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Assessment of United States Industry Structural Codes and Standards for Application to Advanced Nuclear Power Reactors

Final Report

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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

Through out its history, the USNRC has remained committed to the use of industry consensus standards for the design, construction, and licensing of commercial nuclear power facilities. The existing industry standards are based on the current class of light water reactors and as such may not adequately address design and construction features of the next generation of Advanced Light Water Reactors and other types of Advanced Reactors. As part of their on-going commitment to industry standards, the USNRC commissioned this study to evaluate U.S. industry structural standards for application to Advanced Light Water Reactors and Advanced Reactors. The initial review effort included: (1) the review and study of the relevant reactor design basis documentation for eight Advanced Light Water Reactors and Advanced Reactor Designs, (2) the review of the USNRC's design requirements for advanced reactors, (3) the review of the latest revisions of the relevant industry consensus structural standards, and (4) the identification of the need for changes to these standards. The results of these studies were used to develop recommended changes to industry consensus structural standards which will be used in the construction of Advanced Light Water Reactors and Advanced Reactors. Over seventy sets of proposed standard changes were recommended and the need for the development of four new structural standards was identified. In addition to the recommended standard changes, several other sets of information and data were extracted for use by USNRC in other on-going programs. This information included: (1) detailed observations on the response of structures and distribution system supports to the recent Northridge, California (1994) and Kobe, Japan (1995) earthquakes, (2) comparison of versions of certain standards cited in the standard review plan to the most current versions, and (3) comparison of the seismic and wind design basis for all the subject reactor designs. Finally provided is a suggested plan of action to achieve implementation of the recommended industry consensus standard changes.

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Executive Summary

Throughout its history the USNRC has remained committed to the use of industry consensus standards for the design, construction, and licensing of commercial nuclear power facilities. The existing industry standards are based on the current class of Light Water Reactors and as such may not adequately address design and construction features of the next generation Advanced Light Water Reactors and other types of Advanced Reactors. As part of their ongoing commitment to industry standards the USNRC commissioned this study to evaluate US industry consensus structural standards for application to Advanced Light Water Reactors and Advanced Reactors. Throughout the program a special emphasis was placed on those designs which have submitted design certification applications to the USNRC for Part 52 licensing review.

The specific reactor designs covered under this review program are as follows:

ALWR Plants

- Westinghouse Electric Corporation (W) - AP600 [AP600]
- ASEA Brown Boveri/Combustion Engineering (ABB/CE) - System 80$^+$ [Sys.80$^+$]
- General Electric (GE) - Advanced Boiling Water Reactor [ABWR]
- General Electric (GE) - Simplified Boiling Water Reactor [SBWR]

Advanced Reactors

- United States Department of Energy (DOE)/General Atomics (GA) - Modular High Temperature Gas Cooled Reactor [MHTGR]
- ASEA Brown Boveri (ABB) - Process Inherent Ultimate Safety Reactor [PIUS]
- United States Department of Energy (DOE)/General Electric (GE) - Power Reactor Innovative Small Module [PRISM]
- Atomic Energy of Canada Limited (AECL) - CANDU-3U [CANDU-3U]

This program was conducted in two distinct phases, identified as Phase I and Phase II.

The Phase I effort included (1) the review and study of the relevant reactor design basis documentation for the subject reactor designs, (2) the review of the USNRC's design requirements for the Advanced Reactors, (3) the review of the latest revisions of the relevant industry structural consensus codes and standards, and (4) the identification of the need for changes to the affected industry codes and standards. This review is conducted in two distinct tasks. The first task, Task A of Phase I encompasses the review of subject reactor design basis information and identification of the applicable USNRC's Advanced Light Water Reactor and Advanced Reactor design requirements. The second task, Task B encompassed an in depth review of the industry codes and standards which are the subject of this program in relation to the information obtained in Task A.

As part of this Phase I effort, Stevenson and Associates conducted an on-site survey of the performance of structures and distribution system supports subjected to both the 1994 Northridge, California and the 1995 Kobe, Japan earthquakes. Detailed summaries of the surveys were provided in Appendices to the reports and the observations from the studies were considered and used in the Phase II effort.

The second phase (called Phase II) of this effort consisted of using the results of the Phase I effort to determine the changes which are recommended for the subject industry consensus standards and the development of a suggested program to achieve the implementation of the recommended changes.

The primary reactor design definition documents used in this review effort are Standard Safety Analysis Reports (SSAR), or if the SSAR's are not available Preliminary Safety Information Documents (PSID) are used. In addition for some reactor designs Draft USNRC Safety Evaluation Reports (SER) were available for review and use. Together the SSAR's or PSID's and the SER's formed the basis for the review and identification of the design basis and potential unique features associated with each ALWR or Advanced Reactor design.

The documents used to identify and quantify the USNRC positions and criteria concerning the design and construction of ALWR and Advanced Reactors were a series of SECY letters, draft changes to the Code of Federal Regulations, draft changes to Regulatory Guidelines, preliminary and final SER's, and discussions with USNRC Staff members. In addition standard regulatory requirement definition documents such as the current Code of Federal Regulations, Standard Review Plan (SRP), Regulatory Guidelines, and NUREG's and NUREG/CR's were used in this effort.

The standards which are the subject of this review are the current versions of the structural and seismic design standards which are used for the design and construction of Seismic Category I, safety related structures in nuclear power facilities. This includes certain subsections of Section III and Section XI of the ASME Boiler and Pressure Vessel Code, ACI Codes, AISC Specifications, ASCE and ANS Standards and other select industry standards.

The majority of the results of the Phase I efforts are provided in tabular form to consolidate the study results. The resulting tables include:

- Identification of industry consensus standards cited in the SRP
- Identification of the safety classification, seismic category and applicable standards for the safety related structures of each of the subject Advanced Light Water Reactors and Advanced Reactor Designs
- The identification of unique design or construction features for each Advanced Light Water Reactor or Advanced Reactor Design
- Changes required to address existing industry standard deficiencies
- Related and applicable ASME BPVC Code Cases

The results of the Phase I Program were used to develop a set of recommend standard change for application of the standards to ALWR and Advanced Reactor Construction. Over seventy (70) sets of recommended changes are provided as a result of the review effort. The following provides a summary of the standards for which changes were suggested:

•ASME Boiler and Pressure Vessel Code, Section III
- -Division 1, Subsection NE
- -Division 1, Subsection NF
- -Division 1, Appendix N
- -Division 2, Subsection CB
- -Division 2, Subsection CC

•ASME Boiler Pressure Vessel Code, Section XI

•ASME
- -ASME AG-1

•American Concrete Institute
- -ACI 349
- -ACI 530

•American Society of Civil Engineers
- -ASCE 4
- -ASCE 7

•American Institute of Steel Construction
- -N690 Specification
- -Manual of Steel Construction (ASD)

•American Iron and Steel Institute
-Cold Formed Steel Design Manual

•National Fire Protection Association
-NFPA 803
-NFPA 13
-NFPA 14

•American Nuclear Society
-ANS 50.1
-ANS 51.1
-ANS 51.2
-ANS 56.1
-ANS 58.2
-ANS 58.14

•American Welding Society
-AWS D1.1
-AWS D1.3

•Institute of Electrical and Electronics Engineers
-IEEE 628

In addition it was recommended new standards should be developed in the following areas:

•Minimum design loads for nuclear safety related structures in nuclear power facilities

•Minimum design loads for non-safety related structures in nuclear power facilities

•Man-made hazard phenomenon design requirements for safety related and non-safety related facilities in nuclear power plants

•Safety criteria standards for gas cooled, liquid metal, and heavy water reactors.

Also developed as part of the Phase II effort was a suggested course of action for implementation of the recommended changes to industry consensus standards.

In addition to the recommended standard changes, several other sets of information and data were extracted for use by USNRC in other ongoing programs. This data was provided in appendices to the report. It included:

•Detailed observations on the response of structures and distribution system supports to recent strong motion earthquakes (experience data)

•Comparison of versions of certain standards cited in the Standard Review Plan to the most current versions. The subject standards of this review were AISC N690, ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, ACI - 349, and ACI - 359.

•Comparison of the seismic and wind design basis for all the subject Advanced Light Water Reactor and Advanced Reactor designs.

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1.0 Introduction and Program Description

1.1 Introduction

1.2 Program Objective

The objective of this program is to evaluate United States industry consensus structural codes and standards for application to the design and construction of the Seismic Category I , Safety Class structures of currently proposed Advanced Light Water Reactions and Advanced Power Reactors. A special emphasis was placed on those designs that have submitted design certification applications to the USNRC for Part 52 licensing review (the Advanced Light Water Reactor designs). In addition the focus is on those structures or features, such as modular construction, which are unique to the Advanced Light Water Reactor (ALWR) or Advanced Power Reactor designs.

1.3 Program Description

1.3.1 General Scope Description

The subject of this review was the industry codes and standards which are expected to be applied to the design of the Seismic Category I, Safety Class structures in the ALWR and Advanced Power Reactors. The overall focus of the program was the Seismic Category, Safety Class structures and distribution system supports of the ALWR and Advanced Power Reactors shown in Table 1.1. The program was further focused toward the unique aspects of these structures and distribution system supports for which industry codes and standards currently may not provide adequate design and construction guidance or criteria. The emphasis is on design, construction, and inspection aspects of the subject industry consensus codes and standards. For the purposes of this program unique features or uniqueness is defined as attributes or aspects of a particular design that are not adequately addressed in the standards which are the subject of this study. This could include physical attributes, design and analysis attributes, fabrication issues, etc. Therefore "unique features or attributes" applies not only to physical features but also to unique design, analysis, testing, and fabrication features.

The industry codes and standards which were considered in this review are shown in Table 1.2. In addition Table 1.2 provides the general subject area of each of the listed industry codes and standards. The detailed title and reference for each of these is provided in Section 6.11. These industry codes and standards are divided into two categories: primary and secondary. The primary industry consensus codes and standards were the main subject of this review effort. The secondary industry codes and standards are those which are related to the application of the primary industry consensus codes and standards. They may be used in conjunction with or in support of the primary industry codes and standards.

The resulting deliverable of the program is the identification of specific, necessary industry code or standard changes which should be implemented to make the subject industry codes and standards applicable for use in the design and construction of evolutionary and advanced commercial power reactors. This was accomplished by first reviewing the available reactor design basis information and identifying the Seismic Category I, Safety Class items. Next, if available, USNRC Safety Evaluation Reports on specific reactor types were reviewed to determine any licensing guidance applicable to structures and supports. The Seismic Category I, Safety Class items so identified were then screened to identify aspects of those items having unique design features or attributes.

Then USNRC regulatory guidance such as the NUREG-0800 (The Standard Review Plan), regulatory guides, the Code of Federal Regulations (CFR), USNRC staff position papers, and SECY letters were reviewed to identify USNRC issues, requirements and criteria as they relate to the licensing of ALWR and Advanced Power Reactors.

Using these issues, requirements and criteria the subject industry codes and standards were reviewed to identify areas where the codes or standards could be improved or updated to provide more coherent, consistent, design and construction criteria and requirements.

The results of this review were then used to identify suggested industry code and standard changes which should be made to enhance their applicability to the structural features of Advanced Power Reactors shown in Table 1.1.

Table 1.1 -Commercial Power Reactor Structures Reviewed

Concrete Structures
Steel Structures
Containment (Pressure Retaining) Structures
Containment Penetrations and Hatches
Containment Leak Testing
Fire Barriers
Earthen Dams
Masonry Walls
Distribution System Supports
-Piping
-HVAC
-Cable Trays
-Conduit
-Fire Protection

Table 1.2 - Principal Codes and Standards Reviewed

Code and Standard	General Subject
PRIMARY	
ASME BPVC III, Division 1, Subsection NE	Metal Containment Structures
ASME BPVC III, Division 2, Subsection CB	Concrete Reactor Vessels
ASME BPVC III, Division 2, Subsection CC	Concrete Containment Structures
ASME BPVC III, Division 1, Subsection NF	Component Supports - Nuclear
ASME BPVC CODE CASES	Various
ASME BPVC XI-Subarticle IWE, IWL	Leak Rate Testing - Containments
ACI-349	Reinforced Concrete Structures - Nuclear
AISC N690	Steel Structures - Nuclear
IEEE-628	Class 1E Cable Tray Systems
AISI-CFSDM	Cold Formed Steel Design - Commercial
SMACNA Standards	HVAC Design Standards - Commercial
ASCE 7-93	Minimum Design Loads
SECONDARY	
Code and Standard	**General Subject**
ASME BPVC Section IX	ASME Code Welding
ASME AG-1	Code on Nuclear Air and Gas Treatment

Table 1.2 - Principal Codes and Standards Reviewed (continued)	
AISC MSC-ASD	Steel Structures - Commercial
MSS-SP-58	Component Standard Supports
AWS D1.1	Structural Welding
AWS D1.4	Reinforcement Bar Welding
AWS D9.1	Sheet Metal Welding
ASCE 1-82	Dams and Dikes - Nuclear
ASCE 4-86	Dynamic Structural Analysis - Nuclear
NFPA-13	Sprinkler Systems - Commercial
NFPA-14	Standpipe and Hose Systems
NFPA-803	Fire Protection Nuclear Power Plants
ACI-530	Masonry (Block) Walls - Commercial
ACI-318	Reinforced Concrete - Structural
ANS 2.2	Earthquake Instrumentation
ANS 2.3 (Expired)	Tornado Design
ANS 2.8	Flooding Design
ANS 2.11	Geotechnical Parameters
ANS 2.23	OBE Exceedence
ANS 56.8	Containment Leak Testing
ANS 58.1 (Now Appendix to ANS 58.3)	Plant Design Against Missiles
ANS 58.2	Pipe Rupture Protection
ANS 58.3	Physical Protection for SSC - Safety Class
ANSI/ASME B31.1	Power Piping Code - Distribution System Supports
Experience Data	For Distribution System Supports

1.3.2 Reactor Types

The reactor types which were reviewed under this program have been segmented into two categories: Advanced Light Water Reactors (ALWR) and advanced reactors. ALWR designs are those designs which were developed by improving on existing operational reactor designs. These are enhanced or upgraded pressurized water reactor or boiling water reactor designs based on the steam conversion cycle for power generation. ALWR plant designs can be subclassified as "non-passive" and "passive". The major difference is that "passive" reactors require no operator action for near term (approximately 72 hours) post accident mitigation while the "non-passive" reactor design do require such action. This subclassification was not used extensively in the report but it was referred to and discussed for some safety system design aspects. Advanced reactors are any reactors based on safety and operation concepts not currently in any significant commercial use in the United States. These advanced reactor designs include: gas cooled reactors, liquid metal reactors and heavy water reactors.

The specific reactor designs covered under this review program are as follows:

ALWR Plants

- Westinghouse Electric Corporation (W) - AP600 [AP600] (Passive)
- ASEA Brown Boveri/Combustion Engineering- (ABB/CE) System 80+ [Sys.80+]
- General Electric (GE) - Advanced Boiling-Water Reactor [ABWR]
- General Electric (GE) - Simplified Boiling Water Reactor [SBWR] (Passive)

Advanced Reactors

- United States Department of Energy (DOE) /General Atomics (GA) - Modular High Temperature Gas Cooled Reactor [MHTGR]
- ASEA Brown Boveri (ABB) - Process Inherent Ultimate Safety Reactor [PIUS]
- United States Department of Energy (DOE)/General Electric (GE) - Power Reactor Innovative Small Module [PRISM]
- Atomic Energy of Canada Limited (AECL) - CANDU-3U [CANDU-3U]

Section 2.0 provides a brief description of these various reactor designs. Also considered to a lesser degree in this review effort was the Electric Power Research Institute (EPRI), Advanced Light Water Reactor (ALWR) Utilities Requirement Document (URD) as it applied to the ALWR reactor designs.

This program focused the majority of the review effort and suggested modifications of the industry codes and standards on the four (4) ALWR reactor designs. This review included the indepth review of all available design basis documentation and the suggestion of potential code changes. The review of the four (4) advanced reactor designs was conducted at a significantly reduced level of effort. The design basis documents were reviewed in a cursory manner and only highly significant design features were identified. In addition no specific suggested code changes were discussed or prepared in relation to these advanced reactor designs. It should be noted however that some unique features and design concepts of the advanced reactors were common to the ALWR reactors and therefore the suggested code changes would be applicable to several aspects of the design of these advanced reactors.

1.3.3 Codes and Standards

The term "Standard" is used to describe any document which expounds a preference for performing a given activity in a particular manner. In general standards can be Codes, Specifications, Guidelines and Criteria. A Code is a particular type of standard which is prepared by or for a regulatory authority for use within the regulatory authority's jurisdiction. Codes often have the force of law within the jurisdiction. Specifications, Guidelines or Criteria are usually prepared to be adopted as part of a contract between organizations and are enforceable as a matter of contract law. However, in specific instances, regulatory authorities may adopt such standards as a matter of law within their jurisdiction.

