post accident cooling functions from other safety-related RHR system functions. 2. Change may be desirable to address non-safety-related RHR systems which do not provide or support post accident cooling functions.
8. Removed the auxiliary feedwater pumps based on passive RHR function.	1. b,c 2. a,b 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	1. a,b,c, d 2. a,b 3. b,c 4. a	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems or Portions of Systems". 2. IWC-1222, "Components Within Systems or Portions of Systems Other Than RHR, ECC, and CHR Systems".	1. Change may be desirable to specify applicable exemption conditions for Containment shell design and those systems which perform or support the passive CHR functions. 2. Change may be desirable to specify applicable exemption conditions for Containment shell design and associated systems which do not provide or support the CHR functions.
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	1. b,c 2. a,b 3. b,c	N/A	N/A
11. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

GE ABWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	1. a,d 2. a,b 3. a,b,c 4. b	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)". 2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems". 3. IWC-1223, "Integral Attachments".	1. Change may be desired to specify applicable exemption conditions for the flow path of lower drywell flooding from suppression pool, as a severe accident response function. 2. Change may be desired to specify exemption conditions for suppression pools and lower drywell systems providing other than the traditional RHR, ECC, and CHR system functions. 3. Change may be desired to specify the applicable exemption conditions for supports and restraints for the suppression pool flow path for flooding the lower drywell.
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	2. a,b 3. a,b,c 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC			
GE ABWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
3. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and safety depressurization system.</p>	<p>1. a,d 2. a,b 3. a,b,c 4. a,b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p> <p>3. IWC-1223, "Integral Attachments".</p>	<p>1. Change may be desirable to specify applicable exemption conditions for IRWST and associated passive flow path for RHR, ECC, and CHR functions, (i.e., add IRWST to footnote number 3).</p> <p>2. Change may be desirable to specify applicable exemption conditions for IRWST and associated passive flow paths for other than RHR, ECC, and CHR functions.</p> <p>3. Change may be desirable to specify applicable exemption conditions for supports and restraints for the IRWST and associated passive flow paths.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>2. Eliminated the recirculation actuation signal based on adding the IRWST.</p>	<p>1. a,b,c, d 2. a,b 4. b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)", (e.g., may affect the classification of related RHR, ECC, and CHR systems if no longer relied upon for coolant recirculation and cooling).</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p>	<p>1. Change may be desired to specify the applicable exemption conditions for RHR, ECC, and CHR systems classified as non-safety-related based on use of passive safety systems.</p> <p>2. Change may be desired to specify the applicable exemption conditions for passive safety systems that are other than RHR, ECC, and CHR systems.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>3. Dedicated two pumps to shutdown cooling operation.</p>	<p>1. a,b,c, d 2. a,b 4. b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)", (e.g., may affect the classification of related RHR, ECC, and CHR systems if no longer relied upon for coolant recirculation and cooling).</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p>	<p>1. Change may be desired to specify the applicable exemption conditions for RHR, ECC, and CHR systems classified as non-safety-related based on use of passive safety systems.</p> <p>2. Change may be desired to specify the applicable exemption conditions for passive safety systems that are other than RHR, ECC, and CHR systems.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.</p>	<p>1. b 2. a,b 3. b,c 4. b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p>	<p>1. Change may be desired to specify the applicable exemption conditions for ECC systems that utilize only HPSI pumps.</p> <p>2. Change may be desired to specify the applicable exemption conditions for non-safety-related low pressure pumping systems.</p>
<p>5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.</p>	<p>1. b 2. a,b 3. b,c 4. b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p>	<p>1. Change may be desired to specify the applicable exemption conditions for CHR systems classified as non-safety-related.</p> <p>2. Change may be desired to specify the applicable exemption conditions for CHR systems that also provide a non-safety-related shutdown cooling function.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)". 2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related CVCS (Charging Pumps)/ECC system functions from the safety-related CVCS (Charging Pumps)/ECC system functions. 2. Change may be desirable to address non-safety-related CVCS (Charging Pumps)/ECC systems.
7. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	2. a,b 4. a,b	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	2. a,b 3. a 4. a,b	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	1. a,b,c, d 2. a,b 4. a,b	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	2. a,b 3. a,b 4. a,b	N/A	N/A
5. Eliminates reactor core isolation cooling system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	1. b 2. a,b 3. b,c 4. a,b	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)". 2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related RHR or decay heat removal system functions from the safety-related RHR system functions. 2. Change may be desirable to address non-safety-related RHR or decay heat removal systems.
7. Deleted standby gas treatment system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	1. a,d 2. a,b 3. b,c	N/A	N/A
9. Vital AC power is provided by two DC busses with 72 hour capacity.	1. b,c 2. a,b 3. b,c 4. b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	1. b,c 2. a,b 3. b,c 4. b	N/A	N/A
11. High pressure core spray deleted.	1. b,c 2. a,b 3. b,c 4. b	N/A	N/A
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	1. a,b,c, d 2. a,b 3. b,c 4. b	N/A	N/A
13. Deleted low pressure core spray.	1. b,c, 2. a,b 3. b,c 4. a,b	N/A	N/A

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
14. Deleted residual heat removal emergency core cooling system function.	1. b,c 2. a,b 3. b,c 4. a,b	1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)". 2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related RHR system functions from the safety-related RHR system functions. 2. Change may be desirable to address non-safety-related RHR systems.

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
<p>15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.</p>	<p>1. a,d 2. a,b 3. b,c 4. b</p>	<p>1. IWC-1221, "Components Within RHR, ECC, and CHR Systems (or Portions of Systems)".</p> <p>2. IWC-1222, "Components Within Systems (or Portions of Systems) Other than RHR, ECC, and CHR Systems".</p> <p>3. IWC-1223, "Integral Attachments".</p>	<p>1. Change may be desirable to specify applicable exemption conditions for gravity driven cooling system and associated passive flow path for RHR, ECC, and CHR functions, (i.e., add gravity driven cooling system to footnote number 3).</p> <p>2. Change may be desirable to specify applicable exemption conditions for gravity driven cooling system and associated passive flow paths for other than RHR, ECC, and CHR functions.</p> <p>3. Change may be desirable to specify applicable exemption conditions for supports and restraints for the gravity driven cooling system and associated passive flow paths.</p>

TABLE 3 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWC

GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWC SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
16. Uses an isolation condenser containment design.	1. a,b,c, d 2. a,b 3. b,c 4. b	N/A	N/A
17. Deleted standby service water system.	1. b 2. a,b 4. a,b	N/A	N/A
18. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

ALWR IMPACTS ON ASME B&PV CODE SECTION XI, POTENTIAL REVISION CATEGORIES

1. **Safety Classification**
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
2. **Code Terminology**
 - a. concepts
 - b. definitions
3. **Design Life**
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile
4. **Ongoing Code Work**
 - a. code committees
 - b. risk-based approaches

Table 4

**ALWR IMPACTS ON ASME B&PV
CODE SECTION XI, SUBSECTION IWD**

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used for reactor coolant pumps.	2. a,b 3. a,b,c 4. a,b	N/A	N/A
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	1. a,d 2. a,b 3. b,c	N/A	N/A
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	1. a,d 2. a,b 3. a,b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to specify applicable exemption conditions for IRWST and associated passive flow path for RHR, ECC, and CHR functions.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
4. Residual heat removal system does not provide and ECCS function but is used for shutdown and refueling cooling.	1. b,c 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related RHR system functions from the safety-related RHR system functions. A change may be desirable to specify applicable exemption requirements for RHR and associated flow paths that provide shutdown cooling and refueling cooling functions but does not provide a ECC function.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	1. a,d 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to specify applicable exemption conditions for core make-up tanks (CMT) and associated passive flow paths on RCS depressurization.
6. Chemical and volume control system including charging pumps and primary coolant make-up pumps downgraded to non-safety based on CMT addition along with other changes.	1. a,b,c,d 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related CVCS (Charging Pumps)/ECC system functions from the safety-related CVCS (Charging Pumps)/ECC system functions. Change may be desirable to specify the applicable exemption conditions for non-safety CVCS that supports normal plant operation.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	1. a,d 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to distinguish applicable exemptions for RHR flow paths that provide or support post accident cooling functions from other safety-related RHR system functions. Change may be desirable to address non-safety-related RHR systems which do not provide or support post accident cooling functions and passive RHR systems which do.
8. Removed the auxiliary feedwater pumps based on passive RHR function.	1. b,c 2. a,b 4. a,b	N/A	N/A
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	1. a,b,c, d 2. a,b 3. b,c 4. a	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to specify applicable exemption conditions for Containment shell design and those systems which perform or support the passive CHR functions.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IMD			
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IMD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	1. b,c 2. a,b 3. b,c	N/A	N/A
11. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
GE ABWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	1. a,d 2. a,b 3. a,b,c 4. b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desired to specify applicable exemption conditions for the flow path of lower drywell flooding from suppression pool, as a severe accident response function.
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	2. a,b 3. a,b,c 4. a,b	N/A	N/A
3. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWT) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and safety depressurization system.	1. a,d 2. a,b 3. a,b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to specify applicable exemption conditions for IRWT and associated passive flow path for ECC functions and associated flow paths for pressurizer safety and safety depressurization system discharges.
2. Eliminated the recirculation actuation signal based on adding the IRWT.	1. a,b,c,d 2. a,b 4. b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desired to specify the applicable exemption conditions for systems classified as non-safety-related based on use of passive safety systems as well as the passive safety systems and their flow paths.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS*	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
3. Dedicated two pumps to shutdown cooling operation.	1. a,b,c, d 2. a,b 4. b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desired to specify the applicable exemption conditions for the dedicated shutdown cooling system and associated flow paths classified as non-safety-related.
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	1. b 2. a,b 3. b,c 4. b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desired to specify the applicable exemption conditions for ECC systems that utilize only HPSI pumps and no LPSI pumps.
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	1. b 2. a,b 3. b,c 4. b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desired to specify the applicable exemption conditions for Containment Spray pumps that also provide a shutdown cooling function and the associated flow paths.

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related CVCS (Charging Pumps) functions from the safety-related CVCS (Charging Pumps)/ECC system functions. Change may be desirable to address the applicable exemption conditions for non-safety-related CVCS that support normal plant operation.
7. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	2. a,b 4. a,b	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	2. a,b 3. a 4. a,b	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	1. a,b,c, d 2. a,b 4. a,b	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	2. a,b 3. a,b 4. a,b	N/A	N/A
5. Eliminates reactor core isolation cooling system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	1. b 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related RHR or decay heat removal system functions from the safety-related RHR system functions. Change may be desirable to address the applicable exemption conditions for non-safety-related RHR or decay heat removal systems such as reactor water cleanup, shutdown cooling, and pool cooling and cleanup and their flow paths.
7. Deleted standby gas treatment system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	1. a,d 2. a,b 3. b,c	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS*	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
9. Vital AC power is provided by two DC busses with 72 hour capacity.	1. b,c 2. a,b 3. b,c 4. b	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	1. b,c 2. a,b 3. b,c 4. b	N/A	N/A
11. High pressure core spray deleted.	1. b,c 2. a,b 3. b,c 4. b	N/A	N/A
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	1. a,b,c, d 2. a,b 3. b,c 4. b	N/A	N/A
13. Deleted low pressure core spray.	1. b,c, 2. a,b 3. b,c 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
14. Deleted residual heat removal emergency core cooling system function.	1. b,c 2. a,b 3. b,c 4. a,b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to distinguish applicable exemptions for non-safety-related RHR system functions from the safety-related RHR system functions. Change may be desirable to address the applicable exemption conditions for RHR and associated flow paths that are non-safety-related.
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	1. a,d 2. a,b 3. b,c 4. b	1. IWD-1220, "Components Exempt From Examination".	1. Change may be desirable to specify applicable exemption conditions for gravity driven cooling system and associated passive flow path for containment flooding post LOCA cooling.
16. Uses an isolation condenser containment design.	1. a,b,c, d 2. a,b 3. b,c 4. b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTION IWD			
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI	
MAJOR DESIGN DIFFERENCES	POTENTIAL [*] REVISIONS	IWD SECTIONS POTENTIALLY IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
17. Deleted standby service water system.	1. b 2. a,b 4. a,b	N/A	N/A
18. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	N/A	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION XI, POTENTIAL REVISION CATEGORIES

1. Safety Classification
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
2. Code Terminology
 - a. concepts
 - b. definitions
3. Design Life
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile
4. Ongoing Code Work
 - a. code committees
 - b. risk-based approaches

Table 5

**ALWR IMPACTS ON ASME B&PV
CODE SECTION XI, SUBSECTIONS BEYOND IWD**

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
1. Canned motor pumps used for reactor coolant pumps.	2. a,b 3. a,b,c 4. a,b	IWP-1100, "Pump Testing".
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	1. a,d 2. a,b 3. b,c	IWV-1100, "Valve Testing".
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	1. a,d 2. a,b 3. a,b,c 4. a,b	IWV-1100, "Valve Testing".
4. Residual heat removal system does not provide an ECCS function but is used for shutdown and refueling cooling.	1. b,c 2. a,b 3. b,c 4. a,b	IWP-1100, "Pump Testing". IWV-1100, "Valve Testing".
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	1. a,d 2. a,b 3. b,c 4. a,b	IWV-1100, "Valve Testing".
6. Chemical and volume control system including charging pumps and primary coolant make-up pumps downgraded to non-safety based on CMT addition along with other changes.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	IWP-1100, "Pump Testing". IWV-1100, "Valve Testing".

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	1. a,d 2. a,b 3. b,c 4. a,b	IWP-1100, "Pump Testing". IWP-1100, "Valve Testing".
8. Removed the auxiliary feedwater pumps based on passive RHR function.	1. b,c 2. a,b 4. a,b	IWD-1210, "Examination Requirements". IWD-1220, "Items Exempt From Examination".
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	1. a,b,c, d 2. a,b 3. b,c 4. a	IWE-1220, "Components Exempt From Examination".
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	1. b,c 2. a,b 3. b,c	IWP-1100, "Valve Testing", (e.g., closure timing and position verification with DC power only).

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
WESTINGHOUSE AP600	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
11. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	IWE-2411 and Table 2411-1, "Inspection Program A". IWP-1100, "Pump Testing". I WV-1100, "Valve Testing".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
GE ABWR	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	1. a,d 2. a,b 3. a,b,c 4. b	IWV-1100, "Valve Testing".
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	2. a,b 3. a,b,c 4. a,b	IWP-1100, "Pump Testing".
3. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	IWE-2411 and Table 2411-1, "Inspection Program A". IWP-1100, "Pump Testing". IWV-1100, "Valve Testing".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
CE SYSTEM 80+	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and safety depressurization system.	1. a,d 2. a,b 3. a,b,c 4. a,b	IWV-1100, "Valve Testing".
2. Eliminated the recirculation actuation signal based on adding the IRWST.	1. a,b,c, d 2. a,b 4. b	IWP-1100, "Pump Testing". IWV-1100, "Valve Testing".
3. Dedicated two pumps to shutdown cooling operation.	1. a,b,c, d 2. a,b 4. b	IWP-1100, "Pump Testing".
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	1. b 2. a,b 3. b,c 4. b	IWP-1100, "Pump Testing".
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	1. b 2. a,b 3. b,c 4. b	IWP-1100, "Pump Testing".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
CE SYSTEMS 80+	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	1. a,b,c, d 2. a,b 3. b,c 4. a,b	IWP-1100, "Pump Testing". I WV-1100, "Valve Testing".
7. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	IWE-2411 and Table 2411-1, "Inspection Program A". IWP-1100, "Pump Testing". I WV-1100, "Valve Testing".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	2. a,b 4. a,b	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	2. a,b 3. a 4. a,b	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	1. a,b,c, d 2. a,b 4. a,b	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	2. a,b 3. a,b 4. a,b	IWP-1100, "Pump Testing".
5. Eliminates reactor core isolation cooling system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	1. b 2. a,b 3. b,c 4. a,b	IWP-1100, "Pump Testing".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
7. Deleted standby gas treatment system.	1. b,c 2. a,b 3. b,c 4. a,b	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	1. a,d 2. a,b 3. b,c	N/A
9. Vital AC power is provided by two DC busses with 72 hour capacity.	1. b,c 2. a,b 3. b,c 4. b	IWV-1100, "Valve Testing", (e.g., closure timing and position verification with DC power only).
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	1. b,c 2. a,b 3. b,c 4. b	IWV-1100, "Valve Testing", (e.g., closure timing and position verification with DC power only).
11. High pressure core spray deleted.	1. b,c 2. a,b 3. b,c 4. b	N/A

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	1. a,b,c, d 2. a,b 3. b,c 4. b	IWV-1100, "Valve Testing".
13. Eliminated low pressure core spray.	1. b,c, 2. a,b 3. b,c 4. a,b	IWD-1210, "Examination Requirements". IWD-1220, "Components Exempt From Examination".
14. Eliminated residual heat removal emergency core cooling system function.	1. b,c 2. a,b 3. b,c 4. a,b	IWP-1100, "Pump Testing". IWV-1100, "Valve Testing"
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	1. a,d 2. a,b 3. b,c 4. b	IWV-1100, "Valve Testing".
16. Uses an isolation condenser containment design.	1. a,b,c, d 2. a,b 3. b,c 4. b	IWE-1220, "Components Exempt From Examination".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 5 ALWR IMPACTS ON ASME B&PV CODE SECTION XI, SUBSECTIONS BEYOND IWD		
GE SBWR	CATEGORY	ASME B&PV CODE SECTION XI
MAJOR DESIGN DIFFERENCES	POTENTIAL REVISIONS**	SECTIONS BEYOND IWD POTENTIALLY IMPACTED
17. Deleted standby service water system.	1. b 2. a,b 4. a,b	N/A
18. Plant design life extended to 60 years.	2. a,b 3. a,b,c, d,e 4. a,b	IWE-2411 and Table 2411-1, "Inspection Program A". IWP-1100, "Pump Testing". IWW-1100, "Valve Testing".

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION XI, POTENTIAL REVISION CATEGORIES

1. Safety Classification
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
2. Code Terminology
 - a. concepts
 - b. definitions
3. Design Life
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile
4. Ongoing Code Work
 - a. code committees
 - b. risk-based approaches

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**APPLICABILITY OF EXISTING CODES AND
STANDARDS TO ADVANCED REACTORS**

**ANALYSIS OF IEEE STANDARDS
279, 308, 415, and 603**

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Pacific Northwest Laboratory
Richland, Washington 99352

APPLICABILITY OF EXISTING CODES AND STANDARDS TO ADVANCED REACTORS

ANALYSIS OF IEEE STANDARDS 279, 308, 415, and 603

1.0 INTRODUCTION

This report provides an assessment of four Institute of Electrical and Electronics Engineers (IEEE) standards to determine their applicability to the design of advanced light water reactors (ALWRs). Five reactor designs were considered in the analysis of these four standards: the General Electric Advanced Boiling Water Reactor (GE-ABWR), the GE Simplified Boiling Water Reactor (GE-SBWR), the Combustion Engineering System 80+ (CE-80+), the Westinghouse AP600 (W-AP600), and the CANDU 3 Reactor.

This evaluation identifies, to the extent possible, the potential staff application of the following IEEE codes for advanced reactors:

- IEEE Std 279-1971, Criteria for Protection Systems for Nuclear Power Stations.

This older standard was intended to provide criteria to ensure the adequacy of safety-related reactor protection systems functional performance and reliability. But, there were several technical problems with this code version, one was that it did not include the actuator function as part of the systems in the scope of the standard. This code is now obsolete and is no longer endorsed by IEEE. Current IEEE Standards 308 and 603 replace IEEE Std 279 and expand the technical scope to include the actuator functions and they also include many other scope and technical improvements.

- IEEE Std 308-1991 Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

The scope of IEEE Standard 308 includes the Class 1E Power System portion of the reactor safety systems, both ac and dc power sources are included in this code. Sense and command features, execute features, and Class 1E power sources are part of the general elements of safety system criteria covered by this code. The 1991 version of this code is responsive to past NRC exceptions and is technically superior to previous versions.

- IEEE Std 415-1986 Guide for Planning of Preoperational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Stations.

IEEE Std 415 identifies the features of the Class 1E Power System that should be tested during the preoperational stage of the plant. The focus of the standard is on the test methods to achieve a quality testing program consisting of identified test objectives, ensuring that the test methods demonstrate those objectives, and that such demonstration is subject to the review and concurrence of others. Test methods identified in the standard are considered only good examples of

acceptable testing methods and the need for other tests is implied and guidance is furnished to help in the establishment of a test program.

- IEEE Std 603-1991 Standard Criteria for Safety Systems for Nuclear Power Generating Stations.

The scope of IEEE Standard 603 includes the design basis and safety system criteria for reactor safety systems, excluding the Class 1E portions that are covered by Std 308. Sense and command features, execute features, and non-Class 1E power sources are part of the general elements of safety system criteria covered by this code. This code is strongly linked to Std 308 and also is responsive to past NRC exceptions and current regulatory positions and is technically superior to previous versions of Std 603.

1.1 Review Methodology

These IEEE standards fall into a group of standards having direct application to advanced reactors. Since these codes were not developed for advanced reactor use, the staff will have to develop guidance on the application of these codes to make them fully useful. For this task a review of advanced reactor concepts and the code definitions, scope, references, and other detailed requirements was necessary to understand the application problems. Certain problems were discovered, as discussed in Section 2.0 of this letter report.

The review method used was to examine the general and specific electrical and instrumentation design features of the advanced reactors and to compare these findings to the detailed requirements of the four IEEE standards. This report is organized into 1.0 Introduction, 2.0 Discussion, and 3.0 Conclusions. The Conclusions Section contains PNL's recommendations for use of these standards. The Discussion Section lists the advanced reactor type and any affects on code application to that specific reactor design. The Discussion Section then reviews each code and highlights where the specific code text application problems chiefly occur.

Since advanced reactor applications are being considered, the latest code revisions were reviewed. Previous codes were also reviewed, where the staff cited exceptions to certain past code provisions, to determine the differences between the past and current code text changes and the affect on the specific exception(s) cited. In all cases reviewed, the latest codes were responsive to the staff exceptions cited, and in general were technically better than the older codes.

This analysis was based on available information regarding ALWR designs. More detailed design information on ALWR designs, when available and reviewed, may reveal additional information that could alter the identified potential impacts to the IEEE standards.

2.0 DISCUSSION

Evolutionary types of advanced reactors are understood for this review to mean those with ac power to the Class 1E buses. Examples of evolutionary reactor designs are the GE-ABWR, the CE-80+ and as a special case the CANDU 3. In passive reactor plant designs, sometimes referred to as revolutionary designs, the Class 1E power supply usually consists only of dc power from batteries and they have no ac power from diesel generators. Examples of passive reactor designs are the GE-SBWR and the W-AP600. These designs do have diesel generators (ac power) to add to their standby electric power options and to improve safety-in-depth design provisions. However, in these concepts, the diesel-generator ac power output is not connected to the Class 1E buses.

2.1 Standards Applicability

This part of the report is intended to discuss the reactor applications and the important code provisions for each code in this study and highlight those code details that have significant application use or difficulties for advanced reactors. IEEE Std 279-1971 was reviewed and was judged to be deficient and obsolete and is not recommended for future regulatory use for advanced reactors. It has been withdrawn by IEEE and replaced by two good and current codes, IEEE Standards 308 and 603. Based on this analysis, PNL did not believe that it was cost effective to list and discuss the details of the IEEE Std 279-1971 code and its applicability to each reactor type and such analysis was not performed.

2.1.1 Applicability to the ABWR

The IEEE 308, 415, and 603 group of standards can and should be applied to the ABWR evolutionary type of advanced reactor. Design differences exist in power and safety system details, relative amounts of emergency power needed to cope with emergencies, and even the number of expected operational events that would involve the Class 1E power system. Regardless of these differences, all provisions, references, or requirements of these IEEE standards can be applied to the ABWR design.

2.1.2 Applicability to the CE System 80+

The IEEE 308, 415, and 603 group of standards can and should be applied to the System 80+ advanced reactor. The CE System 80+ is an evolutionary reactor type of a large pressurized water reactor with advanced design concepts. Differences exist in electrical design and construction details. Regardless of these differences, all provisions, references, or requirements of these standards do apply to the CE System 80+ advanced reactor design.

2.1.3 Applicability to the SBWR

Certain provisions of the IEEE 308, 415, and 603 group of standards cannot be directly applied to the SBWR advanced reactor. The chief difference that exists in this revolutionary type design concept affecting electrical

standards is that the Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits. Because of these differences many of the provisions, definitions, references, or requirements of these IEEE standards that stipulate the Class 1E System cannot be applied to the SBWR. Most of Standard 603 applies to the SBWR with a few cautions as discussed later in this report.

This applicability limit for the revolutionary design concepts is unfortunate because the IEEE 308, 415, and 603 group of IEEE standards convey sound technical advice on overall safety system, ac, and standby power requirements. Using the intent of these standards, most provisions could be applied and are very useful for both regulation and industry applications. By ignoring the Class 1E or safety system designation in these codes and using the applicable technical provisions, one can extract concise and appropriate guidance. For example, the on-site diesel generator design criteria and testing guidelines are reasonably valid and most can still be directly applied to the revolutionary designs with minor adjustments.

2.1.4 Applicability to the W-AP600

Certain provisions of these standards cannot be directly applied to the W-AP600 revolutionary design advanced reactor. This design also affects application of these standards in that the Class 1E System is supplied by batteries and some reactor systems are considered to be non-safety systems for this design. Diesel generated standby power feeds only non-Class 1E circuits. Because of these differences the same comments and discussion as that for the SBWR applies, including the use of Standard 603 for W-AP600 applications.

2.1.5 Applicability to the CANDU Reactor

The provisions of the IEEE 308, 415, and 603 group of standards can be directly applied to the CANDU-3 advanced reactor. These standards will have some different applications for the CANDU-3. For this design the principal application affects for these standards is that the Class 1E System is somewhat different from typical U.S. designs. For example, the Group 1 equipment, including the main reactor control room, are not included in the Class 1E System. An auxiliary Group 2 equipment reactor control room is connected to the Class 1E system. Thus, some reactor systems that are usually considered to be part of the Class 1E System are non-safety systems in the CANDU design. Diesel generated standby power does feed the Class 1E circuits in this design. Because of the many design differences, the provisions of these codes need to be applied carefully to the CANDU design details. Regardless of these design differences, these three codes do work for the CANDU design.

2.2 DISCUSSION OF STANDARDS

2.2.1 IEEE Std 279-1971, Criteria for Protection Systems for Nuclear Power Stations.

This code has been withdrawn by IEEE and does not reflect current staff or industry technical positions. The current IEEE Standards 308 and 603 replace this code, provide better criteria, and should be cited instead of Standard 279. The newer standards are in agreement with current regulatory staff philosophy and offer improved technical provisions for power and safety systems applications.

2.2.2 IEEE Std 308-1991, Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

This standard is a general revision of IEEE Std 308-1980 and represents the fifth generation of criteria documents tracing back through 1970, the original issue of 308, to IEEE Std 279-1968. IEEE Standard 308 together with Standard 603 and other IEEE standards referenced in these two documents, present a good set of criteria applicable to current operating reactor types. These standards are current, and the code committees have been responsive to past NRC concerns regarding certain provisions in the previous issues of these standards.

As a result, this IEEE Std 308-1991 code has no application difficulties for evolutionary advanced reactor types with ac power to the 1E System buses. This is not true for the passive type of advanced reactors. For reactors with only dc power to the 1E System buses, only the dc system criteria apply. However, diagrams, Class 1E descriptions, and other information make the direct application of this code difficult even for dc system use on passive type reactors. Without ac power to the Class 1E System buses, a direct literal application of this code is not possible. An example is paragraph 6.1.1. describing the essential component parts of the Class 1E System, which includes an alternating current power system component. IEEE Std 308-1991 can be applied to the GE-SBWR advanced reactor which has no ac power to the Class 1E System buses by using Table 1. Table 2 applies to the W-AP600 reactor, which has the same absence of ac power to the Class 1E System.

TABLE 1

GE-SBWR Reactor Design	IEEE Std 308-1991 Application	
MAJOR DESIGN DIFFERENCES	CODE SECTION IMPACTED	EXPLANATION OF IMPACT TO CODE SECTION
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	1. Scope. Includes: (1) Alternating current power systems. Implies that ac power is part of system design.	Exception needed: To state that ac power does not apply to this design, and scope inclusion can be ignored.
Same design difference	Table 1. Some scope items do not apply to this reactor design.	Exception needed: To note that standby generators and transformers (3) do not apply to this design.
Same design difference	Figure 1. Much of this example diagram does not apply to the SBWR design.	Exception needed: To note that only the dc parts apply in Figure 1 to the SBWR design.
Same design difference	4. Definitions. Standby power supply. Typo. Should say: The on-site power supply... . This definition can not be used.	Exception needed: It may correct the definition typo and note that the dc system batteries are the only standby power source for the Class 1E system.
Same design difference	Figure 2. The ac power illustrations on this diagram do not apply to the SBWR.	Exception needed: To note that the ac part does not apply to this design.
Same design difference	Figure 3. The ac power illustrations on this diagram do not apply to the SBWR design.	Exception needed: To note that the ac power part does not apply to this design.
Same design difference	5.4 Design Basis. All of (5), (7) frequency, and parts of (8) (such as standby power supply unit start-up time,) do not apply to design.	Exception needed: To state that these ac power elements do not apply to this passive reactor design.

TABLE 1. Continued

GE-SBWR Reactor Design	IEEE Std 308-1991 Application	
MAJOR DESIGN DIFFERENCES	CODE SECTION IMPACTED	EXPLANATION OF IMPACT TO CODE SECTION
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	5.5 Power Quality. Frequency and waveform do not apply to the dc system design for the Class 1E power system of the SBWR design.	Exception needed: To state that these ac power elements do not apply to the SBWR design.
Same design difference	6.1.1 Description. The Class 1E System includes an ac component description.	Exception needed: To delete ac power elements from the description.
Same design difference	6.1.3 Interaction. This provision for EDG testing does not apply.	Exception needed: To state that this ac power testing paragraph is not applicable to the SBWR.
Same design difference	6.2 Alternating Current Power Systems. This whole section on alternating current power can not be applied to the SBWR passive reactor.	Exception needed: To note that this whole ac power testing section is not applicable to these reactor design concepts.
Same design difference	6.4.1 General. Design Requirement (4) for two or more independent ac power supplies is not applicable.	Exception needed: To state that Design Requirement (4) is not needed for the SBWR design.
Same design difference	6.4.5 AC Supply. Power supply to the I&C power system does not have an ac distribution component for the SBWR.	Exception needed: This whole ac power testing sub-section is not applicable to the design for this reactor.

TABLE 1. Continued

GE-SBWR Reactor Design	IEEE Std 308-1991 Application	
MAJOR DESIGN DIFFERENCES	CODE SECTION IMPACTED	EXPLANATION OF IMPACT TO CODE SECTION
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	Table 3. Illustrative Surveillance Methods. The diesel generator is not Class 1E equipment for this reactor design.	Exception needed: To note this EDG surveillance is not applicable to the design and the ac elements under the battery charger should be retained.
Same design difference	7.3 Preoperation System Test. Test requirements (1), (3), and (5) were written to apply to ac and EDG system components and do not apply.	Exception needed: These three tests can be changed to apply to the dc power system. But, they are not really needed. Deletion recommended.
	8. Multiunit Stations. All of Section 8 is a regulatory problem for the SBWR design.	Exception needed: Recommend a deletion of all of 8.0 for SBWR use. Part 8.4 as revised in this code version, may be acceptable as is, but, it has little practical value.

TABLE 2

AP600 Reactor Design	IEEE Std 308-1991 Application	
MAJOR DESIGN DIFFERENCES	CODE SECTION IMPACTED	EXPLANATION OF IMPACT TO CODE SECTION
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	1. Scope. Includes: (1) Alternating current power systems. Implies that ac power is an essential part of system design.	Exception needed: To state that ac power does not apply to this design, and scope inclusion can be ignored.
Same design difference	Table 1. Some scope items do not apply to this reactor design.	Exception needed: To note that standby generators and transformers (3) do not apply to this design.
Same design difference	Figure 1. Much of this example diagram does not apply to the AP600 design.	Exception needed: To note that only the dc parts apply in Figure 1 to the AP600 design.
Same design difference	4. Definitions. Standby power supply. Typo. Should say: The on-site power supply... . This definition can not be used.	Exception needed: It may correct the definition typo and note that the dc system batteries are the only standby power source for the Class 1E system.
Same design difference	Figure 2. The ac power illustrations on this diagram do not apply to the AP600 design	Exception needed: To note that the ac part does not apply to this design.
Same design difference	Figure 3. The ac power illustrations on this diagram do not apply to the AP600 design.	Exception needed: To note that the ac power part does not apply to this AP600 reactor design.
Same design difference	5.4 Design Basis. All of (5), (7) frequency, and parts of (8) (such as standby power supply unit start-up time,) do not apply to this design.	Exception needed: To state that these ac power elements do not apply to this passive reactor design.

TABLE 2 Continued

AP600 Reactor Design	IEEE Std 308-1991 Application	
MAJOR DESIGN DIFFERENCES	CODE SECTION IMPACTED	EXPLANATION OF IMPACT TO CODE SECTION
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	5.5 Power Quality. Frequency and waveform do not apply to the dc system design for the Class 1E power system of the AP600 reactor design.	Exception needed: To state that these ac power elements do not apply to the AP600 design.
Same design difference	6.1.1 Description. The Class 1E System includes an ac component description.	Exception needed: To delete ac power elements from the description.
Same design difference	6.1.3 Interaction. This provision for EDG testing does not apply.	Exception needed: To state that this ac power testing paragraph is not applicable to the AP600.
Same design difference	6.2 Alternating Current Power Systems. This whole section on alternating current power can not be applied to the AP600 reactor.	Exception needed: To note that this whole ac power testing section is not applicable to this reactor design concept.
Same design difference	6.4.1 General. Design Requirement (4) for two or more independent ac power supplies is not applicable.	Exception needed: To state that Design Requirement (4) is not needed for the AP600 design.
Same design difference	6.4.5 AC Supply. Power supply to the I&C power system does not have an ac distribution component for the AP600.	Exception needed: This whole ac power testing sub-section is not applicable to the design for this reactor.

TABLE 2 Continued

AP600 Reactor Design	IEEE Std 308-1991 Application	
MAJOR DESIGN DIFFERENCES	CODE SECTION IMPACTED	EXPLANATION OF IMPACT TO CODE SECTION
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	Table 3. Illustrative Surveillance Methods. The diesel generator is not Class 1E equipment for this reactor design.	Exception needed: To note this EDG surveillance is not applicable to the design and the ac elements under the battery charger should be retained.
Same design difference	7.3 Preoperation System Test. Test requirements (1), (3), and (5) were written to apply to ac and EDG system components and do not apply.	Exception needed: These three tests can be changed to apply to the dc power system. But, they are not really needed. Deletion recommended.
Same design difference	8. Multiunit Stations. All of Section 8 is a regulatory problem for the AP600 design.	Exception needed: Recommend a deletion of all of 8.0 for AP600 use. Part 8.4 as revised in this code version, may be acceptable as is, but, it has little practical value.

2.2.3 IEEE Std 415-1986, Guide for Planning of Preoperational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Stations.

This revision of IEEE Std 415-1976 identifies the features of the Class 1E Power System that should be tested during the preoperational stage of the plant. This document presents good test program requirements applicable to current operating reactor types. As a result, this code has no application difficulties for evolutionary advanced reactor types (GE-ABWR, CE-80+, and CANDU) with ac power to the 1E System buses.

This is not true for the passive advanced reactors (GE-SBWR and W-AP600). For reactors with only dc power to the 1E System buses, only the dc system criteria applies. However, diagrams, Class 1E descriptions, and other information make the direct application of this code difficult for these dc system applications. Without ac power to the Class 1E System buses, a direct literal application of this code is not possible, because some of the code requirements are based on an ac power component. Direction is needed on how to handle the intertwined ac and dc standards information.

IEEE Std 415-1986 can be applied to advanced reactors without ac power to the Class 1E System buses by using Tables 3 and 4.

2.2.4 IEEE Std 603-1991, Standard Criteria for Safety Systems for Nuclear Power Generating Stations.

This standard is a revision of IEEE Std 603-1980 and represents the fifth generation of systems criteria documents tracing back through 1977. the original issue of 603, to IEEE Std 279-1968. IEEE Standards 308 and 603, together with other IEEE standards referenced in these two documents, now define a consistent set of design criteria for current light water reactors.

As a result, this code has no application difficulties for evolutionary advanced reactor types. This is also true for passive advanced reactors. However, diagrams, Class 1E references, and other information need to be used properly to ensure that the direct application of this code achieves the intended safety goals. To illustrate, the representative examples in Figure 3 may not always apply such as the "diesel start solenoid". Elsewhere in the document, safety systems common to current reactors may not exist in passive advanced reactors or they may not be a safety system.

The primary application problem seems to be that for some passive reactor designs, some current and conventional systems applications for this code are excluded because they are no longer defined as safety systems. Dealing with this technical reality may be a minor problem in that interfaces with non-safety systems are defined and are entirely acceptable. Another possible solution is to retain the criteria to improve confidence in safety, and by a regulatory exception define how IEEE Std 603 is to be applied to these non-safety systems.

TABLE 3

GE-SBWR Reactor Design	IEEE Std 415-1986, Application	
Major Design Differences	Code Section Impacted	Explanation of Impact to Code Section
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	1. Scope. Power systems are defined to include ac supplies. Implies that ac power is an essential part of system design.	Exception needed: To define the Class 1E system so as to include dc power supplies only for the SBWR application.
Same design difference	6.2 Preoperational Tests of Medium-Voltage Class 1E Entire Section applies to testing of ac power systems that do not apply to this reactor design.	Exception needed: To state that this section is not applicable to the SBWR design.
Same design difference	6.3 Preoperational Tests of Low-Voltage Class 1E Section 6.3 refers to the testing of transformers, frequency, phases, and other ac power components as well as useful dc components.	Exception needed: To list the ac components that do not apply, or to define how to handle these non-applicable ac power components for the Class 1E System for the SBWR.
Same design difference	6.5 Preoperational Tests for Standby Generators. Standby generators are not in the Class 1E System for this design.	Exception needed: To show that these tests need to be performed to enhance safety, even though the EDG is not part of the Class 1E system. Need to define the use of these tests for the SBWR application.

TABLE 3 Continued

GE-SBWR Reactor Design	IEEE Std 415-1986, Application	
Major Design Differences	Code Section Impacted	Explanation of Impact to Code Section
Same design difference	6.6 Integrated Systems Functional Tests. Section 6.6 refers to transformers, frequency, phases, standby generator and other ac power components not in the Class 1E System for the SBWR as well as useful dc components.	Exception Needed: These tests need to be defined with care for the SBWR. Some ac system components may be involved in these tests (during loss of preferred power tests). It appears that 6.6.2.1 and 6.6.2.2 tests are not useful for the SBWR reactor design.

TABLE 4

W-AP600 Reactor Design	IEEE Std 415-1986, Application	
Major Design Differences	Code Section Impacted	Explanation of Impact to Code Section
Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.	1. Scope. Power systems are defined to include ac supplies in the Class 1E System. Does not apply to the AP600 reactor.	Exception needed: To define the Class 1E System to include dc power supplies only for AP600 applications. May need to define the special use of EDG tests for the SBWR application.
Same design difference	6.2 Preoperational Tests of Medium-Voltage Class 1E Entire section applies to testing of ac power supplies that do not apply to the AP600 reactor design.	Exception needed: To state that the 6.2 Section is not applicable to the AP600 design.
Same design difference	6.3 Preoperational Tests of Low-Voltage Class 1E This section refers to tests of transformers, frequency, phases, and other non-applicable ac power components as well as useful tests for dc components.	Exception needed: To list the ac components that do not apply, or to define how to handle these non-applicable ac power components.
Same design difference	6.5 Preoperational Tests for Standby Generators. Standby generators are not a Class 1E component for the AP600 design.	Exception needed: These tests need to be performed to enhance safety, even though the EDG is not part of the Class 1E system. Need to define the use of these tests.

TABLE 4 Continued

W-AP600 Reactor Design	IEEE Std 415-1986, Application	
Major Design Differences	Code Section Impacted	Explanation of Impact to Code Section
<p>Class 1E System consists only of dc power furnished by batteries. Diesel generated standby power feeds only non-Class 1E circuits.</p>	<p>6.6 Integrated Systems Functional Tests. This section refers to transformers, frequency, phases, standby generator and other non-applicable ac power components as well as dc components used in the AP600 design.</p>	<p>Exception Needed: These tests need to be defined with care. Some ac system components may be involved in these tests (during loss of preferred power tests). It appears that 6.6.2.1 and 6.6.2.2 tests are not useful for the AP600 reactor design.</p>

3.0 CONCLUSIONS

The IEEE Std 279-1971 code contains many technical provisions that do not reflect current staff or industry positions. This code has been withdrawn and should not be cited. It is not well suited for advanced reactor use because of the many exceptions that would be necessary to define current staff positions.

The IEEE 308, 415, and 603 group of standards do appear to be acceptable for evolutionary types of advanced reactors as written, assuming current NRC requirements are applied. The two 1991 standards (IEEE 308 and 603) contain technical improvements to address aging concerns and regulatory and industry consensus on newer technical requirements. These newer standard revisions especially appear to meet regulatory and industry needs for evolutionary advanced reactors.

For passive reactor plant designs, also referred to as revolutionary designs in this report, problems in applying the provisions of these standards were discussed. Without ac power to the Class 1E bus, parts of the scope, references, design criteria provisions, safety classifications, and other ac power provisions of these standards cannot be directly applied to the passive reactor designs. The defined "Safety Systems" for this design type also are different. This excludes the direct code application of criteria for certain of these non-safety systems. However, this situation is not serious and with proper instructions and regulatory guidelines at least parts of these codes may be utilized by the staff. Tables 1 through 4 defined the application of these codes.

These IEEE standards address general design criteria and provisions, testing, and surveillance found nowhere else in such a concise document. For passive reactors and reactor types with only dc powered Class 1E System buses, it is recommended that the staff use the general intent and existing applicable provisions, as discussed in Tables 1 through 4, that could be applied in the near term until industry has the time and information needed to change these IEEE standards. The industry consensus process to change IEEE standards usually requires three to five years.

A conceivable short term solution recommended is for the regulatory staff to develop written guidance on how to apply the useful provisions in these standards to revolutionary advanced reactors and to the new set of non-safety systems. Other guidance should be developed to describe how to address certain LOCA and Class 1E provisions and references to ac power that would not apply. Various existing NRC document formats could be used to convey this information.

3.1 IEEE Std 279-1971

Criteria for Protection Systems for Nuclear Power Stations.

This code has been withdrawn and should not be cited. This code contains many provisions that do not reflect current staff or industry

technical positions. It is poorly suited for advanced reactor use because of the many exceptions that would be necessary to document current staff positions. The current IEEE Standards 308 and 603 replace this code and generally follow staff positions. IEEE Standards 308 and 603 provide criteria more closely aligned with current regulatory staff philosophy and improved technical provisions for power and safety systems applications.

3.2 IEEE Std 308-1991

Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations.

This code may be cited and used for advanced reactors as published. For some passive reactor designs, some application difficulties occur. These were discussed in this report in Tables 1 and 2.

3.3 IEEE Std 415-1986

Guide for Planning of Preoperational Testing Programs for Class 1E Power Systems for Nuclear Power Generating Stations.

This code may be cited and used for advanced reactors as published. For some passive reactor designs, some application difficulties occur. These were discussed in this report in Tables 3 and 4.

3.4 IEEE Std 603-1991

Standard Criteria for Safety Systems for Nuclear Power Generating Stations.

This code may be applied to advanced reactors as published. For some passive reactor designs, some potential "important to safety" systems applications are not always included because they are no longer defined as "Safety Systems". It is recommended that this code be applied to both the evolutionary designs and the passive advanced reactor designs.

APPLICABILITY OF SELECTED ASME SECTION III
SUBSECTIONS TO ADVANCED REACTORS

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1.0 INTRODUCTION

This report provides an assessment of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC), Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components" to determine the applicability of selected Section III Subsections to the design and construction of advanced light water reactors (ALWRs). Four ALWR designs are considered in this analysis: the General Electric Advanced Boiling Water Reactor (GE ABWR), GE Simplified Boiling Water Reactor (GE SBWR), Combustion Engineering System 80+ (CE 80+), and the Westinghouse AP600.

This evaluation identifies, to the extent possible, the potential need for modification or revisions to ASME B&PVC, Section III, Division 1. A study was performed for Subsections NCA, "General Requirements for Division 1 and Division 2", NB, "Class 1 Components", NC, "Class 2 Components", and ND, "Class 3 Components" to identify potential changes.

Section 2 of this report discusses the analysis approach and applications. Conclusions are provided in Section 3.

2.0. DISCUSSION

The assessment involves two basic steps. The first step identifies and summarizes the unique features or operating modes of the four advanced light water reactor designs being considered. The second step reviews the current ASME Section III, Subsections NCA, NB, NC, and ND for applicability to the ALWRs and identifies potential changes needed to address the ALWRs unique design features. The evaluation is focused on ASME Section III. Details involved in performing these steps are described below.

2.1. Unique Features of the ALWRs

The major changes in ALWR designs involve two aspects: (1) the advanced reactors incorporate passive safety system concepts into their designs for preventing and responding to potential upset and accident conditions, and (2) the design life of the advanced reactor plants is increased to 60 years. Use of passive safety systems allows some systems and components classified as safety-related in current light water reactor designs to be downgraded to non-safety related in the advanced reactor designs. The following summarizes the unique features of the ALWRs.

1. GE ABWR

- a. Added an Automatic Power Regulator to control reactor power in startup and operation by changing rod positions and reactor circulation flow.
- b. Added ability to flood the lower drywell with water from the suppression pool for response to severe accidents where core melt could cause vessel failure.
- c. Increased the scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume. The scram discharge volume is primary water which was released into a tank.
- d. Replaced the jet pumps with canned motor pumps internal to the reactor vessel for forced coolant circulation.
- e. Extends the design life of the plant to 60 years.

2. GE SBWR

- a. Eliminates large nozzle connections to the reactor vessel below the top of the active fuel level. Based on enhanced natural circulation capabilities this allows elimination of forced coolant circulation, coolant circulation piping loops external to the vessel, and deletion of the recirculation flow control system.

- b. Reactor vessel uses forged rings that results in 50% reduction of vessel welds.
- c. Increased the scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume. The scram discharge volume is primary water which was released into a tank.
- d. Accumulators were added to the standby liquid control system and the standby liquid control pumps were eliminated.
- e. Main steam is non-reheat cycle and main feedwater uses adjustable speed electric motor-driven pumps.
- f. Reactor water cleanup/shutdown cooling system purifies reactor coolant during operation and shutdown and provides spent fuel pool cooling functions using sealless pumps.
- g. Process instrument and control systems use multiplexed solid state fiber optics where practical and the process computer performs control functions.
- h. Primary containment is a cylindrical reinforced concrete design with a suppression pool similar to the Mark III BWR-6 design with passive containment cooling provided by the suppression pool, isolation condenser, and gravity driven cooling system.
- i. The standby gas treatment system is deleted.
- j. Neutron monitoring systems use a startup range monitor with auto insert and withdrawal during power maneuvering, eliminating the intermediate range monitor, and incorporates an automatic fixed incore probe that replaces the traversing incore probe.
- k. Uses an Automatic Power Regulator to control reactor power in startup, power operation, and reactor shutdown by appropriate changes to rod positions.
- l. Eliminates need for ac power for Engineered Safety Features (ESF) actuation and uses vital dc power with 72 hour capacity and four divisions of controls and instrumentation power during loss of ac power and LOCA. Non-safety grade diesel generator is provided and can be used for dc power.
- m. Uses automatic depressurization relief through four safety/relief valves and six squib valves for depressurization which allows gravity driven cooling system to flood containment through squib valves to an elevation one meter above the top of active fuel for indefinite post LOCA cooling. The high pressure core spray system, low pressure core spray

system, residual heat removal system, and standby service water system are deleted.

- n. Extends the design life of the plant to 60 years.

3. CE 80+

- a. Incorporates the in-containment refueling water storage tank (IRWST) concept which serves as the initial source of water for ECCS and as a quench tank for releases from the pressurizer safety valves and the safety depressurization system, and eliminates the recirculation actuation signal.
- b. Added two additional pumps which are dedicated to the shutdown cooling mode of operation. This eliminates the need for using the low pressure safety injection pumps for this purpose.
- c. Uses four high pressure safety injection pumps for ECCS. This eliminates two low pressure safety injection pumps.
- d. Reclassifies the chemical and volume control system as a non-safety system based upon safe shutdown functions being performed by the safety injection system.
- e. Extends the design life of the plant to 60 years.

4. AP 600

- a. Uses canned motor pumps for reactor coolant pumps.
- b. Uses an automatic depressurization system to depressurize the reactor coolant system during some accidents to allow passive water injection into the reactor core.
- c. Incorporates the in-containment refueling water storage tank (IRWST) as a large volume of water stored above the reactor core available for passive response to LOCAs upon RCS depressurization.
- d. Eliminates the ECCS function of the residual heat removal system and retains its shutdown and refueling cooling functions.
- e. Adds two core makeup tanks inside containment which float at RCS pressure and provide a quick passive source of water during accidents upon depressurization. This allows the safety function of the chemical and volume control system to be eliminated.
- f. Auxiliary feedwater pumps are eliminated based on use of the passive residual heat removal system which routes primary

water through the in-containment refueling water storage tank during some accidents to cool the primary water.

- g. During some accidents, the AP600 uses a large steel containment shell with passive external cooling from air and water which transfers heat to the ambient atmosphere (ultimate heat sink) and results in reactor coolant being condensed on the inside containment shell and recycled for cooling the reactor core.
- h. Uses safety grade dc power to actuate and reposition some valves for passive systems response which allows downgrading ac power systems to non-safety grade including the standby diesel generators.
- i. Extends the design life of the plant to 60 years.

2.2. Review of ASME Section III and Analysis Approach

1. Overview of the ASME Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components" review for ALWRs:
 - a. A preliminary review of the 1992 Edition, 1993 Addenda, of ASME, Section III, Division 1 was performed. Subsections NCA, NB, NC, and ND that are applicable to the safety related mechanical systems and components affected by the unique ALWR features were considered.
 - b. The ASME portion of the review does not include ASME Section III, Division 2, "Code for Concrete Reactor Vessels and Containments", ASME Section III Division 1 Subsections NE, "Class MC Components", NF, "Component Supports", NG, "Core Support Structures", or electrical, and instrumentation and control systems.
2. Analysis approach and assumptions for identification of potential impacts of unique ALWR features on ASME Section III:
 - a. The analysis is performed to identify potential impacts on ASME Section III to address unique features of the ALWR designs. The approach includes:
 - 1) Identification of the unique features for each ALWR design considered as presented in section 2.1 above.
 - 2) Development of categories for potential revisions to ASME Section III based on the nature of the identified ALWR unique features and ASME Section III structure and content.

The ALWR Standard Safety Analysis Report Design Descriptions were reviewed to identify the major differences between these designs and currently licensed light water reactor designs.

- 3) Identification of the individual articles, subarticles, and subsubarticles that might be affected to address the identified ALWR unique features.
 - 4) Presenting a brief description of the rationale for the potential revisions identified.
- b. The criteria used in performing this analysis include:
- 1) ALWR unique features that interface directly with the reactor coolant pressure boundary and involve a new concept not specifically addressed in ASME Section III create a candidate potential impact.
 - 2) ALWR unique features which involve downgrade of previously safety related systems based upon reliance on passive safety systems create a candidate potential impact.
- c. Assumptions and limitations for this analysis include:
- 1) ALWR unique features were considered conceptually, and subjective judgement was used conservatively to identify potential impacts to ASME Section III. When detailed design specifications for the unique ALWR features are available, the need for some identified potential revisions to ASME Section III may need to be reconsidered.
 - 2) ALWR unique features that involve extended design lives and design concepts which depart from the characteristics of active safety systems may raise degradation or failure mechanisms which were not previously relevant or significant. As such failure mechanisms become known, there may be a need for additional changes to the inspection and testing requirements or acceptance criteria to appropriately address these degradation and failure mechanisms.
3. Applications were performed for Subsections NCA, NB, NC, and ND. Results of the applications are summarized in Tables 1, 2, 3, and 4.
- a. The analysis process was performed for ASME Section III, Division 1, Subsection NCA, "General Requirements for Division 1 and Division 2". This review did not result in the identification of any potential impacts.

- b. The analysis process was performed for ASME Section III, Division 1, Subsection NB, "Class 1 Components". This review resulted in identification of potential impacts to one article, subarticle, or subsubarticle as follows:
 - 1) NB-2540, "Examination and Repair of Forgings and Bars".
- c. The analysis process was performed for ASME Code Section III, Division 1, Subsection NC, "Class 2 Components". This review resulted in identification of potential impacts to two articles, subarticles, or subsubarticles as follows:
 - 1) NC-3900, "0-15 psig Storage Tank Design".
 - 2) NC-6600, "Special Test Pressure Situations".
- d. The analysis process was performed for ASME Section III, Division 1, Subsection ND, "Class 3 Components". This review resulted in identification of potential impacts to two articles, subarticles, or subsubarticles as follows:
 - 1) ND-3900, "0-15 psig Storage Tank Design".
 - 2) ND-6600, "Special Test Pressure Situations".

3.0. SUMMARY AND CONCLUSIONS

The preliminary results show that the existing provisions of ASME Section III are generally applicable to the unique design and operating features of the ALWR designs considered. For the ASME Sections selected for study (NCA, NB, NC, and ND), the results, as shown in Tables 1, 2, 3, and 4 indicate the need for only limited, if any, changes in ASME Section III.

Articles, subarticles, and subsubarticles within ASME Section III may need to be revised in order to completely address the unique features of the ALWRs. ASME Section III may be revised to address the design changes from active to passive safety systems and potential for different failure mechanisms, and the increase in plant design life (60 years).

This preliminary analysis was based on available information regarding ALWR designs. Detailed design information on certified ALWR designs, when available and reviewed, may reveal additional information that could alter or increase the identified potential impacts to ASME Section III.

4.0. REFERENCES

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components", 1992 Edition, 1993 Addenda.

TABLE 1 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NCA			
WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NCA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used as reactor coolant pumps.	N/A	N/A	N/A
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	N/A	N/A	N/A
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	N/A	N/A	N/A
4. Residual heat removal system does not provide ECCS function but is used for shutdown and refueling cooling.	N/A	N/A	N/A
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	N/A	N/A	N/A
6. Chemical and volume control system including charging and primary make-up pumps downgraded to non-safety based on CMT addition along with other changes.	N/A	N/A	N/A
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	N/A	N/A	N/A
8. Removed the auxiliary feedwater pumps based on passive RHR function.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NCA			
WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NCA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	N/A	N/A	N/A
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	N/A	N/A	N/A
11. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NCA			
GE ABWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NCA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	N/A	N/A	N/A
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	N/A	N/A	N/A
3. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
4. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NCA			
CE SYSTEM 80+			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NCA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and depressurization system.	N/A	N/A	N/A
2. Eliminated the recirculation actuation signal based on adding IRWST.	N/A	N/A	N/A
3. Two pumps dedicated to shutdown cooling operation.	N/A	N/A	N/A
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	N/A	N/A	N/A
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	N/A	N/A	N/A
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	N/A	N/A	N/A
7. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NCA			
GE SBWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NCA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	N/A	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	N/A	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	N/A	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	N/A	N/A	N/A
5. Eliminates reactor core isolation cooling system.	N/A	N/A	N/A
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	N/A	N/A	N/A
7. Deleted standby gas treatment system.	N/A	N/A	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	N/A	N/A	N/A
9. Vital AC power is provided by two DC busses with 72 hour battery capacity.	N/A	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	N/A	N/A	N/A
11. High pressure core spray is eliminated.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 1 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NCA			
GE SBWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NCA SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	N/A	N/A	N/A
13. Eliminated low pressure core spray.	N/A	N/A	N/A
14. Eliminated residual heat removal emergency core cooling system function.	N/A	N/A	N/A
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	N/A	N/A	N/A
16. Uses an isolation condenser containment design.	N/A	N/A	N/A
17. Eliminated standby service water system.	N/A	N/A	N/A
18. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
19. Accumulators were added to the standby liquid control system, and its standby pumps were eliminated.	N/A	N/A	N/A
20. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION III, POTENTIAL REVISION CATEGORIES

- 1. Safety Classification**
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
- 2. Code Terminology**
 - a. concepts
 - b. definitions
- 3. Design Life**
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile

N/A - This design difference does not impact this Subsection

*** As compared to closest current licensed LWR design.**

**** See potential revision categories at the end of this table.**

**TABLE 2 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NB
WESTINGHOUSE AP600**

MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NB SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used as reactor coolant pumps.	N/A	N/A	N/A
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	N/A	N/A	N/A
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	N/A	N/A	N/A
4. Residual heat removal system does not provide ECCS function but is used for shutdown and refueling cooling.	N/A	N/A	N/A
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	N/A	N/A	N/A
6. Chemical and volume control system including charging and primary make-up pumps downgraded to non-safety based on CMT addition along with other changes.	N/A	N/A	N/A
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	N/A	N/A	N/A
8. Removed the auxiliary feedwater pumps based on passive RHR function.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NB			
WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NB SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	N/A	N/A	N/A
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	N/A	N/A	N/A
11. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NB			
GE ABWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NB SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	N/A	N/A	N/A
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	N/A	N/A	N/A
3. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
4. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NB			
CE SYSTEM 80+			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NB SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and depressurization system.	N/A	N/A	N/A
2. Eliminated the recirculation actuation signal based on adding IRWST.	N/A	N/A	N/A
3. Two pumps dedicated to shutdown cooling operation.	N/A	N/A	N/A
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	N/A	N/A	N/A
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	N/A	N/A	N/A
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	N/A	N/A	N/A
7. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NB			
GE SBWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NB SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	N/A	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	2.a	NB-2540, "Examination and repair of Forgings and Bars"	NB-2540 may require revision to address reactor vessel forged rings.
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	N/A	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	N/A	N/A	N/A
5. Eliminates reactor core isolation cooling system.	N/A	N/A	N/A
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	N/A	N/A	N/A
7. Deleted standby gas treatment system.	N/A	N/A	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	N/A	N/A	N/A
9. Vital AC power is provided by two DC busses with 72 hour battery capacity.	N/A	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	N/A	N/A	N/A
11. High pressure core spray is eliminated.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 2 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NB**GE SBWR**

MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NB SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	N/A	N/A	N/A
13. Eliminated low pressure core spray.	N/A	N/A	N/A
14. Eliminated residual heat removal emergency core cooling system function.	N/A	N/A	N/A
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	N/A	N/A	N/A
16. Uses an isolation condenser containment design.	N/A	N/A	N/A
17. Eliminated standby service water system.	N/A	N/A	N/A
18. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
19. Accumulators were added to the standby liquid control system, and its standby pumps were eliminated.	N/A	N/A	N/A
20. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION III, POTENTIAL REVISION CATEGORIES

- 1. Safety Classification**
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
- 2. Code Terminology**
 - a. concepts
 - b. definitions
- 3. Design Life**
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile

N/A - This design difference does not impact this Subsection

*** As compared to closest current licensed LWR design.**

**** See potential revision categories at the end of this table.**

TABLE 3 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NC			
WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NC SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used as reactor coolant pumps.	N/A	N/A	N/A
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	N/A	N/A	N/A
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	1. a, 2. a,b	NC-3900, "0-15 psig Storage Tank Design" NC-6600, "Special Test Pressure Situations"	NC-3900 may require revision to address the IRWST. However, the IRWST will probably be designed per NC-3100, "General Design", NC-3200, "Alternative Design Rules for Vessels" or NC-3300, "Vessel Design", if it is of metal construction. NC-6600 may require revision due to the combined internal and external design loading of the IRWST.
4. Residual heat removal system does not provide ECCS function but is used for shutdown and refueling cooling.	N/A	N/A	N/A
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	N/A	N/A	N/A
6. Chemical and volume control system including charging and primary make-up pumps downgraded to non-safety based on CMT addition along with other changes.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 3 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NC			
WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NC SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	N/A	N/A	N/A
8. Removed the auxiliary feedwater pumps based on passive RHR function.	N/A	N/A	N/A
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	N/A	N/A	N/A
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	N/A	N/A	N/A
11. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 3 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NC			
GE ABWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NC SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	N/A	N/A	N/A
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	N/A	N/A	N/A
3. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
4. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 3 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NC			
CE SYSTEM 80+			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NC SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and depressurization system.	1. a, 2. a,b	NC-3900, "0-15 psig Storage Tank Design" NC-6600, "Special Test Pressure Situations"	NC-3900 may require revision to address the IRWST. However, the IRWST will probably be designed per NC-3100, "General Design", NC-3200, "Alternative Design Rules for Vessels" or NC-3300, "Vessel Design", if it is of metal construction. NC-6600 may require revision due to the combined internal and external design loading of the IRWST.
2. Eliminated the recirculation actuation signal based on adding IRWST.	N/A	N/A	N/A
3. Two pumps dedicated to shutdown cooling operation.	N/A	N/A	N/A
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	N/A	N/A	N/A
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	N/A	N/A	N/A
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	N/A	N/A	N/A
7. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 3 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NC**GE SBWR**

MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NC SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	N/A	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	N/A	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	N/A	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	N/A	N/A	N/A
5. Eliminates reactor core isolation cooling system.	N/A	N/A	N/A
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	N/A	N/A	N/A
7. Deleted standby gas treatment system.	N/A	N/A	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	N/A	N/A	N/A
9. Vital AC power is provided by two DC busses with 72 hour battery capacity.	N/A	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	N/A	N/A	N/A
11. High pressure core spray is eliminated.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 3 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION NC			
GE SBWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	NC SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	N/A	N/A	N/A
13. Eliminated low pressure core spray.	N/A	N/A	N/A
14. Eliminated residual heat removal emergency core cooling system function.	N/A	N/A	N/A
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	N/A	N/A	N/A
16. Uses an isolation condenser containment design.	N/A	N/A	N/A
17. Eliminated standby service water system.	N/A	N/A	N/A
18. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
19. Accumulators were added to the standby liquid control system, and its standby pumps were eliminated.	N/A	N/A	N/A
20. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION III, POTENTIAL REVISION CATEGORIES

- 1. Safety Classification**
 - a. passive safety systems and components
 - b. deleted active safety systems and components
 - c. downgraded systems and components
 - d. testing requirements for passive systems and components
- 2. Code Terminology**
 - a. concepts
 - b. definitions
- 3. Design Life**
 - a. aging and failure mechanisms
 - b. inspection frequency and intervals
 - c. maintenance bases and frequency
 - d. plant life extension
 - e. age dependent plant risk profile

N/A - This design difference does not impact this Subsection

*** As compared to closest current licensed LWR design.**

**** See potential revision categories at the end of this table.**

TABLE 4 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION ND WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	ND SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Canned motor pumps used as reactor coolant pumps.	N/A	N/A	N/A
2. Automatic depressurization system added for RCS depressurization to allow for passive system water injection into the core.	N/A	N/A	N/A
3. Added in-containment refueling water storage tank (IRWST) above the reactor core for passive response to RCS depressurization such as a LOCA.	1. a, 2. a,b	ND-3900, "0-15 psig Storage Tank Design" ND-6600, "Special Test Pressure Situations"	ND-3900 may require revision to address the IRWST. However, the IRWST will probably be designed per ND-3100, "General Design", or ND-3300, "Vessel Design", if it is of metal construction. ND-6600 may require revision due to a combined internal and external design loading of the IRWST.
4. Residual heat removal system does not provide ECCS function but is used for shutdown and refueling cooling.	N/A	N/A	N/A
5. Two core makeup tanks (CMT) added for quick response passive source of water upon RCS depressurization.	N/A	N/A	N/A
6. Chemical and volume control system including charging and primary make-up pumps downgraded to non-safety based on CMT addition along with other changes.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION ND			
WESTINGHOUSE AP600			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	ND SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
7. Passive residual heat removal system cools primary water through the IRWST during some accidents.	N/A	N/A	N/A
8. Removed the auxiliary feedwater pumps based on passive RHR function.	N/A	N/A	N/A
9. Large steel shell containment provides passive cooling via heat exchange with passive sources of exterior air and water. Condensate inside containment shell is recycled for core cooling.	N/A	N/A	N/A
10. AC power including the standby diesel generators are downgraded to non-safety based on reliance on passive systems and on DC power to initiate some valve repositioning.	N/A	N/A	N/A
11. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION ND			
GE ABWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	ND SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added capability for flooding the lower drywell with water from the suppression pool for core melt vessel failure severe accidents.	N/A	N/A	N/A
2. Replaced jet pumps with internal canned motor pumps for reactor coolant recirculation.	N/A	N/A	N/A
3. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
4. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION ND			
CE SYSTEM 80+			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	ND SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Added in-containment refueling water storage tank (IRWST) to serve as initial source of ECCS water and quench tank for pressurizer safety discharge and depressurization system.	1. a, 2. a,b	ND-3900, "0-15 psig Storage Tank Design" ND-6600, "Special Test Pressure Situations"	ND-3900 may require revision to address the IRWST. However, the IRWST will probably be designed per ND-3100, "General Design" or ND-3300, "Vessel Design", if it is of metal construction. ND-6600 may require revision due to a combined internal and external design loading of the IRWST.
2. Eliminated the recirculation actuation signal based on adding IRWST.	N/A	N/A	N/A
3. Two pumps dedicated to shutdown cooling operation.	N/A	N/A	N/A
4. Eliminated the two LPSI pumps based on use of four HPSI pumps for ECCS functions.	N/A	N/A	N/A
5. Makes use of containment spray pumps to provide shutdown cooling in addition to containment spray.	N/A	N/A	N/A
6. Chemical and volume control system downgraded to a non-safety system based on safety functions being performed by the safety injection system.	N/A	N/A	N/A
7. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION ND			
GE SBWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	ND SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
1. Eliminated large nozzles in the reactor vessel below the top of active fuel.	N/A	N/A	N/A
2. Reactor vessel uses forged rings which results in a 50% reduction of welds.	N/A	N/A	N/A
3. Eliminates recirculation piping external to vessel based on natural circulation capabilities.	N/A	N/A	N/A
4. Uses adjustable speed electric motor driven feedwater pumps.	N/A	N/A	N/A
5. Eliminates reactor core isolation cooling system.	N/A	N/A	N/A
6. Adds decay heat removal system to perform reactor water cleanup function and shutdown cooling, and pool cooling and cleanup functions using sealless pumps and hollow fiber filters.	N/A	N/A	N/A
7. Deleted standby gas treatment system.	N/A	N/A	N/A
8. Uses passive containment cooling from suppression pool and isolation condenser gravity driven cooling system.	N/A	N/A	N/A
9. Vital AC power is provided by two DC busses with 72 hour battery capacity.	N/A	N/A	N/A
10. Batteries provide power for four divisions of controls and instrumentation during loss of AC power and LOCA.	N/A	N/A	N/A
11. High pressure core spray is eliminated.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

TABLE 4 ALWR IMPACTS ON ASME BP&V CODE SECTION III, SUBSECTION ND			
GE SBWR			
MAJOR DESIGN DIFFERENCES *	POTENTIAL REVISIONS **	ND SECTIONS IMPACTED	EXPLANATION OF SECTIONS POTENTIALLY IMPACTED
12. Automatic depressurization provided for safety/relief valves for overpressure and six squib valves for depressurization.	N/A	N/A	N/A
13. Eliminated low pressure core spray.	N/A	N/A	N/A
14. Eliminated residual heat removal emergency core cooling system function.	N/A	N/A	N/A
15. Added gravity driven cooling system, squib actuated, to flood containment one meter above the top of active fuel for indefinite post LOCA cooling.	N/A	N/A	N/A
16. Uses an isolation condenser containment design.	N/A	N/A	N/A
17. Eliminated standby service water system.	N/A	N/A	N/A
18. Increased scram accumulator nitrogen pressure/volume to eliminate the scram discharge volume.	N/A	N/A	N/A
19. Accumulators were added to the standby liquid control system, and its standby pumps were eliminated.	N/A	N/A	N/A
20. Plant design life extended to 60 years.	N/A	N/A	N/A

N/A - This design difference does not impact this Subsection

* As compared to closest current licensed LWR design.

** See potential revision categories at the end of this table.

ALWR IMPACTS ON ASME B&PV CODE SECTION III, POTENTIAL REVISION CATEGORIES

- 1. Safety Classification**
 - a. passive safety systems and components**
 - b. deleted active safety systems and components**
 - c. downgraded systems and components**
 - d. testing requirements for passive systems and components**
- 2. Code Terminology**
 - a. concepts**
 - b. definitions**
- 3. Design Life**
 - a. aging and failure mechanisms**
 - b. inspection frequency and intervals**
 - c. maintenance bases and frequency**
 - d. plant life extension**
 - e. age dependent plant risk profile**

N/A - This design difference does not impact this Subsection

*** As compared to closest current licensed LWR design.**

**** See potential revision categories at the end of this table.**

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT
DESIGN WITH OTHER PLANT TYPES

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Abstract

The following is a tabular comparison of the CANDU 3 design to that of a typical pressurized light water Westinghouse four loop reactor. This report was prepared under contract DE-AC06-76RLO-1830, NRC FIN L2207, *Applicability of Codes and Standards to Advanced Reactors*, to familiarize the reader with the design, on a systems basis, of the CANDU 3 Standard Nuclear Plant Design.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Reactor	Horizontal pressure tubes within the calandria vessel.	Cylindrical pressure vessel suspended vertically from coolant nozzles with welded hemispherical bottom head and a removable flanged and gasketed hemispherical upper head.
Reactor Coolant	Pressurized heavy water (D ₂ O).	Pressurized light water (H ₂ O) coolant and moderator medium.
Moderator	Heavy water at low pressure.	High pressure light water (H ₂ O)
Fuel Elements	Compacted and sintered natural uranium dioxide pellets sheathed in Zircaloy 4.	Fuel rods contain compacted and sintered slightly enriched uranium dioxide fuel pellets in the shape of right circular cylinders clad in Zircaloy 4 tubing which is plugged and seal welded at both ends to encapsulate the fuel.
Fuel Bundle	Thirty seven fuel elements per bundle enclosed in Zirconium-Nickelium pressure tube.	Fuel assembly consists of 17 by 17 array containing 264 fuel rods, 24 guide thimble tubes, and one centrally located instrument thimble tube all held between a bottom nozzle and a top nozzle and intermittent grids.
Fuel Channel	Twelve fuel bundles per fuel channel.	N/A
Fuel Channel Assembly	Pressure tube, calandria tube, calandria tube extensions, end fittings, and four garter springs which maintain the annular space between the pressure tube and calandria tube.	N/A
Reactor Core	232 fuel channel assemblies connected at each end to feeder pipes of the heat transport system.	264 fuel assemblies comprise the reactor core contained within the reactor pressure vessel between the lower core plate and the upper core plate and inside the core baffle and baffle.
Calandria	Horizontal stainless steel cylindrical vessel shell, end shields, and Zircaloy 2 calandria tubes/tube extensions which contain low pressure, low temperature heavy water moderator.	Cylindrical pressure vessel suspended vertically from coolant nozzles which supports the fuel and contains the light water moderator.
Calandria Vessel	Together with the end shields supports the horizontal fuel channel assemblies and the incore reactivity control unit components.	Pressure vessel supports the internals package, fuel, head, and control rods, and control rod drive mechanisms.