Section 1.3.1 provides a general discussion on the industry codes and standards which were the focus of this program. Table 1.3 provides a detailed list of all the industry codes and standards which were initially considered by this program. An extensive detailed review of the industry codes and standards listed in Table 1.3 was conducted to determine their applicability in the design of the structures and distribution systems supports which were the subject of this program. In the column in Table 1.3 marked "Applicability" one of three designations is provided:

- Primary
- Secondary
- N/A

An industry code or standard designated primary is a primary code or standard that is directly applicable to this program. An industry code or standard marked secondary is applicable to this review in so far as it supports or augments an industry primary code or standard. A code marked N/A indicates a code or standard which was reviewed but it was determined not to have direct application to the scope of this review effort.

Table 1.3 - Editions of Codes/Standards Considered in this Review			
Codes\Standard[2]	**Latest Published Edition**	**Next Expected Edition**	**Applicability**
ACI-349-85 (Rev. 1990)	1990	>1995	Primary
ACI-318-89 (Rev. 1992)	1992	1995	Secondary
ACI-301-89	1989	>1994	Secondary
ACI-530-92	1992	1995	Secondary
AISC - MSC (2nd) [LRFD]	1994	>1995	N/A
AISC - MSC (9th) [ASD]	1989	None Planned	Secondary
AISC - N690[3]	1994	1999	Primary
AISI CFM PART 1	1989	>1995	Primary
AISI CFM PART 2	1989	>1995	Primary
AISI CFM PART 3	1986	>1995	Primary
AISI CFM PART 4	1986	>1995	Primary
AISI CFM PART 5	1986	>1995	Primary
AISI CFM PART 6	1986	>1995	Primary
ANS 2.2	1988	TBD	N/A
ANS 2.3 (Expired)	1983	>1995	Secondary
ANS 2.8	1992	>1997	N/A
ANS 2.11	1978(89)	TBD	Secondary
ANS 2.23	Draft	In Progress	N/A
ANS 56.8[1]	Draft	2nd QTR - 1995	Secondary
ANS 58.1 (58.3)	1992	> 1994	Secondary
ANS 58.2	1988	>1994	Secondary
ASCE 1-82	1992	None Planned	Secondary
ASCE 7-93	1993	None Planned	Secondary
ASCE 4-86	1987	In Progress	Secondary
ASME III-Div.1, NE	1994	1995	Primary
ASME III-Div. 1, NF	1994	1995	Primary
ASME III-Div.2, CB	1994	1995	Primary
ASME III-Div.2, CC	1994	1995	Primary
ASME IX	1994	1995	Secondary
AWS D1.1	1994	1996	Secondary
AWS D1.3	1989	Unknown	Secondary
AWS D1.4	1992	Unknown	Secondary
AWS D9.1	1990	Unknown	Secondary
IEEE-628	1987	>1995	Secondary
IEEE-344	1987	>1995	N/A

Table 1.3 - Editions of Codes/Standards Considered in this Review (continued)			
Codes\Standard[2]	Latest Published Edition	Next Expected Edition	Applicability
NFPA 1	1992	>1994	N/A
NFPA 13	1991	1994	Primary
NFPA 14	1993	>1995	Secondary
NFPA 49	1991	>1995	N/A
NFPA 51B	1989	>1995	N/A
NFPA 80A	1993	>1995	Secondary
NFPA 241	1993	>1995	N/A
NFPA 251	1990	1994	N/A
NFPA 255	1990	1994	N/A
NFPA 256	1993	1994	N/A
NFPA 801	1991	>1994	N/A
NFPA 802	1993	>1995	N/A
NFPA 803	1993	>1995	Secondary
NFPA 850	1992	>1994	N/A
SMACNA, "Seismic Restraint Manual Guidelines for Mechanical Systems"	1991	>1995	Secondary
SMACNA, "HVAC Duct Construction Standards-Metal and Flexible"	1985	>1995	N/A
SMACNA, "Round Industrial Duct Construction Standards"	1997	>1995	Secondary
SMACNA, "Rectangular Industrial Duct Construction Standards"	1980	>1995	Secondary
ANSI/ASME AG-1	1988	>1995	Secondary
Uniform Building Code (UBC)	1994	>1997	N/A
ANSI/ASME B31.1	1993	1996	Secondary
MSS-SP-58	1993	>1995	Secondary

Footnotes for Table 1.3:

(1) These standards were reviewed based on Draft versions of currently proposed changes to the subject standards. This information was received from the applicable code committee organization.

(2) The detailed formal references for these industry codes and standards are provided in Section 6.11.

(3) This standard is due to be published in the last quarter of 1995. (Current Best Estimate)

1.4 Program Plan Description

1.4.1 Program Overview

As previously discussed this effort is conducted in two distinct phases. The first phase (called Phase I) consisted of an in depth review of the ALWR and Advanced Reactor design documentation, and the subject industry codes and standards. The results of this in depth review are used in Phase II of the program to provide the information and data necessary to identify modifications and/or criteria changes which should be initiated to make the subject industry codes and standards applicable to the construction of ALWR and Advanced Reactor designs.

The Phase I effort included (1) the review and study of the relevant reactor design basis documentation for the subject reactor designs, (2) the review of the USNRC's design requirements for the advanced reactors, (3) the review of the latest revisions of the industry codes and standards which are subject to this study, and (4) the identification of the need for changes to the affected industry codes and standards. This review is conducted in two distinct tasks. The first task, Task A of Phase I encompasses the review of subject reactor design basis information and identification of the applicable USNRC's evolutionary and advanced reactor design requirements. The second task, Task B encompassed an in depth review of the industry codes and standards which are the subject of this program in relation to the information obtained in Task A. The detailed discussion of the information reviewed in this section is provided in Section 1.4.3 through 1.4.5.

The second phase (called Phase II) of this effort consisted of using the results of the Phase I effort to (1) determine the changes which are suggested to the industry codes and standards listed in Table 1.2, and (2) the development of a suggested course of action to achieve the implementation of the identified changes. This Phase II review effort is conducted as described, in further detail, in the following paragraphs.

Using the results of the Phase I investigation, it is determined for each primary industry consensus code or standard listed in Table 1.2 what changes or revisions are needed to insure that these industry codes and standards are applicable to the development of subject reactor plant designs.

In addition for the codes and standards listed in Table 1.4, the current revisions were compared to those revisions referenced in the USNRC Standard Review Plan, NUREG-0800 [6.9-13].

Table 1.4 - Codes and Standards Compared in Depth to those cited in the SRP

AISC N690
ACI-349
ACI-359 (ASME BPVC Sec. III, Div. 2)
ASME BPVC, Sec. III, Div. 1, Subsection NE

A detailed tabular comparison of the differences of the two code revisions reviewed was then provided in Appendix B.

1.4.2 Emphasis of the Review

As briefly mentioned in Section 1.3.2 the primary focus of this review effort is the four (4) ALWR plant designs. For these designs the Phase I review is in depth and the results of Phase II were directed toward these reactor designs. For the advanced reactor designs only a cursory Phase I review is conducted and no effort is expended on these reactors during Phase II. Also the review only focuses on the civil structural aspects of the evolutionary or advanced reactor designs. No effort is expended on equipment, components, and distribution systems (other than distribution systems supports), etc.

1.4.3 Applicable Reactor Design Definition Documents

The primary reactor design definition documents used in this review effort are Standard Safety Analysis Reports (SSAR), or if SSAR's are not available Preliminary Safety Information Documents (PSID) are used. In addition for some reactor designs Draft USNRC Safety Evaluation Reports (SER) were available. Together the SSAR's or PSID's and the SER's formed the basis for the review and identification of the design basis and potential unique features associated with each ALWR or advanced reactor design. In addition other vendor documentation such as letters, trip reports, topical reports, design drawings, etc. if applicable and if available were used in the study.

Due to the large quantity of data reviewed and the large amount of data correlated and summarized, specific detailed referencing of each piece of data is not done in this report. The references provided in Section 6, however, were segregated and organized by reactor type, regulatory reference, etc. References provided in Sections 6.1 through 6.9 are the documents used to provide the input for the reactor design basis definition and identification of unique features.

1.4.4 Applicable USNRC Criteria and Position Definition Documents

The documents used to identify and quantify the USNRC positions and criteria concerning the design and construction of ALWR Advanced Power Reactors were a series of SECY letters, draft changes to the Code of Federal Regulations, draft changes to Regulatory Guidelines, preliminary and final SER's, and discussions with USNRC Staff members. In addition the standard regulatory guidance documents such as the current Standard Review Plan (SRP), Regulatory Guidelines, and NUREG's and NUREG/CR's are used in this effort. Also considered were the applicable sections of the current Code of Federal Regulations.

As with the reactor design definition, documentation specifying a specific reference is not provided in the report. Section 6.10 provides a complete listing of all documents used to identify and quantify the USNRC positions, criteria and guidance for advanced and evolutionary light water reactors.

1.4.5 Miscellaneous Applicable Documents

In reviewing the Design Basis and USNRC criteria documentation there are several documents which have significance both to the Phase I review and in developing the Phase II code changes. These documents are in general referenced by name rather than specific reference number. However, they were all listed by name and source in Section 6.12 of the reference section.

1.4.6 Consideration of Recent Earthquake Experience Data

As part of this effort Stevenson and Associates conducted an on-site survey of the performance of distribution system supports in the 1994 Northridge, California earthquake. The focus of this review was the performance of the distribution specific supports at three fossil power stations which experienced high seismic excitation during this recent earthquake. The stations surveyed were the Valley Steam Plant, the Glendale Power Station and the Pasadena Power Station. The detailed results of this survey are provided in Appendix A2 and a summary is provided in Section 3.6. In addition Stevenson and Associates also performed an on-site review of the response of structures and distribution systems to the 1995 earthquake in Kobe, Japan. The detailed results of this review are provided in Appendix A3 and summarized in Section 3.6. The observations from these on-site reviews were considered in developing the suggested changes to industry codes and standards for distribution system supports.

1.4.7 Containment Nomenclature

One source of confusion in reviewing the various reactor types is nomenclature used with respect to the containment design. For the purposes of this report primary containment is defined as the reactor vessel and associated reactor coolant pressure boundary. The secondary containment is the ASME Boiler and Pressure Vessel Code, Section III Division 1, Class MC or Section III Division 2, Class CC structures designed for at least a 5.0 psi internal pressure. The reactor vessel, designated portions of the reactor coolant system and potentially other safety related equipment are located within the secondary containment. A third independent vessel or structure, usually designed to ACI-349, is sometimes present which provides biological shielding and may provide protection from external hazards to the secondary containment. This third vessel or structure may also provide the capability to filter leakage from the secondary containment. This third vessel or structure is generally referred to as the shield building. When the term "containment" is used in this report without modifiers, it is understood that it means the pressure retaining vessel or structure designed and constructed to the ASME Boiler and Pressure Vessel Code, Section III Division 1, Class MC or Section III Division 2, Class CC, ie, the secondary containment.

1.4.8 Other Aspects of the Review

In conducting this review effort the design basis information is reviewed in depth. There are observations concerning the design features and criteria for these reactor designs which although not directly related to changes required in industry codes and standards are felt to be of significance. The items are identified in Appendix C and are provided to USNRC for their consideration in the ongoing ALWR and Advanced Power Reactor licensing process. In addition at the request of the USNRC, a comparative study for four Civil/Structural design standards was conducted. The purpose of this study was to compare the most current revisions of the industry standards to the revisions cited in the Standard Review Plan (NUREG-0800). This was done for AISC N690, ASME BPVC Section III, Division 1, Subsection NE, ACI-349, and ACI-359.

1.4.9 Application of Commercial Standards

In some cases the reactor vendors (owners) have chosen to use commercial (non-nuclear safety related) Standards for the design and construction of certain aspects of a reactor design. For the purposes of this report when this was done suggested changes to the commercial standard required to make it compliant with applicable regulatory requirements and guidelines were provided. However it should be noted that the use of a commercial standard for a nuclear safety class item is permissible if in the SSAR the reactor vendor (owner) commits to any additional requirements or specifications necessary to make the application of the standard compliant with the applicable regulatory requirements. Some examples of these additional requirements which could be specified include extreme load design criteria, necessary quality assurance requirements, appropriate material limitations, etc.

2.0 Overview of the Subject Reactor Designs

2.1 Westinghouse AP600

The AP600, is a simplified, standard 600-MWe nuclear power plant design that combines pressurized water reactor technology with passive safety systems. The simplified, compact plant arrangement has been designed to provide adequate shielding and space for inspection, maintenance, laydown, and removal. For example, the containment is 130 ft (39.6 m) in diameter, compared to the 105 ft. (32 m) containment diameter of a conventional two-loop 600-MWe plant. The additional space helps to reduce traffic congestion and provides valuable laydown area inside the containment during plant outages.

The AP600's passive safety systems use the natural forces of gravity, convection, condensation, and evaporation. These systems are less dependent on operator action or complex, redundant and active emergency equipment. Innovative features include a large volume (500,000 gal) of gravity fed water stored in the containment to eliminate the need for operator action to ensure makeup reactor coolant water, either for small leaks that may occur during normal operation, or a major LOCA. The passive residual heat removal system removes core decay heat if steam generator heat removal is not available. The Automatic Depressurization System (ADS) depressurizes the Reactor Coolant System (RCS) if the core makeup tank level is low.

The Passive Containment Cooling System (PCCS) provides the safety-grade ultimate heat sink that prevents the containment shell from exceeding its design pressure of 45 psig. The PCCS uses natural air circulation between the steel containment shell and the concrete shield building. In accident situations, air cooling is enhanced by the distribution of water onto the steel containment shell. The water is gravity fed from a 350,000 gallon annular tank designed into the roof of the shield building. This tank has sufficient water to provide three days of cooling.

This design also incorporates the use of modular construction for the containment internal floor and wall structures and certain portions of the auxiliary buildings. The purpose of this type of design is to permit significant in-shop (off-site) component fabrication and simplify the on-site assembly and construction.

A significant number of unique or advanced features of this reactor design are in the systems, components, and equipment areas which are outside of the scope of this review effort. The containment design features and criteria and the modular construction methods are unique structural features and the applicability of current industry codes and standards to these features is the focus of this review effort for this evolutionary reactor design.

2.2 ABB/Combustion Engineering System 80+ (Sys 80+)

ABB\Combustion Engineering Nuclear Power's System 80+ Standard Plant is a 1300-MWe advanced pressurized water reactor. It is based upon evolutionary improvements to the standard System 80 Nuclear Steam Supply System (NSSS) and the Cherokee/Perkins Balance of Plant (BOP) design developed by Duke Power Co. Like previous ABB/CE reactors, the System 80+ Reactor Coolant System (RCS) has a two-loop configuration. The System 80+ concept is a complete power plant, including nuclear island, turbine island, and BOP Components.

Active dedicated, four-train safety systems provide emergency core cooling, and feedwater and decay-heat removal. Emergency coolant is piped directly to the reactor vessel, and draws water from an Inside Containment Refueling Water Storage Tank (IRWST)--in conjunction with the emergency core-cooling system, providing an alternative path for decay heat removal through feed and bleed action. The significance of the IRWST being inside containment is its close proximity to the reactor coolant system.

A spherical steel containment provides 75 percent more space on the operating floor than a typical cylindrical containment of equal volume. This simplifies refueling outages and plant maintenance by providing more laydown space and working area. This containment also has been designed specifically to mitigate consequences of severe accidents. Physical placement of each safety train into a separate quadrant of the plant addresses concerns about fire, flood, and sabotage. The containment is surrounded by a cylindrical concrete shield building.

The vast majority of advanced features in the System 80+ are based on upgrades and improvements of existing System 80 fluid systems, instrumentation and control systems and components. There are essentially no unique or new design features in the civil structural area. This review focused on aspects of the spherical containment design, the overall design criteria and the industry consensus codes and standards applicable to the design of distribution systems supports.

2.3 General Electric Advanced Boiling Water Reactor (ABWR)

GE's 1300-MWE (net) Advanced Boiling Water Reactor (ABWR) incorporated the best proven features of BWR designs in Europe, Japan, and the United States, and also uses state-of-the-art electronics, computer, turbine, and fuel technology.

The ABWR eliminates recirculation piping which permits a more compact containment design. It also allows elimination of all large vessel nozzles below the core, and, therefore, it results in the design of a safer, more economic Emergency Core Cooling System (ECCS). Elimination of external recirculation piping and use of vessel forged rings results in a greater than 50 percent reduction of welding joints. The ABWR's internal pumps are an improved version of the wet motor glandless type design. Significant operational experience with these pumps has been accumulated at a number of European BWRs.

The Reactor Pressure Vessel (RPV) is about 7 m in diameter and 21 m in height. It is a standard BWR vessel design except for two items: (1) the annular space between the RPV shroud and the vessel wall is increased to permit positioning of the 10 internal recirculation pumps, and (2) the standard cylindrical vessel support skirt has been changed to a conical skirt, again, to permit use of the 10 internal recirculation pumps.