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MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Calandria Tube	Zircaloy 2 tube enclosing each pressure tube to separate the moderator from the pressure tube outside diameter by creating an annular space which is filled with CO ₂ gas to reduce heat transfer to the moderator.	N/A
Calandria Extension	Maintains the moderator level accommodating shrink and swell due to temperature fluctuations.	N/A
Shutdown Systems	Two diverse systems, each independent of the reactor regulating system and each capable of rendering the reactor subcritical under normal operating or accident conditions. Shutdown system No. 1 is 20 spring assisted gravity drop shutoff rods and shutdown system No. 2 is gadolinium nitrate solution which is injected into the moderator. Parameters used to actuate the shutdown systems are heat transport system pressure, level, and flow, neutron power, reactor building pressure, and steam generator level and pressure.	The reactor trip system automatically shuts down the reactor whenever the limits of safe operation including mechanical and hydraulic limitations of equipment and heat transfer phenomena are approached. These parameters, which include pressure, level, flow, temperature, and others, are monitored by redundant channels which trip the reactor trip breakers when the trip logic is made up and results in dropping all the rod cluster control assemblies into the core by gravity which stops the chain reaction. Additionally the capability exists for the operator to provide refueling water or 4 weight percent boric acid directly to the suction of the charging pumps for emergency boration.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Shutdown System Number 1	Shutdown system No. 1 employs a logic system having three independent channels (2 out of 3 trip logic) which detect the requirement for a reactor trip and de-energize direct current clutches to release the shutoff rods into the moderator.	The reactor trip system is made up of process instrument and control system, nuclear instrumentation system, solid state logic protection system, reactor trip switchgear, and manual actuation system. Each of two reactor trip system trains is capable of opening separate and independent reactor trip breakers whenever necessary to prevent fuel damage for an anticipated operational occurrence, to limit core damage for infrequent faults, to ensure that the energy in the core is compatible with design provisions to protect the reactor coolant pressure boundary for limiting fault conditions.
Shutdown System Number 2	Injects liquid poison solution of gadolinium nitrate into the low pressure moderator through six incore injection nozzle tubes by opening fast acting valves which pressurize the individual poison tanks from a common helium supply tank based on a 2 out of 3 logic or manual signal. Shutdown system No. 2 incorporates design features to prevent unnecessary injections when the shutdown system No. 1 has acted normally.	Additionally the capability exists for the operator to provide refueling water or 4 weight percent boric acid directly to the suction of the charging pumps for emergency boration.
Emergency Core Cooling System	Supplies light water to the heat transport system after a LOCA. Removes residual and decay heat from the fuel following failure of the heat transport system pressure boundary. Includes short term injection and long term recirculation after system actuation on low heat transport system pressure coincident with high reactor building pressure, low pressurizer level, or high moderator level. Analyzed for large break LOCA, small break LOCAs, pressure tube rupture, steam line failure and loss of flow events. System components include Accumulator Water Tanks, Accumulator Gas Tanks, Grade Level Tank, ECCS Pumps (2), and associated heat exchangers and valving.	The emergency core cooling system (ECCS) is designed to remove the stored and fission product decay heat to cool the reactor core and provide shutdown capability for reactor coolant pipe break accidents, rod cluster control assembly ejection accidents, power conversion system pipe break accidents, and steam generator tube rupture accidents.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
High Pressure Injection	<p>Injects light water from water accumulator tanks into the heat transport system using pressurized gas as the motive force. Pressurized gas comes from accumulator gas tank which is pressurized by air compressors with a nitrogen accumulator as backup.</p> <p>Injecting accumulator water requires ECCS actuation which opens gas isolation valves and high pressure injection valves. When water level in accumulator becomes low, high pressure injection valves are closed so gas does not become entrained in core.</p> <p>For a large break LOCA, water is injected to the outlet header of the downstream core pass and the inlet header of the other core pass. For a small break LOCA, all header injection is used (upstream and downstream on both core passes).</p>	<p>Two centrifugal charging pumps automatically, upon receipt of a safety injection signal, deliver borated water flow from the refueling water storage tank to the RCS cold legs at the prevailing pressure during the high head injection phase of emergency core cooling. Two safety injection pumps automatically, upon receipt of a safety injection signal, deliver borated water flow from the refueling water storage tank to the RCS cold legs at an intermediate pressure range of 1650 psig to 715 psig. An accumulator tank that is filled with borated water and pressurized with nitrogen gas (660 psig normal) is connected to each RCS cold leg to provide passive safety injection through two in-series check valves upon RCS depressurization.</p>
Low Pressure Injection	<p>Injects water from a grade level tank, which can be replenished from demineralized water, using emergency core cooling pumps. Low pressure injection begins only after high pressure injection is depleted.</p>	<p>Two residual heat removal pumps (RHR) deliver borated water flow from the refueling water storage tank to the RCS cold legs once the RCS pressure is below 425 psig during the low pressure injection phase of emergency core cooling.</p>
Long Term Recirculation	<p>Recirculates mixture of light water and heavy water from the reactor building floor sump back into the heat transport system via heat exchangers and pumps.</p>	<p>Two RHR pumps and RHR heat exchangers, after injection phase are realigned to remove heat and deliver water from the containment sump to the RCS cold legs and to the safety injection pumps suction for the recirculation phase of emergency core cooling.</p>

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MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Emergency Core Cooling System Initiation (System)	ECCS system is designed to maintain or re-establish sufficient cooling of the fuel and fuel channels for specified loss of cooling accidents. ECCS is initiated on a loss of coolant signal which is conditioned by either high reactor building pressure or low pressurizer level. Actuation initiates injection of the accumulators, starts one ECCS pump, opens the grade level tank isolation valves (supply to ECCS pump), starts group 2 Class III diesels, opens the main steam safety valves, and trips the heat transport pumps.	The engineered safety features actuation system (ESFAS) provides timely initiation signals for actuation of engineered safety features to prevent or mitigate core damage and RCS and containment integrity based on redundant instrument and controls which sense accident situations. ESFAS provides initiation of a reactor trip, cold leg injection valve opening, emergency boration, ECCS pumps start and valve positioning, auxiliary feedwater pump starting, isolation signals, EDG starting, and HVAC alignment.
Containment System	The containment consists of a reinforced concrete structure with a reinforced concrete dome and an internal steel liner. The containment envelope which is the reactor building and metal extensions, and the containment isolation system which automatically isolates all penetrations when an increase in containment pressure or radioactivity is detected. Reactor building air coolers provide a heat sink for containment. The containment system provides isolation, pressure/activity reduction, hydrogen control, and monitoring during accident conditions. Under normal circumstances containment limits the release of radioactive material, provides external shielding, and protects reactor systems against external events.	The containment consists of a steel lined, reinforced concrete structure which houses the nuclear steam supply system and certain portions of the engineered safety features systems. The containment is designed to limit the leakage of radioactive fission products to the environment in the event of an accident inside containment. The containment spray system removes heat from the containment environment after a LOCA or other high energy line break inside containment. The containment isolation system allows normal or emergency passage of fluids through containment boundary while preserving the ability of the boundary to limit the escape of fission products from postulated accidents.
Reactivity Control Units	Vertical and horizontal reactor power measuring devices, neutron absorbing devices (vertical reactivity control units for both the Reactor Regulating System and the shutdown system 1, and the horizontal reactivity control liquid injection nozzles of shutdown system 2).	Power range, intermediate range, source range, and incore flux monitors provide neutron sensing. Rod cluster control assemblies contain full length absorber material (silver-indium-cadmium) to control the reactivity of the core under operating conditions and to control axial power distribution. Single control rod system is actuated by independent systems with diversity provided by manually injecting boric acid.

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MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Shield Tank	Light water filled horizontal cylindrical tank with end shields which enclose the calandria and provide shielding to adjacent areas and transfer heat generated and conducted to the light water for removal by the recirculating cooling water system.	Coolant flowing down the annulus between the reactor vessel wall and the core support structure provides shielding to areas surrounding the reactor vessel.
Shield Tank Extension	Surmounts the shield tank and is topped by the reactivity mechanisms deck which accommodates and supports the vertical reactivity control unit components and also provides shielding and heat removal similar to the shield tank.	N/A
Heat Transport System (RCS)	Circulates pressurized heavy water using forced coolant flow through the reactor fuel channels to remove heat from fission of natural uranium and transfer the heat to steam generators.	Reactor coolant system circulates pressurized light water using forced coolant flow through the reactor vessel and core to remove heat from fission of slightly enriched uranium and transfer the heat to steam generators.
Heat Transfer Parameters	1380 MW(th) transferred to steam generators, 41×10^6 lbm/hr unidirectional core flow, 1500 psi(a) reactor coolant pressure, 590 degrees F reactor coolant temperature, 700 psi(a) steam pressure, 500 degrees F (nominal) steam temperature, 99.75% (minimum) steam quality.	3,000 M(th) transferred to steam generators, 142.0×10^6 lbm/hr core flow, 2,250 psi(a) (nominal) reactor coolant pressure, 590 degrees F (average in vessel) reactor coolant temperature, 1000 psia steam pressure, 548 degrees F (nominal) steam temperature, 99.75% steam quality.
Heat Transport Piping (RCS)	Carbon steel.	Reactor coolant piping and fittings which make up the loops are austenitic stainless steel as are all connections.
Steam Generators	Two vertical inverted U-tube with integral steam drum and preheater, two coolant inlets and one coolant outlet, and Incoloy 800 tubes and carbon steel secondary side shell.	Four vertical inverted U-tube with integral moisture separating equipment one coolant inlet and one coolant outlet, and Inconel-800, a nickel-chromium-iron alloy (ASME-SB-163) tubes and carbon steel secondary side shell.

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MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Heat Transport Pumps (RCPs)	Two vertical centrifugal single suction double discharge with carbon steel casing and stainless steel impeller and shaft. Shaft sealing consists of three mechanical seals and one backup, in series. Each seal is designed to withstand full system pressure. Seals are housed in a removable cartridge located in the gland. The backup seal is located at the top of the gland.	Four identical vertical single speed bottom suction, side discharge centrifugal pumps driven by a top mounted, air cooled, three phase induction motor with shaft flywheel for extended inertia coastdown. The shaft seal section consists of one controlled leakage seal and two rubbing face seals located within the main flange and seal housing.
Heat Transport System (RCS) Overpressure Protection	Overpressure protection is provided by instrumented liquid relief valves which discharge to the bleed condenser as well as by the reactor shutdown systems. The bleed condenser is equipped with spring loaded safety valves set at a pressure above the normal operating pressure of the reactor outlet headers.	Overpressure protection for the RCS is provided by pressurizer safety valves which discharge to the pressurizer relief tank. The pressurizer safety valves are of the totally enclosed pop-type, spring loaded, self activated with back pressure compensation. Power operated relief valves which have upstream isolation valves to prevent excessive leakage are operated automatically or manually to limit system pressure for a large power mismatch.
Inventory Storage Tank	Transfers heavy water inventory into and out of the heat transport system and stores heavy water collected from the heavy water collection system.	The CVCS volume control tank (VCT) collects letdown and provides a suction for RCS injection via the charging pumps. Refueling water storage tank (RWST) and demineralized water tank provide makeup if RCS inventory becomes low.
Pressure and Inventory Control System	Includes the pressurizer, a bleed condenser, feed pumps, and control valves which all together, for all operating modes, control pressure and provide overpressure protection for the heat transport system, accommodates coolant thermal expansion and contraction, provides coolant inventory control, and provides a means of degassing the coolant.	RCS pressure is controlled by use of the pressurizer where water and steam are maintained in equilibrium by electrical immersion heaters and water sprays, where steam is generated by the heaters and condensed by the sprays. Inventory is controlled by the chemical and volume control system using the letdown to the volume control tank and potential for diversion to or make up from the hold up tanks.
Pressurizer	Cylindrical vertical pressure vessel connected to one of the reactor outlet headers, includes electrical immersion heaters, and spray flow to condense steam for heat transport system pressure control during operation at power.	The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. Electrical heaters are installed through the bottom head of the vessel while the spray nozzle, relief, and safety valves connections are located in the top of top head of the vessel.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Defected Fuel Detection and Location System	This delayed neutron detection system is operated in continuous mode sampling from two bulk samples until indication of failed fuel is found. Then the system is placed in location mode and 232 outlet headers are sequentially sampled until the channel containing failed fuel is located.	The process and effluent radiological monitoring and sampling systems includes a failed fuel monitor to provide continuous indication and on demand record of radiation levels of the CVCS letdown line after the letdown heat exchanger to monitor fission product activity.
Moderator System	Circulates heavy water in the calandria vessel through the moderator cooling, purification, and soluble poison control systems.	The reactor coolant system contains the pressurized light water coolant which also is the moderator medium within the reactor pressure vessel.
Moderator Pumps	Two 50% pumps.	Four identical reactor coolant pumps circulate the reactor coolant/moderator.
Moderator Heat Exchangers	Two 50% plate type heat exchangers remove heat from the moderator.	Four U-tube steam generators remove heat from the coolant/moderator.
Moderator Purification	Maintains purity and minimizes corrosion and crud activation via pD(Ph) control and adjusts soluble poison (boron and gadolinium) concentrations. Makeup is from the heat transport D ₂ O system.	The Chemical and Volume Control System (CVCS) maintains required water inventory in the RCS, controls reactor coolant water chemistry conditions, activity level, soluble chemical neutron absorber concentration (boric acid), and makeup.
Moderator Cover Gas	Uses Helium gas and controls concentration of deuterium gas via catalytic recombination.	CVCS chemical addition is used to scavenge oxygen from the coolant during startup and provides hydrogen gas overpressure for dissolved hydrogen to control and scavenge oxygen.
Shutdown Cooling System	Consists of pumps and heat exchangers and connecting piping between the reactor inlet and outlet headers. Capable of operation at full heat transport system pressure and temperature for removal of reactor decay heat. Purpose is to cool the heat transport system to a temperature suitable for maintenance and maintain a cool temperature.	The residual heat removal system (RHRS) transfers heat from the RCS to the component cooling water system to reduce the temperature of the reactor coolant to the cold shutdown temperature at a controlled rate during the second part of normal plant cooldown and maintains this temperature until the plant is started up again. Parts of the RHRS also serve a emergency core cooling function during the injection and recirculation phases after a LOCA.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Fuel Handling System	Stores new fuel, stores irradiated fuel, and performs fuel changing operations on routine basis during full power operation. Includes fully automated fueling machine that refuels the fuel channels at the reactor face and transfers the fuel out of the reactor building for storage or shipment and is operated from the control room.	The fuel handling system consists of equipment and structures with provisions to avoid dropping or jamming fuel assemblies or fuel handling devices, avoid lifting spent fuel above a minimum required depth of water, preserve containment integrity through the fuel transfer tube, preclude criticality during fuel handling, limit accident condition loads on fuel assemblies or core components, provide physical safety to equipment operators. The fuel handling equipment is comprised of equipment needed to refuel the reactor core and includes fuel assemblies, core component and reactor component hoisting and handling equipment, and the fuel transfer tube penetrating containment.
Fueling Machine	Uses horizontal and vertical electric drives with motion controllers to align to the fuel channels. Heavy water, electric power, and control signals are supplied to the fueling machine through flexible catenary assembly. Automatic defueling/refueling sequences are controlled from a dedicated computer in the main control room with operator interfaces and CRT display units.	
Steam System	Produced in two steam generators (41 X 10 ⁶ lbm/hr at 700 psia and 0.25% moisture) and fed to two separate steam mains which pass through containment penetrations to the turbine steam chest. Steam pressure is normally controlled at a constant value by varying reactor power to match turbine generator demand.	The main steam system conveys steam from the outlet of the four steam generators (a total of 15.14 X 10 ⁶ lb/hr at 1000 psia and 0.25% moisture) to the main turbine, the moisture separator reheaters, the steam generator feed pump turbines, the auxiliary feedwater pump turbine, the steam dump system, the turbine shaft seal system, and plant process steam system. Four main steam lines pass through containment and equalize pressure in a header downstream of the steam generator isolation valves.
Turbine Bypass	Four condenser steam discharge valves are sized to handle a turbine trip such that along with atmospheric steam valves safeties will not lift. The condenser steam discharge valves will handle up to 70% full power.	The steam dump system uses 12 air operated valves in parallel off the main steam lines to provide the capability of creating an artificial system load by dumping steam at up to 40% of the total steam flow to the main condensers bypassing the main turbine which enables the nuclear steam supply system to follow turbine load reductions of up to 50% rejection without lifting the main steam line safety valves.

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MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Atmospheric Steam Discharge Valves	Sized for 10% of reactor full power steam and normally controlled by the steam pressure control program, can be positioned manually, and are used during warmup or when the condenser is unavailable.	A power operated relief valve for each main steam line discharging to atmosphere. They are set to open at approximately 1130 psia to provide a means for removal of heat from the nuclear steam supply system when the condenser is not in service and are sized for decay heat removal to pass 10% of the flow at no load steam pressure. These PORVs can also be controlled manually from the control room or operated locally.
Main Steam Safety Valves	Provided in each steam main for steam system overpressure protection. These valves have over 100 percent steam flow at normal steam generator pressure.	Five main steam safety valves are provided on each main steam line outside containment and are rated to pass 105% of the engineered safeguard design steam flow at a pressure not exceeding 110% of the main stem design pressure. These safety valves are ASME B&PV Code certified, spring loaded, double outlet design and provide overpressure protection for the main steam supply system.
Steam Generator Level Control	Maintains the generator level at the setpoint during normal operating conditions and within limits under upset conditions. The steam generator level control system balances feedwater to steam flow for all operating conditions. The water level setpoint is automatically programmed over a set range as a function of load.	Each steam generator is equipped with a three element feedwater flow controller which maintains a programmed water level, which is a function of turbine load, by a regulating valve based on comparing feedwater flow signal, water level signal, programmed water level, and pressure compensated steam flow signal.
Steam Generator Blowdown	The continuous blowdown is flashed to steam into two flash tanks. The flashed steam is dumped to atmosphere and the water is "attenuated" and mixed with condenser cooling water which is discharged into the cooling water discharge tunnel which is sampled.	The steam generator blowdown (SGBD) system is used in conjunction with chemical feed and sampling systems to maintain optimum secondary side water chemistry in the recirculating steam generators and to control radioactivity levels associated with nonvolatile radionuclides. SGBD system is designed to continuously treat through demineralizers (or to bypass the demineralizers) 100% of the blowdown flow which is returned to the condensate or heater drain tank for reuse as secondary coolant. The blowdown is also sampled and passes by a radiation detector.

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MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Feedwater System	Two independent FW Systems are provided (FW and Group 2 FW). The FW System supplies feedwater to the steam generators. The feedwater system comprises the three motor driven main feedwater pumps (50% capacity each) on Class IV power and a diesel-driven auxiliary feedwater pump (4% capacity). The Group 2 FW system supplies feedwater to the steam generators if the FW System is unavailable. The Group 2 FW system contains two motor driven pumps and is independent of the FW system.	The feedwater system is designed to maintain uniform feedwater flow to all steam generators under all conditions and to maintain steam generator water level automatically during steady state and transient conditions. The feedwater system is closed cycle with feedwater heating, and uses two 50% capacity turbine driven variable speed pumps. The auxiliary feedwater system uses two motor driven pumps or a turbine driven pump to provide high pressure feedwater under a loss of normal feedwater condition and for startup, shutdown, and cooldown operations as well as for a small break loss of coolant accident.
Heat Transport (RCS) Pressure Control	Maintains the heat transport system within a narrow fixed pressure band. The pressurizer uses electric heaters and pressurizer sprays to control pressure in the heat transport system.	RCS pressure is controlled by use of the pressurizer where water and steam are maintained in equilibrium by electrical emersion heaters and water sprays, where steam is generated by the heaters and condensed by the sprays.
Heat Transport (RCS) Inventory Control	Maintains the combined mass of heavy water in the heat transport system and pressurizer per program. The D ₂ O feed and bleed system controls pressurizer level at a setpoint which varies with reactor power and heat transport system temperature.	Inventory is controlled by the chemical and volume control system using the letdown to the volume control system. Pressurizer level setpoint is variable dependent on reactor power.
Design Basis Tornado	Provides a generic set of design basis parameters for wind speeds, radius and pressure drop. Also provides a spectrum of Tornado Missiles and their associated mass, dimensions and velocity.	Design basis is based on parameters such as wind speeds, radius and pressure drop. A spectrum of Tornado Missiles and their associated mass, dimensions and velocity is also provided. LWR Tornado loadings are determined from siting locations but are generally more conservative than the values presented by CANDU. Similar missiles to the CANDU are postulated, however, the LWR velocities are substantially higher in most cases. The CANDU values correspond to the least conservative Region III velocity values.

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Design Basis Earthquake	Design Basis Earthquake (DBE) - Peak Horizontal Acceleration 0.30g. Site Design Earthquake (SDE) - Peak Horizontal Acceleration 0.15g.	The seismic design vales are determined for each site and are based on response spectra in addition to accelerations. Designs are presently based on an Operating Basis Earthquake (OBE) and a Safe Shutdown Earthquake (SSE). The SSE is the maximum vibratory ground motion that can reasonably be predicted from geologic and seismic evidence. The OBE is the earthquake that can be reasonably expected to occur during the plant life. For OBE the NSSS is designed to be capable of continued safe operation. The SSE is intended to assure the reactor coolant pressure boundary is not compromised, the reactor can be shutdown and maintained in a safe condition, with the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures is not compromised.
Reactor Auxiliary Building	The Reactor Auxiliary Building surrounds the Reactor Building and accommodates the umbilical which run between the principal structures, the Main Control Room, and the irradiated fuel bay and associated fuel handling facilities. It also house certain ECCS equipment and the Group 2 feedwater tank. This structure is designed to Tornado and Seismic design criteria.	There is no single structure in most LWRs which correspond to this building. However, the Control Building, Fuel Building, and Auxiliary Building provided some of the same services as this single CANDU 3 structure. These structures are designed for Tornado and Seismic Loads.
Chilled Water System	Supplies chilled water at controlled temperatures to the various HVAC systems, coolers in the reactor and Service Buildings and the coolers for the D ₂ O vapor recover system. The chillers are located in the Group 1 Service Building.	A Component Cooling Water system is provided to remove heat from safety-related and non-safety related systems components during normal operation and normal reactor shutdown. Normally heat is rejected to the SWS, but other backup methods are provided to transfer heat to the ultimate heat sink via the Circulating Water system. The CCW System usually provides a Seismic Category I loops to serve the trains of ESF equipment, and has a single Seismic Category II loop to provide cooling to non-ESF equipment.

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Raw Service Water	Two independent once through RSW Systems are provided (the RSW System and the Group 2 RSW System). The RSW Systems cools the Recirculating Cooling Water Systems through the RCW heat exchangers. The Service Water System (SWS) provides cooling to safety related room coolers, CCW heat exchangers, the Diesel-Driven AFW Pump, CCPs and SIP lube oil coolers and EDG jacket water and combustion air coolers. Service water acts as a make-up medium to the CCW System and ESF AFW Pumps. Normal SWS Supply is from an offsite water supply via the Intake structure. Alternative sources of water can be provided from the Circulating Water system and/or the fire protection system. Recirculated Cooling Water	The Service Water System (SWS) provides cooling to safety related room coolers, CCW heat exchangers, the Diesel-Driven AFW Pump, CCPs and SIP lube oil coolers and EDG jacket water and combustion air coolers. Service water acts as a make-up medium to the CCW System and ESF AFW Pumps. Normal SWS Supply is from an offsite water supply via the Intake structure. Alternative sources of water can be provided from the Circulating Water system and/or the fire protection system.
Group 1 Service Building	The Group 1 Service Building houses equipment which serves the normal power production requirement of the NSSS. This includes the Recirculated Cooling Water system Pumps and heat exchangers, the water chillers and Group 1 motor control centers and electrical supplies.	The functions/systems supported in the CANDU 3 Group 1 building are contained in a variety of structures in the LWR. All the structures are seismically and tornado qualified. Structures include the Control Building, Containment, Turbine Building, and Auxiliary and Electrical Building.
Group 2 Service Building	The Group 2 Service Building houses Group 2 systems. These include the essential cooling water supplies, electrical power supplies and the Secondary Control Room. The Secondary Control Room provides control and monitoring of all the systems required for the safe and maintained shutdown of the plant. This structure is seismically and tornado qualified.	The functions/systems supported in the CANDU 3 Group 1 building are contained in a variety of structures in the LWR. All the structures are seismically and tornado qualified. Structures include the Control Building, Containment, Turbine Building, and Auxiliary and Electrical Building. The Remote Shutdown Station (equivalent to the Secondary Control Room) is often found in the Control Building along with the Main Control Room.
Turbine Building	Steel frame structure which houses the turbine generator and its auxiliary process and electrical equipment, the water treatment plant and the station instrument, service and breathing air supplies.	Steel frame structure containing the turbine generator, feedwater, auxiliary process, electrical equipment, and compressed air supplies.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Maintenance Building	Contains mostly shops, plant stores and health physics laboratories. However, Heavy water management and spent resin storage facilities are provided in the basement area. Separate atmospheric control systems are provided for areas containing radioactive components	There is generally no maintenance structure included in the nuclear design of LWRs. Facilities are usually provided on the site and are designed to local building codes. However, liquid and solid radioactive waste management systems are located in the Containment, Control and Auxiliary Buildings.
Pumphouse Group 1	Contains raw service water pumps, screen wash pumps, trash racks, screens and chlorination equipment. Reinforced concrete substructures. The super structure is structural steel, masonry clad and insulated. Not seismically designed to withstand the design basis earthquake.	The Intake Structure contains service water pumps, screen wash pumps, trash racks, screens and chlorination equipment. Seismic Class I structure. NOTE: There is only one Intake Structure.
Pumphouse Group 2	Contains Group 2 raw service water pumps, travelling screen and trash racks. Reinforced concrete substructure and a steel frame superstructure. This structure is seismically qualified. This pumphouse is designed to withstand the design basis earthquake. NOTE: This performs a similar function to the Pumphouse provided for the Group 1 services.	The Intake Structure contains service water pumps, screen wash pumps, trash racks, screens and chlorination equipment. Seismic Class I structure. NOTE: There is only one Intake Structure.
Turbine Generator	Typically consists of one double flow high pressure turbine and two double flow low pressure turbines, operating at 1800 rpm (cold water site). For a warm water site a full speed turbine generator may be used.	Typically consists of a multistage turbine with a high pressure turbine and multiple low pressure turbines, operating at 1800 rpm.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Turbine Generator Auxiliaries	Auxiliaries may vary from site to site depending on cooling water temperatures, but the following subsystems are required. Condenser, turbine extraction steam system for feedwater heating, feedwater system (three 50% capacity electrically driven main feed water pumps, and two 3 percent capacity auxiliary feedwater pumps, one of which is steam turbine driven.)	The Main Steam, Main Feedwater and Condensate and Auxiliary Feedwater systems closely parallel those required in the CANDU 3. In the LWR the purpose of the AFW System is to supply water to the secondary side of the steam generators for reactor decay heat removal when normal feedwater sources are unavailable. The AFW System is equipped with two independent ESF trains, each capable of satisfying 100% of the decay heat removal needs after reactor trip. The AFW pumps are capable of operation in an Station Blackout (SBO) event and are turbine and diesel driven.
Recirculated Cooling Water	Two independent RCW Systems are provided (The RCW System and Group 2 RCW System). The RCW System serves the Nuclear Steam Plant components as well as the turbine generators and auxiliaries. The Group 2 RCW System supplies safety related and special safety systems.	Backups to the SWS include the CW System and/or the fire protection system. See the discussion in RSW above.
Compressed Gases	Compressed air system supplies instrument air, service air and breathing air to both group 1 and group 2 areas of the station. The system provides backup receivers in the group 1 area capable of supplying instrument air requirements during change over from Class IV to Class III power, and backup local air reservoirs capable of supplying instrument air to essential users during accident conditions or on loss of group 1 supply.	Two compressed air systems are provided. A Service Air System provides compressed air for routine maintenance at various stations throughout the plant. The Instrument Air System provides compressed air, which is dried and filtered, for all air-operated instruments and valves.
Gaseous Radioactive Waste Management System	A ventilation system collects potentially active exhaust air from the Reactor Building, irradiated fuel handling and storage area, the decontamination center and the heavy water management area. Facilities for filtration and monitoring are provided.	The Gaseous Waste processing System (GWPS) is designed to perform the following functions: 1) Reduce fission product gas concentrations in the RCS during normal operation; 2) to contain indefinitely the fission product gases removed from the RCS; 3) to collect gases generated by other systems; 4) to maintain a low level of hydrogen gas in the collected gases; 5) to discharge radioactive effluents after monitoring for radioactivity.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Main Power Output	Portion of the station Electrical Power System. Transfers power produced by the turbine-generator to the off-site electrical network. The system includes the turbine-generator with static excitation system, generator neutral grounding, generator metering and protection system, a generator breaker, isolated phase bus-duct, a unit service transformer (UST), a system service transformer (SST) and the main output transformer (MOT)	The offsite portion of the power system provides a connection between the plant power grid for power delivery and as a source of power during periods in which the generator is not operating. During normal plant operating conditions, the offsite power system provides a connection between the generator and the utility grid with the generator supplying power to the grid. A portion of the generated power is supplied to the auxiliary transformer to power plant equipment. During periods in which the plant is not generating power (i.e., startup, shutdown, trip) the offsite power system energizes the onsite power system via the startup transformers. In the event the startup transformer(s) or their circuit(s) are not available, the onsite EDGs can be used to energized selected portions of the onsite power system. An alternate path is available from the offsite supply through the main and auxiliary transformers to the onsite system.
Station Service Power Distribution	Portions of the Station Electrical Power System supplies process and instrumentation loads within the plant. This system is divided into two electrical supply groups. The Group 1 supplies the systems of the plant dedicated to normal power production, while Group 2 supplies safety related loads. Group 1 has Class IV and Class III supplies. Group 2 has Class III and Class II Supplies. Power is supplied to the SSDS from either the off-site network and/or the main generator. On start-up the SSDS is fed from off-site via the SST. During normal operation the SSDS is equally shared between the SST and the UST. Either transformer is sized for the total station service load and can thus supply the total service load in the event of failure of the other.	The onsite AC power systems consistz of the main generator, the main transformers, the unit auxiliary transformers, the unit substation transformers, and the 12.47-kV, 4.16-kV, 480-VAC, and 120/208-V buses. This includes the onsite emergency diesel generators. Two independent trains of AC power are provided throughout the plant. Most equipment is supplied by its associated train of AC power, but selected components can be powered by either train. In addition, the trains can be manually cross-connected at the 4.16-kV bus level.
Class IV Power Supply	AC power to auxiliaries and equipment that can tolerate long duration interruptions. Only supplies Group 1 systems. Supplied from Turbine Generator or offsite Electric Network.	AC power is divided into ESF and non-ESF buses. Non-ESF would generally correspond to Class IV, and has no standby or backup supplies.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES		
MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Class III Power Supply	AC power to auxiliaries necessary for safe shutdown of the reactor and turbine. Short interruptions can be tolerated. Supplies in both Group 1 and Group 2 systems. Normally supplied from the Class IV Power Supply, but can also be supplied from on-site standby diesel generators in either the Group 1 or Group 2 Service Buildings. NOTE: the Group 2 Class III system is seismically qualified.	AC Power is divided into ESF and non-ESF buses. ESF loads would generally fall into Class III electrical loads. The split into ESF and non-ESF occurs at the 4.16-kV System which is designed to provide electric power for operation of both ESF and non-ESF systems and equipment. It consists of four 4.16-kV buses which are normally supplied from the 12.47-kV System. The two ESF buses can also be connected to the standby power supply sources (EDGs) designed to provide power to the ESF loads when power from the 12.47-kV System is not available.
Class II Power Supply	AC uninterruptible power for essential auxiliaries. Class II Supplies in both Group 1 and Group 2 systems. Channelized for redundancy requirements of station instrumentation and control systems. NOTE: The Group 2 Class II system is seismically qualified.	Class 1E AC Uninterruptible Power Systems are provided. The UPS provides continuous, reliable, regulated single path AC power to all Reactor Protection System instrumentation and control equipment required for plant safety. The design bases for these systems is: 1) the inverter shall have the capacity and regulation required for the AC output for proper operation of the equipment supplied; 2) redundant loads shall be assigned to different distribution panels which are supplied from different inverters; 3) Auxiliary devices that are required to operate dependent equipment shall be supplied from the same distribution panel to prevent the loss of eclectic power to one protection set from causing the loss of equipment in another protection set. No single failure shall cause a loss of power supply to more than distribution panel; 4) each of the distribution panels shall have access only to its respective inverter supply and standby power supply.
Class I Power Supply	DC uninterruptible power to essential auxiliaries. Triplicated and channelized for redundancy requirements of control logic and reactor safety circuits. Supplies in both Group 1 and Group 2 systems. NOTE: The Group 2 Class I system is seismically qualified.	The DC system consists of one 250-VDC and two 125-VDC Systems designed to provide an adequate and reliable source of continuous DC power of controls, instrumentation, annunciators, AC inverters, DC motors and emergency lighting. The 250-VDC System and the 125-VDC Systems are comprised of battery chargers, DC control centers and distribution panels and two independent and redundant Class 1E 125-VDC battery systems.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Group 1 Standby Generators	Two diesel generators that provide an alternative power source to specific station loads connected to the Class III system. The Group 1 diesels (if successfully started on failure of normal power sources) will feed both the Group 1 and 2 Class III electrical systems. The standby sets will accept key loads within 35 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days.	The Standby Power Supply System is designed to provide redundant standby power sources of furnishing adequate power to safely shutdown the reactor remove decay heat and maintain the plant in a safe shutdown condition. Two EDG units are provided to generate and supply the required standby power to the 4.16-kV ESF buses. These buses will supply all of the ESF loads and those non-ESF loads for which it is desirable to have manually controlled access to the EDGs. The EDGs are designed to start automatically upon receipt of a start signal, attain rated speed and rated voltage within 10 seconds and to automatically accept loss of coolant ESF loads in sequence. Each EDG is capable of starting all required motors and accelerating them to full load operation in less than 60 seconds.
Group 2 Standby Generators	Two diesel generators that provides an alternative power source to specific station loads connected to the Class III system. The Group 2 diesels only start in the case of unsuccessful Group 1 diesel generators' operation or an earthquake. The Group 2 Diesel generators are seismically qualified. The Group 2 diesels can feed both the Group 1 and 2 Class III Electrical systems. The standby sets will accept key loads within 35 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days.	See discussion above.
Group 1 Station Batteries	Sized to support all loads connected for up to 60 minutes following an interruption in normal Class III source. Capable of providing uninterruptible power supplies (DC and AC) as required to Class II and Class I systems.	Each battery is capable of carrying the essential load continuously for a period of four hours in the event of a total loss of onsite and offsite AC power.
Group 2 Station Batteries	Used for control logic of standby generators and switchgear operation. Seismically qualified. Capable of providing uninterruptible power supplies (DC and AC) as required to Class II and Class I systems.	See discussion above.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Distributed Control System	A collection of channelized local stations linked by dual-redundant data highway used to implement most of the Group 1 control functions. Digital data highways are used for signal transmission and programmable microprocessors for control logic. Sensors and actuators provide input and/or output to local stations located outside the reactor building. Major control functions (reactor regulation, heat transport pressure and inventory control, steam generator pressure and level control, etc.) are distributed among a number of small, DCS micro-processors.	There is no centralized single control system for LWRs. The instrumentation and control systems provide automatic protection and exercise proper control against unsafe and improper reactor operation during steady state and transient power operation. Two major system are provided for the safety-related functions: 1) Reactor Trip system and 2) Engineered Safety Features Actuation System (ESFAS)
Plant Display System	Provides alarm message display, graphic data display, operator machine interface and data logging. Most information is provided via the DCS. The PDS also receives non-DCS information (safety system, meteorological monitoring, clocks, etc.) to provide a complete single source of information for the operators.	There is no centralized single control system for displaying plant information in the Control Room. However, the Safety-Related Display instrumentation does provide the operator with information to enable him/her to monitor the results of engineered safety features. It provides information required to maintain the plant in a hot shutdown condition, or to proceed to cold shutdown. Additional, important information is gather and displayed in the Control Room by a variety of systems. Systems are designed to different interface criteria based on the Information Processing and Display Criteria for three different Categories
Main Control Room	Provides the resources and environment to support the operator in the safe and efficient control of the plant. Located in the Reactor Auxiliary Building. The shift supervisor's office and the Work Control Office are located in close proximity. The computer equipment driving the MCR CRT's is located next to the MCR. The MCR is in the Group 1 area of the plant.	Provides the resources and environment to support the operator in the safe and efficient control of the plant. Located in the Control Building. Displays, indications, alarms and controls are available implementing critical safety functions as well as plant operation and support functions.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Alarm Annunciation	Approximately 300 color coded alarm windows are provided in panels as a supplement to the CRT displayed messages and are completely independent from the CRTs and the PDS. CRTs provide indication of the last forty alarms and clearance messages. Audible signals are also provided to attract the operator.	Multiple color coded alarm systems are provided to ensure information is immediately accessible. Alarm design is also control by the Information Processing and Display Criteria Categories.
Secondary Control Area	Post accident mode of operation is based in the MCR unless the MCR is rendered uninhabitable, then control shifts to the SCA. The SCA has the capability for shutting down the reactor and performing the same long term control and monitoring in the post accident mode as that available in the MCR. The SCA is located in the Group 2 area.	Requirements exist to be able to shutdown the reactor from outside the Control Room. Different plants implement this requirement different. Some have centralized Remote Shutdown Station, others depend on local controls and Hot Shutdown Panels.
Overall Plant Control System	A digital processor based system that performs all major control functions. This system can be used to control the tubing (normal) (reactor follows to maintain constant steam generator secondary side press). Or the reactor can be controlled (alternate) (turbine follows to keep steam generator secondary side pressure constant). Loads used to control secondary side pressure includes the turbine generator, condenser steam discharge valves, and atmospheric steam discharge valves. Other involved control systems include steam generator level control, heat transport pressure control, heat transport inventory control, and reactor control.	No centralized single system is provided for control of the plant. Individual systems are available in the Control Room and at local control panels for control and monitoring of equipment.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Fire Protection Systems	<p>These systems provide a defense-in-depth approach towards fire prevention, fire detection and suppression, and mitigation by separation, layout and structures.</p> <p>Two fire water pumps are provided each with 100 percent design capacity. One pump is diesel driven the other electric on Class III electric power. A fire ring main is arranged to provide water to each building from both paths. Isolation valves are provided such that any break in the main may be isolated without losing water supply to more than one building.</p>	<p>Provides means of detecting, alarming, and extinguishing fires. Two basic subsystems: the Fire Detection System and the fire extinguishing system. the fire detection System is a plant-wide instrumentation system provided to detect fires in various areas of the plant and to alert the Control Room operators of a fire and its location. The fire extinguishing system includes such fire fighting equipment as sprinkler systems, water spray, standpipe and hose stations, halon systems, and portable extinguisher. Three 50 percent capacity fire pumps supply water to the system. One is electric driven the other tow diesel. Siamese fire department connections are provided for emergency fill of the system by a fire truck or a portable auxiliary pump.</p>
Seismic Protection	<p>The reactor building, reactor auxiliary building and all Group 2 structures are designed to withstand a design basis earthquake. The systems and equipment required to maintain reactor coolant pressure boundary integrity, reactor shutdown, decay heat removal, containment isolation and all safety support systems are also designed to withstand a design basis earthquake.</p>	<p>The Containment Building, Safeguards Buildings, Auxiliary Building, Electrical and Control Building, Fuel Building, Service Water Intake Structure, Safe Shutdown Impoundment Dam, Refueling Water Storage Tanks, Reactor Makeup Water Storage Tanks, and the Condensate Storage Tanks are designed to Seismic Category I.</p>
Station Life	<p>Modular design which allows replacement of most components. Non-replaceable components are rated for 100 years.</p>	<p>40 year life with possible plant life extensions. The next generation of plants may be licensed for 60+ years.</p>
Flux Detector Units	<p>Full length guide tube assemblies containing twelve well tubes. Two types of detectors are used. One has a rapid response to gamma and neutron fluxes; the other is slower and sensitive only to neutron flux.</p>	<p>Excore neutron flux measurement system is made up of three systems. The source range system consists of two boron tri-fluoride gas detectors (ionization detectors) which measure gamma and neutrons. The intermediate range system consists of two compensated ion chambers. the power range flux system consists of four channels each containing two uncompensated ion detectors. These channels are used for the Reactor Protection System and Engineered Safety Function Actuation System purposes.</p>

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH OTHER PLANT TYPES

MAJOR SYSTEM/ COMPONENT	CANDU 3	TYPICAL PRESSURIZED LIGHT WATER REACTOR WESTINGHOUSE FOUR LOOP PLANT
Ion Chamber units	Six Lead-shielded housings mounted on the outside of the calandria shell, in which the chamber instruments and calibration shutters are installed. Ion chambers on the sides of the reactor provide signals to the two shutdown system, while only chambers on the moderator room side provide signals to the reactor regulation systems.	Typically six fission chambers can traverse 50 -100 in-core instrument tubes for an axial neutron flux distribution of the core. These channels are used for calibration of excore channels and for post accident monitoring functions.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT
DESIGN WITH THE CANDU 6 DESIGN

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Abstract

The following is a tabular comparison of the CANDU 3 design to that of a CANDU 6 design. This report was prepared under contract DE-AC06-76RLO-1830, NRC FIN L2207, *Applicability of Codes and Standards to Advanced Reactors*, to familiarize the reader with the design, on a systems basis, of the CANDU 3 Standard Nuclear Plant Design.

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Reactor Design*	Nominal 450 MWe, pressurized heavy water cooled and moderated with horizontal tubes within a low-pressure calandria vessel. Single loop, unidirectional cooling flow with single end refueling. New Group 1 and Group 2 equipment concept results in many station layout differences.	Power range at existing stations, 515 to 881 MWe. Pressurized heavy water cooled and moderated with horizontal tubes within a low-pressure calandria vessel. Double loop, bi-directional cooling flow with double ended refueling.
Reactor Coolant*	Pressurized heavy water (D ₂ O) coolant, peak channel power 8.1 MW, higher coolant flow than CANDU 6.	Pressurized heavy water (D ₂ O) coolant, peak channel power 7.3 MW.
Moderator	Heavy water (D ₂ O) at low pressure.	Heavy water (D ₂ O) at low pressure.
Fuel Elements	Compacted and sintered natural uranium dioxide pellets sheathed in Zircaloy 4. Same fuel design.	Compacted and sintered natural uranium dioxide pellets sheathed in Zircaloy 4.
Fuel Bundle*	Thirty seven fuel elements (rods) per bundle enclosed in Zirconium-Niobium pressure tube. 10% higher fuel rating.	Thirty seven fuel elements (rods) per bundle enclosed in Zirconium-Niobium pressure tube.
Fuel Channel	Twelve fuel bundles per fuel channel.	Twelve fuel bundles per fuel channel.
Fuel Channel Assembly*	Pressure tube within a calandria tube, with calandria tube extensions, end fittings, and four garter springs which maintain the annular space between the pressure tube and calandria tube. CANDU 3 is designed for easy assembly replacement.	Pressure tube within a calandria tube, with calandria tube extensions, end fittings, and four garter springs which maintain the annular space between the pressure tube and calandria tube.
Reactor Core*	232 fuel channel assemblies, twelve bundles per channel, and 37 elements (rods) per bundle.	380 to 480 fuel channel assemblies comprise the reactor core contained within the calandria vessel. The number of bundles per channel and the bundle design is the same.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Calandria Vessel*	Horizontal tube, stainless steel cylindrical vessel shell, end shields, and Zircaloy 2 calandria tubes/tube extensions. The calandria contains the low pressure, low temperature heavy water moderator.	The CANDU 6 vessel is larger, other differences in design temperature and construction that are not too significant. Basic moderator design concepts are the same.
Calandria Tube and Gas Space	Zircaloy 2 tube enclosing each pressure tube. Separates the cooler moderator fluid from the hotter Heat Transport System fluid by creating an annular space which is filled with CO ₂ gas to reduce heat transfer to the moderator.	Zircaloy 2 tube enclosing each pressure tube. Separates the cooler moderator fluid from the hotter Heat Transport System fluid by creating an annular space which is filled with CO ₂ gas to reduce heat transfer to the moderator.
Calandria Extension	Maintains the moderator level accommodating shrink and swell due to temperature fluctuations.	Maintains the moderator level accommodating shrink and swell due to temperature fluctuations.
Shutdown Systems	Two diverse systems, each independent of the reactor regulating system and each capable of rendering the reactor subcritical under normal operating or accident conditions. Shutdown system No. 1 is vertical with 20 spring assisted gravity drop shutoff rods. Shutdown system No. 2 is gadolinium nitrate solution which is injected through horizontal tubes into the moderator. Parameters used to actuate the shutdown systems are heat transport system pressure, level, and flow, neutron power, reactor building pressure, and steam generator level and pressure.	Two diverse systems, each independent of the reactor regulating system and each capable of rendering the reactor subcritical under normal operating or accident conditions. Shutdown system No. 1 consists of various numbers of gravity drop shutoff rods. Shutdown system No. 2 is gadolinium nitrate solution which is injected into the moderator into 14 (varies) zones. Parameters used to actuate the shutdown systems are heat transport system pressure, level, and flow, neutron power, reactor building pressure, and steam generator level and pressure.
Shutdown System Number 1*	Shutdown system No. 1 employs a logic system having three independent channels (2 out of 3 trip logic) which detect the requirement for a reactor trip and de-energize direct current clutches to release the shutoff rods into the moderator.	Shutdown system No. 1 employs a logic system having three independent channels (2 out of 3 trip logic) which detect the requirement for a reactor trip and de-energize direct current clutches to release the shutoff rods into the moderator space in the core grid. Number of safety rods vary.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Shutdown System Number 2*	Injects liquid poison solution of gadolinium nitrate into the low pressure moderator through six Incore injection nozzle tubes by opening fast acting valves which pressurize the individual poison tanks from a common helium supply tank based on a 2 out of 3 logic or manual signal. Shutdown system No. 2 incorporates design features to prevent unnecessary injections when the shutdown system No. 1 has acted normally.	Injects liquid poison solution of gadolinium nitrate into the low pressure moderator through various numbers of Incore injection nozzle tubes by opening fast acting valves which pressurize the individual poison tanks from a common helium supply tank based on a 2 out of 3 logic or manual signal. Shutdown system No. 2 incorporates design features to prevent unnecessary injections when the shutdown system No. 1 has acted normally.
Emergency Core Cooling System	Supplies light water to the heat transport system after a LOCA. Removes residual and decay heat from the fuel following failure of the heat transport system pressure boundary. Includes short term injection and long term recirculation after system actuation on low heat transport system pressure coincident with high reactor building pressure, low pressurizer level, or high moderator level. Analyzed for large break LOCA, small break LOCAs, pressure tube rupture, steam line failure and loss of flow events.	Supplies light water to the heat transport system after a LOCA. Removes residual and decay heat from the fuel following failure of the heat transport system pressure boundary. Includes short term injection and long term recirculation after system actuation on low heat transport system pressure coincident with high reactor building pressure, low pressurizer level, or high moderator level. Analyzed for large break LOCA, small break LOCAs, pressure tube rupture, steam line failure and loss of flow events.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
High Pressure Injection*	<p>Injects light water from water accumulator tanks into the heat transport system using pressurized gas as the motive force. Pressurized gas comes from accumulator gas tank which is pressurized by air compressors with a nitrogen accumulator as backup.</p> <p>Injected accumulator water is directed to the specific header which instrumentation indicates would result in the greatest benefit, or result.</p> <p>For a large break LOCA, water is injected to the outlet header of the downstream core pass and the inlet header of the other core pass. For a small break LOCA, all header injection is used (upstream and downstream).</p>	<p>Injects light water from water accumulator tanks into the heat transport system using pressurized gas as the motive force. Pressurized gas comes from accumulator gas tank which is pressurized by air compressors with a nitrogen accumulator as backup.</p> <p>Injection location design not as sophisticated and injection locations are the same for all accidents.</p>
Low Pressure Injection	<p>Injects water from a grade level tank, which can be replenished from demineralized water, using emergency core cooling pumps. Low pressure injection begins only after high pressure injection is depleted.</p>	<p>Injects water from a grade level tank, which can be replenished from demineralized water, using emergency core cooling pumps. Low pressure injection begins only after high pressure injection is depleted.</p>
Long Term Recirculation	<p>Recirculates mixture of light water and heavy water from the reactor building floor sump back into the heat transport system via heat exchangers and pumps.</p>	<p>Recirculates mixture of light water and heavy water from the reactor building floor sump back into the heat transport system via heat exchangers and pumps.</p>

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Emergency Core Cooling System	ECCS system is designed to maintain or re-establish sufficient cooling of the fuel and fuel channels for specified loss of cooling accidents. ECCS is initiated on a loss of coolant signal which is conditioned by either high reactor building pressure or low pressurizer level. Actuation initiates injection of the accumulators, starts one ECCS pump, opens the grade level tank isolation valves (supply to ECCS pump), starts group 2 Class III diesels, opens the main steam safety valves, and trips the heat transport pumps.	ECCS system is designed to maintain or re-establish sufficient cooling of the fuel and fuel channels for specified loss of cooling accidents. ECCS is initiated on a loss of coolant signal which is conditioned by either high reactor building pressure or low pressurizer level. Actuation initiates injection of the accumulators, starts one ECCS pump, opens the grade level tank isolation valves (supply to ECCS pump), starts the diesels, opens the main steam safety valves, and trips the heat transport pumps.
Containment System	The containment consists of a reinforced concrete structure with a reinforced concrete dome and an internal steel liner. The containment envelope which is the reactor building and metal extensions, and the containment isolation system which automatically isolates all penetrations when an increase in containment pressure or radioactivity is detected. Reactor building air coolers provide a heat sink for containment.	The containment consists of a reinforced concrete structure with a reinforced concrete dome and an internal steel liner. The containment envelope is the reactor building and metal extensions. The containment isolation system automatically isolates all penetrations when an increase in containment pressure or radioactivity is detected. Reactor building air coolers provide a heat sink for containment.
Reactivity Control Units*	Vertical and horizontal reactor power measuring devices and neutron absorbing devices. There are vertical reactivity control units for both the Reactor Regulating System and the shutdown system 1, and horizontal liquid reactivity control injection nozzles of shutdown system 2.	The CANDU 6 rod numbers and liquid control zones vary. Vertical and horizontal reactor power measuring devices, neutron absorbing devices (vertical reactivity control units for both the Reactor Regulating System and the shutdown system 1, and the horizontal reactivity control liquid injection nozzles of shutdown system 2).

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Shield Tank	Light water filled horizontal cylindrical tank with end shields which encloses the calandria and provides shielding and waste heat transfer to the recirculating cooling water system.	Light water filled horizontal cylindrical tank with end shields which encloses the calandria and provides shielding and waste heat transfer to the recirculating cooling water system. There are minor design and size differences.
Shield Tank Extension	Surmounts the shield tank and is topped by the reactivity mechanisms deck which accommodates and supports the vertical reactivity control unit components and also provides shielding and heat removal similar to the shield tank.	Surmounts the shield tank and is topped by the reactivity mechanisms deck which accommodates and supports the vertical reactivity control unit components and also provides shielding and heat removal similar to the shield tank. There are minor design and size differences.
Heat Transport System (RCS)*	Circulates pressurized heavy water through the reactor fuel channels to remove heat from fission of natural uranium and transfer the heat to steam generators. CANDU 3 has higher coolant flow and more heat transfer capability for each channel.	Circulates pressurized heavy water through the reactor fuel channels to remove heat from fission of natural uranium and transfer the heat to steam generators.
Heat Transfer Parameters*	1380 MW(th) transferred to steam generators, 41 X 10 ⁶ lbm/hr unidirectional core flow, 1500 psia reactor coolant pressure, 600 degrees F reactor coolant temperature, outlet, 700 psi(a) steam pressure, 500 degrees F (nominal) steam temperature, 99.75% (minimum) steam quality.	(See Note) Mw(th) transferred to steam generators, (See Note) X 10 ⁶ lbm/hr core flow, 1500 psi(a) (nominal) reactor coolant pressure, 600 degrees F outlet reactor coolant temperature, 700 psia steam pressure, 500 degrees F (nominal) steam temperature, 99.75% steam quality. Note: These numbers will vary because of variations in CANDU 6 power and power densities.
Heat Transport Piping (RCS)	Carbon steel.	Carbon steel.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Steam Generators*	Two vertical inverted U-tube with integral steam drum and preheater, two coolant inlets and one coolant outlet, and Incoloy 800 tubes and carbon steel secondary side shell.	Four vertical inverted U-tube with integral steam drum and preheater, two coolant inlets and one coolant outlet, with differences in tube materials, but same carbon steel secondary side shell material.
Heat Transport Pumps (RCPs)*	Two vertical centrifugal single suction double discharge with carbon steel casing and stainless steel impeller and shaft.	Four to 16 pumps are used in one or two loops.
Heat Transport System (RCS) Overpressure Protection	Overpressure protection is provided by instrumented liquid relief valves which discharge to the bleed condenser as well as by the reactor shutdown systems. The bleed condenser is equipped with spring loaded safety valves set at a pressure above the normal operating pressure of the reactor outlet headers.	Overpressure protection is provided by instrumented liquid relief valves which discharge to the bleed condenser as well as by the reactor shutdown systems. The bleed condenser is equipped with spring loaded safety valves set at a pressure above the normal operating pressure of the reactor outlet headers.
Heavy Water Storage Tank (RCS Inventory)	Transfers heavy water inventory into and out of the heat transport system and stores heavy water collected from the heavy water collection system.	Transfers heavy water inventory into and out of the heat transport system and stores heavy water collected from the heavy water collection system.
Pressure and Inventory Control System	Includes the pressurizer, a bleed condenser, feed pumps, and control valves which control pressure and provide overpressure protection for the heat transport system, accommodates coolant thermal expansion and contraction, provides coolant inventory control, and provides a means of degassing the coolant.	Includes the pressurizer, a bleed condenser, feed pumps, and control valves which control pressure and provide overpressure protection for the heat transport system, accommodates coolant thermal expansion and contraction, provides coolant inventory control, and provides a means of degassing the coolant.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Pressurizer	Cylindrical vertical pressure vessel connected to one of the reactor outlet headers, includes electrical emersion heaters, and spray flow to condense steam for heat transport system pressure control during operation at power.	Cylindrical vertical pressure vessel connected to one of the reactor outlet headers, includes electrical emersion heaters, and spray flow to condense steam for heat transport system pressure control during operation at power.
Defected Fuel Detection and Location System	This delayed neutron detection system is operated in continuous mode sampling from two bulk samples until indication of failed fuel is found. Then the system is placed in location mode and the 232 outlet headers are sequentially sampled until the channel containing failed fuel is located.	This delayed neutron detection system is operated in continuous mode sampling from two bulk samples until indication of failed fuel is found. Then the system is placed in location mode and the outlet headers are sequentially sampled until the channel containing failed fuel is located.
Moderator System	Circulates heavy water in the calandria vessel through the moderator cooling, purification, and soluble poison control systems.	Circulates heavy water in the calandria vessel through the moderator cooling, purification, and soluble poison control systems.
Moderator Pumps	Two 50% pumps.	Two to four pumps.
Moderator Heat Exchangers	Two 50% plate type heat exchangers remove heat from the moderator heavy water.	Two to four heat exchangers remove heat from the moderator heavy water.
Moderator Purification	Maintains purity and minimizes corrosion and crud activation via pD(Ph) control and adjusts soluble poison (boron and gadolinium) concentrations. Makeup is from the heat transport D ₂ O system.	Maintains purity and minimizes corrosion and crud activation via pD(Ph) control and adjusts soluble poison (boron and gadolinium) concentrations. Makeup is from the heat transport D ₂ O system.
Moderator Cover Gas	Uses Helium gas and controls concentration of deuterium gas via catalytic recombination.	Uses Helium gas and controls concentration of deuterium gas via catalytic recombination.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Shutdown Cooling System	Consists of pumps and heat exchangers and connecting piping between the reactor inlet and outlet headers. Capable of operation at full heat transport system pressure and temperature for removal of reactor decay heat. Purpose is to cool the heat transport system to a temperature suitable for maintenance and maintain a cool temperature.	Consists of pumps and heat exchangers and connecting piping between the reactor inlet and outlet headers. Capable of operation at full heat transport system pressure and temperature for removal of reactor decay heat. Purpose is to cool the heat transport system to a temperature suitable for maintenance and maintain a cool temperature.
Fuel Handling System*	Stores new fuel, stores irradiated fuel, and performs fuel changing operations. Includes a single face, fully automated refueling machine that is operated from the control room and refuels the fuel channels from the outlet end on a routine basis during full power operation. Transfer of the fuel out of the reactor building is made for storage or shipment.	Stores new fuel, stores irradiated fuel, and performs fuel changing operations. Includes fully automated refueling, from two faces of the reactor. Refueling machines are operated from the control room and refuel the fuel channels from the outlet end on a routine basis during full power operation. Transfer of the fuel out of the reactor building is made for storage or shipment.
Fueling Machine*	Uses horizontal and vertical electric drives with motion controllers to align to the fuel channels. Heavy water, electric power, and control signals are supplied to the fueling machine through flexible catenary assembly. Automatic defueling/refueling sequences are controlled from a dedicated computer in the main control room with operator interfaces and CRT display units. Some design differences for improved reliability.	Two machines are used. Uses horizontal and vertical electric drives with motion controllers to align to the fuel channels. Heavy water, electric power, and control signals are supplied to the fueling machine through a flexible catenary assembly. Automatic defueling/refueling sequences are controlled from a dedicated computer in the control room.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Steam System*	One loop in the Heat Transport System feeds two steam generators (41×10^6 lbm/hr at 700 psia and 0.25% moisture) and these feed two separate steam mains which pass through containment penetrations to the turbine steam chest. Steam pressure is normally controlled at a constant value by varying reactor power to match turbine generator demand.	One to two loops in the Heat transport System feed two to four steam generators at various flow rates (at 700 psia and 0.25% moisture) and these feed two separate steam mains which pass through containment penetrations to the turbine steam chest. Steam pressure is normally controlled at a constant value by varying reactor power to match turbine generator demand.
Turbine Bypass	Four condenser steam discharge valves are sized to handle a turbine trip such that, along with atmospheric steam valves, safeties will not lift. The condenser steam discharge valves will handle up to 70% full power.	Condenser steam discharge valves are sized to handle a turbine trip such that, along with atmospheric steam valves, safeties will not lift. The condenser steam discharge valves will handle up to 70% full power.
Atmospheric Steam Discharge Valves	Sized for 10% of reactor full power steam and normally controlled by the steam pressure control program, can be positioned manually, and are used during warmup or when the condenser is unavailable.	Sized for 10% of reactor full power steam and normally controlled by the steam pressure control program, can be positioned manually, and are used during warmup or when the condenser is unavailable.
Main Steam Safety Valves	Provided in each steam main for steam system overpressure protection. These valves designed for over 100 percent steam flow at normal steam generator pressure.	Provided in each steam main for steam system overpressure protection. These valves have over 100 percent of rated steam flow at normal steam generator pressure.
Steam Generator Level Control	Maintains the generator level at the setpoint during normal operating conditions and within limits under upset conditions. The steam generator level control system balances feedwater to steam flow for all operating conditions. The water level setpoint is automatically programmed over a set range as a function of load.	Maintains the generator level at the setpoint during normal operating conditions and within limits under upset conditions. The steam generator level control system balances feedwater to steam flow for all operating conditions. The water level setpoint is automatically programmed over a set range as a function of load.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Steam Generator Blowdown	The continuous blowdown is flashed to steam into two flash tanks. The flashed steam is dumped to atmosphere and the water is "attenuated" and mixed with condenser cooling water which is discharged into the cooling water discharge tunnel which is sampled.	The continuous blowdown is flashed to steam into flash tanks. The flashed steam is dumped to atmosphere and the water is "tempered" and mixed with condenser cooling water which is discharged into the cooling water discharge tunnel which is sampled.
Feedwater System*	Two independent FW Systems are provided (FW and Group 2 FW). The FW System supplies feedwater to the steam generators. The feedwater system comprises the three motor driven main feedwater pumps (50% capacity each) on Class IV power and a diesel-driven auxiliary feedwater pump (4% capacity). The Group 2 FW system supplies feedwater to the steam generators if the FW System is unavailable. The Group 2 FW system contains two motor driven pumps and is independent of the FW system.	The FW System supplies feedwater to the steam generators. The feedwater system comprises the motor driven main feedwater pumps (50% capacity each) on Class IV power and auxiliary feedwater pumps. The Group 2 concept is not used.
Heat Transport (RCS) Pressure Control	Maintains the heat transport system within a narrow fixed pressure band. The pressurizer uses electric heaters and pressurizer sprays to control pressure in the heat transport system.	Maintains the heat transport system within a narrow fixed pressure band. The pressurizer uses electric heaters and pressurizer sprays to control pressure in the heat transport system.
Heat Transport (RCS) Inventory Control	Maintains the combined mass of heavy water in the heat transport system and pressurizer per program. The D ₂ O feed and bleed system controls pressurizer level at a setpoint which varies with reactor power and heat transport system temperature.	Maintains the combined mass of heavy water in the heat transport system and pressurizer per program. The D ₂ O feed and bleed system controls pressurizer level at a setpoint which varies with reactor power and heat transport system temperature.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Design Basis Tornado*	Provides a generic set of design basis parameters for tornadoes. Also provides a spectrum of Tornado Missiles and their associated mass, dimensions and velocity. U.S. LWR Tornado loadings are determined from site locations but are generally more conservative than the values presented by CANDU. The CANDU values correspond to the least conservative Region III velocity values. CANDU 3 as designed could not be sited in U.S. tornado regions I and II.	Provides a generic set of design basis parameters for tornadoes. Also provides a spectrum of Tornado Missiles and their associated mass, dimensions and velocity which are generally less conservative than U.S. criteria.
Design Basis Earthquake*	U.S.A. Designs are presently based on an Operating Basis Earthquake (OBE) and a Safe Shutdown Earthquake (SSE). The CANDU is designed to; Design Basis Earthquake (DBE) - Peak Horizontal Acceleration 0.30g. Site Design Earthquake (SDE) - Peak Horizontal Acceleration 0.15g. The CANDU 3 is not designed for U.S. seismic regions I and II.	The CANDU is designed to; Design Basis Earthquake (DBE) - Peak Horizontal Acceleration <0.30g. Site Design Earthquake (SDE) - Peak Horizontal Acceleration <0.15g. Design values vary somewhat.
Reactor Auxiliary Building*	The Reactor Auxiliary Building is designed to tornado and seismic design criteria and surrounds the Reactor Building and accommodates the umbilical which runs between the principal structures, the Main Control Room, and the irradiated fuel bay and associated fuel handling facilities. It also house certain ECCS equipment and the Group 2 feedwater tank.	The Reactor Auxiliary Building and the Reactor Building layout is different as are the other principal structures, the Main Control Room, and the irradiated fuel bay and associated fuel handling facilities. The reactor auxiliary building also houses certain ECCS equipment and the feedwater tank. This structure is designed to Canadian Tornado and Seismic design criteria.

* = Design Differences

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COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Chilled Water System*	Supplies chilled water at controlled temperatures to the various HVAC systems, coolers in the reactor and Service Buildings and the coolers for the D ₂ O vapor recovery system. The chillers are located in the Group 1 Service Building.	Supplies chilled water at controlled temperatures to the various HVAC systems, coolers in the reactor and service buildings and the coolers for the D ₂ O vapor recovery system.
Group 1 Service Building*	The Group 1 Service Building is built to UBC standards only and houses equipment which serves the normal power production requirement of the NSSS. This includes the Recirculated Cooling Water system Pumps and heat exchangers, the water chillers and Group 1 motor control centers and electrical supplies.	The functions/systems supported in the CANDU 3 Group 1 building are contained in a variety of structures in the CANDU 6 design.
Group 2 Service Building*	The Group 2 Service Building houses Group 2 systems. These include the essential cooling water supplies, electrical power supplies and the Secondary Control Room. The Secondary Control Room provides control and monitoring of all the systems required for the safe and maintained shutdown of the plant. This structure is seismically and tornado qualified.	The functions/systems supported in the CANDU 3 Group 2 building are contained in a variety of structures in the CANDU 6. Structures are seismically and tornado qualified as needed in the CANDU 6 design.
Turbine Building	Steel frame structure which houses the turbine generator and its auxiliary process and electrical equipment, the water treatment plant and the station instrument, service and breathing air supplies.	Steel frame structure which houses the turbine generator and its auxiliary process and electrical equipment. May contain the water treatment plant and the station instrument, service and breathing air supplies.
Maintenance Building	Contains mostly shops, plant stores and health physics laboratories. However, Heavy water management and spent resin storage facilities are provided in the basement area. Separate atmospheric control systems are provided for areas containing radioactive components.	Contains mostly shops, plant stores and health physics laboratories. Separate atmospheric control systems are provided for areas containing radioactive components.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Pumphouse Group 1*	Contains raw service water pumps, screen wash pumps, trash racks, screens and chlorination equipment. Reinforced concrete substructures. The superstructure is structural steel, masonry clad and insulated. Not seismically designed to withstand the design basis earthquake.	Group 1 concept not used. The Intake Structure contains service water pumps, screen wash pumps, trash racks, screens and chlorination equipment. Seismic resistant structure. NOTE: There is only one Intake Structure.
Pumphouse Group 2*	Contains Group 2 raw service water pumps, travelling screen and trash racks. Reinforced concrete substructure and a steel frame superstructure. This structure is seismically qualified. This pumphouse is designed to withstand the design basis earthquake. NOTE: This performs a similar function to the Pumphouse provided for the Group 1 services.	(see above).
Turbine Generator	Typically consists of one double flow high pressure turbine and two double flow low pressure turbines, operating at 1800 rpm (cold water site). For a warm water site a full speed turbine generator may be used. Stop valve temperature is approximately 257 (495) degrees centigrade (Fahrenheit).	Typically consists of a multistage turbine with a high pressure turbine and multiple low pressure stages, operating at 1800 rpm. Stop valve temperature range 245 (473) to 255 (491) degrees centigrade (Fahrenheit).
Turbine Generator Auxiliaries	Auxiliaries may vary from site to site depending on cooling water temperatures, but the following subsystems are required. Condenser, turbine extraction steam system for feedwater heating, feedwater system (three 50% capacity electrically driven main feed water pumps, and two 3 percent capacity auxiliary feedwater pumps, one of which is steam turbine driven.)	Auxiliaries may vary from site to site depending on cooling water temperatures, but the following subsystems are required. Condenser, turbine extraction steam system for feedwater heating, feedwater system with main feed water pumps, and two auxiliary feedwater pumps, one of which is steam turbine or engine driven.)