The ABWR uses a pressure suppression containment which is a hybrid of earlier GE Boiling Water Reactors (BWR) Mark II and Mark III pressure suppression type containment designs. This containment incorporates both drywell and wetwell scheme but incorporates several modified concepts for implementation of severe accident prevention. The containment vessel is a cylindrical steel lined reinforced concrete structure enclosing the reactor vessel and integrated with the reactor building.

As with the previous reactor designs discussed the majority of advanced and unique features are fluid and instrument and control systems and component related as opposed to structural modifications. The containment design and design criteria along with distribution system supports were the focus of our review effort.

2.4 General Electric Simplified Boiling Water Reactor (SBWR)

The 640-MWe Simplified Boiling Water Reactor (SBWR), incorporates the best proven features of BWR designs in Europe, Japan, and the United States, and also uses state-of-the-art electronics, computer, turbine, and fuel technology. Elimination of external recirculation piping in the SBWR permits a more compact containment design. It also allows elimination of all large vessel nozzles below the core, and, therefore, design of a safer, more economic Emergency Core Cooling System (ECCS). Elimination of external recirculation piping and use of vessel forged rings results in a greater than 50 percent reduction in welded joints.

The natural circulation used to accomplish the core coolant flow in the SBWR technology is not new to BWRs. The small (60 MWe) Dodewaard plant in the Netherlands which uses natural circulation core cooling has operated since the 1960's at a lifetime capacity factor of 84 percent. The small size of the SBWR allows use of this feature. Larger BWRs (Leibstadt and Vermont Yankee, among others) have operated at 50 percent power levels in natural circulation mode which further verifies the SBWR's natural circulation feature. The selection of natural circulation as the means for providing coolant flow through the reactor, coupled with a 42-kW/litre core power density, results in a number of benefits. Compared to existing forced circulation plants, the natural circulation SBWR offers low fuel cycle costs, fewer operational transients, and increased thermal margin for transients expected to occur. In addition, elimination of the recirculation loops, the pumps, and the controls needed for forced circulation substantially simplifies the design.

The SBWR power cycle includes a single high-pressure turbine and a single two flow low pressure turbine with 52-in. last stage buckets. To achieve further simplification, the SBWR steam conversion power cycle is a non-reheat cycle. The large moisture-separator reheaters are replaced by compact high velocity separators, whose performance has been demonstrated through use in France. The SBWR employs a pressure suppression containment design with a passive emergency response system. The SBWR uses the suppression pool, a gravity driven cooling pool, an isolation condenser, and a passive containment cooling system (PCCS). For accident response, the PCCS provides long-term passive cooling capability for the containment using natural convection processes. No active pumps or diesels are needed for heat removal, resulting in no operator action required for at least three days after a LOCA.

Based on discussions with members of the USNRC staff and some supplemental information from GE, it appears that aspects of the SBWR civil/structural design will include modular construction. However the current status of the design and the implementation of modular construction techniques is in a very preliminary and essentially undefined design stage. This limited the ability to provide an indepth review of the modular construction aspects of the SBWR design.

The majority of the other civil structural elements of the SBWR design are typical of current operating BWR plants. This review focused on design criteria, containment, and concrete design and construction, distribution system supports and limited aspects of the potential use of modular construction for the SBWR.

2.5 DOE/GA Modular High Temperature Gas Reactor (MHTGR)

The modular high-temperature gas-cooled reactor (MHTGR) uses a single phase helium coolant and a high-heat-capacity graphite moderator. This concept has been augmented through use of refractory-coated particle fuel, and reactor size, shape, and power density chosen to provide for passive heat removal.

The reference MHTGR plant consists of four identical 350-MWt reactor modules whose output together totals 538 MWe (net). Each module is housed in a vertical cylindrical concrete silo embedded underground. Each silo serves as an independent, vented confinement structure. The four reactor structures form part of the nuclear island along with other structures that house systems for helium purification; shutdown cooling; hot cell maintenance; power conditioning; and heating, ventilating, and air conditioning. The energy conversion area, or turbine island, is nonsafety-related and it is separated from the nuclear island so that commericial (verses nuclear safety related) standards can be used in its construction and operation.

A recent design modification replaced the secondary steam electric generation turbine with a primary gas turbine unit. The design modifications associated with the implementation of this change are still in the process of conceptual design. This could have significant effect on systems design and system design criteria but the effect on the structural design should be minimal. Therefore the Phase I review effort proceeded with the currently available design data and criteria.

The active region of the core consists of fuel blocks arranged in three annular rings. The center and outer portions of the core are made from unfueled reflector blocks. The core assembly is surrounded by a steel core barrel and contained inside the uninsulated reactor vessel. Helium flows downward through the core to a plenum at the bottom of the core. The hot helium flows through the inner part of the steam generator vessel and downward through the tube bundle. Cool helium flows upward around the outside of the tube bundle to a single-stage, axial compressor driven by an electric motor equipped with magnetic bearings and then through the outer part of the cross duct.

The RCCS is the safety-related heat removal system used by the MHTGR. The RCCS is the only safety grade heat removal system and is totally passive. Heat is transferred by means of conduction, convection, and radiation from the core to the RCCS. The system has no controls, valves, circulating fans, or other active components.

The most advanced or unique structural features of the MHTGR are in the area of safety design criteria and concepts. The overall building structural design is based on a probabilistic confinement concept in lieu of the classical nuclear plant containment concept. The use of a confinement concept for relatively small commercial gas cooled nuclear power plants was permitted in the Fort Saint Verain Plant. Therefore the primary confinement structures use nuclear concrete and structural steel design codes in lieu of ASME BPVC Section III containment design criteria. Further the safety classification of systems, structures, and components is also based on a probabilistic approach and criteria versus more traditional deterministic methods. The majority of the reactor building is underground which require a complex soil structure interaction analysis.

This is one of the limited review scope reactor designs and therefore the review effort focus on the potential effects of probabilistic safety classification approach and the design and evaluation of the deep soil embedment confinement structure on the applicable industry consensus codes and standards. Some discussion is provided on the potential changes in these industry consensus codes and standards which may be necessary to address this probabilistic approach.

2.6 ABB Process Inherent Ultimate Safety (PIUS)

The PIUS (Process Inherent Ultimate Safety) Reactor is an effort to develop a nuclear power plant design in which safety against severe accidents is a built-in feature of the reactor configuration and cannot be compromised by malfunctioning equipment or human intervention. The PIUS design is based on well-established LWR technology and infrastructure in which demonstrated component technology is used to the maximum extent. Compared with current LWR designs, the primary system configuration has been rearranged.

The core has 213 fuel assemblies located near the bottom of the reactor pool, a high-boron-content water mass enclosed in a vessel. Reactivity is controlled by coolant boron concentration and temperature. Control rods are not used.

From the core, the coolant passes up through a riser pipe and leaves the reactor vessel in the upper portion. There are four steam generators of the straight-tube once-through type. Main coolant pumps are glandless, wet-motor design. An open natural-circulation path through the core is always available - from the pool through a lower density lock of the core, and then through the core itself, the riser, the passage from the upper riser plenum, and the upper density lock back to the pool.

The prestressed concrete reactor vessel has a cavity diameter of about 12 m and contains some 3300 m^3 of water. The vessel monolith has a cross-section of about 27 m and a height of about 43 m. Pressure-retaining capability is ensured by a large number of horizontal and vertical prestressing tendons, and by reinforcement bars. The inside of the cavity has a stainless-steel liner. Also, there is a second barrier (an embedded steel membrane) from the bottom of the reactor to a height above the upper density lock. This liner insures the reactor pool water below this level cannot be lost by liner leakage.

On top of the concrete vessel, there is a steel vessel extension, anchored to the bottom of the concrete vessel by separate tendons. A steel dome closes the shaft above the reactor vessel. The concrete vessel and reactor system are enclosed in a large containment structure of pressure-suppression type. All equipment containing reactor loop or reactor pool water at high pressure and high temperature is located inside containment. The structure is made of reinforced concrete strong enough to withstand the impact of an aircraft crash. Tornado hazards have not been explicitly considered but should be enveloped by the aircraft crash design basis. The whole containment is provided with a steel liner to ensure leaktightness.

The concrete reactor vessel, with a steel liner and the upper containment design along with some aspects of the design basis are considered unique features for this reactor design. These areas were the major focus of our limited review on this reactor. The basis for identifying these as unique features is discussed further in subsequent sections.

2.7 DOE/GE Power Reactor Innovative Small Module (PRISM)

The PRISM reactor design is being developed by an industrial team led by General Electric Corporation and funded by the Department of Energy. Several United States National Laboratories are providing support in the areas of technology development and validation. PRISM is designed as a safe, reliable, and economically competitive liquid-sodium-cooled reactor power plant, with the following key features:

- Each compact reactor module is sized to enable factory fabrication and shipment to either inland or water-side sites.

- Passive reactivity reduction during undercooling and over-power transients with failure to scram.

- Passive decay heat removal for loss-of-heat-sink accidents.

- Protection against severe accidents by simple and passive safety features.

- Optional capability to use - as fissile material for startup - either plutonium or actinide wastes from light-water reactor spent fuel.

- Flexibility of core design to use either the reference metal fuel cycle or, alternatively, an oxide fuel cycle.

An attractive feature of the metal fuel cycle is the capability to recycle high-level, long half-life actinides back into the reactor core. This capability of PRISM to fission its own actinides can be extended to use as startup fuel actinides produced in other reactors. Thus PRISM offers an attractive waste management benefit. Additional benefits include the low operational pressure of the reactor coolant system and good heat transfer characteristics of liquid sodium.

The reactor module is about 20 ft in diameter and has a shipping weight of about 800 tons, not including removable internal components shipped separately. Reactor and containment vessels have no penetrations below the top head. The reactor is a pool design, with primary sodium recirculated within the reactor vessel by four submerged, self-cooled electromagnetic pumps. Two Intermediate Heat Exchangers (IHX) transfer heat to the Intermediate Heat Transport System (IHTS) sodium, which in turn transfers heat to the steam generator to produce steam. The tall, slender reactor geometry enhances uniformity and stability of internal flow distribution and natural circulation for shutdown heat removal.

The reactor and its safety-related systems are seismically isolated horizontally by an array of seismic bearings made of alternating layers of steel and natural rubber. The designers claim that vertical isolation is not required because the structure is stiff (rigid) in the vertical direction. PRISM has been designed with passive heat removal and reactivity shutdown features that bring the reactor to a safe, stable state in the unlikely event of failure of active systems. The primary system boundary consists of the reactor vessel, seal-welded reactor head closure, associated isolation valves, control rod drive housings, instrument drywells, and the tube surfaces of the IHXs.

The containment boundary completely surrounds the primary system and consists of the containment vessel that surrounds the reactor vessel and the upper containment dome that encloses the head closure. There are no penetrations in the containment vessel. Isolation valves or air locks are provided on all containment dome penetrations.

Other unique design features include high temperature issues for the concrete and steel containment and the containment seismic isolation system. These items were the focus of the limited review. The potential use of modular construction is not discussed in sufficient detail to permit any significant evaluation under this review program.

2.8 Atomic Energy of Canada Limited (CANDU-3U)

The CANDU-3U pressurized heavy-water reactor is the latest evolution in CANDU designs. Entirely reengineered from current CANDU 600 MWe designs, it is a compact 450-MWe power plant. The proposed use of extensive modular construction and prefabrication is a highly unique aspect of this reactor design.

The CANDU-3U has two steam generators and 232 fuel channels. The station uses only one fueling machine involving a simpler, single-ended on-line refueling system. The plant has a design life in excess of 40 years for the reactor building structures and the calandria/shield tank assembly and other key components. The unit uses modular construction - the layout enables the plant to be built either by conventional methods or using shop-assembled modules.

The CANDU-3U based on existing documentation is being designed and constructed to CSA (Canadian standards) which are not directly applicable to this review program. This fact plus the de-emphasis currently placed on this reactor design limited the review effort to a few high level aspects of this design.

2.9 Electric Power Industry Utility Requirements Document (EPRI-URD)

The United States utilities are leading an industry wide effort to establish the technical foundation for the design of the Advanced Light Water Reactor (ALWR). The cornerstone of the ALWR Program is a set of utility design requirements which are contained in the ALWR Utilities Requirements Document.

The purpose of the Requirements Document is to present a clear, complete statement of utility desires for their next generation of nuclear plants. The Requirements Document consists of a comprehensive set of design requirements for future LWRs. The requirements are grounded in proven technology of 30 years of commercial United States and international LWR experience.

The anticipated uses of the Requirements Document are threefold:

- Establish a stabilized regulatory basis for future LWRs which includes the USNRC's agreement on resolution of outstanding licensing issues and severe accident issues, and which provides high assurance of licensibility;

- Provide a set of design requirements for a standardized plant which are reflected in individual reactor and plant supplier certification designs;

- Provide a set of technical requirements which are suitable for use in an ALWR investor bid package for eventual detailed design, licensing and construction, and which provide a basis for strong investor confidence that the financial risks associated with the initial investment to complete and operate the first ALWR are minimal.

The Requirements Document covers the entire plant up to the grid interface. It therefore is the basis for an integrated plant design, i.e., nuclear steam supply system and balance of plant, and it emphasizes those areas which are most important to the objective of achieving an ALWR which is excellent with respect to safety, performance, constructibility, and economics. The document applies to both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs).

Accordingly, implementation scenarios for the Evolutionary and Passive ALWRs have been developed. Though uncertainties still exist at this point, these scenarios are plausible enough to provide reasonable understanding of the relationships noted above.

This document is somewhat of a general design criteria document. The majority of the suggested changes are in the systems and control systems area. The use of modular construction and design simplification is discussed. The review of this document was limited and focused on the design aspects currently being implemented in the ALWR plant designs which are reviewed under this program.

3.0 Phase I Review Effort

3.1 The Phase I Review Effort

The Phase I Review Effort consisted of the review of the currently proposed ALWR and advanced reactor design bases including USNRC Requirements for these reactors and the review of industry consensus codes and standards which would be applicable to the design of these reactors. Sections 1.4.3 through 1.4.6 list and provide reference to the documents used in this review effort. Due to the significant quantity of vendor and USNRC input information reviewed, a two pass review process is used. The first pass review which is conducted by one senior investigator focused on identifying the following items:

(a) Seismic Category I, Safety Class Structures for each of the reactor designs which were the subject of this review.

(b) Sections of Vendor Documents such as Standard Safetv Analysis Reports (SSAR), Safety Evaluation Reports (SER), etc., directly applicable to this program which would require further indepth review.

(c) Industry consensus codes and standards applicable to the various aspects of a given ALWR or Advanced Reactor design.

(d) Correlation of the information in (a), (b), and (c) above and development of concise information packages for each specific reactor type for review and evaluation by various cognizant program team investigators.

The second pass review is conducted by an assigned investigator(s) with expertise in the subject design or construction area and the industry consensus code and standard related to this design and construction area. The second pass review was an in-depth review of all aspects of the structural design codes which are the subject of this review including (1) vendor design input, (2) USNRC requirements, and (3) miscellaneous data, etc. Further, as part of the Phase I review the investigator assigned to review specific sections of the vendor design documentation and USNRC criteria is also assigned to review the industry consensus code or standards which had applicability to specific reactor design sections which the investigator reviews.

While other supporting reports and documents are used, the major sources of information about the various reactor model design basis and criteria were the Standard Safety Analysis Reports or Preliminary Safety Information Documents (PSID) and the corresponding USNRC Safety Evaluation Report (if they exist) of such reports. All these document types, the Standard Safety Analysis Reports or Preliminary Safety Information Document and the Safety Evaluation Reports in general follow the format and content of the Standard Review Plan (SRP), (NUREG-0800). Therefore in developing the initial scoping review effort the USNRC SRP is first reviewed on a detailed scoping basis to identify (1) those sections of the SRP which are directly applicable to this review, (2) those sections of the SRP which have some limited applicability to this review, (3) the industry consensus codes and standards applicable to that section of the SRP, and (4) the USNRC Regulatory Guides applicable to the subject SRP section. The results of this review and categorization effort are shown in Table 3.1. This table is then used as a guideline and the basis for conducting the initial scoping reviews of the remaining design basis documents. Further, it provided a basis for the identification of appropriate sections of the USNRC Standard Review Plan and Regulatory Guidelines, which required in depth review.