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Raw Service Water	Two independent once through RSW Systems are provided (the RSW System and the Group 2 RSW System). The RSW Systems cools the Recirculating Cooling Water Systems through the RCW heat exchangers.	Two independent once through RSW Systems are provided. The RSW Systems cools the Recirculating Cooling Water Systems through the RCW heat exchangers.
Recirculated Cooling Water	Two independent RCW Systems are provided (The RCW System and Group 2 RCW System). The RCW System serves the Nuclear Steam Plant components as well as the turbine generators and auxiliaries. The Group 2 RCW System supplies safety related and special safety systems.	Two independent RCW Systems are provided. The RCW System serves the Nuclear Steam Plant components as well as the turbine generators and auxiliaries.
Compressed Gases	Compressed air system supplies instrument air, service air and breathing air to both group 1 and group 2 areas of the station. The system provides backup receivers in the group 1 area capable of supplying instrument air requirements during change over from Class IV to Class III power, and backup local air reservoirs capable of supplying instrument air to essential users during accident conditions or on loss of group 1 supply.	Compressed air system supplies instrument air, service air and breathing air to areas of the station. The system provides backup receivers capable of supplying instrument air requirements during change over from Class IV to Class III power, and backup local air reservoirs capable of supplying instrument air to essential users during accident conditions.
Gaseous Radioactive Waste Management System	A ventilation system collects potentially active exhaust air from the Reactor Building, irradiated fuel handling and storage area, the decontamination center and the heavy water management area. Facilities for filtration and monitoring are provided.	A ventilation system collects potentially active exhaust air from the Reactor Building, irradiated fuel handling and storage area, the decontamination center and the heavy water management area. Facilities for filtration and monitoring are provided.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Main Power Output	Portion of the station Electrical Power System. Transfers power produced by the turbine-generator to the off-site electrical network. The system includes the turbine-generator with static excitation system, generator neutral grounding, generator metering and protection system, a generator breaker, isolated phase bus-duct, a unit service transformer (UST), a system service transformer (SST) and the main output transformer (MOT)	Portion of the station Electrical Power System. Transfers power produced by the turbine-generator to the off-site electrical network. The system includes the turbine-generator with static excitation system, generator neutral grounding, generator metering and protection system, a generator breaker, isolated phase bus-duct, a unit service transformer (UST), a system service transformer (SST) and the main output transformer (MOT)
Station Service Power Distribution	Portions of the Station Electrical Power System supplies process and instrumentation loads within the plant. This system is divided into two electrical supply groups. The Group 1 supplies the systems of the plant dedicated to normal power production, while Group 2 supplies safety related loads. Group 1 has Class IV and Class III supplies. Group 2 has Class III and Class II Supplies. Power is supplied to the SSDS from either the off-site network and/or the main generator. On start-up the SSDS is fed from off-site via the SST. During normal operation the SSDS is equally shared between the SST and the UST. Either transformer is sized for the total station service load and can thus supply the total service load in the event of failure of the other.	Portions of the Station Electrical Power System supplies process and instrumentation loads within the plant. This system is divided into two electrical supply groups. The normal power supplies the systems of the plant dedicated to normal power production. Safety related loads are supplied by qualified systems. Power is supplied from either the off-site network and/or the main generator.
Class IV Power Supply*	AC power to auxiliaries and equipment that can tolerate long duration interruptions. Only supplies Group 1 systems. Supplied from Turbine Generator or offsite Electric Network.	AC power to auxiliaries and equipment that can tolerate long duration interruptions. Supplied from turbine generator or offsite electric network.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Class III Power Supply*	AC power to auxiliaries necessary for safe shutdown of the reactor and turbine. Short interruptions can be tolerated. Supplies in both Group 1 and Group 2 systems. Normally supplied from the Class IV Power Supply, but can also be supplied from on-site standby diesel generators in either the Group 1 or Group 2 Service Buildings. NOTE: the Group 2 Class III system is to be seismically qualified.	AC power to auxiliaries necessary for safe shutdown of the reactor and turbine. Short Interruptions can be tolerated. Normally supplied from the Class IV Power Supply, but can also be supplied from on-site standby diesel generators. Seismically qualified to Canadian standards.
Class II Power Supply*	AC uninterruptible power for essential auxiliaries. Class II Supplies in both Group 1 and Group 2 systems. Channelized for redundancy requirements of station instrumentation and control systems. NOTE: The Group 2 Class II system is seismically qualified.	AC uninterruptible power for essential auxiliaries. Class II Supplies are channelized for redundancy requirements of station instrumentation and control systems.
Class I Power Supply*	DC uninterruptible power to essential auxiliaries. Triplicated and channelized for redundancy requirements of control logic and reactor safety circuits. Supplies in both Group 1 and Group 2 systems. NOTE: The Group 2 Class I system is seismically qualified.	DC uninterruptible power to essential auxiliaries is designed for redundancy requirements of control logic and reactor safety circuits. System is seismically qualified where applicable.
Group 1 Standby Generators	Two diesel generators that provide an alternative power source to specific station loads connected to the Class III system. The Group 1 diesels (if successfully started on failure of normal power sources) will feed both the Group 1 and 2 Class III electrical systems. The standby sets will accept key loads within 35 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days.	Diesel generators provide an alternative power source to specific station loads connected to the various systems. There are no Group 1 and 2 electrical system divisions. The standby sets will accept key loads within 35 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Group 2 Standby Generators	Two diesel generators that provides an alternative power source to specific station loads connected to the Class III system. The Group 2 diesels only start in the case of unsuccessful Group 1 diesel generators' operation or an earthquake. The Group 2 Diesel generators are seismically qualified. The Group 2 diesels can feed both the Group 1 and 2 Class III Electrical systems. The standby sets will accept key loads within 35 seconds, and full load within two minutes. The fuel system has the capacity to supply the diesel generators for seven days.	(see above).
Group 1 Station Batteries*	Sized to support all loads connected for up to 60 minutes following an interruption in normal Class III power source. Capable of providing uninterruptible power supplies (DC and AC) as required to Class II and Class I systems.	No Group 1 designation. Normal service batteries are sized to support all needed loads for up to 60 minutes following an interruption in a normal Class III source. Capable of providing uninterruptible power supplies (DC and AC) as required to Class II and Class I systems.
Group 2 Station Batteries*	Used for control logic, standby safety equipment, and switchgear operation. Seismically qualified. Capable of providing uninterruptible power supplies (DC and AC) as required to Class II and Class I systems.	No Group 2 designation. Seismically qualified batteries are used for control logic, standby safety equipment, and other safety functions. Capable of providing uninterruptible power supplies (DC and AC) as required to Class II and Class I systems.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Distributed Control System*	A collection of channelized local computers linked by dual-redundant data highway used to implement most of the Group 1 control functions. Digital data highways are used for signal transmission and programmable microprocessors for control logic. Sensors and actuators provide input and/or output to local stations located outside the reactor building. Major control functions (reactor regulation, heat transport pressure and inventory control, steam generator pressure and level control, etc.) are distributed among a number of small, DCS micro-processors.	CANDU 6 uses more conventional technology for control functions since these are older units.
Plant Display System*	Provides alarm message display, graphic data display, operator machine interface and data logging. Most information is provided via the DCS. The PDS also receives non-DCS information (safety system, meteorological monitoring, clocks, etc.) to provide a complete single source of information for the operators.	Provides alarm message display, graphic data display, operator machine interface, and data logging. The PDS also receives information from safety system, meteorological monitoring, clocks, etc. to provide a complete single source of information for the operators.
Main Control Room*	Provides the resources and environment to support the operator in the safe and efficient control of the plant. Located in the Reactor Auxiliary Building. The shift supervisor's office and the Work Control Office are located in close proximity. The computer equipment driving the MCR CRT's is located next to the MCR. The MCR is in the Group 1 area of the plant.	Older technology used in the typical CANDU 6 design. Provides the resources and environment to support the operator in the safe and efficient control of the plant. The shift supervisor's office and the Work Control Office are located in close proximity.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Alarm Annunciation*	Approximately 300 color coded alarm windows are provided in panels as a supplement to the CRT displayed messages and are completely independent from the CRTs and the PDS. CRTs provide indication of the last forty alarms and clearance messages. Audible signals are also provided to the operator.	Older technology used in the typical CANDU 6 design. Provides indication of alarms and clearance messages. Audible signals are also provided to the operator.
Secondary Control Area*	Post accident mode of operation is based in the Main Control Room unless the MCR is rendered uninhabitable, then control shifts to the SCA. The SCA has the capability for shutting down the reactor and performing the same long term control and monitoring in the post accident mode as that available in the MCR. The SCA is located in the Group 2 area.	Post accident mode of operation is based in the Main Control Room unless the MCR is rendered uninhabitable, then control shifts to the SCA. The SCA has the capability for shutting down the reactor and performing the same long term control and monitoring in the post accident mode as that available in the MCR.
Overall Plant Control System*	A digital processor based system that performs all major control functions. Preferred control is with the turbine leading and the reactor following by maintaining the secondary steam pressure constant. Alternate control allows the reactor to lead with the turbine following by maintaining the secondary steam pressure constant. Loads used to control secondary side pressure include the turbine generator, condenser steam discharge valves, and atmospheric steam discharge valves. Other involved control systems include steam generator level control, heat transport pressure control, heat transport inventory control, and reactor control.	Older technology used in the typical CANDU 6 design. Preferred control is with the turbine leading and the reactor following by maintaining the secondary steam pressure constant. Alternate control allows the reactor to lead with the turbine following by maintaining the secondary steam pressure constant. Loads used to control secondary side pressure include the turbine generator, condenser steam discharge valves, and atmospheric steam discharge valves. Other involved control systems include steam generator level control, heat transport pressure control, heat transport inventory control, and reactor control.

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN

MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Fire Prevention Systems	<p>These systems provide a defense-in-depth approach towards fire prevention, fire detection and suppression, and mitigation by separation, layout, and structure resistance to fire.</p> <p>Two fire water pumps are provided each with 100 percent design capacity. One pump is diesel driven the other electric on Class III electric power. A fire ring main is arranged to provide water to each building from both paths. Isolation valves are provided such that any break in the main may be isolated without losing water supply to more than one building.</p>	<p>These systems provide a defense-in-depth approach towards fire prevention, fire detection and suppression, and mitigation by separation, layout, and structure resistance to fire.</p> <p>Two fire water pumps are provided each with 100 percent design capacity. One pump is usually diesel driven the other electric. A fire main is arranged to provide water to each building. Isolation valves may be provided such that breaks in the main may be isolated without losing water supply to more than one building.</p>
Seismic Protection*	<p>The reactor building, reactor auxiliary building and all Group 2 structures are designed to withstand a Design Basis Earthquake (which is approximately equivalent to U.S. region III). The systems and equipment required to maintain reactor coolant pressure boundary integrity, reactor shutdown, decay heat removal, containment isolation and all safety support systems are also designed to withstand a design basis earthquake.</p>	<p>The reactor building, reactor auxiliary building and all safety structures are designed to withstand a Canadian Design Basis Earthquake. The systems and equipment required to maintain reactor coolant pressure boundary integrity, reactor shutdown, decay heat removal, containment isolation and all safety support systems are also designed to withstand a design basis earthquake.</p>
Station Life*	<p>Modular design which allows replacement of most components. Non-replaceable components are rated for 100 years.</p>	<p>Modular design allows replacement of most components. Design is for 30 year life with possible plant life extensions.</p>
Flux Detector Units	<p>Full length guide tube assemblies containing twelve well tubes. Two types of detectors are used. One has a rapid response to gamma and neutron fluxes; the other is slower and sensitive only to neutron flux.</p>	<p>Two types of detectors are usually in the design. One detector type has a rapid response to gamma and neutron fluxes.</p>

* = Design Differences

COMPARISON OF CANDU 3 STANDARD NUCLEAR PLANT DESIGN WITH THE CANDU 6 DESIGN		
MAJOR SYSTEM/ COMPONENT	CANDU 3 REACTOR	TYPICAL CANDU 6 REACTOR DESIGN
Ion Chamber units	Six Lead-shielded housings are mounted on the outside of the calandria shell, in which the chamber instruments and calibration shutters are installed. Ion chambers on the sides of the reactor provide signals to the two shutdown systems, while only chambers on the moderator room side provide signals to the reactor regulation systems.	Lead-shielded housings mounted on the outside of the calandria shell, in which the chamber instruments and calibration shutters are installed. Ion chambers on the sides of the reactor provide signals to the shutdown systems.

* = Design Differences

APPLICABILITY OF SELECTED ASME SECTION III
SUBSECTIONS TO THE CANDU 3

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1.0 INTRODUCTION

This report provides an assessment of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (B&PVC), Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components," to determine the applicability of selected Section III Subsections to the design and construction of the Atomic Energy of Canada Limited (AECL) CANDU 3 reactor.

This evaluation identifies, to the extent possible, the potential need for modification or revisions to the ASME B&PV Code, Section III, Division 1. A study was performed for Subsections NCA, "General Requirements for Division 1 and Division 2," NB, "Class 1 Components," NC, "Class 2 Components," and ND, "Class 3 Components" to identify potential changes.

Section 2.0 of this report discusses the analysis approach and applications. Conclusions are provided in Section 3.

2.0. DISCUSSION

The assessment involves five basic steps. The first step identifies and summarizes the unique design features of the CANDU 3. The second step reviews the applicable Canadian nuclear codes (ref. 3 through 15) for inclusion of ASME Section III, Subsections NCA, NB, NC, and ND requirements. The third step reviews the current ASME Section III, Subsection NCA, NB, NC, and ND for applicability to the CANDU 3 and identifies potential changes needed to address the CANDU 3 unique design features. The fourth step reviewed CANDU 6 and CANDU 3 design differences for potential revisions to CAN/CSA standards. The fifth step reviewed CANDU 3 design features against a currently licensed four loop PWR. Details involved in performing these steps are described below.

2.1. Unique Design Features of the CANDU 3

The following summarizes the unique design features of the CANDU 3 that are not presently addressed in the ASME B&PV Code.

1. The fuel channel pressure tube to end fitting connection is a roll-expanded tube joint.
2. The calandria assembly lattice tube to calandria tube sheet weld is not 100% radiographed.
3. The calandria tube sheet has hubs machined from rolled plate.
4. The calandria assembly lattice tube to fueling tubesheet joint is not a full penetration weld.
5. The calandria assembly lattice tubes are not radiographed within 20% of the final machined thickness.
6. The calandria assembly calandria tubes diametral internal support is unique to the CANDU design.
7. The calandria assembly calandria tubes use a rolled joint.
8. The CANDU design uses threaded coils or solid inserts.
9. The fuel handling equipment uses reinforced elastomer hoses.
10. The CANDU design uses fuel handling equipment channel closure and fueling machine safety locks.
11. The fuel handling equipment fueling machine mobile supports are unique to the CANDU design.
12. The fuel channel pressure tube, fuel channel pressure tube to end fitting, reactivity control unit liquid injection shutdown system

nozzle, reactivity control rod, thin-walled-large diameter zirconium alloy tubing, and zirconium alloy wire for fuel channel spacer materials are not addressed in ASME section III, Subsections NB/NC/ND.

2.2. Review of Canadian Nuclear Codes for ASME Class 1, 2 and 3 Systems and Components

1. Canadian Standards Association CAN/CSA-N285.0-M91, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants," (ref. 2) addresses the general requirements for the design, fabrication, and installation of pressure retaining systems, components, and their supports. CAN/CSA-N285.0-M91 also includes requirements of a regulatory nature that are commonly applied to CANDU power plants located in Canada. It is similar in scope to ASME Section III, Subsection NCA.
2. Canadian Standards Association CAN/CSA-N285.1-M91, "Requirements for Class 1, 2, and 3 Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants," (ref. 3) requires CANDU nuclear power plant Class 1, 2 and 3 pressure retaining systems and components to be constructed of materials that comply with ASME Section III, Articles NB/NC/ND-2000, "Materials." Design is to comply with ASME Section III, Articles NB/NC/ND-3000, "Design." Fabrication and installation are to comply with ASME Section III, Articles NB/NC/ND-4000, "Fabrication and Installation." Examination is to be performed according to ASME Section III, Articles NB/NC/ND-5000, "Examination." Testing is to be performed in accordance with ASME Section III, Articles NB/NC/ND-6000, "Testing." Overpressure protection is to comply with ASME Section III, Articles NB/NC/ND-7000, "Overpressure Protection." Compliance to ASME Section III, Articles NB/NC/ND-1000, "Introduction" and NB/NC/ND-8000, "Nameplates, Stamping, and Reports" is not required in CAN/CSA-N285.1-M91 (ref. 3). These ASME subjects are addressed in CAN/CSA-N285.0 (ref. 2).
3. Canadian Standards Association CAN/CSA-N285.2-M89, "Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants," (ref. 4) addresses pressure containing components of CANDU nuclear power plants that, because of their design concept, the rules of the ASME B&PV Code do not exist, are not applicable, or are not sufficient.
4. Canadian Standards Association CAN/CSA-N285.6 Series-88, "Material Standards for Reactor Components for CANDU Nuclear Plants," (refs 5 through 14) provide reference design data for alloys and component conditions not included in the ASME B&PV Code.

2.3. Review of ASME Section III and Analysis Approach

1. A preliminary review of the 1992 Edition, 1993 Addenda, of ASME, Section III, Division 1 was performed. Subsections NCA, NB, NC, and ND that are applicable to the safety related mechanical systems and components affected by the unique CANDU 3 features were considered.
2. The ASME portion of the review does not include ASME Section III, Division 1, Subsection NCA-4000, "Quality Assurance," ASME Section III, Division 2, "Code for Concrete Reactor Vessels and Containments," ASME Section III Division 1 Subsections NE, "Class MC Components," NF, "Component Supports," NG, "Core Support Structures," or electrical, and instrumentation and control systems.
3. The analysis was performed to identify potential impacts on ASME Section III to address unique features of the CANDU 3 designs. The approach included:
 - a) Identification of the unique features for each CANDU 3 design considered as presented in section 2.1 above. The CANDU 3 Technical Description (ref. 16) and CAN/CSA-N285.2-M89 (ref. 4) were reviewed to identify the major differences between this design and currently licensed light water reactor designs.
 - b) Identification of the individual articles, subarticles, subsubarticles, paragraphs, subparagraphs, and subsubparagraphs that might be effected to address the identified CANDU 3 unique features.
 - c) Presenting rationale for the potential revisions identified.
4. Assumptions and limitations for this analysis include:
 - a) CANDU 3 unique features were considered conceptually, and subjective judgement was used conservatively to identify potential impacts to ASME Section III. When detailed design specifications for the unique CANDU 3 features are available, the need for some identified potential revisions to ASME Section III may need to be reconsidered.
 - b) CANDU 3 unique features that involve extended design lives may raise degradation or failure mechanisms that were not previously relevant or significant. As such failure mechanisms become known, there may be a need for additional changes to the inspection and testing requirements or acceptance criteria to appropriately address these degradation and failure mechanisms.
5. Applications were performed for Subsections NCA, NB, NC, and ND. Results of the applications are summarized in Table 1.

- a) The analysis process was performed for ASME Section III, Subsection NCA, "General Requirements for Division 1 and Division 2." CAN/CSA-N285.0-M91, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants" (ref. 2) is a Canadian code similar in scope to ASME Section III, Subsection NCA, but is intended for construction of CANDU nuclear plants located in Canada. The review of ASME Section III, Subsection NCA did not result in the identification of any potential impacts.
- b) The analysis process was performed for ASME Section III, Division 1, Subsection NB, "Class 1 Components." This review resulted in identification of potential impacts to seven articles or subparagraphs as follows:
 - 1) NB-2000, "Material"
 - 2) NB-3327.1, "Positive Locking Devices"
 - 3) NB-3671.2, "Expanded Joints"
 - 4) NB-3000, "Design"
 - 5) NB-4000, "Fabrication and Installation"
 - 6) NB-5000, "Examination"
 - 7) NB-6000, "Testing"
- c. The analysis process was performed for ASME Code Section III, Division 1, Subsection NC, "Class 2 Components." This review resulted in identification of potential impacts to seventeen articles, paragraphs, subparagraphs or subsubparagraphs as follows:
 - 1) NC-1100(b), "Scope"
 - 2) NC-2000, "Material"
 - 3) NC-2572.1(b)(1), "Acceptance Examinations"
 - 4) NB-3000, "Design"
 - 5) NC-3133.3, "Cylindrical Shells and Tubular Products"
 - 6) NC-3352.4(d), "Attachment of Nozzles Using Partial Penetration Welds"
 - 7) NC-3327.1, "Positive Locking Devices"
 - 8) NC-3358.4, "Flat Heads and Tubesheets With Hubs"

- 9) NC-3671.2, "Expanded or Rolled Joints"
- 10) NB-4000, "Fabrication and Installation"
- 11) NC-4243.1, "Flat Heads and Tubesheets With Hubs"
- 12) NC-4244(d), "Attachment of Nozzles Using Partial Penetration Welds"
- 13) NC-4265.1, "Flat Heads and Tubesheets With Hubs"
- 14) NC-4266(d), "Attachment of Nozzles Using Partial Penetration Welds"
- 15) NB-5000, "Examination"
- 16) NC-5254, "Category D Welded Joints"
- 17) NB-6000, "Testing"

d. The analysis process was performed for ASME Section III, Division 1, Subsection ND, "Class 3 Components." This review resulted in identification of potential impacts to six articles, or subparagraphs as follows:

- 1) ND-2000, "Material,"
- 2) ND-3000, "Design"
- 3) ND-3327.1, "Positive Locking Devices"
- 4) ND-4000, "Fabrication and Installation"
- 5) ND-5000, "Examination"
- 6) ND-6000, "Testing"

3.0. SUMMARY AND CONCLUSIONS

The preliminary results show that the existing provisions of ASME Section III are generally applicable to the unique design features of the CANDU 3 design considered. For the ASME Sections selected for study (NCA, NB, NC, and ND), the results, as shown in Table 1, suggest the need for limited changes to ASME Section III.

Articles, subarticles, subsubarticles, paragraphs, subparagraphs, and subsubparagraphs within ASME Section III will need to be revised to completely address the unique features of the CANDU 3. ASME Section III, Division 1, may be revised to address the design changes for different failure mechanisms and the increase in plant design life (100 years).

This preliminary analysis was based on available information regarding the CANDU 3 design. Detailed design information on a certified CANDU 3 design, when available and reviewed, may reveal additional information that could alter or increase the identified potential impacts to ASME Section III.

4.0. REFERENCES

1. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, "Rules for Construction of Nuclear Power Plant Components," 1992 Edition, 1993 Addenda.
2. Canadian Standards Association CAN/CSA-N285.0-M91, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants."
3. Canadian Standards Association CAN/CSA-N285.1-M91, "Requirements for Class 1, 2, and 3 Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants."
4. Canadian Standards Association CAN/CSA-N285.2-M89, "Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants."
5. Canadian Standards Association CAN/CSA-N285.6-M88, "Material Standard for Reactor Components for CANDU Nuclear Power Plants."
6. Canadian Standards Association CAN/CSA-N285.6.1-M88, "Seamless Zirconium Alloy Tubing for Fuel Channels."
7. Canadian Standards Association CAN/CSA-N285.6.2-M88, "Seamless Zirconium Alloy Tubing for Reactivity Control Rods."
8. Canadian Standards Association CAN/CSA-N285.6.3-M88, "Annealed, Seamless Zirconium Alloy Tubing for Liquid Injection Shutdown System Nozzles."
9. Canadian Standards Association CAN/CSA-N285.6.4-M88, "Thin-Walled, Large-Diameter Zirconium Alloy Tubing."
10. Canadian Standards Association CAN/CSA-N285.6.5-M88, "Zirconium Alloy Wire for Fuel-Channel Spacers."
11. Canadian Standards Association CAN/CSA-N285.6.6-M88, "Inspection Criteria for Zirconium Alloys."
12. Canadian Standards Association CAN/CSA-N285.6.7-M88, "Zirconium Alloy Design Data."
13. Canadian Standards Association CAN/CSA-N285.6.8-M88, "Martensitic Stainless Steel for Fuel-Channel End Fittings."
14. Canadian Standards Association CAN/CSA-N285.6.9-M88, "Materials for Supports for Pressure-Retaining Items."
15. Canadian Standards Association B51-M1991, "Boiler, Pressure Vessel, and Pressure Piping Code."

16. AECL, CANDU Operations, "CANDU 3 Technical Description," revision 3, Sept. 1989, 74-01371-TED-001.
17. AECB Regulatory Document R-77, "Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems," 10/20/1987.
18. AECL, "CANDU 3 Conceptual Safety Report," March 1989.
19. 10CFR50.55a, "Codes and Standard".
20. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants".
21. NUREG 0800, "Standard Review Plan".

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Fuel Channel pressure tube to end fitting roll-expanded tube joint.	NB-3671.2, "Expanded Joints"	<p>ASME Section III, Subparagraph NB-3671.2 states expanded joints shall not be used. Subsection NB only addresses full penetration welded and flanged joints. The CANDU designed joint between the stainless steel end fitting and the zirconium alloy pressure tube cannot be welded. Geometry precludes a flanged connection.</p> <p>ASME Section III, Subparagraph NB-3671.2 will require revision to address the fuel channel pressure tube roll-expanded joint.</p> <p>ASME Section III, Articles NB-5000 and NB-6000 will require revision to address examination and testing of the pressure tube to end fitting roll-expanded tube joint.</p>
Fuel channel pressure tube to end fitting material.	NB-2000, "Material"	<p>ASME Sections II and III do not contain rules for the pressure tube to end fitting modified Type 403 martensitic material.</p> <p>ASME Section III, Article NB-2000 will require revision to allow use of the fuel channel pressure tube to end fitting material and/or revise the ASME B&PV Code, Section II to include a material specification and zirconium alloy design data similar to CAN/CSA-N285.6.5-88 and CAN/CSA-N285.6.8-88, respectively.</p>
Effects of radiation on the fuel channel pressure tube.	NC-1100(b), "Scope"	<p>ASME Subparagraph NB-1100(b) excludes radiation effects from the scope of ASME III, Subsection NB.</p>

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Fuel channel pressure tube material.	NB-2000, "Material"	<p>ASME Sections II and III do not contain rules for the zirconium pressure tube material.</p> <p>ASME Section III, Article NB-2000 will require revision to allow use of the fuel channel pressure tube material and/or revise the ASME B&PV Code, Section II to include a material specification and zirconium alloy design data similar to CAN/CSA-N285.6.1-88 and CAN/CSA-N285.6.8-88, respectively.</p>
Calandria assembly lattice tube to calandria tube sheet.	<p>NC-5254, "Category D Welded Joints"</p> <p>NC-4243.1, "Flat Heads and Tubesheets With Hubs"</p> <p>NC-3358.4, "Flat Heads and Tubesheets With Hubs"</p> <p>NC-4265.1, "Flat Heads and Tubesheets With Hubs"</p>	<p>ASME Section III, Paragraph NC-5254 requires 100% radiography of all full penetration welds. Current CAN/CSA-N285.2-M89 practice, due to geometry constraints, is to radiograph only a statistical sampling when automatically welded.</p> <p>ASME Section III, Paragraph NC-5254 will require revision to address the current CAN/CSA-N285.2-M89 statistical sampling practice.</p> <p>ASME Section III, Subparagraphs NC-4243.1, NC-4265.1 and NC-3358.4 do not allow hubs, for butt welding, to be machined from rolled plate. Literal interpretation of ASME Section III would suggest that forged hubs are required. This is impractical for the CANDU design due to distortion caused by the large number of welds. CAN/CSA-N285.2-M89 allows, as an alternative to ASME Section III, Subparagraph NC-4243.2,** hubs to be machined onto the calandria tube sheet.</p> <p>ASME Section III, NC-4243.1 may require revision to include the current CANDU practice.</p> <p>**ASME Section III, Subparagraph NC-4243.2 could not be found in the 1983, 1986, 1989, or 1992 Editions or Addenda to the ASME B&PV Code.</p>

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Calandria assembly lattice tube to fueling tubesheet joint.	<p>NC-3352.4(d), "Attachment of Nozzles Using Partial Penetration Welds"</p> <p>NC-4266(d), "Attachment of Nozzles Using Partial Penetration Welds"</p> <p>NC-4244(d), "Attachment of Nozzles Using Partial Penetration Welds"</p>	<p>ASME Section III, Subparagraph NC-3352.4 requires full penetration welds for nozzles greater than two inch nominal pipe size. The CANDU geometry precludes full penetration welds.</p> <p>ASME Section III, Subparagraph NC-3352.4 will require revision to address the CANDU design.</p> <p>ASME Section III, Paragraphs NC-4244 and NC-4266 may require revision to address the CANDU design.</p>
Calandria assembly lattice tubes.	NC-2572.1(b)(1), "Acceptance Examinations"	<p>ASME Section III, Subsubparagraph NC-2572.1(b)(1) requires radiography to be performed within 20% of the final machined thickness. Stepped lattice tubes are machined after installation. At this time, because of access limitations, radiography is not possible. CAN/CSA-N285.2-M89 allows, as an alternate to ASME Section III, Subsubparagraph NC-2577.2(a)1,** radiography to be performed in a partially machined condition that exceeds 120% of the finished thickness. ASME Section III, NC-2572.1(b)(1) will require revision to address this CANDU practice.</p> <p>** ASME Section III, Subsubparagraph NC-2277.2(a)1 could not be found in the 1983, 1986, 1989, or 1992 Editions or Addenda to the ASME B&PV Code.</p>

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Calandria assembly calandria tubes diametral internal supports.	NB/NC/ND/NF-3000, "Design," 4000, "Fabrication and Installation," 5000, "Examination," and 6000, "Testing"	ASME Section III does not contain rules for diametral internal supports. ASME Section III, Articles NB/NC/ND-3000, 4000, 5000, and 6000 will require revision to address the design, fabrication, examination and testing of diametral internal supports.
Calandria assembly calandria tube rolled joint.	NC-3671.2, "Expanded or Rolled Joints"	ASME Section III, Subparagraph NC-3671.2 may require revision to address the calandria tube rolled joint.
Reactivity control units liquid injection nozzle.	NB/NC/ND-3000, "Design," 4000, "Fabrication and Installation," 5000, "Examination," and 6000, "Testing"	ASME Section III, Subsection NB does not contain rules for the threaded joint connecting the zircalloy reactivity control unit liquid injection nozzle to a stainless steel tube. ASME Section III, Articles NB/NC/ND-3000, 4000, 5000, and 6000 will require revision to address the design, fabrication, examination and testing of the zircalloy injection nozzle to stainless steel tube screwed joint.
Reactivity control units liquid injection shutdown system nozzles material.	NB/NC-2000, "Material"	ASME Sections II and III do not contain rules for the liquid injection shutdown system zirconium nozzle material. ASME Section III, Article NB-2000 or NC-2000 (dependent upon classification) will require revision to allow use of the reactivity control units liquid injection shutdown system nozzle material and/or revise the ASME B&PV Code, Section II to include a material specification and zirconium alloy design data similar to CAN/CSA-N285.6.3-88 and CAN/CSA-N285.6.8-88, respectively.
Threaded coil or solid inserts	NB/NC/ND-3000, "Design," 4000, "Fabrication and Installation," 5000, "Examination," and 6000, "Testing"	ASME Section III does not contain rules for threaded coil or solid inserts. ASME Section III, Articles NB/NC/ND-3000, 4000, 5000, and 6000 will require revision to address the design, fabrication, examination and testing of threaded coil or solid inserts.

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Joints between tubular components.	NB/NC/ND-3000, "Design," 4000, "Fabrication and Installation," 5000, "Examination" and 6000, "Testing"	ASME Section III does not contain rules for all types and classification of joints. ASME Section III, Articles NB/NC/ND-3000, 4000, 5000, and 6000 will require revision to address the design, fabrication, examination and testing of other tubular joints required by the CANDU design.
Fuel handling equipment reinforced elastomer hoses.	NB/NC/ND-2000, "Material," 3000, "Design," 4000, "Fabrication and Installation," 5000, "Examination," and 6000, "Testing"	ASME Section III does not contain rules for nonmetallic hoses. ASME Section III, Articles NB/NC/ND-2000, 3000, 4000, 5000, and 6000 will require revision to address the material, design, fabrication, examination and testing of fuel handling equipment reinforced elastomer hoses.
Fuel handling equipment channel closure safety lock.	NB/NC/ND-3327.1, "Positive Locking Devices"	ASME Section III does not contain rules for CANDU channel closure locks. ASME Section III, subparagraphs NB/NC/ND-3327.1, may require revision to address the CANDU channel closure locks.
Fuel handling equipment fueling machine safety lock.	NB/NC/ND-3327.1, "Positive Locking Devices"	ASME Section III does not contain rules for the fueling machine safety lock. ASME Section III, subparagraphs NB/NC/ND-3327.1, may require revision to address the fueling machine safety lock.
Fuel handling equipment fueling machine supports.	NB/NC/ND-3000, "Design," 4000, "Fabrication and Installation," 5000, "Examination" and 6000, "Testing" or Subsection NF	ASME Section III does not contain rules for the fueling machine mobile supports. ASME Section III, Articles NB/NC/ND-2000, 3000, 4000, 5000, and 6000 and/or ASME Subsection NF, "Component Supports," will require revision to address the material, design, fabrication, examination and testing of fueling machine mobile supports.

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Reactivity control rod material.	NC/ND-2000, "Material"	<p>ASME Section III and Section II do not contain rules for the zirconium reactivity control rod material.</p> <p>ASME Section III, Articles NC/ND-2000 (dependent upon ASME classification) will require revision to allow use of the reactivity control rod material and/or revise the ASME B&PV Code, Section II to include a material specification and zirconium alloy design data similar to CAN/CSA-N285.6.2-88 and CAN/CSA-N285.6.8-88, respectively.</p>
Thin-walled, large diameter zirconium alloy tubing material.	NC-2000, "Material"	<p>ASME Section III and Section II do not contain rules for the thin-walled, large diameter zirconium alloy tubing material.</p> <p>ASME Section III, Article NC-2000 will require revision to allow use of thin-walled, large diameter zirconium alloy tubing material and/or revise the ASME B&PV Code, Section II to include a material specification and zirconium alloy design data similar to CAN/CSA-N285.6.4-88 and CAN/CSA-N285.6.8-88, respectively.</p>
Zirconium alloy wire for fuel-channel spacer material.	NB-2000, "Material"	<p>ASME Section III and Section II do not contain rules for the zirconium alloy wire for fuel-channel spacers material.</p> <p>ASME Section III, Article NB-2000 will require revision to allow use of zirconium alloy wire for fuel-channel spacer material and/or revise the ASME B&PV Code, Section II to include a material specification and zirconium alloy design data similar to CAN/CSA-N285.6.5-88 and CAN/CSA-N285.6.7-88, respectively.</p>

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Inspection criteria for zirconium alloys.	NB/NC/ND-2000, "Material"	<p>ASME Section III and Section II do not contain rules for inspection criteria for zirconium alloys.</p> <p>ASME Section III, Article NB/NC/ND-2000 (dependent upon ASME classification) will require revision to include inspection criteria for zirconium alloys and/or revise the ASME B&PV Code, Section II to include a specification similar to CAN/CSA-N285.6.6-88.</p>
Materials for supports for pressure retaining components.	NF-2000, "Material"	<p>ASME Section III, Subsection NF does not contain rules for some CANDU materials used for pressure retaining supports.</p> <p>ASME Section III, Article NF-2000 will require revision to allow use of CANDU materials for supports for pressure retaining components and/or revise the ASME B&PV Code, Section II to include a material specification similar to CAN/CSA-N285.6.9-88.</p>
Calandria tube determination of thickness due to external pressure.	NC-3133.3, "Cylindrical Shells and Tubular Products"	<p>ASME Section III, Subparagraph NC-3133.3 will require revision to utilize the CANDU practice of showing by test or analysis that the deformation of the calandria tube onto the pressure tube, due to external pressure under all service conditions, will not cause rupture of the calandria tube.</p>

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Component Classification.	NCA-2110(c), "Scope"	<p>ASME Section III NCA-2110(c) states guidance for the selection of specific component classifications is not provided. This guidance is found in engineering standards or in the requirements of regulatory and enforcement authorities.</p> <p>10CFR50.55a, "Codes and Standard," Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and NUREG 0800, "Standard Review Plan," Section 3.2.2, "System Quality Group Classification" address classification of components.</p> <p>CAN/CSA-N285.0 and the CANDU 3 Conceptual Safety Report (ref. 18), Safety Design Guide SDG-006, "Code Classification," address classification of CANDU 3 components.</p>
Component Classification (con't).		<p>10CFR50.55a, SRP 3.2.2 and Regulatory Guide 1.26 base all classification of components on 'importance to safety.' CAN/CSA-N285.0 and SDG-006 base Class 1 classifications on safety function of the major safety related systems and Class 2, 3 and 6 on the health effects of the release of radioactive materials contained in the systems due to a pressure boundary failure. This difference in classification philosophy would appear to result in differences in US versus Canadian classification of like components. Until such time as detailed CANDU 3 component classifications are provided this is an assumption. However, as an example, review of CANDU 3 classifications would classify Emergency Diesel Generator (EDG) support systems (fuel oil, starting system etc.) as Class 6 (similar to Regulatory Guide 1.26 Quality Group D). NUREG 0800, Section 3.2.2 reviews EDG support systems for classification as ASME Class 3 (Quality Group C).</p>

TABLE 1
CANDU 3 IMPACTS ON SELECTED ASME B&PV CODE SECTION III SUBSECTIONS

CANDU 3 DESIGN DIFFERENCES	ASME SECTION III IMPACTED	EXPLANATION OF ASME SECTION III POTENTIALLY IMPACTED
Shutdown systems as part of an integrated overpressure protection system.	ASME-NB-7110(d), "Scope," NB-7700, "Relieving Capacity"	<p>ASME Section III Subparagraph NB-7110(d) specifically omits design of reactor shutdown systems and their application as an integrated part of system overpressure protection. Guidance is referred to engineering standards and the appropriate regulatory and enforcement authorities.</p> <p>CAN/CSA-N285.1-M91 paragraph 7.2, "Overpressure Protection," allows capacity requirements to be determined by methods other than those permitted by the ASME B&PV Code for shutdown systems which are an integral part of an overpressure protection system. ASME Section III Subarticle NB-7300 and Subparagraph NB-7110(d) may require revision to address this Canadian practice. An alternative would be to revise NUREG 0800, "Standard Review Plan, " Section 5.2.2, "Overpressure Protection," and/or NUREG 0800, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports , and Core Support Structures," to address this Canadian practice.</p>

COMPARISON OF CANADIAN STANDARD CSA B51-M1991 TO ANSI/ASME
B31.1-1992 - (General Topic)

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1.0 INTRODUCTION

This report provides a comparison of the American National Standards Institute (ANSI) and American Society of Mechanical Engineers (ASME) B31.1, "Power Piping," to Canadian Standard CSA B51-M1991, "Boiler, Pressure Vessel, and Pressure Piping Code," which is cited for use in the design and construction of the Atomic Energy of Canada Limited (AECL) CANDU 3 reactor.

The scope of CSA B51 includes all boilers, pressure vessels, fittings, and piping. The scope of ANSI/ASME B31.1 includes the design, materials, fabrication, erection, test, and inspection of power and auxiliary service piping systems for electric generation stations, industrial and institutional plants, central and district heating plants and district heating systems. Piping as used in ANSI/ASME B31.1 includes pipe, flanges, bolting, gaskets, valves, relief devices, fittings, and the pressure containing portions of other piping components. U.S. constructed boilers and pressure vessels are addressed in ASME Section I, "Power Boilers," Section IV, "Heating Boilers," Section VII, "Pressure Vessels," or Section X, "Fiber-Reinforced Plastic Pressure Vessels."

This evaluation identifies, to the extent possible, the potential need for modification or revisions to ANSI/ASME B31.1.

2.0 DISCUSSION

The assessment involves two basic steps. The first step reviews the scope of CSA B51. The second step reviews the current ANSI/ASME B31.1 for applicability to the CANDU 3 and identifies potential changes needed to address the CANDU 3 design features.

The Scope of CSA B51-M1991, "Boiler, Pressure Vessel, and Pressure Piping Code" includes the following systems and components:

- 2.1 CSA B51 paragraph 5.8 "Anhydrous Ammonia, Liquefied Petroleum Gas, Liquefied Natural Gas and Compressed Natural Gas Service."
 - 5.8.1.1 *Anhydrous ammonia pressure vessels and piping used to store, handle and transport are to be designed, constructed and installed in accordance with ANSI K61.1 and CSA Standard B620.*
 - 5.8.2.1 *Pressure vessels and piping for liquefied gases must meet one of the following codes:*
 - o *Highway Vehicle Engine Fuel Tanks - ASME Code Section VIII, Division 1*
 - o *Ground Storage Tanks - ASME Code Section VIII, Division 1*
 - o *Bulk Transport Tanks (greater than 454 L water capacity) - ASME Code Section VIII, Division 1*
 - o *Portable Cylinders (water capacity 454 L or less) - CSA Standard CAN/CSA-B339*

5.8.3 *Liquefied natural gas containers design, materials, manufacturing procedures, testing inspection, and design approval registration are to meet CAN/CSA-B339.*

5.8.4 *Compressed Natural Gas (CNG) refuelling station pressure piping systems and containers are to meet CSA-B51, Appendix G. ANSI/ASME B31.1 is referenced throughout Appendix G for details of design and construction.*

ANSI/ASME B31.1 does not address the anhydrous ammonia, liquefied petroleum gas, liquefied natural gas and compressed natural gas service components of CSA B51 paragraph 5.8. However, these components are not within the scope of ANSI/ASME B31.1. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1.

2.2 CSA B51 paragraph 5.9 "Pressure Vessels for Highway Transportation of Dangerous Goods"

5.9 *Pressure vessels for highway transportation of dangerous goods are to be designed, constructed, installed, inspected and tested to CSA Standards B620 or CAN/CSA-B339.*

ANSI/ASME B31.1 does not address the pressure vessels for highway transportation of dangerous goods components of CSA B51 paragraph 5.9. However, these components are not within the scope of ANSI/ASME B31.1. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1

2.3 CSA B51 paragraph 6. "Boilers and Related Equipment"

6.1 *Boilers and related equipment are to be designed, constructed, installed, inspected, tested and repaired as required in the ASME Code:*

- o Section I - Power Boilers*
- o Section II - Material Specifications, Parts A, B, and C*
- o Section IV - Heating Boilers*
- o Section V - Nondestructive Examination*
- o Section VIII - Pressure Vessels, Division 1 and 2*
- o Section IX - Welding and Brazing Qualifications*
- o Part CW of ASME Safety Code CSD-1 - Controls and Safety Devices for Automatically Fired Boilers*

6.5 *Blow-Off vessels and piping, which is discharged into a sewer system, is to be designed and constructed in accordance with CSA B51.*

6.6 *Thermal fluid heaters and piping are to be designed in accordance with the ASME Code Sections I, IV or VIII. Thermal fluid piping is to be designed in accordance with ANSI/ASME B31.3.*

ANSI/ASME B31.1 does not address the boilers and related equipment components of CSA B51 paragraph 6. However, these components are not within the scope of ANSI/ASME B31.1. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1

2.4 CSA B51 paragraph 7. "Pressure Vessels"

7.1 *Pressure vessels are to be designed, constructed, installed, inspected, tested and repaired as required in the ASME Code:*

- o Section II - Material Specifications, Parts A, B, and C*
- o Section V - Nondestructive Examination*
- o Section VIII - Pressure Vessels, Division 1 and 2*
- o Section IX - Welding and Brazing Qualifications*
- o Section X - Fiber-Reinforced Plastic Pressure Vessels*

7.4.1 *Hot-water tanks are to be designed to meet the requirements of the ASME Code Section VIII, Division 1, lined potable hot-water tanks may be designed to the ASME Code Section IV (Part HLW).*

7.4.2 *Hydropneumatic tanks are to be designed to meet the requirements of the ASME Code Section VIII, Division 1.*

7.4.3 *Cushion tanks when meeting pressure and size limitations will be designed to meet the requirements of the ASME Code Section VIII, Division 1.*

ANSI/ASME B31.1 does not address the pressure vessels of CSA B51 paragraph 7. However, these components are not within the scope of ANSI/ASME B31.1. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1

2.5 CSA B51 paragraph 8. "Piping and Fittings"

8.1 *Piping and fittings are to be designed, constructed, installed, inspected, tested, and repaired in accordance with applicable ANSI Standards: B31.1, B31.2, B31.3, B31.4, B31.5, and B31.9; applicable CSA standards CAN/CSA-Z183, CAN/CSA-Z184, CAN/CSA-Z305.1, CAN/CSA-Z305.3, CAN/CSA-Z305.4; and applicable CGA Standards CAN1-B149.1 and CAN1-B149.2.*

Power piping and fittings are to meet all of the requirements of ANSI/ASME B31.1. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1

2.6 CSA B51 paragraph 9. "Refrigeration Equipment"

9.1 *Refrigeration equipment is to designed, constructed, installed, inspected, tested, and repaired In accordance with CSA Standard B52, "Mechanical Refrigeration Code".*

ANSI/ASME B31.1 does not address the refrigeration equipment components of CSA B51 paragraph 9. However, these components are not within the scope of ANSI/ASME B31.1. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1.

2.7 CSA B51 paragraph 10. "Pressure Coils in Petroleum and Chemical Plant Fired Heaters"

10.1 *Pressure coils in petroleum and chemical plant fired heaters, except those designed and fabricated to the ASME Code Section I or ANSI/ASME B31.1, are designed, constructed, installed, inspected, and tested in accordance with ANSI/ASME B31.3 or API RP530.*

Pressure coils in petroleum and chemical plant fired heater components of CSA B51 paragraph 10 are to meet all of the requirements of ANSI/ASME B31.1, ANSI/ASME B31.3 or API RP530. Therefore, the results of this review do not suggest a need for changes to ANSI/ASME B31.1

3.0 SUMMARY AND CONCLUSIONS

With the exception of liquefied natural gas containers, compressed natural gas refuelling station pressure piping systems and containers, pressure vessels for highway transportation of dangerous goods, refrigeration equipment, and blow-off vessels and piping discharged into a sewer system CSA B51 requires all other systems and components within its scope to meet U.S. Codes and/or Standards. None of the above are within the scope of ANSI/ASME B31.1.

CAN/CSA-N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants", Class 6 systems and components, are to be constructed to CSA B51. The comparison of CSA B51-M1991 and ANSI/ASME B31.1 shows that the existing provisions of ANSI/ASME B31.1 are applicable to the design features of the CANDU 3. As used in CSA B51, power piping and fittings are to be designed, constructed, installed, inspected, tested, and repaired in accordance with ANSI/ASME B31.1. CSA B51 does not relax the requirements of ANSI/ASME B31.1. The results of this review do not suggest a need for changes to ANSI/ASME B31.1.

4.0 REFERENCES:

CSA Standards

1. B52-M1991, "Mechanical Refrigeration Code"
2. CAN/CSA-N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants"
3. CAN/CSA-B339, "Cylinders, Spheres, and Tubes for the Transportation of Dangerous Goods"
4. B620, "Highway Tanks and Portable Tanks for the Transportation of Dangerous Goods"
5. CAN/CSA-Z183, "Oil Pipeline Systems"
6. CAN/CSA-Z184, "Gas Pipeline Systems"
7. Z305.1, "Nonflammable Medical Gas Piping Systems"
8. CAN/CSA-Z305.1, Not found
9. CAN/CSA-Z305.3, "Pressure Regulators, Gauges, and Flow-Metering Devices for Medical Gases"
10. CAN/CSA-Z305.4, "Qualification Requirements for Agencies Testing Nonflammable Medical Gas Piping Systems"

ANSI Standard

11. K61.1, "Safety Requirements for Storage and Handling of Anhydrous Ammonia"

ANSI/ASME Standards

12. B31.1, "Power Piping"
13. B31.2, "Fuel Gas Piping"
14. B31.3, "Chemical Plant and Refinery Piping"
15. B31.4, "Liquid Transportation Systems for Hydrocarbons, Liquid Petroleum Gas, Anhydrous Ammonia, and Alcohols"
16. B31.5, "Refrigeration Piping"
17. B31.9, "Building Services Piping Code"

ASME Boiler and Pressure Vessel Code

18. Section I, "Power Boilers"
19. Section II, "Material Specifications, Part A, Ferrous Materials"
20. Section II, "Material Specifications, Part B, Nonferrous Materials"
21. Section II, "Material Specifications, Part C, Welding Rods, Electrodes and Filler Metals"
22. Section IV, "Heating Boilers"
23. Section V, "Nondestructive Examination"
24. Section VIII, "Pressure Vessels, Divisions 1 and 2"
25. Section IX, "Welding and Brazing Qualifications"
26. Section X, "Fiber-Reinforced Plastic Pressure Vessels"
27. Part CW of ASME Safety Code CSD-1, "Controls and Safety Devices for Automatically Fired Boilers"

American Petroleum Institute

28. RP530, "Calculation of Heater Tube Thickness in Petroleum Refineries"

Canadian Gas Association

29. CAN1-B149.1, "Natural Gas Installation Code"

30. CAN1-B149.2, "Propane Installation Code"

REVIEW OF CANDU 3 AND U.S. REGULATORY AUTHORITY PIPE RUPTURE
PROTECTION REQUIREMENTS (System Design Guide - SDG-010)

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1.0 INTRODUCTION

CANDU 3 System Design Guide (SDG)-010 [1] defines and identifies high-energy systems of a CANDU 3 plant in which pipe ruptures must be postulated to occur, also systems that must be protected from the dynamic effects of such ruptures. This review attempts to locate U.S. regulatory documents similar in scope to SDG-010 and other documents that specify regulatory positions on postulated pipe ruptures.

2.0 DISCUSSION

The RECALL 94-01.all database was searched to determine which U.S. regulatory documents are similar in scope to SDG-010. The following search strings were used in succession:

1. "pipe rupture or pipe whip or jet impingement or restraint"
2. "standard review plan or srp w/3 "3.6.1" or "3.6.2""
3. "branch technical position or btp and (emeb or splb or asb or meb w/2 "3-1")"

Approximately eighteen hundred documents were identified, some of which were not pursued because they are not used by the NRC to distribute regulatory positions, e.g., NRC Bulletins, Licensee Event Reports. The remaining documents were reviewed to determine if they provided a regulatory position or were a basis document for a regulatory position. Regulatory position documents include documents that define systems to be analyzed, and systems that must be protected from pipe ruptures (See 4.0 REFERENCES).

2.1 US REGULATORY POSITION ON POSTULATED PIPING RUPTURES

10CFR50 Appendix A, General Design Criterion (GDC) 4 [2] and SRP Section 3.6.2 [4] pertaining to postulated double-ended guillotine breaks (DEGB) of the largest pipes and postulated arbitrary intermediate pipe breaks was changed to include more realistic criteria and to allow consideration and acceptance of validated analysis methods. The previous requirements of GDC 4 led to a situation where protective devices were added to forestall events that are extremely unlikely. These protective devices were designed for the extremely unlikely events could, however, reduce safety and increase worker radiation exposure under normal operations and design basis events.

In NUREG 1061, Volume 3 [5], the NRC Piping Review Committee outlined the limitations and general technical guidance on Leak Before Break (LBB) analyses to justify mechanistically that breaks in high energy fluid system piping need not be postulated.

Rulemaking changes to GDC 4 were made to redefine the need to consider the dynamic effects of pipe breaks. A proposed rule to modify GDC 4 was published [6] in July 1985 and codified leak-before-break technology, but was limited only to the primary loop piping of PWRs; the final rule was published in April 1986. A proposed broad scope rule dealing with all high energy piping in LWRs was published [7] in July 1986; the final rule was published [8] in October 1987. With the issuance of these revised rules, revisions to SRP Sections

3.6.1 [3] and 3.6.2 were needed to eliminate the postulation of arbitrary intermediate breaks. In addition, Generic Letter No. 87-11 [9] was issued to licensees detailing the relaxation in arbitrary intermediate pipe rupture requirements (BTP MEB 3-1) [10].

With the modification to GDC 4, the NRC has formalized the use of the LBB approach to justify the elimination of pipe whip restraints and jet impingement barriers as design requirements for a hypothetical double-ended guillotine break (DEGB) in high energy piping systems. Thus there is NRC recognition the LBB concept provides added margin above the ASME Code piping design structural margins.

2.2 SDG-010 POSITION ON POSTULATED PIPE RUPTURES

SDG-010 defines and identifies high-energy systems of a CANDU 3 plant in which pipe ruptures must be postulated to occur, also systems that must be protected from the dynamic effects of such ruptures.

Methods of protection are similar to SRP 3.6.1 and BTP SPLB 3-1 [11] and include:

- o Separation by distance of the required systems from the potential rupture location.
- o Provision of protective barriers between the potential rupture location and essential systems.
- o Provision of redundant systems/components so that if one is damaged, the other system/component is sufficient to carry out the required safety function.
- o Provision of pipe whip restraints.

SDG-010 does not require protection when, by performing a probabilistic evaluation, it can be shown that the acceptance criteria for Category B events, as specified in the CANDU 3 Standard Plant Licensing Basis, are met. An exception of this nature could not be located in the U.S. regulatory documents reviewed.

SDG-010 allows an LBB approach when a catastrophic pipe failure has a very low probability of occurrence and there is a basis to assert that a pipe would leak a detectable amount for a prolonged time before any crack could grow to a size that would result in a break.

3.0 CONCLUSION

GDC 4 and SRP Sections 3.6.1 and 3.6.2 provide the U.S. Regulatory basis for postulated pipe ruptures. SDG-010 is the Canadian basis for postulated pipe ruptures. SDG-010 does not provide the level of detail as to break location and analysis techniques as do SRP Sections 3.6.1 and 3.6.2. SRP Sections 3.6.1 and 3.6.2 do contain the requirements of SDG-010 except SDG-010 does not require protection when, by performing a probabilistic evaluation, it can be shown that the acceptance criteria for Category B events are met.

4.0 REFERENCES

1. CANDU 3 Conceptual Safety Report, System Design Guide SDG-006, rev. 0, "Pipe Rupture Protection"
2. 10CFR50 Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4, "Environmental and Dynamic Effects Design Bases"
3. Standard Review Plan - 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," Revision 1, October 1990
4. Standard Review Plan - 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping," Revision 1, July 1981
5. NUREG 1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," Volume 3, "Evaluation of Potential for Pipe Breaks," 1984.
6. Federal Register Notice 50 FR 27006, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," July 1, 1985.
7. Federal Register Notice 51 FR 26393, "10 CFR Part 50, Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," July 23, 1986.
8. Federal Register Notice 52 FR 41288, "10 CFR Part 50, Modification of General Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures," October 27, 1987.
9. Generic Letter 87-011, "Relaxation in Arbitrary Intermediate Pipe Rupture Requirements," June 1987
10. Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," June 1987
11. Branch Technical Position SPLB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," October 1990

Documents Providing Regulatory Guidance or Basis for Regulatory Positions

- o NUREG 0933, "A Prioritization of Generic Safety Issues," ITEM A-18: "Pipe Rupture Design Criteria," Revision 3, July 1991
- o NUREG 0933, "A Prioritization of Generic Safety Issues," ITEM B-6: "Loads, Load Combinations, Stress Limits," Revision 3, July 1991

- o NUREG 0933, "A Prioritization of Generic Safety Issues," ITEM 119.1: Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads," Revision 3, July 1991
- o Generic Letter 81-014, "Seismic Qualification of Auxiliary Feedwater Systems," February 1981
- o Generic Letter 86-001, "Safety Concerns Associated With Pipe Breaks in the BWR Scram System," January 1985
- o Generic Letter 89-010 Sup. 3, "Consideration of the Results of NRC-Sponsored Tests of Motor-Operated Valves," October 1990
- o NRC Inspection Manual Chapter 2512/13, "Inspection or Replacement of BWR Recirculation Piping," June 1984
- o Policy Statements - Power Plants, "Policy Statement on Additional Applications of Leak-Before-Break Technology, May 1989
- o Regulatory Guide - 1.57 Rev. 0, "Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components," June 1973
- o Standard Review Plan - 3.8.3, "Concrete and Steel Internal Structures of Steel or Concrete Containments," Revision 1, July 1981
- o Standard Review Plan - 3.8.4, "Other Seismic Category I Structures" Revision 1, July 1981

Documents Defining Regulatory Positions and Systems to be Analyzed

- o Regulatory Guide 1.70 Rev. 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, November 1978
- o Regulatory Guide 1.96 Rev. 1, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants," Revision 1, June 1976
- o Standard Review Plan - 3.8.2, "Steel Containment," Revision 1, July 1981
- o Standard Review Plan - 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment," Revision 2, July 1981
- o Standard Review Plan - 5.4.6, "Reactor Core Isolation Cooling System (BWR)," Revision 3, April 1984
- o Standard Review Plan - 5.4.7, "Residual Heat Removal (RHR) System," Revision 3, April 1984
- o Standard Review Plan - 6.2.1.1.C, "Pressure-Suppression Type BWR Containments Review Responsibilities," Revision 6, August 1984

- o Standard Review Plan - 6.2.3, "Secondary Containment Functional Design," Revision 2, July 1981
- o Standard Review Plan - 6.2.4, "Containment Isolation System," Revision 2, July 1981
- o Standard Review Plan - 6.3, "Emergency Core Cooling System," Revision 2, April 1984
- o Standard Review Plan - 9.1.2, "Spent Fuel Storage," Revision 3, July 1981
- o Standard Review Plan - 9.2.1, "Station Service Air System," Revision 4, June 1986
- o Standard Review Plan - 9.2.2, "Reactor Auxiliary Cooling Water Systems," Revision 3, June 1986
- o Standard Review Plan - 9.3.2, "Process and Post-Accident Sampling Systems Review Responsibilities," Revision 2, July 1981
- o Standard Review Plan - 9.3.4, "Chemical and Volume Control System (PWR) (Including Boron Recovery System)," Revision 2, July 1981
- o Standard Review Plan - 9.3.5, "Standby Liquid Control System (BWR)," Revision 2, July 1981
- o Standard Review Plan - 9.5.4, "Emergency Diesel Engine Fuel Oil Storage and Transfer System," Revision 2, July 1981
- o Standard Review Plan - 9.5.1, "Fire Protection Program," Revision 3, July 1981
- o Standard Review Plan - 9.5.5, "Emergency Diesel Engine Cooling Water System," Revision 2, July 1981
- o Standard Review Plan - 9.5.6, "Emergency Diesel Engine Starting System Review Responsibilities," Revision 2, July 1981
- o Standard Review Plan - 9.5.7, "Emergency Diesel Engine Lubrication System," Revision 2, July 1981
- o Standard Review Plan - 9.5.8, "Emergency Diesel Engine Combustion Air Intake and Exhaust System Review Responsibilities," Revision 2, July 1981
- o Standard Review Plan - 10.2, "Turbine Generator," Revision 2, July 1981
- o Standard Review Plan - 10.3, "Main Steam Supply System," Revision 3, April 1984
- o Standard Review Plan - 10.4.3, "Turbine Gland Sealing System Review Responsibilities," Revision 2, July 1981

- o Standard Review Plan - 10.4.4, "Turbine Bypass System," Revision 2, July 1981
- o Standard Review Plan - 10.4.6, "Condensate Cleanup System," Revision 2, July 1981
- o Standard Review Plan - 10.4.8, "Steam Generator Blowdown System (PWR),"
- o Standard Review Plan - 10.4.9, "Auxiliary Feedwater System (PWR)," Revision 2, July 1981
- o Standard Review Plan - 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," Revision 2, July 1981
- o Standard Review Plan - 15.2.8, "Feedwater System Pipe Breaks Inside and Outside Containment (PWR)," Revision 1, July 1981

REVIEW OF CANDU 3 AND THE ASME BOILER AND PRESSURE CODE
SECTION III COMPONENT CLASSIFICATION REQUIREMENTS - (SYSTEM
DESIGN GUIDE 6)

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1.0 INTRODUCTION

This report provides a comparison of the CANDU 3 Conceptual Safety Report, System Design Guide SDG-006, "Code Classification," and CAN/CSA-N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants" to the ASME Boiler and Pressure Code (B&PV) Section III classification criteria for systems and components. This evaluation identifies, to the extent possible, the potential need for modification or revisions to the ASME B&PV Code Section III.

2.0 DISCUSSION

Classification of CANDU 3 components is defined in Clause 5, "Classification," of CAN/CSA-N285.0 as follows:

Clause

5.3.1 Class 1 - Nuclear

Sections of systems, and systems connected thereto, containing fluid that directly transports heat from nuclear fuel, the failure of which would cause a loss of coolant accident as defined in the safety report, shall be classified as Class 1.

Clause

5.3.2 Class 2 - Nuclear

Sections of process systems that penetrate the containment structure and form part of the containment boundary shall be classified as Class 2 where required by Clause 5.4.1.1.

Clause 5.4.1.1 deals with components and extensions of the containment system.

Clause

5.3.3 Class 3 - Nuclear

Sections of systems, not classified as Class 1 or 2, that contain radioactive substances with a tritium concentration exceeding 0.4 TBq/kg (0.01 Ci/g), or a concentration of other radionuclides having a combined equivalent radioactivity, shall be classified as Class 3.

Clause

5.7 Class 1C, 2C, and 3C Components

Components that would otherwise be classified as Class 1, 2, or 3, but for which the requirements are not given by CSA Standard CAN/CSA-N285.1, or to which the requirements of CSA Standard CAN/CSA-N285.1 cannot be applied, shall be classified as Class 1C, 2C, or 3C, respectively.

Per SDG-006, CAN/CSA-N285.0 bases Class 1 and 1C classifications on safety function of the major safety related systems and Class 2, 2C, 3, and 3C on the health effects of the release of radioactive materials contained in the systems due to a pressure boundary failure.

CAN/CSA-N285.1 requires systems and components classified as Class 1, 2, or 3 to be designed, fabricated, installed, examined, tested, overpressure protected and materials to be according to the respective ASME B&PV Code Section III,

Articles (NX-2000, 3000, 4000, 5000, 6000, and 7000). The ASME B&PV Code Section III, Articles NX-1000, "Introduction," and NX-8000, "Nameplates, Stamping, and Reports" are addressed in CAN/CSA-N285.0.

The Preface of CAN/CSA-N285.2, "Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants" states in part:

This Standard is intended to provide uniform rules for the design, fabrication, examination, and inspection for such components and supports where rules are needed to complement those specified in the ASME Boiler and Pressure Vessel Code.

The specific objective of this Standard is to establish rules relating to design, fabrication, examination, testing, inspection, and documentation, where such rules differ from those specified in CSA Standards CAN3-N285.0 and CAN3-N285.1.

The Foreword of CAN/CSA-N285.2 states in part:

It is recognized that the ASME Boiler and Pressure Vessel Code constitutes a very valuable set of requirements for nuclear plant construction; however, although trying to be general, the ASME rules basically address a pressure vessel reactor such as the PWR or BWR type rather than a pressure tube reactor such as CANDU. Modification of the ASME rules as well as additional rules are therefore required to address both the administrative system and the design concepts, which are uniquely Canadian. CSA Standard CAN3N285.0 provides the general requirements, most of which govern the Canadian administrative system of Classification, Registration, and Quality Assurance, where they differ from ASME Boiler and Pressure Vessel Code Section III, Subsection NCA.

This Standard (N285.2) provides the technical rules to complement those of ASME Section III. In some cases ASME Code requirements are modified to properly relate the construction of specific CANDU components to the intent of the ASME Code. In other cases, additional rules are provided in this Standard where the ASME Code does not address Canadian needs.

... it was intended to have separate classification (Class Special or Class 5) for components that did not strictly comply with the ASME Boiler and Pressure Vessel Code. A change has been made to discontinue Class Special and designate such components as Class 1C, 2C, or 3C, with Class 1C covering components for installation in Class 1 systems that meet Class 1 requirements as complemented by the rules of this Standard. Similarly, Class 2C, and Class 3C cover components for Classes 2 and 3 systems, respectively.

... The criteria that were adopted state that the purpose of N285.2 is to provide rules for the nuclear pressure-retaining components of CANDU nuclear power plants where, because of the design concept, the rules of the ASME Boiler and Pressure Vessel Code as required by CSA Standard CAN3-N285.1, (a) do not exist, or (b) are not applicable, or (c) are not

sufficient.

An example of the ASME B&PV Code Section III rules which do not exist for components is from the CAN/CSA-N285.2 Foreword and states in part as follows:

It was decided that this Standard should provide rules for elastomeric components, where the consequence of their failure is a significant release of fluid. Items such as hoses are addressed, but seal components such as "O" rings and gaskets, the failure of which results in a leak rather than rupture, are not covered.

An example of the ASME B&PV Code Section III rules that are not sufficient is provided in the Forward to CAN/CSA-N285.2 and is stated in part as follows:

Fuelling machine supports are addressed in this Standard. Portions of them have a mobility not usually found in supports for pressure-retaining components. As an example, an elevating bridge and carriage is commonly used to move a fuelling machine from one reactor channel to another for on-power fuelling. A mechanism such as a ball screw and nut assembly is often used to produce bridge and carriage motion and to provide support. Rules in this Standard primarily cover the construction of these mechanisms and the control and interlock requirements necessary to prevent overstressing of the pressure boundary, or its support, caused by inappropriate support movement. Reference is made to the ASME Code, Section III, Subsection NF, for structural support elements.

The ASME B&PV Code Section III, Article NCA-2000, "Classification of Components," subarticle NCA-2110(c), "Scope," states in part:

... this Section does not provide guidance in the selection of a specific classification to fit a component in a given system. Such guidance is derived from systems safety criteria for specific types of nuclear power systems, such as pressurized water reactors, . . . , and may be found in engineering standards or in the requirements of regulatory and enforcement authorities having jurisdiction at the nuclear power plant site.

NCA-2110(d) states in part:

The Owner of a nuclear power plant shall be responsible for applying system safety criteria to classify the equipment in the nuclear power plant . . .

Regulatory authority requirements are found in 10CFR50.55a, "Codes and Standards," and guidance in Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and NUREG 0800, "Standard Review Plan," Section 3.2.2, "System Quality Group Classification."

10CFR50.55a, SRP 3.2.2 and Regulatory Guide 1.26 base all classification of components on importance to safety. As an example of the potential difference in U.S. and Canadian classification criteria, review of CANDU 3 classification criteria would classify Emergency Diesel Generator (EDG) support systems (fuel

oil, starting system etc.) as Class 6 (similar to Regulatory Guide 1.26 Quality Group D). However, NUREG 0800, Section 3.2.2 reviews EDG support systems for classification as the ASME B&PV Code Section III Class 3 (Quality Group C).

3.0 SUMMARY AND CONCLUSIONS

The ASME B&PV Code Section III does not presently provide classification criteria for components. It is the Owner's responsibility to classify components in accordance with enforcement and regulatory requirements. CAN/CSA-N285.0 Classes 1, 2 and 3 are systems and components which can be constructed to the current ASME B&PV Code Section III. CAN/CSA-N285.0 Classes 1C, 2C and 3C are for components that would be classed as Class 1, 2, or 3, except that the requirements for those classes are either not suitable or not sufficient in the ASME B&PV Code Section III.

The results of this review do not suggest a need for changes to the classification criteria in the ASME B&PV Code Section III. However, technical revisions to the ASME B&PV Code Section III and/or regulatory documents will be required to address the uniquely designed and constructed CANDU 3 Class 1C, 2C and 3C components.

4.0 REFERENCES:

1. ASME Boiler and Pressure Vessel Code Section III, "Rules for Construction of Nuclear Power Plant Components," Subsection NCA, "General Requirements for Division 1 and Division 2," 1992, 1993 Addenda
2. ASME Boiler and Pressure Vessel Code Section III, Article NX-1000, "Introduction"
3. ASME Boiler and Pressure Vessel Code Section III, Article NX-2000, "Materials"
4. ASME Boiler and Pressure Vessel Code Section III, Article NX-3000, "Design"
5. ASME Boiler and Pressure Vessel Code Section III, Article NX-4000, "Fabrication and Installation"
6. ASME Boiler and Pressure Vessel Code Section III, Article NX-5000, "Examination"
7. ASME Boiler and Pressure Vessel Code Section III, Article NX-6000, "Testing"
8. ASME Boiler and Pressure Vessel Code Section III, Article NX-7000, "Overpressure Protection"
9. ASME Boiler and Pressure Vessel Code Section III, Article NX-8000, "Nameplates, Stamping, and Reports"
10. 10CFR50.55(a), "Codes and Standards"
11. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
12. NUREG 0800, "Standard Review Plan," Section 3.2.2, "System Quality Group Classification"
13. CANDU 3 Conceptual Safety Report, System Design Guide SDG-006, rev. 2, "Code Classification"
14. CAN/CSA-N285.0, "General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants"
15. CAN/CSA-N285.2, "Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants"

**APPLICABILITY OF THE ASME CODE
SECTION XI (INSERVICE INSPECTION)
TO THE CANDU 3 POWER PLANT DESIGN**

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1.0 INTRODUCTION

1.1 Purpose and Basic Approach

The purpose of this report is to assist with the NRC review in evaluating the applicability of Section XI of the ASME Code (also called "ISI Code", because it covers Inservice Inspection) to the CANDU 3 design. The intent of this review is to:

1. Give a concise structural comparison between Section XI of the ASME Pressure Vessel Code and the corresponding Canadian ISI Codes N285.4, N285.5 and SDG-007 (Safety Design Guide 007); this is presented, partly in tabular form, in Section 3 of this report.
2. Identify the special features of the CANDU-3 design which might not directly fit into the existing Section XI requirements; this is presented, mostly in tabular form, in Section 4 of this report.
3. Provide an evaluation of the Canadian Regulatory Policy Statement AECB R-7, as it relates to ISI requirements; this evaluation is provided in Section 5 of this report.

Certain Parts of the ASME/ANSI O&M Code (also called "IST Code", because it covers Inservice Testing) are presently part of Section XI. According to industry sources this is a temporary condition; the O&M items will be removed from Section XI in the near future. With respect to plant safety and reliability, the basic intent and structure of the ASME Code addresses three distinctly different issues. The first issue is the quality of the plant as initially designed and built. This issue is addressed by Section III which assures the quality for all plant structures and equipment at the time of the initial installation. The second and the third issues come into play during the lifetime of the plant. These issues are code maintenance of active equipment (performance testing) and inspection of the passive components of the equipment. Performance testing is addressed by the IST Code (O&M Code), as an integral part of the maintenance program. Performance testing by the IST Code assures not only the basic operability of the Class 1 and Class 2 plant systems, but it demonstrates actual performance levels (flow rates, pump heads, pressure drops, vibration readings, etc). The third issue, addressed by Section XI, is called the ISI Code. It covers potential aging degradation such as erosion, corrosion, crack formation and crack growth. ISI is focused on passive components: integrity of welds, wall thickness of pipes, flaw detection, incipient crack detection, etc. For the primary pressure boundary

and for the containment, ISI deals with two aspects: leak tightness and structural integrity (avoidance of erosion, fatigue, plastic deformation, etc). ISI is basically a pressure boundary examination. The most important pressure boundaries are the ones of Class 1 equipment and the containment. ISI is typically performed on a 10 year cycle, while IST is performed on a shorter cycle (quarterly or monthly). This report does not address the IST items which are presently incorporated (by reference) in Section XI (1992 Edition); this report only addresses ISI issues.

The N285 Series of the Canadian Codes has incorporated much of Section III of the ASME Code. However, this is not the case with Section XI and the "Periodic Inspection" portion of the Code Series N285. The area covered by Section XI is codified in the Canadian Codes N285.4 and N285.5 and in SDG-007 (Safety Design Guide 007); although SDG-007 is formally called a "Guide", compliance with this Guide is mandatory (as stated in the Guide), and as such SDG-007 is de facto part of the Canadian Codes.

Upon first examination, the scope and the intent of the Canadian Codes appears to be the same as scope and intent of Section XI, although the Canadian approach is structured differently. Some of the readily apparent differences are:

1. Compared to Section XI, the Canadian Codes N285.4 and N285.5 are generally performance based and consequently less prescriptive and shorter.
2. The ASME Code uses a fairly complex structure of Sections, Subsections, Parts, Subparts, Articles, Subarticles, Subsubarticles, Mandatory and Nonmandatory Appendices which are marked with articulated identifiers composed of Roman numerals, Arabic numerals, and Latin letters. The Canadian Code simply consists of Clauses which are ordered in a plain decimal numbering pattern.
3. The Canadian Codes N285.4 and N285.5 are strictly ISI codes, with no training issues mixed in. Section XI contains detailed training, examination and certification requirements for NDE personnel. When specifying personnel qualification requirements, the Canadian Codes reference the "CGSB 48-GP" Series of Standards, which deals with training and requalification requirements for NDE personnel. The Canadian Codes address training in a training and requalification code, while the inservice inspection code deals with inservice inspection.
4. The Canadian Codes N285.4 and N285.5 are strictly ISI codes, with no maintenance (O&M) issues mixed in. As mentioned above, Section XI (1992 Edition) contains O&M items (inservice performance testing) in Subarticles IWF-5200, IWP-1100 and IWV-1100. According to industry sources, the ASME Code Committee is planning to remove all O&M items

from Section XI, so that performance testing requirements will then reside in the O&M Code only.

5. Both the Canadian and the ASME Codes are based upon sampling of the inspection items (welds, bolts, flanges, etc). The Canadian Code allows certain sample sizes to be reduced when performing the inspection on the second, third, or fourth unit of a set of identical plant units, of which the lead unit has been inspected first, if all the NDE indications on the lead unit (the oldest one of the set) are normal. Such multi-unit reduction factors are not allowed in Section XI, and they might not be justifiable for the U. S. plant population.
6. The inspection pattern in Section XI is based on the Code Classes of the systems. By contrast, the Canadian Codes N285.4 and N285.5 do not use Code Classes for this purpose; they use Inspection Categories instead. There are four Inspection Categories, and plant systems are assigned to these Categories with no regard to the Code Classes they are in.

When taking a closer look at N285.4, N285.5 and SDG-007, it becomes somewhat difficult to ascertain the equivalence of the Canadian Codes and Section XI. Because the structured approach in the Canadian Periodic Inspection Codes differs significantly from the structure of Section XI, this applicability evaluation has been performed in several steps.

The remainder of Section 1.1, "Purpose and Basic Approach", provides a short summary of the general design features of the CANDU-3 design. Section 2 of this report attempts to increase the reader's knowledge of the ISI subject by giving various perspectives. Section 2.1 shows that the basic definition of Code Classes is largely identical between the two countries, except for Canadian Classes 4 and 6. However, the philosophy of assignment of plant systems to the various Code Classes is different, with several CANDU-3 systems being in Class 1 which by U. S. standards would be in Class 2. Section 2.2 shows that the appropriate section of the Code of Federal Regulations is directly applicable, without modification, to the CANDU-3 design. Section 2.3 then provides a short discussion of the structure of ASME Code Section XI. Section 2.4 of this report discusses the Canadian Code series N285. Section 2.5 addresses the related Safety Design Guides SDG-006 and SDG-007, and Section 2.6 provides a short evaluation of Canadian Policy Statement AECB R-7.

Section 3 of this report was originally intended to be written in tabular format only, as in Table 3.1, which shows a side by side comparison of the main technical issues between the Canadian and the ASME Codes. When it became apparent that the inspection methodology in N285.4 and N285.5 is not based on Code Classes (as Section XI is), it was decided that some of the material had to be left in text format in order to retain important procedural details. The purpose of Section 3 is to answer the question: What are the fundamental differences between ASME Code Section XI and the corresponding Canadian Codes?

Section 4 of this report consists mainly of Table 4.1. This Table lists special design features of the CANDU-3 design and quotes the applicable portion of Section XI which corresponds to the quoted design feature, or it points out the lack of a corresponding portion of Section XI. The purpose of Section 4 is to answer the question: To what degree can ASME Code Section XI be directly applied to the CANDU-3 Design?

Section 5 of this report addresses Canadian Policy Statement AECB R-7. Section 6 provides a few final thoughts and conclusions. References are listed in Section 7.