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
2.2 [2.2.1-2.2.3]	Hazards/Accidents	Secondary	Primarily Input Definition for Use in Sections 3.3 to 3.11	None	•ANS 2.12	Reg. Guide 1.91 Reg. Guide 1.70
2.3 [2.3.1 only]	Climatology	Primary	Input definition for use in Section 3.4 and 3.5 (Snow, Ice, Wind, Tornado) Affects subject standards.	None	•ASCE 7-93 •ANS 2.3 •ANS 2.5	Reg. Guide 1.76 Reg. Guide 1.117 RS 705-4
2.4 [2.4.1-2.4.10]	Hydrology	Cursory	General Hydrology Definitions	None	•ANS 2.8	Reg. Guide 1.27 Reg. Guide 1.29 Reg. Guide 1.59 Reg. Guide 1.102
2.5 [2.5.1-2.5.3]	Geology/Seismic	Secondary	Seismic Input Ground Motion Definition	None	•ANS 2.11	Reg. Guide 1.12 Reg. Guide 1.132 Reg. Guide 1.138
3.2.1	Seismic Classification	Primary	Seismic Design	None	None	Reg. Guide 1.29
3.2.2	Quality Group Classification	Secondary	Primary Equipment Related	None	•ASME III,-NE,-NF •AISC N690 •ASME IX	Reg. Guide 1.26 Reg. Guide 1.28
3.3.1	Wind Loads	Primary	Wind Design	None	•ASCE 7-93 •ASCE Paper 3269 •ASCE Paper 4933 •ANS 58.1	None

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
3.3.2	Tornado Loads	Primary	Tornado Design	None	•ASCE 7-93 •ASCE Paper 3269 •ANS 2.3 •ANS 58.1	Reg. Guide 1.76 Reg. Guide 1.117
3.4.1 [3.4.1-3.5.1.6]	Flood Protection	Secondary	Not a Major Structural Concern	None	•ANS 2.8	Reg. Guide 1.59 Reg. Guide 1.102
3.5.1 [3.4.1-3.4.2]	Missile/Aircraft	Primary	Missile and Aircraft Design Basis (Input to Demand Criteria)	None	•ANS 58.1 •ANS 58.2 •ANS 58.3 •ASCE 7-93 •Stanford Paper •BRL Paper •NDRC Paper •ASCE Impact and Impulse Com. Rpt.	Reg. Guide 1.115 Reg. Guide 1.117 Reg. Guide 1.76 Reg. Guide 1.91
3.5.2	SSC Missile protection Identification	Secondary	Missile Target Identification (Input to Barrier Design)	SRP 3.5.3	•ANS 58.1 •ANS 58.2 •ANS 58.3 •ASCE 7-93 •Stanford Paper •BRL Paper •NDRC Paper •ASCE Impact and Impulse Com. Rpt.	Reg. Guide 1.27 Reg. Guide 1.115 Reg. Guide 1.117

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)						
SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
3.5.3	Barrier Design Procedures	Primary	Structural Barriers Design Criteria	None	•ANS 58.1 •ASCE 7-93 •Stanford Paper •BRL Paper •NDRC Paper •ASCE Impact and Impulse Com Rpt. •ACI 349 App. C •AISC N690	Reg. Guide 1.76
3.6.1	Pipe Break - OC[(1)(2)] Only •Pipe Supports •Protective Structures	Cursory	Primarily Break Location Criteria (only as it effects support design)	SRP 3.8.4 SRP 3.6.2	•ANS 58.2	None
3.6.2	Pipe Break - GEN[(3)] Only •Pipe Supports •Whip Restraints •Protective Structures	Cursory	Primarily Break Location Criteria	SRP 3.8.3 SRP 3.8.4	•ANS 58.2	None
3.7.1	Seismic Design Parameters	Primary	Primary Seismic Demand Criteria	None	•ASCE 4-86	Reg. Guide 1.60 Reg. Guide 1.61
3.7.2	Seismic System Analysis	Primary	Primary Seismic Demand Criteria	None	•ASCE 4-86	Reg. Guide 1.92 Reg. Guide 1.122
3.7.3	Seismic Subsystem Analysis	Primary	As it Relates to Distribution System Supports.	SRP 3.7.2	•ASCE 4-86	Reg. Guide 1.122

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(1) Outside Containment Criteria

(2) The criteria was reviewed only to evaluate its impact on piping (distribution) system supports and pipe whip restraints.

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
3.8.1	Concrete Containment	Primary	Structural Design Criteria	None	•ASME III-CC •ASME IX •AWS D1.4	Reg. Guide 1.35 Reg. Guide 1.90 Reg. Guide 1.94 Reg. Guide 1.107 Reg. Guide 1.136
3.8.2	Steel Containment	Primary	Structural Design Criteria	None	•ASME III-NE •ASME IX	Reg. Guide 1.57
3.8.3	Concrete and/or Steel Structures Inside Containment	Primary	Structural Design Criteria	SRP 3.7.3 SRP 3.7.2	•ASCE 4-86 •ACI-349 •AISC N690 •AISC-ASD •AWS D1.1 •AWS D1.4	Reg. Guide 1.57 Reg. Guide 1.94 Reg. Guide 1.142
3.8.4	Other Seismic Category I Safety Class Concrete and/or Steel Structures	Primary	Structural Design Criteria	SRP 3.8.3 SRP 3.6.1 SRP 3.7	•ASCE 4-86 •ACI-349 •AISC N690 •AISC ASD •ACI-531(530) •AWS D1.1 •AWS D1.4	Reg. Guide 1.69 Reg. Guide 1.91 Reg. Guide 1.94 Reg. Guide 1.115 Reg. Guide 1.117 Reg. Guide 1.124 Reg. Guide 1.142 Reg. Guide 1.143 Reg. Guide 1.29 Reg. Guide 1.60 Reg. Guide 1.61 Reg. Guide 1.76 Reg. Guide 1.92
3.8.5	Foundations	Primary	Structural Design Criteria	SRP 3.8.4 SRP 3.2.1 SRP 3.2.2 SRP 3.6.1	•ASCE 4-86 •ACI-349 •AISC N690 •AWS D1.1 •AWS D1.4	Reg. Guide 1.132 Reg. Guide 1.136 Reg. Guide 1.142

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
3.9.1	Mechanical Components -Supports	Secondary	Possible Distribution Support Design Issues Only	SRP 3.9.2	•AISC N690 •AISC-ASD •ASME III-NF •ASME IX •MSS-SP-58 •AWS D1.1 •AWS D1.3 •AWS D9.1 •AISI-CFSDM •SMACNA STDS. •ANSI/ASME AG-1 •IEEE-628 •ACI-349, App. B	Reg. Guide 1.100
3.9.2	Testing/Analysis of Systems/Components - Only Supports	Cursory	Possible Distribution Support Design Issues	SRP 3.7.3	None	Reg. Guide 1.61 Reg. Guide 1.92
3.9.3	ASME Class 1,2,3 Components, Supports, Core Supports (Only Component Supports)	Primary	ASME Safety Class Distribution System Support Designs	SRP 3.6.2 SRP 3.9.2 SRP 3.10	•AISC N690 •AISC-ASD •ASME III-NF •ASME IX •MSS-SP-58 •AWS D1.1 •AWS D1.3 •AISI-CFSDM •SMACNA STDS. •ANSI/ASME AG-1 •IEEE-628 •ACI-349, App. B	Reg. Guide 1.124 Reg. Guide 1.130
3.10	Seismic and Dynamic Qualification of Equipment - Only Supports	Cursory	Possible Support Issues Only	None	•IEEE 344 (out of scope of this review)	None

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
5.2.1	Compliance with Codes/Standards	Primary	Defined Code and Standard Compliance Requirement	None	•All	Reg. Guide 1.26 Reg. Guide 1.84 Reg. Guide 1.85 Reg. Guide 1.147
5.4.14	RCS Component Supports	Cursory	ASME Safety Class 1 Support Design	SRP 3.6 SRP 3.7 SRP 3.9 SRP 3.10	•ASME III-NF •ASME IX	Reg. Guide 1.124 Reg. Guide 1.130
6.2.4	Containment Isolation System	Cursory	Isolation of Fluid Systems which penetrate the containment boundary	SRP 3.2.1 SRP 3.2.2 SRP 3.8 SRP 3.9 SRP 16.0	•ANSI/ANS 56.2	Reg. Guide 1.141
6.2.6	Containment Leakage Testing	Primary	Containment Leakage Testing Program (Type A, B & C Tests)	SRP 3.2	•ANSI/ANS N45.4 •ANSI/ANS 56.8	MS 021-5
9.1 [9.1.1-9.1.5]	Fuel Storage and Handling System	Primary	Design Requirements Spent and New Fuel Structures	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASCE 4-86 •ASME III-NF •ASME IX •AISC N690 •AWS D1.1 •AWS D1.3	Reg. Guide 1.29 Reg. Guide 1.13 Reg. Guide 1.115 Reg. Guide 1.117

Table 3.1 – Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)						
SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
9.2 [9.2.1-9.2.6]	Auxiliary Systems	Cursory	Uses Structural Design Standards for Distribution System Supports	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •MSS-SP-58 •AISC N690 •AISC-ASD •AISC-CFSDM •SMACNA Standards •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	Reg. Guide 1.29
9.3 [9.3.1, 9.3.3, 9.3.4, 9.3.5]	Auxiliary Systems	Cursory	Uses Structural Design Standards for Distribution System Supports	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •MSS-SP-58 •AISC N690 •AISC-ASD •AISC-CFSDM •SMACNA Standards •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	Reg. Guide 1.29

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
9.4 [9.4.1-9.4.5]	Auxiliary Systems	Cursory	Uses Structural Design Standards for Distribution System Supports	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •AISC N690 •AISC-ASD •AISC-CFSDM •SMACNA Standards •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	Reg. 1.29
9.5.1	Fire Protection Program	Primary Secondary	As it Relates to Structures/Buildings, Distribution System Supports, and Fire Barriers	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.8.4 SRP 3.8.5	•NFPA Standards •AISC N690 •AISC-CFSDM •AWS D1.1 •AWS D1.3 •AWS D9.1	None
9.5.5 9.5.6 9.5.7	Emergency Diesel Generator	Cursory	Uses Structural Design Standards for Distribution Systems	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •AISC N690 •AISC-ASD •AISC-CFSDM •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	None

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)						
SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
10.3	Main Steam Supply System	Cursory	Uses Structural Design Standards for Distribution Systems	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •AISC N690 •AISC-ASD •AISC-CFSDM •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	Reg. Guide 1.26 Reg. Guide 1.29 Reg. Guide 1.115 Reg. Guide 1.117
10.4.7	Condensate and Feedwater System	Cursory	Uses Structural Design Standards for Distribution Systems	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •MSS-SP-58 •AISC N690 •AISC-ASD •AWS D1.1 •AWS D1.3 •ACI 349, App. B	Reg. Guide 1.29
10.4.8	Steam Generator Blowdown System	Cursory	Uses Structural Design Standards for Distribution Systems	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •MSS-SP-58 •AISC N690 •AISC-ASD •AWS D1.1 •AWS D1.3 •ACI 349, App. B	Reg. Guide 1.26 Reg. Guide 1.29 Reg. Guide 1.143

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)						
SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
10.4.9	Auxiliary Feedwater System	Cursory	Uses Structural Design Standards for Distribution System	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •MSS-SP-58 •AISC N690 •AISC-ASD •AWS D1.1 •AWS D1.3 •ACI 349, App. B	Reg. Guide 1.29
11.2	Liquid Waste Management Systems	Cursory	May Use Structural Codes and Standards, but not Specifically in Scope for this Review	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •MSS-SP-58 •AISC N690 •AISC-ASD •AWS D1.1 •AWS D1.3 •ACI 349, App. B	Reg. Guide 1.143
11.3	Gaseous Waste Management Systems	Cursory	May Use Structural Codes and Standards, but not Specifically in Scope for this Review	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •AISC N690 •AISC-ASD •AISC-CFSDM •SMACNA Standards •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	Reg. Guide 1.143

SRP Section	Topic or Title	Review Level		Reference Primary Section	Subject Industry Consensus Codes and Standards	Reference Regulatory Guide (Reg. Guide)
		Type	Basis			
11.4	Solid Waste Management Systems	Cursory	May Use Structural Codes and Standards, but not Specifically in Scope for this Review	SRP 3.3.1 SRP 3.3.2 SRP 3.5.3 SRP 3.7.1 SRP 3.7.4 SRP 3.8.4 SRP 3.8.5	•ASME III-NF •ASME IX •AISC N690 •AISC-ASD •AISC-CFSDM •SMACNA Standards •IEEE-628 •AWS D1.1 •AWS D1.3 •AWS D9.1 •ACI 349, App. B	Reg. Guide 1.143
15.1.5	Steam System Piping Failures	Secondary	Input Load Definition	SRP 3.6 SRP 3.9	•All	N/I[4]
15.2.8	Feedwater Piping Systems Failures	Secondary	Input Load Definition	SRP 3.9	•All	N/I[4]
15.6.5	LOCA Events	Secondary	Input Load Definition	SRP 3.6 SRP 3.9	•All	N/I[4]
15.8	ATWS	Secondary	Possible Input Load Definition	SRP 3.6 SRP 3.9	•All	N/I[4]

Table 3.1 - Results of the USNRC Standard Review Plan (NUREG-0800) Scoping Review (continued)

Footnote for Table 3.1:

[4] N/I = None Identified

3.2 Identification of Seismic Category I Safety Class Structures

After the identification of the applicable Standard Safety Analysis Report Sections, etc., the next step in the review is the identification and tabulation of the Seismic Category I, Safety Class Structures for each of the evolutionary and advanced power reactor designs which are the subject of this study. In addition, if available on the vendor documentation the industry codes and standards applicable to the design of each of the Seismic Category I, Safety Class Structures are also identified. This provides a definitive list of the structures which required in depth review for uniqueness in design, analysis, construction, etc. The results of this review are provided in Tables 3.2.1 thru 3.2.9. For each reactor design one table is provided listing the structure type, seismic category, safety class and the codes and standards listed as being the applicable by each reactor design vendor to the design of the subject structure. These data are obtained from the industry documents identified in Section 1.4.3. These tables (3.2.1 through 2.9) in conjunction with Table 3.1 are then used as part of the input and the basis of the in depth review effort discussed in the following sections. Further it clearly defines for each of the evolutionary and advanced power reactor designs what structures are the subject of this review program.

If no specific industry consensus code and standard information is provided in the vendor documentation, the applicable industry consensus code and standards is identified as N/D (Not Defined). Also in many cases due to the preliminary nature of the design information, a safety classification is not available or provided. In these cases as with the industry codes and standards the Safety Classification is identified as N/D. If specific design data or information could not be located in the design basis documentation then it is identified by N/I in the tables which signifies none identified.

Table 3.2.1 - Safety Class, Seismic Category Structures for Westinghouse AP600[7]					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Nuclear Island Basemat (Foundations)	Reinforced Concrete Basemat	I	N/D[2]	ACI-349 ACI-301	Yes
Nuclear Island Containment Interior	Reinforced Concrete, Composite Construction, and Structural Steel	I	N/D[2]	AISC N690 ACI-349 AWS D1.1 AWS D1.4	Yes
Nuclear Island Shield Building	Reinforced Concrete with a Conical Roof Supported on a Basemat	I	N/D[2]	ACI-349 ACI-301	Yes
Nuclear Island Auxiliary Building	Typical of Existing Operating Plants	I	N/D[2]	AISC N690 ACI-349 AWS D1.1	Yes
Nuclear Island Containment Air Baffle	Baffles are Parallel Gauge Galvanized Sheet Plate Attached to the Containment Shell	I	N/D[2]	AISC N690 AWS D1.1	Yes
Nuclear Island Containment Vessel -Structure	Metal (Steel) Containment Vessel with Ellipsoidal Top and Base Heads	I	SC-2	ASME-III-NE ASME-IX ASME CC N-284	Yes
Turbine Building	Steel Column and Beam Structure on Reinforced Concrete Slabs	NS[1]	N/D[2]	Commercial STDS	No
Annex Building I	Steel Column and Beam Structure on Reinforced Concrete Slabs	NS[1]	N/D[2]	Commercial STDS	No
Annex Building II	Steel Column and Beam Structure on Reinforced Concrete Slabs	NS[1]	N/D[2]	Commercial STDS	No
RCL & RPV Supports	Supports the Reactor Coolant Loop and the Reactor Pressure Vessel	I	SC-1	ASME-III-NF ACI-349 ASME-IX	Yes

Table 3.2.1 - Safety Class, Seismic Category Structures for Westinghouse AP600[7] (continued)					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Cable Tray Supports	Seismic Category I Cable Tray Supports (Linear Structural Steel)	I	SC-1	AISI-CFSDM AISC N690 AWS D1.3 AWS D1.1 ACI 349, Appendix B[6]	Yes
HVAC Supports	Seismic Category I HVAC Supports (Linear Structural Steel)	I	SC-3	AISC N690 AWS D1.1 AWS D1.3 ACI 349, Appendix B[6]	Yes
Piping Supports	Safety Class System (Linear Structural Steel)	I	SC-1,2,3	ASME-III-NF AISC N690 ASME-IX ACI 349, Appendix B[6]	Yes
	Non-safety Class Systems	NS	N/S[4]	B31.1 ASME-IX	No
Fire Protection Systems/Supports	Fire Barriers	II	N/S[4]	NFPA-803	No
	Fire Protection Piping Supports	I/II	N/S[4]	NFPA-13 NFPA-14 ANSI B31.1 ASME-IX	Yes[5]
Fuel Storage Racks	Free Standing Fuel Racks	I	SC-3	ASME-III-NF	Yes
Containment Leak Testing	Periodic Leak Testing System	N/A[3]	N/S	ANS 56.8	Yes
Missile Barriers	Safety Class Missile Barriers	II	N/S	For Steel: BRL or Stanford Formula For Concrete: Modified NDRC	Yes

Footnotes for Table 3.2.1:

[1] NS = Non Seismic Category
[2] N/D = Not Defined
[3] N/A = Not Applicable
[4] N/S = Non Safety Class
[5] Safety Class Portion of the Systems
[6] Embedded portions of supports and concrete expansion anchors
[7] The information presented in this table is based on Revision 2 of the SSAR for the AP600.