There are a few subtle vocabulary differences between the Canadian and the U. S. Codes, but they are very minor. A few examples are:

<u>Canadian Term</u>	<u>U. S. Term in ASME Code</u>
Authorized Inspector	Qualified Examiner (the title of "Inspector" is reserved for the ANI, the Authorized Nuclear Inspector)
Periodic inspection	Inservice inspection
Inaugural inspection	Preservice inspection
Pressure containing surfaces	Pressure retaining components
Definitions	Glossary
Clause	Subsection, Subsubsection, Paragraph, Subparagraph, etc.
Bq (Bequerel)	Ci (Curie); 1 Ci = 37.0 GBq
Sv (Sievert)	rem; 1 Sv = 100 rem

1.2 General CANDU-3 Design Features

One of the most unusual features of the CANDU-3 reactor is the pressure tube, which is made of zirconium-niobium alloy and forms the primary pressure boundary. Another unusual feature is the rolled joint connection between the pressure tube and its two endpieces which are of martensitic stainless steel. The ASME Code requires that the pressure tube be of steel, and that the two end joints be welded joints. A steel pressure tube which the ASME Code requires would not permit the operation of the CANDU-3 reactor with natural uranium fuel. This fuel has an enrichment level of only about 0.71 weight

percent of the fissionable isotope U-235, and steel has too large a neutron absorption cross section to allow satisfactory neutron flux levels with such fuel.

The calandria, classified as an ASME Class 1 component in the Canadian design, is also unique. It is not part of the primary pressure boundary, and therefore it should be ASME Class 2 as far as leak inspection is concerned (Subsection IWC). The calandria (particularly the two tube sheets forming the end shield at each end) serves as a core support structure, which means that for structural purposes certain parts the calandria should be assigned to ASME Class 1 (Subsection IWF). The designers of the CANDU-3 reactor have classified the leak inspection aspect of the calandria also as Class 1 (Subsection IWB), possibly because the heavy water is very expensive and therefore warrants the increased leak inspection effort. Leak detection equipment is used extensively wherever heavy water is present; even moisture from the containment atmosphere is recondensed to preserve the valuable inventory of deuterium oxide. Another reason for having the calandria assigned to ASME Class 1 is based on the Canadian Code Class assignment philosophy described in SDG-006 (see Section 2.5 of this report).

The level of functional redundancy achieved by the Canadian Group 1, Group 2 configuration is unique to the CANDU-3 design. Group 1 consists of commercial grade equipment and structures, while Group 2 consists of ASME Class 1, seismically qualified equipment and structures. The Group 2 service building contains a full size second feedwater system (in addition to an auxiliary feedwater system), a complete second main control room, a second set of diesel generators (in addition to the Group 1 diesel generators, each set of which is sufficient), a second residual heat removal system (called Group 2 Shutdown Heat Removal System), and the Group 2 liquid reactor shutdown system (gadolinium nitrate injection). There is also a Group 2 Plant Service Water Pumphouse in addition to the Group 1 Service Water Pumphouse.

The CANDU-3 design for component stresses is also more conservative in some aspects. For instance, the Group 2 structures and equipment are designed to a seismic peak ground acceleration of 0.3 g, where the pressure boundary equipment is limited to Service Level C, as defined in ASME Code Section III, under seismically induced loads. By comparison, the U. S. design philosophy allows Service Level D for this type of scenario.

A few of the basic design data for the CANDU-3 reactor are summarized in Table 1.1 and contrasted with the corresponding data for a Westinghouse four loop PWR (Pressurized Water Reactor).

Table 1.1 Primary System Data, CANDU-3 versus Westinghouse 4 Loop

<u>Item</u>	<u>CANDU-3</u>	<u>WX 4 Loop</u>
Type of Reactor	Heavy Water Reactor	Light Water Reactor
Configuration of Reactor	232 Parallel Pressure Tubes	Single Pressure Vessel
Type of Coolant	Deuterium Oxide	Hydrogen Oxide
Reactor Outlet Coolant Temperature	590 F	590 F
Saturation Pressure at 590 F	1,450 psi	1,430 psi
Reactor Outlet Coolant Pressure	1,450 psi	2,250 psi
Reactor Thermal Power	1.306 MBtu/s	3.233 MBtu/s
Reactor Coolant Flow	11.49 Klb/s	39.48 Klb/s
Specific Coolant Flow	8.79 lb/KBtu	12.2 lb/KBtu
Number of Primary Coolant Pumps	2	4
Number of Steam Generators	2	4
Reactor Pressure Boundary Material	Zr-Nb Alloy	Steel

2.0 DISCUSSION

2.1 Canadian versus ASME Code Classification Assignments

The concept of Code Classes is used to organize and standardize, in a prioritizing pattern, the multitude of inspection, fabrication and maintenance requirements into distinct activity templates. The Code defines such a set of templates, called Code Classes, each with its specific set of requirements. Using the Code then requires assignment of plant systems to the various Code Classes. The plant owner surveys the plant systems and assigns each system to its appropriate Code Class by following a decision pattern based on a specific Code Class Assignment Philosophy. ASME Classes 1, 2 and 3 are used in the Canadian Code, with Canadian Classes 4 and 6 added; this is the set of templates (Code Classes) of the Canadian Code. The assignment philosophy used in assigning plant systems to these templates (Code Classes) is different between the Canadian and the U. S. in practice.

In the U. S. nuclear industry, there is a general (historically based) perception that the plant designer and/or owner is free to assign the ASME Code classification to the plant systems, while for the Canadian industry the Code Class assignment is prescribed in SDG-006 and Clause 5.3 of N285.0. In practical reality, the Code Class assignment in the U. S. industry has been codified also, although this is somewhat less obvious than with the Canadian Code.

The Code of Federal Regulations (Reference 1) provides the direct basis for the assignment to Code Class 1. 10CFR50.55a, Paragraph (c) (1) requires that the primary pressure boundary of the reactor plant (Quality Group A) shall meet the requirements of Class 1 as defined in ASME Code Section III. Paragraph (c)(2) then specifies minimum requirements for Code boundaries where Class 2 systems connect: two automatically actuated valves in series, or a pipe diameter small enough that coolant makeup capability for the reactor is not affected in case of pipe break. 10CFR50.55a, Paragraph (d) imposes Class 2 requirements for systems and components of Quality Group B; and finally, 10CFR50.55a, Paragraph (e) imposes Class 3 (ASME Section III) requirements for systems and components of Quality Group C.

Regulatory Guide 1.26 (Reference 2) defines the four Quality Groups A, B, C and D. These Quality Groups are defined such that Quality Group A has to meet the ASME Class 1 requirements, Quality Group B has to meet the ASME Class 2 requirements, Quality Group C has to meet the ASME Class 3 requirements, and Quality Group D has to meet the requirements of ANSI Standard B31.1. Since Quality Group A has already been (by law) assigned to ASME Class 1 in 10CFR 50.55a, Regulatory Guide 1.26 provides recommended system assignments for Quality Groups B thru D.

Quality Group B is designated for reactivity shutdown systems, residual heat removal systems, emergency core cooling systems, post accident containment heat removal systems, and post accident fission product removal systems. For PWRs, the in-containment portions of the secondary system are also assigned to Quality Group B.

The next group, Quality Group C (ASME Class 3 requirements), is for systems which are not directly involved in the reactivity shutdown or heat removal functions specified for Group B, but are still connected with Group B systems in some auxiliary fashion, plus any systems which contain radioactive material in such a way that a postulated system failure might result in conservatively calculated potential offsite doses exceeding 0.5 rem (no time interval specified). Finally, any system not assigned to Group A, B or C, which may contain radioactive material, is assigned to Quality Group D, which means that it has to meet the requirements of ASME/ANSI Standard B31.1 (the basic Power Piping Code).

For the Canadian industry, Section "3.0 CODE CLASSIFICATION PHILOSOPHY" in SDG-006 (Reference 12) describes the rationale underlying the classification framework defined in N285.0. Canadian Code Classes 1, 2 and 3 are used as defined in Section III of the ASME Code. Canadian Code Classes 1C, 2C and 3C are defined to be analogous to ASME Section III Code Classes 1, 2 and 3, for the purpose of covering systems which the ASME Code does not address (expansion joint testing for zirconium-niobium pressure tube, radiography requirements for lattice tube weld on calandria tubesheet, etc). In order to keep things comparable between the Canadian Code and the ASME Code, the Canadian Code Committee has assigned the material for Classes 1C, 2C and 3C to Code N285.2, while Classes 1, 2 and 3 are addressed in N285.1; this makes N285.1 more analogous to Section III. In addition to these three Classes, Section 3.0 of SDG-006 defines Canadian Code Class 4 for containment systems (corresponding to ASME Class MC, Subsection NE of Section III and Subsection IWE of Section XI) and Canadian Code Class 6 for systems containing only limited amounts of radioactive materials so that acceptable dose limits would not be exceeded in case of system failure.

While the definition of the Code Classes is largely identical or parallel to the U. S. approach, system assignments to these Code Classes (described in Section "4.0 CLASSIFICATION REQUIREMENTS" of SDG-006) is significantly different from the U. S. assignment methodology. First, not only the primary pressure boundary is assigned to Code Class 1, but also the emergency core cooling systems and the reactivity shutdown systems are in Class 1; both of the latter systems would be in Class 2 by U. S. philosophy. The Canadian approach here is more conservative; as mentioned in the Introduction of this report, the high cost of heavy water may have provided the motivation for imposing additional inspection requirements for systems which normally would be in Class 2. Second, the Canadian assignments to Classes 2, 3 and 6 are based on potential worker exposure considerations (instead of plant functions or coolant inventory considerations). Class 2 systems are those not in Class 1 which

have the potential of causing a plant worker exposure of more than 500 mSv (= 50,000 mrem = 50 rem) in case of postulated failure. Similarly, Class 3 systems are those not in Classes 1 or 2 which have the potential of causing a plant worker exposure of more than 100 mSv (= 10,000 mrem = 10 rem) in case of postulated failure. This means that failure of a Class 3 system can cause more than 100 mSv but less than 500 mSv of exposure. Assignments for Classes 1C, 2C and 3C are strictly analogous to the assignments for Classes 1, 2 and 3. The containment, by definition, is assigned to Canadian Class 4. Canadian Class 6 is for systems which, in case of postulated failure, cannot cause worker exposures above 100 mSV (10 rem).

Clause 5.3.3 of N285.0 gives an additional definition for a Class 3 system assignment: a system which is not in Classes 1 or 2, which carries radioactive substances with a specific activity in excess of 0.4 TBq/Kg (= 0.4 GBq/g = 11 mCi/g) are assigned to Code Class 3. Similarly, Clause 5.3.4 of N285.0 assigns a system to Class 6 based on specific activity: a system is assigned to Code Class 6 if it carries radioactive substances with a specific activity of less than 0.4 TBq/Kg (= 0.4 GBq/g = 11 mCi/g).

Section "5.0 CODE CLASS BOUNDARY INTERFACES" in SDG-006 defines, in a performance based fashion, two requirements which have to be met for the design of Code boundaries (also called "Code breaks"): a dose limit criterion and a coolant inventory criterion.

2.2 Applicability of the U. S. Code of Federal Regulations

Inservice testing and inspection requirements for nuclear power plants are provided by Federal Law, within the Code of Federal Regulations. The requirements are codified in Title 10 of CFR, Part 50, Paragraph 55a: Codes and Standards (Reference 1); Subparagraph (f): Inservice Testing Requirements; and Subparagraph (g): Inservice Inspection Requirements (Reference 2). Subparagraphs (f) and (g) of 10CFR50.55a specify that Section XI of the ASME Boiler and Pressure Vessel Code is the basis for these testing and inspection activities. The Federal Code allows for a certain amount of flexibility:

10CFR50.55a (f) (5) (iii):

If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit ... information to support the determination.

10CFR50.55a (g) (5) (iii):

If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit ... information to support the determination.

In both of these cases, the Federal Code states that the Commission will evaluate the specific cases and may grant relief. This means that, in principle, no particular power plant design is excluded from being accepted under the requirements of 10CFR50.55a. The CANDU-3 design would fit into the category of "pressurized water cooled nuclear power facility" in the Federal Code; the Federal Code (as opposed to the ASME Code) does not specify "light-water cooled"; it simply specifies "water cooled". Consequently, it appears that the present framework of the Code of Federal Regulations does accommodate the CANDU-3 design without any modification. Subparagraph (f), Inservice Testing, refers to the O&M Code, while Subparagraph (g), Inservice Inspection, refers to Section XI.

2.3 ASME Code Section XI and the ASME/ANSI O&M Standard

Section XI of the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components", is divided into three Divisions:

- Division 1: Rules for Inspection and Testing of Components of Light Water Cooled Plants,
- Division 2: Rules for Inspection and Testing of Components of Gas Cooled Plants, and
- Division 3: Rules for Inspection and Testing of Components of Liquid-Metal Cooled Plants.

Division 1 is the applicable portion of Section XI for this report. Using the Code Edition of July 1992, the following subsections of Division 1 are covered in this review:

Subsection IWA:	General Requirements
Subsection IWB:	Requirements for Class 1 Components of Light Water Cooled Power Plants
Subsection IWC:	Requirements for Class 2 Components of Light Water Cooled Power Plants

- Subsection IWD: Requirements for Class 3 Components of Light Water Cooled Power Plants
- Subsection IWE: Requirements for Class MC ... Components of Light Water Cooled Power Plants
- Subsection IWF: Requirements for Class 1, 2, 3 and Component Supports of Light Water Cooled Power Plants

It is a peculiarity of Section XI of the ASME Code to refer to "Light Water" instead of just "Water". As far as leak detection or inspection is concerned, there is no reason for this distinction. The Code of Federal Regulations does not make such a distinction.

A few explanatory words are in order about the historical background of ISI (Inservice Inspection) versus IST (Inservice Testing). ISI has historically been encoded in Section XI of the ASME Code, written by an ASME Code Committee. The purpose of Section XI is to assure, over the lifetime of the plant, that the initial level of quality (as installed to Section III requirements) is maintained for both the primary pressure boundary and containment. This covers two aspects: leak tightness and structural integrity (avoidance of erosion, fatigue, plastic deformation, etc). By contrast, the purpose of the O&M Code is to demonstrate by testing (not just inspection) the actual performance of the Class 1 and Class 2 safety systems. IST (performance testing) is performed on a short cycle (quarterly or monthly), while ISI (pressure boundary examination) is typically performed on a 10 year cycle. ISI is concerned with possible degradation of passive components caused by aging, while IST is concerned with maintenance and testing of active equipment.

IST was originally handled by the ANSI N45 Committee (The ANSI Committee N45 on Reactor Plants and Their Maintenance). The N45 Committee produced National Standards, not mandatory Codes. The N45 Committee was replaced in 1975 by the O&M Committee (The ASME/ANSI Committee on Operation and Maintenance of Nuclear Power Plants). While the O&M Committee produced the National Standards ASME/ANSI O&M Part 6 (for inservice testing of pumps) and ASME/ANSI O&M Part 10 (for inservice testing of valves), the ASME Code Committee produced Subsections IWP (for inservice testing of pumps) and IWV (for inservice testing of valves) within Division 1 of Section XI of the ASME Code. Some of these overlaps were then eliminated by deleting the text from Subsections IWP and IWV in Division 1 of Section XI and incorporating by reference Standard ASME/ANSI OM Part 6 as Subsection IWP and Standard ASME/ANSI OM Part 10 as Subsection IWV.

One partial overlap presently exists in Article IWF-5000 (Code Edition of July 1992); it covers snubbers, which are also addressed by ASME/ANSI OM Part 04. Rather than transferring all requirements into ASME/ANSI OM Part 04 and incorporating Part 04 by reference, with all text in Article IWF-5000 deleted, some requirements have been left in IWF-5000, which then also references ASME/ANSI OM Part 04.

Legally speaking, the ASME/ANSI O&M Standard is presently located in the third tier: 10CFR50.55a is the directly applicable Federal Law, which incorporates by reference Section XI of the ASME Code, which in turn incorporates by reference the ASME/ANSI OM Standard.

There are two reasons why it is being planned to restructure this area:

1. It is not desirable to have the O&M Code implicitly (rather than explicitly) incorporated in 10CFR50.55.
2. As explained in Section 1.1 of this report, the O&M Code addresses the performance requirements of active systems, while the ISI Code (Section XI) addresses long term aging degradation of passive components and structures. These are two distinctly different areas, and as such they will be kept in two different Codes.

The plan is for future editions of 10CFR50.55a to reference the O&M Code directly, in parallel to the present reference to Section XI. The references to the O&M Code which are now located in Section XI will be removed. Section XI will then be again the ISI Code it was intended to be, rather than a mixture of ISI and O&M issues. For practical purposes, all Parts of the ASME/ANSI O&M Standard are considered mandatory, even if they are not referenced in Section XI; for this reason, the ASME/ANSI O&M Standard is commonly called "the O&M Code" or "the IST Code", although it started out as a "Standard", just like the ASME Pressure Vessel Code. In the general usage of terms, a "code" is a standard that has been made a legal requirement. As of 1990, the following Parts of the ASME/ANSI OM Standard had been issued:

ASME/ANSI OM Part 01	Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices
ASME/ANSI OM Part 02	Requirements for Performance Testing of Nuclear Power Plant Closed Cooling Water Systems
ASME/ANSI OM Part 03	Requirements for Preoperational and Initial Startup Vibration Testing of Nuclear Power Plant Piping Systems
ASME/ANSI OM Part 04	Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints (Snubbers)

ASME/ANSI OM Part 05	Inservice Monitoring of Core Support Barrel Axial Preload in Pressurized Water Reactors
ASME/ANSI OM Part 06	Inservice Testing of Pumps in Light Water Reactor Power Plants
ASME/ANSI OM Part 07	Requirements for Thermal Expansion Testing of Nuclear Power Plant Piping Systems
ASME/ANSI OM Part 08	Startup and Periodic Performance Testing of Electric Motor Operators on Valve Assemblies in Nuclear Power Plants
ASME/ANSI OM Part 10	Inservice Testing of Valves in Light Water Reactor Power Plants
ASME/ANSI OM Part 13	Requirements for Periodic Performance Testing and Monitoring of Power Operated Relief Valve Assemblies
ASME/ANSI OM Part 16	Inservice Testing and maintenance of Diesel Drives in Nuclear Power Plants

Parts 01, 04, 06 and 10 are referenced in Subarticle IWA-1600. A general reference to IST of pumps and valves is made in Paragraph (f) of Subsubarticle IWA-2430. Part 04 is referenced in Article IWF-5000; Part 06 is referenced in Subarticle IWP-1100; Part 10 is referenced in Subarticle IWV-1100 (July 1992 Code Edition).

As stated before, the scope of this report does not include any O&M issues. This short excursion into the O&M area was necessary to clarify the fact that O&M issues are different and separate from ISI issues, and to explain why the O&M portions presently incorporated in Section XI (ISI Code) will be removed.

2.4 The Canadian Nuclear Code Series N285

Canadian Nuclear Code N285.0 (Reference 5), General Requirements for Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants, lays out the general requirements for the design, fabrication, installation and periodic inspection of pressure retaining systems, components, and their supports. N285.0 also includes requirements of a legal nature which are applicable to CANDU power plants located in Canada.

N285.0 is similar in scope to ASME Code Section III, Subsection NCA, General Requirements for Code Section III Division 1 and Division 2. The ASME Code is extensively referenced. Clause 5.2.2 specifies the separation requirement for

a Code Class boundary. Clauses 5.3.1 thru 5.3.3 define Code Classes 1, 2 and 3. The Code Class 1 definition is identical to the corresponding U. S. definition. The Class 2 definition is not entirely clear. Clause 5.3.2 designates "Sections of process systems that penetrate the containment structure ... as Class 2 where required by Clause 5.4.1.1"; but Clause 5.4.1.1 then designates those same items "as Class 4 except as required by Clause 5 of N285.3...". A review of N285.3 was not intended within the scope of this project.

The definition for Class 3 is basically parallel to the definition in RG 1.26, except that the exposure limit (in RG 1.26) has been replaced by the specific activity limit of 0.4 TBq/Kg (0.4 GBq/g). Clause 5.3.4 defines a Class 6, which appears to be similar to Quality Group D (ASME/ANSI B31.1). Clause 5.4.2 allocates Class 1 to the Emergency Cooling Injection Systems; in the U. S. classification philosophy, these are assigned to Class 2, not to Class 1. Clause 5.4.3 allocates Class 1 to the "Shutdown Systems", where it is not clear whether reactivity shutdown systems, shutdown cooling systems, or both are addressed here; for all three possible interpretations, in the U. S. classification philosophy, Class 2 would apply, not Class 1. Clause 5.7 allocates Classes 1C, 2C and 3C to those unique CANDU specialty items which would be residing in Classes 1, 2 and 3 but are not covered in N285.1; those specialty items are covered in N285.2, so that N285.1 is kept parallel to ASME Section III. Clause 7.1.3.1(c) defines the Service Levels A thru D. Clause 11.4.2 invokes ASME Section V as a requirement for examination methods. Clause 13 of N285.0 covers requirements for inservice inspection, component replacement, repair and modification of plant systems. Inservice inspection requirements are addressed by referencing N285.4 and N285.5. It appears that N285.0 is generally equivalent to Subsection NCA, and in some places it is more restrictive.

Canadian Nuclear Code N285.1 (Reference 6), Requirements for Class 1, 2, and 3 Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants, is extensively based on Section III of the ASME Code. In order to keep N285.1 parallel with ASME Section III, all CANDU specialty items (Classes 1C, 2C and 3C) have been left out of N285.1; they are covered in N285.2. Clause 7.1 of N285.1 invokes Articles NB 3000, NC 3000 and ND 3000 as the basis for design of Class 1, Class 2 and Class 3 equipment, respectively. Clause 7.2.1.1 references Articles NB 7000 (Class 1), NC 7000 (Class 2) and ND 7000 (Class 3) for overpressure protection requirements. Clause 7.2.2.2 invokes subsubarticle NCA 2140, Design Basis, for definition of service limits during overpressure events (Service Levels A, B, C and D). Clause 7.2.6.4 requires that testing of overpressure protection devices shall be based on ASME Code Section XI, Subsection IWV, Valves, and further references ASME/ANSI OM-1, Part 1, Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices.

N285.1 Clause 8 requires that all materials used for pressure retaining systems shall comply with Articles NB 2000 (Class 1), NC 2000 (Class 2) and ND

2000 (Class 3) of Section III; it specifies N285.6 for those materials which are not covered in ASME Code Section III (zirconium, etc). Clause 9 establishes Articles NB 4000 (Class 1), NC 4000 (Class 2) and ND 4000 (Class 3) of Section III as the basis for fabrication and installation. Clause 11.1 establishes Articles NB 5000 (Class 1), NC 5000 (Class 2) and ND 5000 (Class 3) of Section III as the basis for examination of systems and components, and Clause 11.2 establishes Articles NB 6000 (Class 1), NC 6000 (Class 2) and ND 6000 (Class 3) of Section III as the basis for testing.

Articles NB 1000, NC 1000 and ND 1000 of Section III, and Articles NB 8000, NC 8000 and ND 8000 of Section III are not referenced in N285.1, because these items are already addressed in Code N285.0. It appears that the requirements of N285.1 are generally equivalent to the requirements of Section III, and a special effort has been made to keep the comparison simple, by relegating specialty items (Classes 1C, 2C and 3C) to N285.2.

Canadian Nuclear Code N285.2 (Reference 7), Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants, addresses CANDU-specific components which cannot be covered by referencing the ASME Code, because these components are different from the BWR or PWR type components, which the ASME Code addresses. CANDU-specific Class 1 components are classified as Class 1C, CANDU-specific Class 2 components are Class 2C, and CANDU-specific Class 3 components are Class 3C. The Foreword to N285.2 states that N285.2 is intended "to maintain the calibre (sic) of the requirements established by the ASME". N285.2 addresses the zirconium fuel channel (Clause 5.1) and the rolled expansion joint between the zirconium fuel channel and the stainless steel endpieces (Clause 5.2). In addition to the pressure tubes, the calandria assembly is addressed, radiography of calandria tubesheet welds, control rod drives, liquid injection nozzles, elastomeric hoses on the fueling machine and fueling machine supports.

Canadian Nuclear Code N285.3 (Reference 8), Requirements for Containment System Components in CANDU Nuclear Power Plants, was not included in the scope of this review project.

Canadian Nuclear Code N285.4 (Reference 9), Periodic Inspection of CANDU Nuclear Plant Components, is analogous to Section XI without Subsections IWE (metal containments) and IWL (concrete containments); containment requirements are addressed in N285.5. While N285.1 incorporates many aspects of Section III, N285.4 shows very few references to Section XI. A detailed discussion of N285.4 is provided in Section 3 of this report. Here the most significant issues are highlighted:

- (1) While the Section XI inspection methodology is based on Code Class, the Canadian Code is structured by Inspection Category instead. Assignment of Inspection Category is not tied to Code Class; it is based on

- Hypothetical Failure Size (Energy Release)
- Fatigue Usage Factor
- Stress Ratio

(2) The Section XI inspection methodology assigns the required inspection method individually for each single type of inspection item, within each Examination Category Table, repeated for each Code Class. The Canadian Code handles inspection methods summarily by groups:

- Group 1: Visual Inspection
- Group 2: Dimensional Inspection
- Group 3: Surface Examination and Volumetric Examination
- Group 4: Integrative Examination

The inspection method is assigned by Inspection Group, within a given Inspection Category.

- (3) While Section XI contains detailed NDE inspector training and requalification requirements, the Canadian Code references an NDE training Code to establish training and requalification requirements.
- (4) While Section XI also covers containment inspection requirements (Subsections IWE and IWL), N285.4 addresses plant equipment only. Containment inspection requirements are covered in N285.5.
- (5) While N285.4 allows for inspection sample size reduction in case of multiple identical plant units, Section XI does not allow for such reductions.

Canadian Nuclear Code N285.5 (Reference 10), Periodic Inspection of CANDU Nuclear Plant Containment Components, covers inspection for structural integrity and operability of the containment system. This Code corresponds to Subsection NE (Class MC Components) of ASME Code Section III for preservice inspections and Subsection IWE (Metal Containments and Metallic Liners of Concrete Containments) of Section XI of the ASME Code for inservice inspections, except for one basic difference:

- N285.5 does not address leak testing
- N285.5 addresses inspection for structural integrity

As stated in Table 1 (Page 27) of N285.5, leak testing is addressed in N287.7 and in AOPs (Approved Operating Procedures). Neither N287.7 nor the AOPs are within the scope of this report. As with N285.4, this code is also sufficiently different from Section XI of the ASME Code so that N285.5 is not directly comparable.

Canadian Nuclear Code N285.6 (Reference 11), Material Standards for Reactor components for CANDU Nuclear Plants, consists of a series of 9 separate Codes (see individual titles under Section 5.0, "REFERENCES", in this report). The purpose of N285.6 is to cover those materials which are not addressed by the ASME Code: zirconium for the reactor pressure tubes, zirconium for the reactivity control rods, zirconium for the gadolinium nitrate injection nozzles, etc. The codes specify impurity limits for the alloys, thermal treatment of the base material, manufacturing methods, hardness tests and other examinations, radiographic standards, inspection requirements and acceptance criteria. The various Codes in N285.6 contain a large number of references to the ASME Code. N285.6.8 specifies the quenching and tempering requirements for the modified Type 403 martensitic stainless steel of which the endpieces of the fuel channel pressure tubes are manufactured. (The steel for the calandria, moderator heat exchangers, pumps and piping is austenitic Type 304L.)

2.5 The Canadian Safety Design Guides SDG-006 and SDG-007

The Canadian Safety Design Guides appear to be in purpose and format similar to the U. S. Regulatory Guides, with some policy material added; Section 3 in SDG-006 and SDG-007 contains material which in U. S. methodology would more likely be placed into a Policy Statement. The material is formatted in terms of guidelines; however, in Section 2 (Compliance Section) of SDG-006 and SDG-007 it is made clear that compliance with these guidelines is mandatory.

SDG-006, Code Classification, provides much of the technical and philosophical basis for N285.0, N285.1 and N285.2. The most distinguishing features can be summarized as follows:

- ASME Section III Code Classes are adopted, but for assignment of Code Class,

- The plant safety function of the system is the main discriminant for assigning a system to Class 1 and Class 4, while
- The health effects from a radioactive release caused by postulated rupture of the system are the main discriminant for assigning a system to Classes 2, 3 and 6.

This highlights the Canadian Code Class assignment philosophy which was discussed in Section 2.1 of this report. This also explains why the reactor shutdown systems and the emergency core cooling systems cannot be in Code Class 2, as they would be under the U. S. philosophy. The primary design functions of the emergency core cooling systems and reactor shutdown systems are reactivity control and heat removal. Therefore, these systems cannot be assigned to Canadian Code Class 2; with the safety function being their primary purpose, they have to be assigned to Class 1. It should be emphasized, though, that the requirements for the various Code Classes are the original ASME Section III requirements; only the assignment of Code Class differs from the American practice.

SDG-007, Periodic Inspection, provides much of the technical and philosophical basis for N285.4 and N285.5. Section 3.0 of SDG-007 specifies the three selection criteria for the specimens making up the inspection samples:

- a) Select all components constituting a pressure boundary which is important for safety, both preventive and mitigative of accidents.
- b) Select all components whose failure would significantly impair the function of a Special Safety System (Engineered Safety Feature).
- c) From the population of components collected under a) and b), derive a biased stratified sample with specimens containing the severest service conditions in terms of stress, fatigue, erosion and corrosion. Then add any specimens which in previous inspections have shown marginal indications.

Section 3.0 of SDG-007 also specifies a maximum of 6 years for the first inspection interval, as compared with a 4 year interval given in Clause 7.6.1 of N285.4. Section 4.7 addresses minimizing the radiation exposure for inspection personnel, and Section 5.0 discusses Supplementary Requirements for the reactor pressure tubes, for the fuel channel feeder piping, and for the steam generator tubes.

Appendices A and B of SDG-007 provide details on the inspection sample selection methodology. Appendix C deals with accessibility and inspectability of the inspection sites themselves. Sections C-3.0, C-4.0 and C-5.0 contain a

number of diagrams similar to the Figures in IWB-2500, IWC-2500, and IWD-2500. Appendix D provides system design recommendations for optimum inspectability. Appendix E provides guidelines for selecting inspection specimens for erosion and corrosion considerations.

2.6 The Canadian Regulatory Policy Statement R-7

The AECB Policy Statement R-7, Requirements for Containment Systems for CANDU Nuclear Power Plants, presents a concise summary of containment functional requirements. It covers all active and passive aspects, from leakage criteria to electrical separation requirements to seismic design requirements. The main characteristics of this document can be summarized as follows:

- AECB R-7 is not a "Policy Statement" in the sense that the NRC uses this term; AECB R-7 is functionally like an SRP Section with Regulatory Guide material added for detail, except that
- AECB R-7 is a specimen of performance based regulation; requirements are expressed in terms of criteria which have to be met rather than in terms of required implementation details.

Only a short Section of R-7 is devoted to inservice inspection requirements. Section 5 of this report provides an evaluation of R-7 with respect to ISI aspects.

3.0 STRUCTURAL COMPARISON OF SECTION XI VERSUS CANADIAN ISI CODES

The correlation between the Codes is as follows:

<u>System or Component</u>	<u>ASME Code Section XI</u>	<u>Canadian Code</u>
Containment, Structural	Subsection IWE	Code N285.5
Containment, Leak Test	Subsection IWE with Appendix J	Code N287.7
All other Components	Other Subsections	Code N285.4

3.1 Comparison of Section XI versus Canadian Code N285.4

Part of this report section has been written in tabular form. For easy item by item scanning, see Table 3.1. The first four items in Table 3.1 are of a general nature, and the remaining items are arranged according to the structure of Section XI of the ASME Pressure Vessel Code.

The initial intent was to present the entire Section 3 of this report in tabular form. However, some aspects of this Code comparison cannot easily be analyzed in tabular form because of the significant differences in the basic structure of methodology between the two Codes. For example:

- In the ASME Code, the importance hierarchy for ISI purposes is directly based on the Code Class assignments (Classes 1, 2, 3); the Canadian Code handles the importance hierarchy (in N285.4) by assigning Inspection Categories (Categories A, B, C1, C2), with no reference to Code Classes.
- In the ASME Code, the Code Classes are generally assigned in an administrative way: the primary pressure boundary is Class 1, engineered safety feature systems are Class 2, etc. The Canadian Codes, rather than basing assignment of Inspection Category on a similarly straightforward administrative rule, are using an individual system analysis which considers:
 1. Maximum possible energy release in case of postulated, hypothetical catastrophic rupture.
 2. Fatigue Usage Factor, as defined in the ASME Code. This factor accounts for cyclic loading of the system.
 3. Stress Ratio; this is the reciprocal of the safety factor; it measures how much of the static load carrying capability of the component is being used.
- In the ASME Code, the inspection method is prescribed in the Examination Category Tables (Tables IWB-2500-1, IWC-2500-1, etc), for each individual type of inspection item. The Canadian Code handles the inspection methods in terms of groups, assigned to Inspection Categories:
 - Group 1: Visual Inspection
 - Group 2: Dimensional Inspection

- Group 3: Surface Examination and Volumetric Examination
- Group 4: Integrative Examination; this is the most extensive examination. This examination can be used to confirm the design adequacy of a component.

Tabulating these items without additional explanation would not provide a clear picture, because some of the procedural details which define the underlying process would be lost. In the following paragraphs an attempt is made to present a short but complete sketch of the basic features of each methodology.

In the ASME Code, a basic process pattern is applied to each Code Class separately: Class 1, Class 2, Class 3, Class MC. The basic tenet is that Code Classes translate into levels of importance, with Class 1 and Class MC being the most important, followed by Class 2 and Class 3. In each Class, Examination Categories are defined: nineteen in Class 1, seven in Class MC, eight in Class 2 and two in Class 3. These Examination Categories contain the various sets of Items (welds, flanges, bolts, etc) to be examined.

For each Examination Category a table enumerates the Items (grouped in Item Groups) and indicates for each Item the Examination Requirement, Examination Method, Acceptance Standard, etc. To illustrate the process, we take Subsection IWB-2500, Examination and Pressure Test Requirements. Paragraph (a) indicates that Table IWB-2500-1 is the applicable Item table; paragraph (b) lists the 19 Examination Categories. We select Category "B-J Pressure Retaining Welds in Piping" and go to the appropriate section of Table IWB-2500-1, which shows 4 groups of Items: B9.10 for nominal pipe diameter of 4 inch or larger, B9.20 for nominal pipe diameter of less than 4 inch, B9.30 for Branch Pipe Connection Welds and B9.40 for socket welds. Since this is Subsection IWB, we are dealing with primary loop pressure boundary piping welds (Class 1 piping).

From the B9.10 Item group, we select subgroup "B9.11 Circumferential Welds". Note (1)(d) indicates that the inspection sample has to cover 25 percent of all Items of this type: circumferential butt welds in the reactor coolant piping system (the same sample size applies to subgroup B9.21 which covers circumferential welds in primary piping of less than 4 inch nominal diameter). This 25 percent sample is not a random sample. Rather, the welds with the highest stress intensity and/or highest fatigue use factor are selected; technically this is called a biased cluster sample. Table IWB-2500-1 indicates that a surface examination and a volumetric examination is required, and Note (3) indicates that essentially 100 percent of the weld length has to be examined.

By contrast, the corresponding Class 2 Item sample is selected such that the entire range of pipe sizes and different stress conditions are covered;

technically this is called a balanced stratified sample. This Item is shown in Table IWC-2500-1, Examination Category "C-F-2 Pressure Retaining Welds in Carbon Steel or Low Alloy Steel Piping", Item Group C5.50 for welds in piping of more than 0.375 inch nominal wall thickness and more than 4 inch nominal pipe diameter, subgroup "C5.51 Circumferential Welds". Note (2) indicates that the inspection sample has to cover 7.5 percent of all Items of this type, but no less than 28 welds. Table IWC-2500-1 indicates that a surface examination and a volumetric examination is required, and that 100 percent of each weld length has to be examined (the same as in Subsection IWB, for Class 1 welds).

The Canadian methodology in N285.4 does not utilize the Code Classes for inservice inspection. It starts out by looking at the entire system population, all Code Classes included, and it assigns a system (or portion of a system) to one of 4 Inspection Categories: A, B, C1, C2. These Inspection Categories then translate into levels of importance: The most intensive inspection is applied to items in Category A, the least intensive in Category C2. Systems are assigned to these Inspection Categories without regard to the Code Classes which they are in. When comparing Section XI to the Canadian Codes, one has to keep in mind that the U. S. Examination Category is a set of physical items to be inspected, while the Canadian Inspection Category is a set of inspection requirements to be applied to the items within a given plant system area.

Assignment of the Inspection Category depends on three parameters:

- Failure Size (small, medium, large)
- Fatigue Usage Factor (low, medium, high)
- Stress Ratio (low, medium, high)

Each one of these parameters can have only 3 discrete values. Failure Size is either Small, Medium or Large. It is determined (by Clause 7.1.2 of N285.4) from the maximum possible energy release for a given component when postulating a hypothetical catastrophic rupture of the component. If the Failure Size is ranked as Small, the component is automatically assigned to Inspection Category C2. If the Failure Size is Medium or Large, the two other parameters (Fatigue Usage Factor and Stress Ratio) have to be evaluated.

Calculation of the Fatigue Usage Factor (by Clause 7.1.3 of N285.4) is based on Section III of the ASME Code. Fatigue Usage values between 0 and 0.01 are ranked Low, values between 0.01 and 0.1 are ranked Medium, and values between 0.1 and 1.0 are ranked High. The Stress Ratio (calculated by Clause 7.1.4 of N285.4) is basically the ratio of calculated stress (for Service Level A and B) divided by allowable stress; this ratio is the reciprocal of the safety

factor. The stress calculation for the Stress Ratio can be done by using Section III of the ASME Code or by using more refined methods (finite element models). Stress Ratio values between 0 and 0.33 are ranked Low, values between 0.33 and 0.67 are ranked Medium, and values between 0.67 and 1.0 are ranked High.

With the values (Low, Medium, High) for Fatigue Usage Factor and for Stress Ratio known, Table 1(a) in N285.4 gives the Inspection Category for the case of Medium Failure Size, and Table 1(b) in N285.4 gives the Inspection Category for Large Failure Size. Clause 7.4.3 of N285.4 gives the requirements for Inspection Category A. In each pipe run, the joint with the highest Fatigue Usage Factor and/or highest Stress Ratio is selected for examination. On the surface, just one joint per run sounds like a small sample, but the number of pipe runs can be fairly high. For pressure vessels, the minimum number of welds to be inspected is $N_{\text{MINIMUM}} = N_1 * F_A$, where N_1 is the number of identical type welds on a representative pressure vessel, and F_A is a factor determined from a curve in Figure 2 of N285.4. This is similar to Section XI; for example, if a plant has two identical Residual Heat Removal heat exchangers, only one is inspected. On mechanical couplings, all basic components of the coupling are examined. On pumps and valves, all pressure retaining welds and internal surfaces are inspected.

Clause 7.4.4 of N285.4 gives the requirements for Inspection Category B. In Category B, a pipe joint is selected the same way as in Category A, except that a pipe run needs no Category B inspection if it already had a Category A joint being inspected. For welds on pressure vessels, the minimum number of identical type welds to be inspected is the lesser of either one third of all Category B welds or a number determined with a factor F_B , analogous to the factor F_A mentioned above, taken from the same Figure off a different curve. For mechanical couplings, only 10 percent of bolts and flange ligaments are inspected (instead of 100 percent). For pumps and valves, the sample size goes down to 33 percent of all Category B welds.

Erosion and Corrosion inspection is handled as a separate, additional activity (Clause 7.4.7) for Inspection Categories A, B and C1. Another separate, additional activity is a supplemental inspection of the reactor pressure tubes (minimum of 4 tubes, Clause 12.3.2), the pressure tube feeder pipes (minimum of 10 feeder pipes, Clause 13.3.2), and the steam generator tubes (minimum of 1 percent of all tubes, Clause 14.3). As mentioned before, the multiunit sample reduction factors, to be used if all indications are normal on the oldest specimen of a given set of identical units, cannot be used for the U. S. industry because of lack of standardization in plant design and maintenance procedures.

The various examination methods to be applied are specified in Section XI in Table IWB-2500-1 (for Class 1), Table IWE-2500-1 (for Class MC), Table IWC-2500-1 (for Class 2) and Table IWD-2500-1 (for Class 3). Table 1 in N285.4 and Table 1 in N285.5 (one page each, as compared to some 30 pages in Section

XI) accomplish basically the same purpose. Inspection methods are grouped into four Groups (Clause 4.1 of N285.4): Group 1 = Visual (corresponding to the VT type methods in Section XI), Group 2 = Dimensional, Group 3 = Surface and Volumetric (includes liquid penetrant, magnetic particle, ultrasonic, radiographic, eddy current, acoustic holography), Group 4 = Integrative. The Integrative group is very thorough. It basically amounts to verification of design adequacy, including design analysis. Qualification of NDE personnel is defined by simply referencing the 48-GP Series of the CGSB (Canadian General Standards Board) Standards.

Within the scope of this review project it is not possible to make a determination as to which one of the Codes is the more conservative one: Section XI or the N285 series. With the amount of detail covered so far, it appears that the Canadian approach might be more conservative than the ASME approach. If this question is to be pursued further, it might require two or three specific case studies to arrive at a decision.

3.2 Comparison of Section XI, Subsection IWE, versus Canadian Code N285.5

As stated before, Subsection IWE (together with Appendix J) covers both structural integrity and leak testing of the containment, while N285.5 covers structural integrity only. N287.7, which covers leak testing, is not within the scope of this report.

In Subsection IWE, as in the previous Subsections (IWB, IWC, IWD), Inspection Program B is favored over Inspection Program A (see Table IWE-2412-1). Inspection Program B is based on a simple 10 year inspection cycle, with:

- Completion level at 16 percent to 34 percent by the end of 3 years,
- Completion level at 50 percent to 67 percent by the end of 7 years, and
- Completion level at 100 percent at the end of 10 years.

The Canadian Code N285.5 specifies in Clauses 8.6.1 and 8.6.2:

- First inspection within 5 years of first criticality, and
- Subsequent periodic inspections within the shorter of these periods:
 - (a) one third of the design life of the plant, or

(b) 10 years.

Here the Canadian Code is more conservative for the first inspection period; the subsequent periods are 10 years, assuming that the design life of the facility is always greater than 30 years.

References to Inspection Categories or Code Classes are not needed here, because all containment components are in the same category of importance (all are Canadian Code Class 4, although Code Class is not relevant to Inspection Category). Additionally, instead of four Groups of inspection methods we have simply four inspection methods stated in Clause 5.2:

1. Visual Inspection
2. Surface Inspection
3. Volumetric Inspection
4. Thickness Measurement

Clause 4.3 defines the components subject to inspection:

- (a) containment boundary components with penetration areas greater than the cross section of a 1 inch pipe (nominal diameter),
- (b) components required for pressure suppression within the containment,
- (c) fluid boundaries of systems which form an extension of the containment envelope if the cross section is larger than that of a nominal 1 inch pipe and if the system is shown in Figure 1 of N285.5 as being subject to inspection. Figure 1 shows four different cases of piping extending out from the containment; it basically indicates that any pipe extending out of the containment has to be inspected up to the outboard isolation valve, unless the pipe is extending from a system inside containment which is closed in itself (no connection to containment atmosphere).

Note: The piping sketches in Figure 1 of N285.5 show outboard isolation valves (called "credited isolation barriers") but no inboard isolation valves. This by itself does not mean that inboard isolation valves are not required; but from the information in Figure 1 it is not obvious whether they are required or not.

Clause 4.3 defines components to be exempt from inspection if:

1. the component (or a sample of these components) is being inspected already under N285.4, or
2. it has been demonstrated to and accepted by the regulatory authority that the limits of radioactive release in case of accidents are still met without that component (or system).

Figure 2 in N285.5 shows a flow chart which performs the assignment of inspection methods for a given component. Figure 3 in N285.5 gives the required sample size for populations of identical components or welds to be inspected. If the population has 10 specimens, the chart indicates a required sample size of 2.1, which is rounded to 3 specimens. For population sizes of up to 65 specimens, the chart gives 3 specimens for the minimum sample size.

The following Table 3.1 gives an item by item comparison for both N285.4 and N285.5.

Table 3.1 Comparison between Canadian Codes and ASME Code Section XI

ASME Code Item	Canadian Code Item	Comment
Inservice testing items (IWF-5200, IWP-1100, IWV-1100) of snubbers, pumps, valves are presently inside Section XI; these are maintenance issues which are not part of ISI. Section XI is intended to be ISI Code, not IST Code.	The Canadian Code (N285.4 for plant systems, N285.5 for containment) is strictly an ISI (inservice inspection) code, it does not contain O&M items.	The ASME Code is presently in a state of transition. System performance testing items will be removed from Section XI and will be located only in the O&M Code. The O&M Code will then be referenced in 10CFR50.55a, just as Section XI is now.
With respect to primary pressure boundary leakage, the calandria would be ASME Class 2 by U. S. standards; only certain structural aspects (tube sheets) would be in Class 1	By the Canadian Code the entire calandria is assigned to ASME Class 1 (SDG-006, N285.0)	The Canadian Code uses Class 1 classification on all systems which contain heavy water. See also Section 2.5 of this report for Code Class assignment philosophy.
The ECCS systems would be ASME Class 2 by U. S. standards.	By the Canadian Code the ECCS systems are in ASME Class 1	The Canadian Code appears to use the Class 1 classification on all systems which contain heavy water. See also Section 2.5 of this report for Code Class assignment philosophy.
The reactivity shutdown systems would be ASME Class 2 by U. S. standards	By the Canadian Code both reactivity shutdown systems are in ASME Class 1	Section 2.5 of this report provides the rationale for the underlying Code Class assignment philosophy.

Table 3.1 Comparison between Canadian Codes and ASME Code Section XI

ASME Code Item	Canadian Code Item	Comment
<p>Systems subject to inspection</p> <p>Subsubarticle IWA-1320 (a):</p> <p>Rules of IWB for Class 1</p> <p>Rules of IWC for Class 2</p> <p>Rules of IWD for Class 3</p>	<p>N285.4, Clause 3.3.1</p> <p>(a) covers Class 1 systems;</p> <p>(b) covers Class 2 and Class 3 systems;</p> <p>(c) covers rotating components which might generate a missile</p>	<p>The Canadian Code does not have a separate subsection (IWB, IWC, etc) for each Class of equipment. The Canadian and U. S. Codes appear to be equivalent, except that N285.4 includes rotating components.</p>
<p>Qualifications of Examination Personnel: IWA-2310, 2311, 2312, 2313, and 2314</p> <p>Qualification Examinations: IWA-2321, 2322, 2323</p> <p>Level I Responsibilities: IWA-2330</p> <p>Level III Education: IWA-2340</p> <p>Limited Certification: IWA-2350</p> <p>Level I and Level II Training and Experience: IWA-2360</p> <p>Level III Experience: IWA-370</p> <p>NDE Instructor: IWA-2380</p>	<p>N285.4, Clause 5.2.1</p> <p>Requirements:</p> <p>(a) certified with CGSB 48-GP Series of Standards for the NDE technique to be employed, or</p> <p>(b) considered by the regulating authority to have sufficient knowledge ... to perform in a competent manner certain designated inspections for which CGSB Standards are not issued.</p> <p>Note: CGSB = Canadian General Standards Board</p>	<p>The format of the Canadian Code is functionally simpler: training requirements are dealt with in a training and requalification Code, while inservice inspection is dealt with in the inservice inspection Code.</p> <p>Note: Review of the Canadian CGSB 48-GP Series of Standards for NDE personnel qualification requirements was not part of this project.</p>
<p>Article IWA-9000, Glossary</p>	<p>N285.4, Clause 2.0, Definitions</p>	<p>The Canadian Code presents definitions at the beginning, while Section XI locates them at the end of the first Subsection.</p>

Table 3.1 Comparison between Canadian Codes and ASME Code Section XI

ASME Code Item	Canadian Code Item	Comment
Subarticle IWB-2200 <u>Preservice Examination</u> basically requires examination of all Class 1 pressure boundary welds for new components (either initially installed or replaced)	Clause 6. <u>Inaugural Inspection</u> of N285.4 formulates basically the same requirement	The Canadian Code appears to be equivalent to the ASME Code
Subarticle IWB-2400 <u>Inspection Schedule</u> defines Inspection Program A (accelerated program) and Inspection Program B (normal program). Program B specifies 10 years for the first inspection interval and 10 years for successive intervals.	Clause 7.6.1 of N285.4 specifies 4 years for <u>first</u> inspection interval and a maximum of 10 years for successive intervals, for operating systems. [Sec 3.0 of SDG-007 gives 6 yr interval for first inspection] Clause 7.6.2 specifies <u>one half</u> of the above values for systems which are dormant at Service Levels A and B, but have to operate at Service Levels C and D	According to U. S. industry sources, Inspection Program B is used, nobody uses A; the ASME Code does not comment on this. The Canadian requirement appears to be more conservative for operating systems, and considerably more conservative for standby (dormant mitigative) systems.
Table IWB-2412-1 <u>Inspection Program B</u> specifies, in terms of percent of all inspections required, how the work is to be spread over the inspection interval.	Clause 7.6.1.1 of N285.4 specifies that the "inspection should be distributed evenly, as far as it is practical, over the specified period".	The Canadian Code does not specify percentage numbers as a function of time, but the intent of both Codes is obviously the same.

ASME Code Item	Canadian Code Item	Comment
<p><u>Subsubarticle IWB-2430 Additional Examinations</u> (a) requires that the sample size be doubled if flaws are found; (b) specifies that in case more flaws are found with the doubled sample, the inspection has to be expanded to cover the entire population of the type of specimen being inspected.</p> <p>Additional inspections under (a) shall be done during the current inspection period, while additional inspections under (b) shall be done during the current outage.</p>	<p><u>Clause 8.3 Additional Inspection</u> of N285.4 also requires doubling of the sample in case of flaw indications; the Clause also specifies the additional inspections have to be performed within 20 percent of the original target time.</p>	<p>The Canadian Code appears to be equivalent to the ASME Code; although the Canadian Code only specifies doubling of the sample in case of flaws, successive doubling will quickly become equivalent to covering the total population of a given type of specimen. The time constraint is somewhat less tight on the Canadian Code.</p>
<p><u>Subsection IWE, Requirements for Class MC Components and for Metallic Liners of Class CC Components...</u></p>	<p>Code N285.5, Code N287.7</p>	<p>Subsection IWE incorporates Appendix J to 10CFR50 by reference and covers containment leak testing, see Table IWE-2500-1, Examination Category E-P, Pressure Retaining Components. In the Canadian Codes, N285.5 covers structural integrity <u>only</u>, and leak testing is addressed in N287.7.</p>
<p>Program B gives inspection period of 10 years (Table IWE-2412-1)</p>	<p>First inspection period is 5 years (N285.5, Clause 8.6.1), subsequent inspection periods are 10 years (Clause 8.6.2)</p>	<p>Canadian inspection period is shorter for <u>first</u> inspection and is equal to the ASME Program B requirement for subsequent inspections</p>
<p>Program B specifies inspection rate during the inspection period, see Table IWE-2412-1</p>	<p>N285.5 (Clauses 8.6.1 and 8.6.2) only specifies 100 percent completion within the period</p>	<p>Canadian Code allows for more leeway in planning the inspection activities over the required inspection period</p>

4.0 EVALUATION OF SECTION XI VERSUS CANDU-3 DESIGN FEATURES

The following is a short summary of CANDU-3 design features which appear unique in view of the ASME Code (Reference 3).

1. The fuel channel pressure tube is connected to its stainless steel end fittings by means of rolled expansion joints.
2. The fuel channel pressure tube is made of a zirconium niobium alloy (2.5 percent niobium).
3. Both the Group 1 shutdown absorber rods and the reactivity control rods reside outside of the primary pressure boundary. These items are classified as Class 1 components; however, by commonly accepted U. S. application of the ASME Code, they should be Class 2 components; the moderator liquid in the calandria is not pressurized.
4. The Group 2 reactivity shutdown (gadolinium nitrate injection) system is classified as Class 1. This system never penetrates the primary pressure boundary; therefore, by commonly accepted U. S. application of the ASME Code, this system should be a Class 2 system.
5. The calandria tubes (made of zirconium alloy) are attached to the steel tubesheets with rolled joints. The calandria is classified as a Class 1 structure; however, the calandria is not part of the primary pressure boundary and therefore, by commonly accepted U. S. application of the ASME Code, it should be a Class 2 component. On the other hand, the tube sheets of the calandria (2 tube sheets at each end, forming the end shield) serve as reactor support structure, which is a function properly classified as Class 1. Consequently, some structural portions of the calandria should be Class 1, while all aspects of leak tightness should properly be classified as Code Class 2.
6. The weld connecting the lattice tube to the outboard calandria tube sheet is not 100 percent radiographed; also, this weld is not a full penetration weld. However, this weld is not part of the primary pressure boundary and therefore is not a Class 1 component. The main function of the stainless steel lattice tubes is to serve as a spacer between the inboard and outboard tubesheets.
7. The CANDU fuel handling equipment uses reinforced elastomer hoses. This equipment is in Canadian Code Class 1. It would be classified as either Code Class 2 or Code Class 3 under U. S. standards, except for those portions of the equipment which form the primary pressure boundary during at-power refueling.

These features are the most obvious ones, taken from the currently available design information. Based on the limitation of the currently available material, this summary might not be complete. The following Table 4.1 addresses these unique features and relates them to Subsections in Section XI of the Code which might require modification in order to cover those unique features. The purpose of Table 4.1 is to answer the question: What are the areas in the CANDU-3 Design which the ASME Code Section XI might not cover or might be incompatible with?

Unique CANDU-3 Design Feature	ASME Code Section XI Impact	Comment on Compatibility of ASME Section XI With CANDU-3 Design Feature
<p>Fuel channel pressure tube connected to end pieces by roll-expanded tube joints. The pressure tube is a Class 1 component.</p>	IWA-2500, "Extent of Examination"	Code Section XI, Subarticle IWA-2500 addresses weld joints. This Subarticle would have to be expanded to include rolled tube joints.
	IWA-2600, "Weld Reference System"	Code Section XI, Subarticle IWA-2600 addresses a weld reference system. This Subarticle would have to be rephrased to also invoke a rolled tube joint reference system.
	IWA-3300, "Flaw Characterization"	Code Section XI, Subarticle IWA-3300 characterizes flaw images for welded components. When examining a rolled tube joint, flaws might generate images sufficiently different from weld flaws to require separate flaw characterization.
	IWA-4000, "Repair and Replacement"	Code Section XI, Article IWA-4000 addresses repair and replacement in terms of welding. The article would have to be expanded to also address repair and replacement for rolled tube joints.
	IWA-4621, "Class 1 Piping Systems"	Paragraph IWA-4621 of Subsubsection IWA-4620, "Mechanical Joints", states in Subparagraph (b): "Expanded joints shall not be used in Class 1 piping systems". The subparagraph would have to be rephrased to allow for the use of rolled joints on reactor pressure tubes.
	IWB-2200, "Preservice Examination"	Subparagraph (a) of Subarticle IWA-2200 addresses preservice examination of welds in Class 1 systems. This subparagraph would have to be expanded to also address preservice examination of rolled tube joints in Class 1 systems.

Unique CANDU-3 Design Feature	ASME Code Section XI Impact	Comment on Compatibility of ASME Section XI With CANDU-3 Design Feature
Fuel channel pressure tube is connected to end pieces by roll-expanded tube joints. ** Continued **	IWB-2500, "Examination and Pressure Test Requirements"	Subparagraph (b) of Subarticle IWB-2500 organizes examination areas into coded "Examination Category" items. Examination Categories are then listed in Table IWB-2500-1, where "Parts Examined" are further encoded by a 4 or 5 digit alpha numeric "Item Number", which is correlated with Examination Method, Acceptance Standard, etc. This Subparagraph, including the Table, would have to be expanded to include the reactor pressure tubes. The set of illustrative Figures (Fig. IWB 2500-1 thru Fig. IWB 2500-20) might also have to be expanded to show details of the reactor pressure tubes. In addition to Examination Category B-A "Pressure Retaining Welds in Reactor Vessel", the additional Category "Pressure Retaining Expansion Joints on Reactor Pressure Tubes" would have to be added.
	IWB-3500, "Acceptance Standards"	This Subarticle contains various subsubarticles, paragraphs and subparagraphs which specify acceptable flaws within examination areas. The Subarticle would have to be expanded to include limits for acceptable flaws in reactor pressure tubes made of zirconium-niobium alloy, as well as flaws in the tube expansion joint between the pressure tube and its end pieces.

Unique CANDU-3 Design Feature	ASME Code Section XI Impact	Comment on Compatibility of ASME Section XI With CANDU-3 Design Feature
<p>The fuel channel pressure tube is made of zirconium-niobium alloy, not of steel. The fuel channel pressure tube is a Class 1 component.</p>	<p>IWA-4000, "Repair and Replacement"</p> <p>IWB-3500, "Acceptance Standards"</p> <p>IWB-3600, "Analytical Evaluation of Flaws"</p> <p>IWB-3700, "Analytical Evaluation of Plant Operating Events"</p> <p>IWB-5000, "System Pressure Tests"</p>	<p>Code Section XI, Article IWA-4000 addresses repair and replacement of steel components. The article would have to be expanded to also address repair and replacement for zirconium components.</p> <p>This Subarticle contains various subsubarticles, paragraphs and subparagraphs which specify acceptable flaw sizes within the examination areas for specific materials, mostly for ferritic steels. This Subarticle would have to be expanded to include limits for acceptable flaws in zirconium-niobium pressure tubes. The acceptance criteria in Canadian Code N285.6.6 (Ref. 11) might be used for formulating this addition.</p> <p>This Subarticle would have to be examined for possible changes to accommodate analytical flaw evaluation for zirconium-niobium material.</p> <p>This Subarticle would have to be examined for possible changes to accommodate effects of operating events on zirconium components.</p> <p>The test pressure versus test temperature given in Table IWB-5230-1 might have to be examined for suitability in testing of zirconium components as opposed to testing of steel components</p>
<p>Group 1 shutdown absorber rods and reactivity control rods reside outside of the primary pressure boundary. Therefore, these items are Class 2 components. But control rod welds on U. S. reactors are Class 1 components.</p>	<p>IWA, IWB, IWC</p>	<p>These systems could either be reclassified as Class 2 systems and then inspected by Subsection IWC, or they could be left in their Class 1 status and be inspected by Subsection IWB.</p>

Table 4.1 Compatibility of CANDU-3 Design With Section XI of ASME Code Requirements

Unique CANDU-3 Design Feature	ASME Code Section XI Impact	Comment on Compatibility of ASME Section XI With CANDU-3 Design Feature
Group 2 reactivity shutdown system is a Code Class 1 system.	IWA, IWB, IWC	This system could either be reclassified as a Class 2 system and then inspected by Subsection IWC, or it could be left in its Class 1 status and be inspected by Subsection IWB.
The calandria tubes, made of zirconium, are attached to the steel tubesheets with rolled joints.	Same as with rolled joints on fuel pressure tubes, see above.	Same as with rolled joints on fuel pressure tubes, see above.
Weld on lattice tube to out-board tube sheet is not full penetration weld.	Possibly IWC	Weld might be reclassified as Class 2 weld, since lattice tubes only serve as spacers and are not part of primary pressure boundary. Lattice tubes are horizontal, do not carry the reactor weight. Strength of lattice tubes might come into play in seismic scenarios.
Fuel handling equipment, fueling machine supports, elastomer hoses, except for those parts which manipulate the primary pressure boundary during refueling.	Possibly IWC, IWD	This equipment would be Class 2 or Class 3 equipment under the U. S. Code assignment philosophy, except for those portions of the refueling machine which affect the primary pressure boundary during refueling.
Those parts of the refueling equipment which form the primary pressure boundary during at-power refueling.	IWB	These parts of the equipment would be in Code Class 1 under the U. S. Code assignment philosophy.

5.0 EVALUATION OF SECTION XI VERSUS CANADIAN POLICY STATEMENT R-7

AECB Policy Statement R-7, Requirements for Containment Systems for CANDU Nuclear Power Plants, provides, in a generic way, a concise summary of

- Design Requirements
- Availability Requirements (probabilistic)
- Functional Requirements
- Testing Requirements (preservice and inservice)
- Inspection Requirements (preservice and inservice)

for containments and metal extensions of the containment envelope. It covers all active and passive aspects, from leakage criteria to electrical separation requirements to seismic design requirements. It quotes an upper (probabilistic) limit for containment unavailability. AECB R-7 is not a Policy Statement in the NRC usage of this term.

In the NRC usage of the term, a Policy Statement is a position paper. It is focused on a specific issue; for example:

- In what situations may probabilistic arguments be utilized for licensing decisions?

or:

- In In LOCA analysis, where may the Leak Before Break concept be applied?

The NRC Policy Statement analyzes and dissects the issue, then provides the official interpretation which defines the regulatory position. It then presents the technical basis for that position. The NRC Policy Statement sets a guide post; it defines the official direction where professional opinions otherwise might diverge, on a specific technical question.

The AECB Policy Statement is a "Make Sure You do Everything Right" document. It might be compared to a NUMARC Policy Statement. It does not present any decision or interpretation, nor does it set any rules, as the Codes do. It simply calls attention to all the existing rules and requirements which apply to the topic (containment systems in this case). It basically provides a checklist of all the requirements that need to be met. Since the AECB Policy Statement is intended to be an instrument of performance based regulation

(just like a NUMARC Policy Statement), it carries no quantitative information. Three short excerpts might illustrate this:

3.3 Dose Limits under Accident Conditions

The containment system shall be capable of limiting the release of radioactive material such that the reference dose limits are not exceeded.

No quantitative limit is stated, only a qualitative criterion is given. Another example:

3.5 Leakage Criteria

3.5.1 *The maximum allowable leakage rate from the containment envelope shall be the value used in the safety analyses which demonstrate that the reference dose limits are not exceeded.*

Again, only a qualitative criterion is given. A third example:

3.14 Seismic Requirements

All parts of the containment system credited in the safety analysis following a design basis ground motion for that plant site shall be designed to remain fully functional following such an event.

This criterion does not state the CANDU-3 design value for maximum horizontal ground acceleration, which is 0.3 g. Only a generic criterion is given. The seismic design basis ground motion is what in NRC terminology is called SSE (safe shutdown earthquake, usually at twice the maximum acceleration value of the OBE). Out of a total of 5 Sections, only Subsection 5.2 is devoted to in-service inspection.

In terms of types of documents which the NRC uses, the AECB Policy Statement is like an SRP Section: a checklist to assure technical completeness. The Standard Review Plan is generally qualitative; but in places it does state quantitative information, which the Canadian Policy Statement generally does not do.

There is an exception to being consistently qualitative in the body of AECB R-7. In Clause 3.7.1, the maximum allowable annual unavailability for the containment function is quantitatively stated at a cumulative value of $1.0E-3$:

3.7 Availability Requirements

3.7.1 *The containment system shall be designed such that the fraction of time for which it is not available can be demonstrated to be less than $1.0E-3$ years per year. The system shall be considered available only if it can be demonstrated to meet all of the minimum allowable performance standards as defined in accordance with subsection 3.2.2*

In the Appendix to AECB R-7 there is also a minor amount of quantitative material. The following Table 5.1 provides a short item by item discussion, with emphasis on those items which relate to inservice inspection.

Table 5.1 Evaluation of Canadian Policy Statement AECB R-7

AECB R-7 Item	Corresponding ASME Code Item	Comment
<p>Section 1.0 Definitions</p> <p>Terms like "closed system", "fuel cooling system", "special safety system" are defined</p>	<p>This item is not specific to in-service inspection (Section XI). Section XI of the ASME Code uses "Glossary" (see Article IWA-9000) instead of "Definitions".</p>	<p>The purpose of this Section is partly to establish a reference between the English and the corresponding French terms.</p>
<p>Section 2.0 Basic Requirements</p> <p>This section enumerates all systems and components which have to be located within the containment envelope. It designates the containment to be a Special Safety System.</p>	<p>This item is not specific to in-service inspection (Section XI). Containment design requirements can be found in ASME Code Section III, in 10CFR50 GDCs (General Design Criteria), and in various Reg Guides.</p>	<p>"Special Safety System" is the equivalent of "Engineered Safety Feature"</p>
<p>Section 3.0 Design Requirements</p> <p>3.1 Containment Envelope 3.2 Design Information 3.3 Dose Limits under Accident Conditions 3.4 Structural Integrity 3.5 Leakage Criteria 3.6 Environmental Requirements 3.7 Availability Requirements 3.8 Separation and Independence Requirements 3.9 Requirements for Containment Penetrations 3.10 Containment Atmosphere Control 3.11 Shielding Requirements 3.12 Status Monitoring Requirements 3.13 Codes and Standards 3.14 Seismic Requirements</p>	<p>Items are not specific to ISI</p> <p>GDCs, Reg Guides, ASME Sec III GDCs, Reg Guides GDCs, Reg Guides</p> <p>GDCs; Reg Guides, ASME Sec III</p> <p>GDCs, Reg Guides, 10CFR50 Apndx J GDCs, Reg Guides No counterpart in U. S. Codes GDCs, IEEE Standards</p> <p>GDCs, Reg Guides, IEEE Standards</p> <p>GDCs, 10CFR50, Reg Guides</p> <p>GDCs, Reg Guides GDCs, Reg Guides</p> <p>10CFR50.55a GDCs, Reg Guides</p>	<p>None of the items in Section 3.0 of AECB R-7 relates to inservice inspection (ASME Section XI).</p> <p>3.7 specifies maximum allowable annual unavailability, is a probabilistic requirement</p>

AECB R-7 Item	Corresponding ASME Code Item	Comment
Section 4.0 Operating Requirements 4.1 Requirements for Normal Operation 4.2 Requirements for Accident Conditions	Item is not specific to inservice inspection. General requirements for maintaining containment. General requirements for adequate instrumentation and procedures	Very general requirements for containment operability under all conditions.
Section 5.0 Testing Requirements 5.1 Commissioning Tests 5.2 Inservice Tests and Inspections 5.3 Availability Tests	Item relates specifically to ISI Preservice test, ASME Sec III ASME Sec XI, Subsections IWE and IWL No U. S. counterpart, see also Section 3.7 above	The actual Code requirements for containment testing and inspection are contained in Code N285.5. AECB R-7 only invokes the applicable test and inspection requirements.
Appendix: Requirements for Metal Extensions of the Containment Envelope A-1.0 Code Requirements A-2.0 Isolation A-2.1 Primary Heat Transport Auxiliary Systems Penetrating Containment A-2.2 Systems Connected to Containment Atmosphere A-2.3 Closed Systems A-2.4 Small Lines	Items in Appendix are not specific to ISI ASME Sec III, also B31.1 ASME Sec III, Sec XI, GDCs ASME Sec III, Sec XI, GDCs ASME Sec III, Sec XI, GDCs No U. S. counterpart	A-1.0 quotes ASME Sec III, N285.0 and CSA B51 (Canadian Power Piping Code) Specifies inboard and outboard isolation valves Specifies two isolation barriers for lines going to containment atmosphere Requirements decreased if system has closed envelope A-2.4 allows crimping as means of isolation for small bore lines

6.0 SUMMARY AND CONCLUSIONS

We can summarize the results from this investigation in the following seven points:

- (1) The correspondence between the Canadian Codes and ASME Section XI can be summarized as follows:

<u>Plant System</u>	<u>ASME Code Section XI</u>	<u>Canadian Code</u>
Containment, Structural	Subsection IWE	Code N285.5
Containment, Leak Test	ASME IWE with Appendix J	Code N287.7
All other Plant Components	Other Subsections in Sec XI	Code N285.4

N285.4 and N285.5 have been evaluated in this report, while N287.7 is not within the scope of this evaluation.

- (2) Section XI currently contains some O&M items; these will be removed in the near future, so that Section XI will be strictly an ISI Code. The ASME O&M Code will be kept as a separate Code, referenced in 10CFR50.55a in parallel to Section XI. The O&M Code is concerned with maintenance issues, while the ISI Code addresses aging issues. Canadian Codes N285.4 and N285.5 are strictly ISI Codes. O&M issues are not within the scope of this report.
- (3) While the Section XI inspection methodology is based on Code Class, the Canadian ISI Code is structured by Inspection Category instead. Assignment of Inspection Category is not related to Code Class; it is based on
 - Hypothetical Failure Size (Maximum Energy Release)
 - Fatigue Usage Factor
 - Stress Ratio

The Inspection Categories are A, B, C1 and C2. The concept of "Inspection Category" is functionally similar to "Code Class", it is not

similar to the ASME "Examination Category". The higher classes of inspection cover more equipment in the Canadian Code than they do in the corresponding ASME Code. This may mean that the Canadian ISI Code is more conservative than the ASME Code. However, because of the dissimilarity in the approaches, the amount of investigation performed so far is not sufficient for drawing a firm conclusion.

- (4) The Canadian Codes use ASME Code Classes 1, 2 and 3. Instead of Code Classes MC (metal containment) and CC (concrete containment), Canada uses Canadian Code Class 4. In addition, Canada defines Canadian Code Class 6, which is at the B31.1 level. Although the set of Canadian Code Classes (as a set of working templates) is in principle not different from the set of U. S. Code Classes, the assignment philosophy is different:

- In the U. S., all Code Classes are assigned based on the thermal hydraulic function of a plant system, as it relates to plant safety. Regulatory Guide 1.26 is the basis for Code Class assignment (see Section 2.1 of this report).
- In Canada, only Code Classes 1 and 4 are assigned based on plant safety, while Classes 2, 3 and 6 are assigned based on potential radioactive release in case of postulated system failure (see Section 2.5 of this report). This explains why the engineered safety feature systems in the CANDU design are Code Class 1, while in the U. S. they would clearly be Code Class 2.

In summary it can be stated that the Code Classes are basically the same between Canada and the United States. However, the Canadian Code Class assignment philosophy is more conservative than the U. S. assignment philosophy.

- (5) The Canadian Codes are generally performance based while the ASME Code is generally in a prescriptive format.
- (6) The Canadian SDGs (Safety Design Guides) function in the same way as Regulatory Guides. However, the SDGs are explicitly mandatory (see Section 2.5 of this report), while the Regulatory Guides are provided for guidance.

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- (7) The Canadian Policy Statement functions like an SRP Section, also somewhat comparable to a NUMARC Policy Statement. It is different from the NRC Policy Statement (see Section 5.0 of this report).

In terms of conclusions, we can state with respect to applicability of the U. S. inservice inspection methodology to the Canadian reactor plant design that:

1. 10CFR50.55a is directly applicable to the CANDU-3 design, see Section 2.2 of this report.
2. Section XI of the ASME Code does not cover all of the design features of the CANDU-3 design, see Section 4.0 of this report. Also, it might be desirable to consider changing the title of Division 1 of Section XI from "Light-Water Cooled Plants" to "Water Cooled Plants". This would put the Code more in line with 10CFR50.55a, where no distinction is made between light water and heavy water.

7.0 REFERENCES CITED

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3. ASME Code Section XI Rules for Inservice Inspection of Nuclear Power Plant Components, Edition of July 1992, with Addenda of December 1993. American Society of Mechanical Engineers, United Engineering Center, New York, New York.
4. ASME/ANSI OMC-1990 Operation and Maintenance of Nuclear Power Plants (also called the Inservice Test Standard or Inservice Test Code). Edition of 1987, with Addenda up to 1990. American Society of Mechanical Engineers, United Engineering Center, New York, New York.
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