Table 3.2.2 - Safety Class, Seismic Category Structures for the ABB/CE-System 80+					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Reactor Building Internal Structures	Reinforced Concrete Founded on Base Slab over Containment Steel	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes
Reactor Building Shield Building	Cylindrical Concrete Shear Wall Structure	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes
Reactor Building Containment	Spherical Steel Structure	I	SC-2	ASME III-NE ASME-IX ASCE 4-86(1)	Yes
Reactor Building Equipment Hatch	Metal	I	SC-2	ASME III-NE ASME-IX	Yes
Reactor Building Personnel Airlocks	Metal	I	SC-2	ASME III-NE ASME-IX	Yes
Reactor Building Subsphere	Portion of Reactor Building below 91'9"	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes
Nuclear Annex Control Area	Monolithic Attachment to Shield Building	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes
Nuclear Annex Spent Fuel Pool	Steel Lined Integral Part of the Nuclear Annex	I	SC-3	ACI-349	Yes
Nuclear Annex Fuel Racks	High Density Stainless Steel	I	SC-3	N/D	Yes
Nuclear Annex Valve House Areas	Integral with Nuclear Annex	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes
Nuclear Annex Emergency Diesel Generator Areas	Integral with Nuclear Annex	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes

Table 3.2.2 - Safety Class, Seismic Category Structures for the ABB/CE-System 80+ (continued)					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Nuclear Annex CVCS/Maintain-ance Area	Integral with Nuclear Annex	I	SC-3	ACI-349 AISC N690 AWS D1.1 ASCE 4-86(1)	Yes
Foundations	Reinforced Concrete	I	N/D	ACI-349 ASCE 4-86(1)	Yes
Station Service Water Pump Structure	Reinforced Concrete Structure on a Mat Foundation	I	SC-3	ACI-349 ASCE 4-86(1)	Yes
Component Cooling Water Heat Exchanger Structure	Reinforced Concrete	I	SC-3	ACI-349 ASCE 4-86(1)	Yes
Diesel Fuel Storage Structure	Reinforced Concrete	I	SC-3	ACI-349 ASCE 4-86(1)	Yes
Turbine Building		II	NNS(3)	Commercial STDS	No
Radwaste Building		II	NNS(3)	Commercial STDS	No
Dike - Main	Earthen Structure	II	NNS(3)	Commercial STDS	No
Dike - Aux.	Earthen Structure	II	NNS(3)	Commercial STDS	No
Piping Supports	Safety Class Systems (Linear Structural Steel)	I	SC-1,2,3	ASME-III-NF(6) ASME-IX ANSI B31.1(7) ASCE 4-86(1) ACI-349, App. B(8)	Yes
	Non-Safety Class Systems	II	NNS(3)	ANSI B31.1(7) ASME-IX ASME 4-86(1)	No
Safety Class HVAC Supports	Seismic Category I Supports, (Linear Structural Steel)	I	SC-3	AISC-N690(9) AISC-ASD AISI-CFSDM AWS D1.1 AWS D1.3	Yes

Table 3.2.2 - Safety Class, Seismic Category Structures for the ABB/CE-System 80+ (continued)					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Safety Class Cable Tray Supports	Seismic Category I Supports, (Linear Structural Steel)	I	SC-3	AISC-N690[9] AISC-ASD AISI-CFSDM AWS D1.1 AWS D1.3	Yes
Fire Protection Structures and Distribution Systems	Barriers/Doors	II	NNS[3]	NFPA-803	No
	Piping Supports[10]	I/II	SC-2 NNS[3]	NFPA-13 ANSI B31.1	Yes[5]
	Hose/Standpipe[10] Systems	I/II	SC-2 NNS[3]	NFPA-14 ANSI B31.1	Yes[5]
Containment Leak Testing	Periodic Leak Testing System	N/A[2]	N/A	ANS 56.8	Yes

Footnotes for Table 3.2.2:

(1) ASCE 4-86 is listed as a reference for the seismic design section of the SSAR (Section 3.7). However it is not specifically referenced in the text of Section 3.7. Instead the actual analysis methodology employed in seismic analysis is described in detail.

(2) N/A = Not Applicable

(3) NNS = Non-nuclear Safety

(4) Intentionally Left Blank

(5) These systems which are Seismic Category I and/or Safety Class.

(6) Supplemented by AWS D1.1 for A500 Grade B Tube Steel.

(7) Specified for minimum spacing requirements.

(8) Including amendments and requirements to meet I&E 79-02 criteria. Note that ACI-349 is specified for Seismic Category I Structures in general and piping supports in particular but not for other distribution systems supports.

(9) As amended by Section 3.8.4.5 of the SSAR.

(10) Seismic Category I, Safety Class systems are in the Reactor Building and the Control Building.

(11) N/D = Not Defined.

Table 3.2.3 - Safety Class, Seismic Structures for the GE ABWR					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Primary Containment Vessel	Reinforced Concrete Structure	I	SC-2	ASME III-CC	Yes
Vent System	Steel Shell Construction (The Stack)	NS	NNS(1)	Commercial STDS	No
RCCV Penetrations and Drywell Head	Primary Containment Vessel Penetrations and Drywell Head	I	SC-2	ASME III-NE ASME-IX AISC-SCM	Yes
RPV Stabilizer Truss	Steel Frame Structure	I	SC-3	AISC N690 AWS D1.1	Yes
Diaphragm Floor	Reinforced Concrete Slab	I	SC-3	ACI-349	Yes
Lower Drywell Equipment and Personnel Tunnels	Steel Structure Carbon and Stainless	I	SC-2	AISC N690 AWS D1.1	Yes
RPV Pedestal and Shield Wall	Composite Steel and Concrete Construction of Two Concentric Cylinders	I	SC-3	AISC N690 AWS D1.1 ACI-301	Yes
Foundation Work	Basemats (2) - One under the Reactor Building and one under the Control Building	I	SC-3	ACI-359(4) ACI-349(4)	Yes
Reactor Building	Unlined Reinforced Concrete - Similar to Current Layouts	I	SC-3	ACI-349 AISC N690 AWS D1.1	Yes
Control Building	Reinforced Concrete with Steel Roof	I	SC-3	ACI-349 AISC N690 AWS D1.1	Yes
Radwaste Building	Reinforced Concrete Structure	I	SC-3	ACI-349 AISC N690 AWS D1.1	Yes
Containment Internal Steel	Structural Steel	I	SC-3	AISC N690 AWS D1.1	Yes
DEPPS	Drywell Equipment and Pipe Support Structure	I	SC-3	AISC N690 AWS D1.1	Yes

Table 3.2.3 - Safety Class, Seismic Structures for the GE ABWR (continued)					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Containment Internal Concrete	Reinforced Concrete	I	SC-2	ACI-349	Yes
Safety Class Conduit, Cable Trays, and Conduit and Cable Tray Supports	Distribution System Supports (Linear Structural Steel)	I	SC-3	AISC N690 AISI-CFSDM AWS D1.1 AWS D1.3 NEMA	Yes
Piping Supports	Safety Class Distribution System Supports (Linear Structural Steel)	I	SC-1,2,3	ASME III-NF(7) AISC-ASD(8) ASME-IX AWS D1.1 AWS D1.3	Yes
	Non Safety Class Distribution Systems Supports	II; NS(6)	NNS;N/S(1),(5)	ANSI B31.1 ASME IX	No
Safety Class HVAC Supports	Distribution System Supports (Linear Structural Steel)	I	SC-3	AISC N690 ASME AG-1 AWS D1.1 AWS D1.3	Yes
Fire Protection	Hose, Stand Pipe, and Sprinkler System Supports Within the Containment Boundary	I	SC-2	ANSI B31.1 NFPA-13 NFPA-14	Yes
	Barriers	II	NNS(1)	NFPA-803	Yes
Fuel Storage Racks	Free Standing Fuel Racks	I	NNS(1)	ASME III-NF ASME-IX	Yes
Containment Leak Testing	Periodic Leak Testing System	N/A(2)	N/S(5)	ANS 56.8	Yes

Footnotes for Table 3.2.3:

(1) NNS = Non-Nuclear Safety System

(2) N/A = Not Applicable

(3) Intentionally Left Blank

(4) ACI-359 is used for the containment building foundation and ACI -349 is used for all other safety class building foundations.

(5) N/S = Non Safety Class System

(6) NS = Non Seismic

(7) Augmented by ASME Code Case N-476, Supplement 89.1 and "Torsional Analysis of Steel Members", AISC Publication T1142/83. Also IE Bulletin 79-02.

(8) Specified for pipe supports using supplementary steel (building structure component supports).

Table 3.2.4 - Safety Class, Seismic Category Structures for the GE SBWR					
Structure	Description	Seismic Category	Safety Class	Applicable Code and Standard	Covered by this Review
RCCV	Reactor Building Containment Vessel (Reinforced Concrete with Steel Liner)	I	SC-2	ASME-III-CC ASCE 4-86	Yes
Reactor Building Structure	Reinforced Concrete and Steel Structure	I	SC-2 NS	ACI-349 AISC N690 AWS D1.1 AWS D1.4 ASCE 4-86 AISC ASD	Yes
Reactor Pedestal	Reinforced Concrete	I	SC-2	ACI-349 ASCE 4-86	Yes
Reactor Shield Wall	Structural Steel Shield Wall	I	SC-2	AISC N690 AWS D1.1 ASCE 7-93 ASCE 4-86	Yes
Basemat	Under RPV Pedestal Supports Entire Reactor Building	I	SC-2	ACI-349 ASCE 4-86	Yes
RVST	Reactor Vessel Stabilizer Truss	I	SC-2	AISC N690 AWS D1.1 ASCE 7-93 ASCE 4-86	Yes
DGPSS	Support Platforms/Steel For Piping, Equipment, etc.	I	SC-2	AISC N690 AWS D1.1 ASCE 7-93 ASCE 4-86	Yes
Drywell Airlocks	Upper and Lower Steel	I	SC-2	ASME-III-NE ASME-IX ASCE 4-86	Yes
Drywell Head	Steel	I	SC-2	ASME-III-NE ASME-IX ASCE 4-86	Yes
Diaphragm Floor	Barrier between Drywell and Suppression Chamber (Modular Construction)	I	SC-2	AISC N690 ASCE 7-93 ACI-349 AWS D1.1 AWS D1.4 ASCE 4-86	Yes

Table 3.2.4 - Safety Class, Seismic Category Structures for the GE SBWR (continued)					
Structure	Description	Seismic Category	Safety Class	Applicable Code and Standard	Covered by this Review
GDCS Pools	Gravity Driven Cooling System Pools - Structural Steel Square Tanks	I	SC-2	AISC N690 AWS D1.1 ASCE 7-93 ASCE 4-86 AISC-ASD	Yes
PCCS	Passive Containment Cooling System	I	SC-2	N/D(3)	Yes
CAT I HVAC	Category I HVAC Distribution System Supports (Linear Structural Steel)	I	SC-2	ASCE 4-86 N/D(3)	Yes
CAT I Cable Trays	Category I Cable Tray Supports (Linear Structural Steel)	I	SC-2	ASCE 4-86 N/D(3)	Yes
Fire Protection	Barriers	N/D(3)	N/D(3)	NFPA-803	Yes
	Fire Protection Piping System Supports	N/D(3)	N/D(3)	N/D(3)	Yes
Piping System Supports	Safety Class Piping Supports - Linear Structural Steel	I	SC-1,2,3	ASME-III-NF ASME-IX AISC N690 AWS D1.1	Yes
	Non Safety Class Piping Supports	II, NS(4)	N/S(1)	ANSI B31.1 AISC-ASD	No
Fuel Storage Racks	Spent Fuel Storage Racks	I	N/S(1)	ASME-III-NF ASME-IX	Yes
Containment Leak Test	Periodic Leak Testing System	N/A(2)	N/S(1)	ANS 56.8 ANSI N45.4	Yes

Footnotes for Table 3.2.4:

(1) N/S = Non Safety Class System
(2) N/A = Not Applicable
(3) N/D = Not Defined in Design Basis Documentation
(4) NS = Non Seismic

Table 3.2.5 - Safety Class, Seismic Category Structures for DOE/GA - MHTGR					
Structure	Description	Seismic Category(2),(5)	Safety Class(1)	Applicable Codes and Standard	Covered by This Review
Reactor Vessel Support	Structural Steel	SDR(5)	S	ASME-III-NF ASME-IX	Yes
Reactor Building(3)	Reinforced Concrete Cylinder with a Flat Concrete Slab Base and Top	SDR(5)	S	ACI-349 AISC-ASD AWS D1.1	Yes
Reactor Service Building	Reinforced Concrete and Structural Steel	UBC(6) or SI(2)	N/S(7)	ACI-349 AISC-ASD AWS D1.1 AWS D1.4	Yes
Reactor Auxiliary Building	Reinforced Concrete and Structural Steel	SDR(5)	S	ACI-349 AISC-ASD AWS D1.1 AWS D1.4	Yes
Reactor Cavity Cooling Panels	Intake/Exhaust Structures	SDR(5)	S	N/D(4)	Yes
Reactor Cavity Cooling Panels	Plenum Structures	SDR(5)	S	N/D(4)	Yes
Reactor Cavity Cooling Panels	Ducting Supports	SDR(5)	S	N/D(4)	Yes
Essential Uninterruptable Power System Supply - Supports	Cables Trays and Conduit Supports (Linear Structural Steel)	SDR(5)	S	N/D(4)	Yes
Essential DC Power System Supports	Cable Trays & Conduit Supports (Linear Structural Steel)	SDR(5)	S	N/D(4)	Yes
Fire Protection System	Piping Supports	UBC(6) or SI(2)	N/D(4)	N/D(4)	Yes
	Fire Barriers	N/D(4)	N/D(4)	N/D(4)	Yes
Piping Supports	Piping Distribution System Supports - Linear Structural Steel	SDR(5)	S	ASME III-NF	Yes
Containment Leak Testing	Confinement Structure verses Pressure Retaining Containment	N/A(8)	N/A(8)	N/D(4)	Yes

Footnotes For Table 3.2.5:

(1) MHTGR does not use a traditional Safety Classification System; Systems, Structures and Components are either classified as Safety Related (S) or Non-Safety Related (N/S).

(2) Seismic Category in the traditional definition is not used with the MHTGR, per Reference (6.4-[8]) all SC items are to be seismically designed. For non safety items they are seismically designed to the Uniform Building Code (UBC) Zone 2B. If however a non safety related SSC could be postulated to cause the failure in a safety related SSC then it will be designated as Safety Impact (SI) and a seismic interaction design will be required.

(3) MHTGR does not have a traditional (classical) containment vessel or structure.

(4) N/D = Not Defined

(5) SDR = Seismic Design Required, essentially a Seismic Category I component but not explicitly defined as such.

(6) UBC = Design to UBC per footnote (2).

(7) N/S = Non Safety

(8) N/A = Not Applicable

Table 3.2.6 - Safety Class/Seismic Category Structures for ABB-PIUS					
Structure	Description	Seismic Category(1)	Safety Classification	Applicable Codes and Standard(2)	Covered by This Review
Reactor Building	Contains Safety Related Structures (Prestressed Concrete Cylinder with Concrete Flat Bottom and Top)	I	SC-3	See Note (2)	Yes
Control Service Building	Equivalent of US Control Building	I	SC-3	See Note (2)	Yes
Concrete Vessel	Monolith (Reinforced Concrete)	I	SC-1	ASME III-CB	Yes
Concrete Vessel Liner	Steel	P	N/S(3)	ASME III-CB	Yes
Concrete Containment	Pressure Suppression Containment - Part of the Reactor Building	I	SC-2	ASME III-CC ASME III-NE See Note (2)	Yes
Reactor Supports	Safety Related RPV Supports	I	SC-2	See Note (2)	Yes
Fuel Racks	N/A(5)	P	N/S(3)	See Note (2)	No
Reactor Pool	N/A(5)	N/D(4)	SC-2	See Note (2)	Yes
Cooling Towers for Reactor Pool Coolers	N/A(5)	I	SC-3	See Note (2)	Yes
Control Building Ventilation System Supports	Not Classified as Safety Structures	N	SC-3	See Note (2)	No
Emergency Ventilation System Supports	Not Classified as Safety Structures	N	SC-3	See Note (2)	No
Piping System Supports	Seismic Category I Supports	I	SC- 2,3	See Note (2)	Yes
		P		See Note (2)	
		N	N/S(3)	See Note (2)	No
Cable Tray Supports	Seismic Category I Supports	I	SC-2 SC-3	See Note (2)	Yes
		P	N/S(3)	See Note (2)	No
Containment Leak Testing	Periodic Leak Testing System	N/A(5)	N/S(3)	10CFR 50, Appendix J	Yes

Footnotes for Table 3.2.6:

(1) PIUS uses 3 Seismic Classifications I, P, or N, where I is an active SSC, P is a structure for which only the Structural Integrity of SSC is maintained, N is a non-seismic SSC.

(2) For design, qualification, and construction of the PIUS reactor ABB normally uses Swedish and/or European Codes and Standards. Several times in the PSID it is stated the PIUS design will comply with the proper US industry codes and standards. However other than the certain items no specific US industry codes and standards are explicitly specified. It is stated this conversion to US industry codes and standards will be done during the detailed design stage.

(3) N/S = Non Safety

(4) N/D = Not Defined

(5) N/A = Not Applicable

Table 3.2.7 - Safety Class Seismic Category Structures for DOE/GE PRISM					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Reactor Vessel	Reactor Pressure Vessel	I	SC-1	N/A(2)	No
Containment Vessel	Steel	I	SC-1	ASME-III-NB ASME-IX	Yes
Reactor Closure Head	Steel	I	SC-1	ASME-III-NB ASME-IX	Yes
Reactor Building HAAC	Head Access Area Closure, Reinforced Concrete Structure	I	SC-3	ACI-349 AWS D1.4	Yes
Reactor Building Reactor Silo	Reactor Silo, Reinforced Concrete Structure	I	SC-3	AISC-ASD ACI-349 AWS D1.4 AWS D1.1	Yes
Reactor Building E&IV	Electrical & Instrument Vaults	I	SC-3	N/D(3)	No
Reactor Building PSP & SDTV	Primary Sodium Processing and Sodium Drain Tank Vaults	I	SC-3	ACI-349 AWS D1.4	Yes
Reactor Building RVACS - Duct Supports	Inlet & Outlet Duct Supports	I	SC-3	ANSI SMACNA	Yes
Reactor Building RVACS	Shielding Concrete	I	SC-3	ACI-349 AWS D1.4	Yes
Reactor Building Seismic Isolators	Seismic Isolators	I	SC-3	N/D(3)	Yes
Steam Generator Building	Steam Generator Building	NS(4)	N/S(1)	N/D(3)	No
Radioactive Waste Buildings	Ground Floor & Curbs - Reinforced Concrete Structures	I	SC-3	ACI-349	Yes
Mobile Refueling Enclosure	Wall & Roof Structural Steel Frame	I	SC-3	AISC-ASD AWS D1.1	Yes
SC Cable Trays	Cable Tray Supports	I	SC-1	N/D(3)	Yes
SC Piping Systems	Piping Supports - Linear Structural Steel	I	SC-1,2,3	ASME III-NF ASME-IX ASME CC N47	Yes
Fuel Storage Racks	No On-site Spent Fuel Storage Currently in the Design Basis	N/A(2)	N/A(2)	N/A(2)	No

Table 3.2.7 - Safety Class Seismic Category Structures for DOE/GE PRISM (continued)					
Structure	Description	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by this Review
Fire Protection Systems	Piping Supports	N/D(3)	N/D(3)	NFPA	Yes
	Fire Barriers	N/D(3)	N/D(3)	NFPA	Yes
Containment Leak Testing	No Specific Leak Testing; Relies on Negative Pressures to Prevent Leakage	N/I(5)	N/I(5)	N/I(5)	Yes

Footnotes for Table 3.2.7:

(1) N/S = Non Safety System or Structure
(2) N/A = Not Applicable
(3) N/D = Not Defined
(4) NS = Non Seismic Category Structure
(5) N/I = None Identified

Table 3.2.8 - Safety Class/Seismic Category Structures for the AECL CANDU-3U					
Structure	Description[5]	Seismic Category[2]	Safety Classification[1]	Applicable Code and Standard[3]	Covered by This Review
Special Internals	20800	S	S	See Note (3)	Yes
Reactor Building	21100	S	S	See Note (3)	Yes
Reactor Aux. Building	21200	S	S	See Note (3)	Yes
Turbine Building	22000	NS[7]	S	See Note (3)	No
Group 2 Pumphouse	This includes Pumphouse (23500), Intake Channel & Structures (23600) and Outfill Channel and structures (23700) Intake and Discharge Ducts, (23800) and Recirculation Structure(23900)	S	S	See Note (3)	Yes
Group 2 Building	24200	S	S	See Note (3)	Yes
Maintenance Building	25000	S	S	See Note (3)	Yes
Radioactive Waste Storage Structure	26700	NS[7]	S	See Note (3)	No[4]
Fuel Storage Structures	35200	S	S	See Note (3)	Yes
Safety Class Cable Trays Supports	57400	S	S	See Note (3)	Yes
Safety Class Piping System Supports	Safety Class Piping (Distribution System Supports)	S	S	See Note (3)	Yes
Reactor Building Ventilation System Supports	HVAC Supports 73120	S	S	See Note (3)	Yes
Irradiated Fuel Storage Bay Ventilation System-Supports	HVAC 73160	S	S	See Note (3)	Yes

Table 3.2.8 - Safety Class/Seismic Category Structures for the AECL CANDU-3U (continued)					
Structure	Description[5]	Seismic Category[2]	Safety Classification[1]	Applicable Code and Standard[3]	Covered by This Review
Group 2 Service Building	HVAC 73310 73320 7330	S	S	See Note (3)	Yes
Fire Protection Systems	Barriers	N/D[6]	N/D[6]	See Note (3)	No
	Supports 74200	S	S	See Note (3)	Yes
Containment Leak Testing	While not clearly defined appears to differ significantly from current US requirements.	N/D[6]	N/D[6]	See Note (3)	Yes

Footnotes for Table 3.2.8:

(1) At this point in the design, systems, structures and components are only identified as Safety or Non-safety, no detailed subclassification has yet been done. Therefore systems with S in the SC column identify safety related systems, structures, and components.

(2) At this point in the design detailed seismic classification has not been done. Safety systems which require a seismic design basis are identified as S; NS indicates non seismic.

(3) Currently all design codes are Canadian National Codes and Standards. Recent communications between the AECL and USNRC imply that the CANDU-3U intends to meet the intent of US industry codes and standards for Group 2 SSC. Group 2 SSC are those SSC designed to mitigate the effects of design basis accidents. The actual mechanism via which this will be accomplished is unclear at the time of this report.

(4) This structure appears to have a limited simplified seismic analysis therefore it was removed from scope.

(5) The numbers given represent the GSI numbers assigned in the conceptual Safety Report, Appendix D1, Volume 1.

(6) N/D = Not Defined

(7) NS = Non-Seismic

Table 3.2.9 - Safety Class/Seismic Category Structures for the EPRI URD					
Structure	Description[1]	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by This Review
Steel Containment	N/A[2]	I	SC-2	ASME III-NE ASME-IX ASCE 4-86	Yes
Concrete Containment	N/A[2]	I	SC-2	ASME III-CC ASCE 4-86	Yes
Building Foundations	N/A[2]	I	SC-2	ASME III-CC ACI 349 ASCE 4-86	Yes
	N/A[2]	I	SC-3	ACI 349 ASCE 4-86	Yes
	N/A[2]	II	NNS[4]	ACI-318 ASCE 4-86	No
Steel Super Structures	N/A[2]	I	SC-3	AISC N690 AWS D1.1 ASCE 4-86	Yes
	N/A[2]	II	NNS[4]	AISC-ASD AISC-LRFD AWS D1.1 ASCE 4-86	No
Concrete Super Structures	N/A[2]	I	SC-3	ACI 349 AWS D1.4 ASCE 4-86	Yes
	N/A[2]	II	NNS[4]	ACI 318 AWS D1.4 ASCE 4-86	No
Turbine Building (BWR Only)	Concrete/Steel Frame Building	I	SC-2	N/D[3]	Yes
Piping Supports	N/A[2]	I	SC-1,2,3	ASME III-NF AISC N690 ASME-IX ASCE 4-86	Yes
	N/A[2]	II	NNS[4]	ANSI B31.1 ASME-IX AISC-ASD AWS D1.1	No
Cable Tray Supports	Safety Class/Seismic Category I Cable Tray Supports	I	N/D[3]	N/D[3]	Yes

Table 3.2.9 - Safety Class/Seismic Category Structures for the EPRI URD (continued)					
Structure	Description[1]	Seismic Category	Safety Classification	Applicable Code and Standard	Covered by This Review
HVAC Supports	Safety Class/Seismic Category I Duct Supports	I	N/D[3]	ASME AG-1 ANSI N509	Yes
Fire Protection Systems	Supports	I	N/D[3]	NFPA-13	Yes
	Fire Barriers	N/D[3]	N/D[3]	N/D[3]	Yes
Containment Leak Testing	Periodic Leak Monitoring	N/A[2]	N/A[2]	ANS 56.8	Yes

Footnotes for Table 3.2.9:

(1) The EPRI - URD provides only general functional design requirements, therefore in most cases structural descriptions are not possible and this item is marked as N/A (not applicable).

(2) N/A = Not Applicable.

(3) N/D = Not Defined

(4) NNS = Non Nuclear Safety Component Classification

3.3 Identification of Unique Design Features or Attributes

The ALWR plant designs which are the primary focus of this review are upgrades, advancements, and simplifications to existing (currently operational) reactor designs. The majority of the advancement and simplification is in the areas of systems, components and operations. This advancement includes the use of passive safety systems, reduction in the number of components (pumps, valves, tanks), reduction in the amount of piping required, and the use of digital distributed control systems. The evolutionary reactor structural design basis and the structural components, although in some cases different in appearance, are very similar in nature to today's existing United States domestic operating nuclear power plants. There have been attempts to enhance the structural design and construction process including (1) offsite prefabrication (called modular construction by the reactor vendors), (2) the elimination of the OBE as a design basis event, and (3) the use in some cases of more recent industry consensus codes and standards for Safety Class design and construction applications (AISC N690, AISI CFSDM, NFPA-13). However, not withstanding these features, the majority of the design and construction criteria, methods, etc. are very similar to what was done in recent vintage domestic United States commercial power reactors.

The unique design features of the advanced reactor group include (1) the operational cycles (gas cycles, liquid sodium cycles, etc.) for heat and power generation and (2) primarily in the system components, and operational areas. Also the safety classification and seismic categorization in some cases uses probablistic methods verses the more deterministic approach which has been used in previous commercial power reactor designs. This approach results in power reactor designs which do not have "containments" designed to ASME-III-Division 1, Subsection NE and/or ACI-359 as currently seen in United States operating nuclear power plants. While these reactors do provide some unique structural design aspects, in general, their structural design and construction processes are very similar to the ALWR reactors and to existing operating nuclear power plants.

This lack of revolutionary design concepts must be kept in mind when reviewing the results of the reactor design unique attributes evaluation summarized in this section. The items cited and code changes which are proposed for the ALWR plants are based on the incremental design feature changes which exist in these reactor designs versus distinctly new features. In the tables associated with Section 3.3 in those cases where unique features or attributes are not identified in the design basis data, the table identifies this an N/I, None Identified.

3.3.1 Westinghouse Electric Corporation AP600

As has been previously discussed the majority of the unique features or attributes associated with the Westinghouse AP600 design are in the systems and equipment areas. A significant portion of the structural design and construction is typical of currently operating domestic United States nuclear power plants. There are however some novel features and code applications which should require some modifications to the referenced and existing industry consensus codes and standards.

Table 3.3.1 provides a tabular summary of the unique features and/or attributes associated with the Seismic Category I, Safety Class Structures. The most significant of the unique features and/or attributes are:

- •Elimination of the Operating Basis Earthquake (OBE) from the design basis.
- •Use of modular construction (pre-fabrication) in the containment internal walls and floors and for selected equipment skids or platforms.
- •The passive containment cooling system including such features as: external cooling, weir distribution system, concrete water storage vessels, depressurization loads, etc.
- •The use of the AISI-Cold Formed Steel Design Manual for design of cable tray, tray supports, HVAC ducting and possibly HVAC supports.
- •The use of the AISC N690 for structural steel design and HVAC Supports.
- •Building filtered hydrodynamic loadings due to the operation and discharge of the IRWST storage tank.
- •Hydrodynamic loadings due to the seismic evaluation of the PCCS Storage Pool.
- •Hydrodynamic loadings due to the Automatic Depressurization Event (ADS) event from IRWST.
- •The use of free standing new and spent fuel storage racks in the fuel storage pools.

•No integral connection between the steel containment structure and the reinforced concrete basemat.

The features listed above will be discussed in greater depth in the following subsections. The remaining items listed in Table 3.3.1 are not explicitly discussed but were considered in the identification of the suggested code changes provided in Section 4.0 and 5.0 of this document.

3.3.1.1 Elimination of the OBE

The USNRC is in the process of issuing revisions to 10CFR100.23 and issuing a new Appendix S to 10CFR50 which will state that if the review level earthquake (OBE) is defined as less than 1/3 of safe-shutdown earthquake no explicit design analysis for the OBE level earthquake shall be required. This change is tailored toward new reactor designs and has been incorporated in the SSAR (design basis) for the AP600. This unique design feature will require changes to all the primary industry codes and standards which are the subject of this study. The changes are wide ranging but include the elimination of the OBE from load combinations, and thereby introduce the potential need to address seismic anchor motion effects explicitly for the SSE event, etc. This change will require consideration of code modifications to provide for control of primary plus secondary stress limits for thermal and SSE loading conditions when placed in ASME Service Levels C and D for the metal containment structure. This will include the need to evaluate fatigue control for such loadings. The code required changes are discussed in more detail in Section 4.

3.3.1.2 Modular Construction

The use of modular construction techniques are discussed throughout the AP600 SSAR, the major structural modules that will be implemented are "M" wall modules, "L" wall modules, floor modules and the use of prefabricated concrete reinforcing steel frames. For the containment internal walls below 98′ the construction is reinforced concrete. Above 98′ elevation is where implementation of the "L" and "M" modules begins. However, as discussed below some of these modules extend below the 98′ level to fulfill various design functions.

The "M" modules will be used in the construction of containment internal wall structures above elevations of 83′-98′ depending on their location within the containment. These modules are prefabricated structural steel box sections with steel plates on each face stiffened by vertical diaphragms and horizontal angle stiffeners assembled from subunits which are constructed offsite. Wall modules are anchored to the concrete base by means of anchor bolts and dowels embedded in the concrete below el. 98′. After erection, the walls are filled with concrete. Concrete is used where radiation shielding is required. There is no reinforcing steel used in the concrete since, in general, the concrete will not be required to carry loads. In addition the concrete will not be designed to act compositely with the structural steel except for vertical compression loads. Certain design load combinations for some walls of the "M" modules result in 2D and 3D stress states currently not addressed in the AISC N-690 Standard. Additionally, effective width and width/thickness provisions do not adequately cover the plate and stiffener configuration of the concrete in-filled steel walls for these modules.

Wall modules without concrete fill are also utilized inside containment. The west wall of the refueling water storage tank is this type of module which is constructed solely from structural steel. Structural steel modules are constructed from carbon steel plates and shapes (A36). Stainless steel (A304L) plates are used on the surface of modules that are in contact with water.

"L" modules are permanent steel forms used for the containment internal base concrete structures below elevation 98′. The steel modules consist of steel plates (A36) reinforced with 2″ x 2″ angle stiffeners and 4″ WT sections on the concrete side of the plate. The L wall modules are used in lieu of removable concrete forms. The advantage is that these wall modules can be fabricated and preassembled offsite in parallel with other ongoing construction activities. This reduces construction efforts at the site which results in cost savings to the project. In addition, savings are achieved by eliminating curing time and the need to strip forms, clean-up and patch exposed concrete surfaces.

The concrete floors in the containment interior above elevation 98′ consist of steel tee-sections welded to horizontal steel plates. The steel plates are stiffened by angle stiffeners. Support is provided using deep girders whose webs pass through the horizontal plate. Reinforcing bars are placed above the top flange. After erection, concrete is poured on top of the horizontal plates embedding the upper section of the beams and reinforcing bars. The concrete will be designed to act compositely with the tee sections for vertically downward (not upward) loads. However for upward loads the concrete together with angle plate stiffeners provides stability to the plates. The above also applies to the operating deck above the IRWST; however, the remainder of the operating floor consists of 12" concrete on "Q" decking supported by structural steel. Design of these slabs for composite action references the AISC N-690 standard provisions for fully encased beams, although the described configuration varies from that specified in AISC N-690. Due to the presence of the horizontal plate and angle stiffeners, the intent of the current provisions for fully encased beams is considered to be met, and a "better" bond between the steel and concrete than obtained by the standard configuration is claimed. Suggested changes to the AISC N-690 Standard to address this situation are provided in Section 4.

Floors located above the main control room and instrumentation and control rooms in the auxiliary building are designed as finned floor modules. The purpose of the finned floor modules is to provide a passive heat sink for each room. A finned floor is comprised of a 24 inch thick reinforced concrete slab poured over a stiffened steel plate ceiling. Composite action of the steel and concrete is developed using shear studs welded to the steel plate and embedded in the concrete. The horizontal steel plates are stiffened by welding 1/2" x 9" steel plate perpendicular to the ceiling plates. The steel fins project into the room and act as thermal fins to enhance the transfer of heat from the air to the concrete. Several modules cut to the room width are prefabricated in a shop. On site they are installed side by side perpendicular to the room length. Adjacent panels are made continuous by welding a flat bar along the interface of two panels.

Prefabricated, component rebar support frames will be used in the nuclear island basemat. The frames consist of a bi-direction top and bottom rebar mesh welded to the top and bottom of structural members. These frames are fabricated in sections off site, laid in place on site, and then concrete fills the forms covering the rebar frames. This concept may also be adopted in portions of the containment building interior and exterior concrete placement. The subject industry codes and standards currently do not adequately address these construction techniques. Therefore the use of these designs will require code changes to Section Q1.11 of AISC N690 and probably to ACI-349 and possibly ACI-347. In addition some testing may be required to verify the composite actions of these components.

In addition the AP600 SSAR also defines prefabricated equipment packages which will be built off site and installed as modular units. The majority of the equipment on these packages is outside the scope of this study, however the structural steel used in these packages and the distribution supports will be designed to codes and standards which are the subject of this review. Explicit design requirements sections of these codes and standards are not expected to change significantly. However, items such as code jurisdictions, N-stamping, and the need to consider transportation and fabrication loads result in some suggested code changes.

3.3.1.3 Passive Containment Cooling System (PCCS)

The Passive Containment Cooling System (PCCS) provides a mechanism to maintain the post accident containment pressure and temperature within design limits for at least 72 hours without external power and/or explicit operator actions to initiate or activate safety systems.. The containment is an ASME BPVC Section III, Division 1 Class MC (Subsection NE) steel cylindrical, eliptial head containment vessel. It is surrounded by cylindrical reinforced concrete shield building that results in an annular space between the containment vessel and the shield building. Initial post accident cooling is provided by natural circulation and free air convective heat transfer in this annular region. Should additional heat removal be required a containment wall cooling system is automatically initiated. The containment vessel has a weir structure attached to the top of the vessel which is used for water distribution during this portion of the cooling process. The top of the shield building has a annular shaped water storage tank. This tank discharges water into the weir system which allows it to flow over the containment vessel achieving the necessary cooling.

The PCCS will require changes to ASME-III Subsection NE to provide design criteria for the weir structure and evaluation criteria (including fatigue) for the convective air and water effects on the vessel. In addition changes may be required to ACI-349 to address the design of the concrete water storage tank. During an earthquake event hydrodynamic loads due to the water in the storage tank should be considered in the building design analysis and the generation of in-structure response spectra.

3.3.1.4 Use of AISI-Cold Formed Steel Design Manual

The is a commercial grade standard being applied in Safety Class, Seismic Category I application. Changes to the standard are suggested for its use in a nuclear safety related application.

3.3.1.5 Use of ANSI/AISC N690

This specification has not been generically accepted by the USNRC. Further as discussed in Section 3.4 the USNRC has several concerns with the application of the current version. Therefore changes to the specification should be made to address these concerns.

3.3.1.6 Use of ASCE 4-86

The standard has not been generically accepted by the USNRC. It is however referenced several times in Section 3.7.2 of the Standard Review Plant as a basis for seismic design. The Standard was directly referenced by Westinghouse in Revision 0 of the AP600 SSAR. However revision 2 of the SSAR has dropped direct reference to this Standard and has listed the actual seismic analysis basis in the SSAR. However, since it was originally referenced by Westinghouse and is also referenced by the SRP the Standard was reviewed versus current USNRC SRP requirements and existing regulatory guidelines and suggested changes to insure conformance with these regulatory requirements were developed.

3.3.1.7 Seismic Hydrodynamic Loadings

In the AP600 design the top of the reactor shield building contains an annular shaped storage tank which stores 350,000 gallons of water for potential use with the Passive Containment Cooling System. During an earthquake there will be significant seismic excitation of the tank which will cause excitation of the water within the tank resulting in direct and building filtered hydrodynamic loadings being applied to the shield building and to the balance of the nuclear island. Analysis techniques for these type of seismic loads do not exist in the current SRP nor in ASCE 4-86 Standard. Criteria for this type of analysis is currently being developed by the ASME (Appendix N), ASCE (Dynamic Analysis Committee) and Department of Energy (Tank Seismic Experts Panel). Changes will be required to the SRP and/or to ASCE 4-86 to address these issues. Further the effect of these loadings must be incorporated in the applicable design loadings for the Shield Building, Containment and Containment Internal Structures as part of the seismic input loads.

3.3.1.8 Building filtered IRWST Operation Hydrodynamic Loads

During an Automatic Depressurization Event (ADS) significant dynamic loads are imposed on the In-containment Reactor Water Storage Tank (IRWST). This input causes excitation (vibration) of the IRWST which in turn may excite the building structure housing the tank (the containment and containment internal structures). This could result in direct hydrodynamic loads on the containment structure and building and building filtered hydrodynamic loads on the balance of the nuclear island structures. This effect is similar to the hydrodynamic loadings in pressure suppression containments. While some test data is available, the magnitude of the loads are not yet defined and the overall effects of the significance of this load can not be quantified. The code changes being suggested for evaluation of pressure suppression containment hydrodynamic loads would also be applicable to these loadings.

3.3.1.9 Use of Free Standing Fuel Racks

Generically there is not a clear definition of the applicable standard which should be used for the design of free standing fuel racks. Previous designs have used ASME III, Subsection NF but it is not necessarily the most appropriate for this application since fuel racks and fuel elements are not pressure retaining components. Replacement fuel racks in US operating nuclear plants have used SRP 3.8.4, Appendix D as the design basis input. These racks require evaluation for potential sliding, impact and water sloshing loads during an earthquake event. Consideration should be given to the use of AISC/ANSI N690 for this application.

3.3.1.10 Foundation and Containment Interface

The containment vessel shell (Class MC containment) is supported on the base mat without structural connection. Interaction and containment restraint is assumed to be achieved due to the ellipsoidal shape of the containment vessel. Both ASME III BPVC, Division 1 Subsection NE and ACI-349 should be modified to identify the need to verify this interaction and restraint.

3.3.1.11 Concrete Placement Issues

Specific concrete placement procedures for the "M" wall modules need to be developed for the following reasons:

a) Height may require careful placement and consolidation of wet concrete (vibration)

b) Vertical diaphragms and horizontal stiffeners may interfere with complete consolidation

c) Access for concrete workers inside module may be restricted.

There may also be potential problems in concrete placement for the conical roof. The ACI-349 code should be modified to address this issue either directly or by reference to other applicable ACI concrete placement standards.

3.3.1.12 Stability of the Nuclear Island

In the current SSAR (Section 3.8.5.5.4) to evaluate the overturning potential of the nuclear island during a SSE event, it is indicated that the energy balance method is to be used. Due to USNRC concerns and questions, a change to the moment balance method is being considered. None of the subjected codes and standards currently provide design guidance on this issue.

3.3.1.13 Concrete Codes

While the current SSAR does not reference ACI-318, it is the authors understanding that the detailing, placing, anchoring, and splicing of reinforcing bars will be in accordance with Chapter 21 of ACI-318. ACI-349 should be modified to provide the necessary design guidance.

In addition ACI-349, Appendix B should include enhanced design rules for expansion anchors and base plate flexure consistent with I&E 79-02.

3.3.1.14 Concrete Missile Barrier Design

The use of the Modified NDRC is specified in SAR 3.5.3. Appropriate standardized missile design criteria should be developed and incorporated into an appropriate industry standard.

3.3.1.15 Differentiation Between Design Basis and Severe Accident Loads

The current SSAR in Section 3.8.2.4.2 discusses the evaluation of ultimate capacity but does not explicitly state to which severe accident loads this evaluation will be applicable. To thoroughly understand the AP600 containment design basis it is necessary the severe accident loads being evaluated be identified (ie. hydrogen deflagration, steam explosion, etc.)

3.3.1.16 Subsequent SSAR Revisions

The review and assessment provided of the Westinghouse AP600 Design presented in this document was based on Revision 2 of the SSAR. Subsequent to the completion of the review effort and issuance of the final draft report Westinghouse issued Revisions 3 and 4 to the SSAR. These revisions were not included in this program but appear to contain some significant structural design changes which could alter some of the conclusions and recommendations of this study.

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2]

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Nuclear Island</u> Basemat (Foundations)	Reinforced Concrete Basemat Common to Containment Vessel, Shield Building and Auxiliary Building	1. Interaction Requirements between steel containment shell type vessel and reinforced concrete basemat 2. Mat is 6'-0" thick under the Auxillary Building. This is shallow in comparison to similiar structures (basemats) in the SYS 80+ and the ABWR (≈ 40% less).	N/I[1]	1. Elimination of the OBE 2. Interaction with containment vessel 3. Building Filtered Hydrodynamic loads from Seismic Excitation of Shield Building Storage Tanks 4. Building Filtered IRWST Hydrodynamic Loads	1. Finite element analysis includes effect of walls and interior structures to resist mat bearing pressures.	N/I[1]	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2] (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Nuclear Island</u> Containment Interior	Concrete at the base of steel containment floor and equipment compartments	1. L modules 2. M modules 3. Floor modules	1. Use of AISC N690 for structural steel designs	1. Elimination of the OBE 2. Sandwich panels (modular construction) 3. Missile barrier capacity of sandwich panels 4. Building Filtered Hydro-dynamic loadings from Seismic Excitation of the Shield Building Storage Tank 5. Building Filtered IRWST Hydrodynamic Loads 6. Direct IRWST Hydrodynamic Loadings	1. Stress Criteria for sandwich panels 2. Potential for testing requirements to demonstrate adequacy of composite design approach	1. Modular construction 2. Concrete Placement	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2] (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Nuclear Island</u> Shield Building	Cylindrical concrete supported on basemat with conical roof	1. Modular rebar design 2. Concrete storage water tanks on top of shield building	N/I[1]	1. Elimination of the OBE 2. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks 3. Building Filtered IRWST Hydrodynamic Loadings	1. Finite element analysis includes effect of walls and interior structures to resist mat bearing pressures.	1. Modular construction and poured in place 2. Potential problems for concrete placement for conical roof concrete.	N/I[1]
<u>Nuclear Island</u> Auxiliary Building	Typical of existing operating plant layouts except that basemat is common with containment	1. Modular Construction	1. Use of AISC N690 for structural steel designs	1. Elimination of the OBE 2. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks 3. Building Filtered IRWST Hydrodynamic Loadings	1. Finite element analysis includes effect of walls and interior structures to resist mat bearing pressures.	1. Modular construction 2. Concrete Placement	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2] (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Nuclear Island</u> Containment Air Baffle	Steel Plates	1. These baffles are unique feature for cooling containment vessel	1. Use of N690 for baffle design	1. Applicable loads 2. Design code criteria	N/I[1]	1. Applicable codes and standards must be determined; also jurisdictional boundary between Class MC and N690	N/I[1]
Containment Vessel	Steel vessel	1. External water cooling flow post accident 2. Weir water distribution system 3. Bottom head not structurally connected to concrete basemat	N/I[1]	1. Elimination of OBE loads 2. Thermal Loads to weir system 3. Interaction with basemat 4. Depressurization loads 5. Building Filtered and Direct IRWST Hydrodynamic Loadings 6. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks		N/I[1]	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2] (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
RCL & RPV Supports	RCL Equipment Supports & RPV Supports	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Building Filtered IRWST Hydrodynamic Loads 3. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks	N/I[1]	N/I[1]	
Cable Tray Supports	Category I Cable Tray Supports	N/I[1]	1. Use of AISI-CFSDM for Safety Class supports	1. Elimination of the OBE 2. Building Filtered IRWST Hydrodynamic Loads 3. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks	N/I[1]	1. Use of AISI-CFSDM for Safety Systems	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2] (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
HVAC Supports	Category I HVAC Supports	N/I[1]	1. Use of AISC N690 for supports	1. Elimination of the OBE 2. Building Filtered IRWST Hydrodynamic Loads 3. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks	N/I[1]	N/I[1]	N/I[1]
Piping Supports	Seismic Category I Piping Supports	N/I[1]	1. Use of Support Deflection Criteria 2. Use of N690 for Piping Supports	1. Elimination of the OBE 2. Support deflection criteria 3. Building Filtered IRWST Hydrodynamic Loads 4. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks	1. Use of N690 for Piping Supports	1. Use of N690 for Piping Supports	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[2] (continued)							
		Unique Features and Attributes					
Structure	Description	Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Fire Protection Systems/Supports	Fire Barriers	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
	Piping Supports	N/I[1]	1. Use of B31.1 in conjunction with NFPA-13 and the use of NFPA-14	1. Elimination of the OBE 2. Building Filtered IRWST Hydrodynamic Loads 3. Hydrodynamic loadings from Seismic Excitation of the Shield Building Storage Tanks	1. Use of ANSI B31.1.	1. Use of ANSI B31.1.	N/I[1]
Fuel Storage Racks	Racks which are free standing	1. Free standing fuel racks for new design.	N/I[1]	1. Elimination of the OBE 2. Free standing fuel racks for new designs.	1. Free standing fuel racks for new designs.	N/I[1]	N/I[1]

Table 3.3.1 - Unique Features and Attributes of the Westinghouse AP600[3] (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Missile Barriers	Concrete and Steel	N/I[1]	N/I[1]	N/I[1]	1. For steel use of BRL or Stanford Formulas 2. For reinforced concrete barrier the use of Modified NDRC formula.	N/I[1]	N/I[1]
Containment Leak Testing/Evaluation	Leak Rate Testing	N/I[1]	N/I[1]	1. Use of ANS 56.8.	N/I[1]	N/I[1]	1. Use of ANS 56.8

Footnotes for Table 3.3.1:

[1] N/I= Not Identified.

[3] The information summarized in this table is based on Revision 2 of the SSAR for the AP600.

3.3.2 ABB/CE System 80+

Of all the evolutionary reactor designs the ABB/CE System 80+ has the fewest unique or innovative features in the structural design, fabrication, and testing area. The majority of the innovative features are in the systems, control systems, and operational areas. Table 3.3.2 provides a tabular summary of the unique features and/or attributes associated with the Seismic Category I, Safety Class Structures. The most significant of the unique features and/or attributes are:

- Elimination of the Operating Basis Earthquake (OBE) from the design basis.
- The support for the containment vessel is a concrete subsphere structure.
- The consideration of NUREG-0800 selected severe accidents in the containment and containment internal structure design basis.
- The use of the AISI-Cold Formed Steel Design Manual (CFSDM) for cable tray and HVAC supports.
- The use of AISC N690 for structural steel design and cable tray and HVAC supports.
- Seismic Hydrodynamic Loading.
- The use of free standing new and spent fuel storage racks in the fuel storage pools.

The features listed above will be discussed in greater depth in the following subsections. The remaining items listed in Table 3.3.1 will not be explicitly discussed but were considered in the identification of the code and standard changes provided in Section 4.0 and 5.0 of this document. It should be noted that the nuclear island contains the reactor building and surrounding nuclear annex, all of which are Seismic Category I structures.

3.3.2.1 Elimination of the OBE

The USNRC is in the process of issuing revisions to 10CFR100.23 and issuing a 10CFR50, Appendix S which will state that if the inspection level earthquake (OBE) is defined as less than 1/3 of safe-shutdown earthquake no explicit design analysis for the OBE level earthquake will be required. This change is tailored toward new reactor designs and has been incorporated in the SSAR (design basis) for the System 80+. This unique design feature will require changes to all the primary industry consensus codes and standards which are the subject of this study. The changes are wide ranging but include the elimination of the OBE from load combinations, the potential need to address seismic anchor motion effects for the SSE event, etc. This change will require consideration of code modifications to provide for control of primary plus secondary stress limits for thermal and SSE loading conditions when placed in ASME Service Levels C and D for the metal containment structure. This will include the need to evaluate fatigue control for such loadings. The code required changes are discussed in more detail in Section 4.

3.3.2.2 Containment Vessel Support

The containment vessel is a sphere with a diameter of 200 feet. The lower segment below El. 91′-9″ is supported by concrete. Radially extending shear bars are welded to interior and exterior surfaces of the containment vessel in the embedded region. Above El. 91′-9″, the containment behaves as a free standing vessel with no constraints. In the transition region a compressible material is provided. Past practice has indicated that in this transition region the containment vessel may be subject to water accumulation and corrosion. In SAR 3.8.2.4.3H the safety factor against sliding is 2.4 while in SAR 3.8A, 5.2.4 and SAR 3.8.2.5F a safety factor of 1.1 is given for load combination which includes safe shutdown earthquake (SSE)

3.3.2.3 Consideration of Severe Accident Loadings

The System 80+ has considered some NUREG-0800 extreme accident loads as design basis loads for the containment and containment internals. The loads considered appear to be somewhat selective and arbitrary and do not encompass either the NUREG-0800 events or the requirements put forth in the SECY- 93-087 criteria document. Specifically in Section 3.8.2.3 of the SAR Hydrogen Deflagration is considered as a design load with the design load combinations and allowable stress limits given in Tables 3.8-2 and Table 3.8-3A. Changes should be considered to the containment design standards used for design and construction of the internal structure(s) to address load combinations and allowable stress for this severe accident event(s).

3.3.2.4 Use of AISI-Cold Formed Steel Design Manual (AISI-CFSDM)

Seismic Category I, Safety Class HVAC and Cable Tray systems cold formed steel supporting members are being designed using the AISI-CFSDM. This is a commercial grade standard being applied in Safety Class, Seismic Category I application. Changes to the standard should be considered for its use in this application. This is especially true in the materials and Quality Assurance areas. Further it should also be noted that the HVAC ducting and the cable trays themselves are also being designed with the AISI-CFSDM. Review of the mechanical systems themselves is beyond the scope of this program. However it is suggested that the use of AISI-CFSDM in lieu of IEEE-628 for cable tray design should be reviewed and evaluated.

3.3.2.5 Use of AISC N690

This specification has not been generically accepted by the USNRC. Further, as will be discussed in section 3.4 the USNRC has several concerns with the application of the current version which are provided in NUREG-1462 (System 80+ SER). Therefore changes should be considered to this specification to address USNRC concerns.

3.3.2.6 Use of Free Standing Fuel Racks

Generically there is not a clear definition of the applicable standard which should be used for the design of free standing fuel racks. Previous designs have used ASME III, Subsection NF but it is not necessarily the most appropriate for this application since fuel racks and fuel elements are not pressure retaining components and Replacement fuel racks in US operating nuclear plants have used SRP 3.8.4, Appendix D as the design basis input. These racks require evaluation for potential sliding, impact and water sloshing loads during an earthquake event. Consideration should be given to the use of AISC/ANSI N690 for this application.

3.3.2.7 Use of ACI-318, Chapter 21

ABB/CE has committed to aspects of Chapter 21 of ACI 318-89, Revised 1992; however the reference section to Chapter 21 is somewhat ambiguous. It is believed that this was done to take advantage of the enhanced ductile design features which are provided in Chapter 21 of ACI-318. These features are not currently available in ACI-349 and therefore ACI-318 was used. This is an enhancement to the overall plant design. Changes should be made to ACI-349 to provide the necessary design information, consistent with the intent of Chapter 21 of ACI-318-89.

3.3.2.8 Concrete Missile Barrier

For missile penetration in concrete ABB/CE intends to use the Modified Petry or Modified NDRC methodology. Only one method should be selected and used consistently. In either case the appropriate industry standards (ANS) should be modified to incorporate these approaches into them.

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Reactor Building</u> Shield Building	Cylindrical concrete structure with hemispheric dome enclosing containment	Part of nuclear island.	N/I[1]	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]
<u>Reactor Building</u> Internal Structures	Reinforced Concrete Over the Bottom of the Containment Shell (Note: The Containment Shell is Founded on the Basemat)	1. Structures not supporting Seismic Category I items are Seismic Category II	N/I[1]	1. Elimination of the OBE 2. Severe accident criteria 3. Use of AISC N690 4. Use of ACI-318, Chapter 21	1. Seismic I/II criteria and analysis methods	N/I[1]	N/I[1]
<u>Reactor Building</u> Containment	Spherical Steel Structure	Except for base support below El. 91'-9", it is structurally independent	N/I[1]	1. Elimination of the OBE, will require new fatigue and fatigue ratcheting criteria for SSE 2. Missile design criteria 3. Severe accident criteria	N/I[1]	N/I[1]	N/I[1]

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+ (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Reactor Building</u> Equipment Hatch		Part of the steel containment	N/I[1]	1. Elimination of the OBE will require new fatigue and fatigue ratchet criteria for the SSE	N/I[1]	N/I[1]	N/I[1]
<u>Reactor Building</u> Personnel Airlocks		Part of the steel containment	N/I[1]	1. Elimination of the OBE will require new fatigue and fatigue ratchet criteria for the SSE	N/I[1]	N/I[1]	N/I[1]
<u>Reactor Building</u> Subsphere	Portion of reactor building below 91′- 9″ which is in direct contact with containment vessel.	1. Transition zone design criteria between the sphere and the subsphere 2. Compressible material is used.	N/I[1]	1. Elimination of the OBE 2. Interface design between containment vessel and basemat 3. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]
<u>Nuclear Annex</u> Control Area	Monolithic Attachment to Shield Building on Common Basemat	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+ (continued)							
		Unique Features and Attributes					
Structure	Description	Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
<u>Nuclear Annex</u> Spent Fuel Pool	Steel lined pool Integral part of the Nuclear Annex	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]
<u>Nuclear Annex</u> Fuel Racks	High density stainless steel racks (Limited Design Data Available)	1. Free standing fuel racks for new plant design	N/I[1]	1. Elimination of the OBE 2. Free standing fuel racks	1. Free standing fuel racks 2. Only single rack analysis verses consideration of multiple racks	N/I[1]	N/I[1]
<u>Nuclear Annex</u> Valve House Areas	Integral part of the nuclear annex	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]
<u>Nuclear Annex</u> Emer. Diesel Generator Areas	Integral part of the nuclear annex	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]
<u>Nuclear Annex</u> CVCS/Main Area	Integral part of the nuclear annex	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I[1]	N/I[1]	N/I[1]

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+ (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Station Service Water Pump Structure	Reinforced Concrete Structure	N/I(1)	N/I(1)	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I(1)	N/I(1)	N/I(1)
Component Cooling Water Heat Exchanger Structure	Reinforced Concrete Structure	N/I(1)	N/I(1)	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I(1)	N/I(1)	N/I(1)
Diesel Fuel Storage Structure	Reinforced Concrete Structure	N/I(1)	N/I(1)	1. Elimination of the OBE 2. Use of ACI-318, Chapter 21	N/I(1)	N/I(1)	N/I(1)
Foundations	Reinforced Concrete Structure	N/I(1)	N/I(1)	1. Elimination of OBE 2. Use of ACI-318, Chapter 21	N/I(1)	N/I(1)	N/I(1)
Piping Supports	Seismic Category I Pipe Supports	N/I(1)	N/I(1)	1. Elimination of OBE	N/I(1)	N/I(1)	N/I(1)
HVAC Supports	Seismic category portion of distribution supports	N/I(1)	N/I(1)	1. Elimination of OBE 2. Use of AISC N690 3. Use of AISI-CFSDM	N/I(1)	N/I(1)	N/I(1)

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+ (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Cables/Conduit Supports	Seismic Category I portion of distribution supports	N/I[1]	N/I[1]	1. Elimination of OBE 2. Use of AISC N690 for Supports 3. Use of AISI-CFSDM for supports 4. Use of AISC-ASD for Seismic Category I Supports	N/I[1]	N/I[1]	N/I[1]

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+ (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Fire Protection	Barriers/Doors	N/I[1]	N/I[1]	1. Use of NFPA-803	N/I[1]	N/I[1]	N/I[1]
	Piping Supports	1. Allowable support types an issue (Cast Iron, etc.)	N/I[1]	1. SSE design criteria 2. Use of non nuclear codes and standards for SC Items (NFPA-13) 3. Use of ANSI B31.1 for Sprinkler Design	N/I[1]	N/I[1]	N/I[1]
	Hose/Standpipe Systems	1. Allowable support types an issue (Cast Iron, etc.)	N/I[1]	1. SSE design criteria 2. Use of non nuclear codes and standards for SC Items (NFPA-14) 3. Use of ANSI B31.1 for Sprinkler Design	N/I[1]	N/I[1]	N/I[1]
Containment Leak Testing	Periodic Leak Rate Testing	N/I[1]	1. No criteria for secondary containment leak testing specified	N/I[1]	N/I[1]	N/I[1]	N/I[1]

Table 3.3.2 Unique Features and Attributes of the ABB/CE System 80+ (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication/ Construction	Testing
Missile Barrier	Concrete	N/I[1]	N/I[1]	N/I[1]	1. For concrete use modified Petry or modified NDRC methodology	N/I[1]	N/I[1]

Footnotes for Table 3.3.2:

[1] N/I - None Identified

3.3.3 GE ABWR

From a structural standpoint General Electric's ABWR is similar in design to Mark II and Mark III pressure suppression containment based BWR designs currently operating in the United States. While the actual design layout and physical appearance differs from Mark II and III designs, it is a pressure suppression type containment with a drywell, suppression pool, enclosed and attached monolithically to a reinforced concrete reactor building. As with the other evolutionary reactors the majority of the unique features are in the system components and operational areas. Table 3.3.3 provides a tabular summary of the unique features and/or attributes associated with the seismic Category I, Safety Class Structures. The most significant of these features and/or attributes are:

- Elimination of the Operating Basis Earthquake (OBE) from the design basis.
- Use of modular construction in the area of the RPV pedestal and shield wall.
- The use of the AISI-Cold Formed Steel Design Manual for cable tray supports.
- The use of AISC N690 for structural steel design including HVAC and Cable Tray Supports
- Seismic Hydrodynamic Loadings.

The features listed above will be discussed in greater depth in the following subsections. The remaining items listed in Table 3.3.1 will be explicitly discussed but were considered in the identification of the code changes provided in Section 4.0 and 5.0 of this document.

3.3.3.1 Elimination of the OBE

The USNRC is in the process of issuing revisions to 10CFR100.23 and issuing a 10CFR50, Appendix S which will state that if the review level earthquake (OBE) is defined as less than 1/3 of safe-shutdown earthquake no explicit design analysis for the OBE level earthquake will be required. This change is tailored toward new reactor designs and has been incorporated in the SSAR (design basis) for the GE ABWR. This unique design feature will require changes to all the primary industry consensus codes and standards which are the subject of this study. The changes are wide ranging but include the elimination of the OBE from load combinations, the potential need to address seismic anchor motion effects for the SSE event, etc. This change will require consideration of code modifications to provide for control of primary plus secondary stress limits for thermal and SSE loading conditions when placed in ASME Service Levels C and D for the metal containment structure. This will include the need to evaluate fatigue control for such loadings. The code required changes are discussed in more detail in Section 4. It should be noted that OBE was included in a load combination for spent fuel storage racks.

3.3.3.2 Limited Use of Modular Construction

The RPV pedestal and shield wall structures are made up of two concentric steel shells joined by horizontal and vertical steel plate diaphragms. These steel structures are first set in place and then filled with concrete. Based on the information provided in the SSAR, it is stated that all loads are resisted by the integral action of the inner and outer steel shells. It is further stated that the concrete placed in the annulus between the inner and outer shells acts to distribute loads between the steel shells and provides compressor load stability. The design of this concrete is per ACI-301 which is not a safety related design code nor does it address any possible composite action between the concrete and the steel members. Modifications should be made to ACI-301 to address the safety related aspects of this application. Further the applicable steel and concrete design codes should be modified to insure that there is adequate concrete corrosion protection and adequate structure composite action. The industry consensus codes and standards suggested changes developed for the W AP600 (to address composite design) are reviewed to insure they cover this particular structure.

3.3.3.3 Use of AISI-Cold Formed Steel Design Manual (CFSDM)

The AISI CFSDM is specified for the design of Seismic Category I, safety class cable tray cold formed steel supports, while the trays themselves are being designed to NEMA standards. These are commercial grade standards being applied in a nuclear safety class, Seismic Category I application. Changes to these standards are recommended for their use in this application.

3.3.3.4 Use of AISC N690

This specification has not been generically accepted by the USNRC. Further as will be discussed in Section 3.4 the USNRC has several concerns with the application of the current version as documented in Appendix G of NUREG-1503. Therefore specification changes are suggested to address the concerns expressed by the USNRC.

Roof beams have connections to the concrete slabs to prevent uplift during a tornado. The available documentation is not clear on the nature of these connections. However the standard changes suggested to address the Westinghouse AP600 modular constructions features should address this issue.

3.3.3.5 Seismic Hydrodynamic Loadings

The ABWR uses a pressure suppression type of containment which includes a water suppression pool inside the reactor building. During an earthquake there will be significant seismic excitation of the reactor building which will cause excitation of the water in the suppression pool resulting in direct and building filtered hydrodynamic loadings being applied to the reactor building and internal structures. Analysis techniques for these types of loadings currently do not exist in ASCE 4-86 or in the SRP. Criteria for this type of analysis is currently being developed by the ASME (Appendix N), ASCE (Dynamic Analysis Committee) and the Department of Energy (Tank Seismic Experts Panel). Therefore modifications to ASCE 4-86 or the development of a new standard is suggested to provide an industry code and standard with the necessary analysis criteria. Further these loadings should be incorporated in the applicable design loadings for the Reactor Building, Containment and Containment Internal Structures as part of the seismic input loadings.

3.3.3.6 Hydrodynamic Loads

The evaluation requirements for hydrodynamic loads resulting from the use of a pressure suppression containment are discussed in Section 3.5.2 and shown in Table 3.5.2.1.

3.3.3.7 Concrete Missile Barrier

In SAR 3.5.3.1.1, either the modified Petry formula or the TM5-855-1 formula may be used. In SAR Appendix 3H.1.5.7 only TM 5-855-1 was used. This should be clarified; however, the modified NDRC formulas have been generally used in past practice.

3.3.3.8 Concrete Codes

While only ACI-318 is listed in SAR 3.8.4.2.1 for reactor buildings, SAR Appendix 3H.1.4.11 lists ACI-318, ACI-349 and ACI-359 for the reactor building and discusses no specific application of ACI-318. This leaves in doubt the implementation of the seismic detailing requirements as specified in Chapter 21, ACI-318. Increases in minimum steel reinforcing requirements are being considered for both ACI-318 and ACI-349. IE 79-02 is used as the design basis for concrete anchor bolt and pipe support base plate analysis.

3.3.3.9 Spent Fuel Storage Racks

In SAR 9.1.4.3 no details are provided except that the spent fuel storage racks are purchased equipment. SRP 3.8.4 Appendix D is listed as the design basis.

3.3.3.10 Stability Requirements

In SAR 3.8.5.5, 3.7.2.14 and 3H.1.5.6 only energy considerations are used for overturning instead of moment equilibrium.

3.3.3.11 Concrete Containment and Reactor Building Interconnection

In a departure from past practice, the containment structure is structurally connected to the reactor building. The connections are made by the following:

a) Common basemat
b) Six reinforced concrete slabs which monolithically connect the reactor building to the containment
c) The containment top slab is integral with the fuel pool girders and the containment wall

In spite of the fact that the combined structure will respond to all loads, SAR Fig. 3H.1-2 indicates a division of code applicability; such that only the containment walls and basemat within the containment walls are covered by ACI-359/ ASME III, Div.2. The remainder of the combined structure; ie, the reactor building, is presumably covered by ACI-349 and to some unknown extent by ACI-318. This situation is rather confused in that the basemat and reactor building participates with the containment and therefore, must be considered as adding strength and thus treated as containment.

3.3.3.12 Control and Radwaste Building, Composite Construction

While structural steel frames support reinforced concrete slabs, no specifics are provided to indicate that the reinforced concrete slabs and structural steel beams are to be designed and detailed for composite action. However, roof beams are provided with welded studs to prevent uplift, which could provide for some composite action for downward (as opposed to uplift) loads.

Table 3.3.3 - Unique Features and Attributes of the GE ABWR							
Structure	**Description**	**Unique Features and Attributes**					
		Physical	**Application**	**Design (Inc. Loads)**	**Analysis**	**Fabrication / Construction**	**Testing**
Containment Vessel	Reinforced Concrete Structures	Structurally interconnected to reactor building. Current ACI codes do not cover this interaction.		1. Elimination of the OBE, Fatigue & Fatigue Ratchet Criteria for SSE for liner 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	Combined analysis is required.	N/I[1]	N/I[1]
Vent System		N/I[1]	N/I[1]	1. Elimination of the OBE 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	N/I[1]	N/I[1]	N/I[1]
PCV Pene. & Drywell Head	Primary Containment Vessel Penetrations & Drywell Head	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Fatigue & Ratchet Criteria for SSE 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	N/I[1]	N/I[1]	N/I[1]

Table 3.3.3 - Unique Features and Attributes of the GE ABWR (continued)							
Structure	**Description**	**Unique Features and Attributes**					
		Physical	**Application**	**Design (Inc. Loads)**	**Analysis**	**Fabrication / Construction**	**Testing**
RPV Stabilizer Truss	Steel Frame Structure	N/I(1)	N/I(1)	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	N/I(1)	N/I(1)	N/I(1)
Diaphragm Floor	Reinforced Concrete Slab	Supported by reactor pedestal and containment wall	N/I(1)	1. Elimination of the OBE 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	ASME III BPVC, Div. 2 for intersection with containment wall	N/I(1)	N/I(1)
Lower Drywell Equipment & Personnel Tunnels	Steel Structure (Stainless & Carbon Steel)	N/I(1)	N/I(1)	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	N/I(1)	N/I(1)	N/I(1)

Table 3.3.3 - Unique Features and Attributes of the GE ABWR (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
RPV Pedestal & Shield Wall	Pedestal consists of two concentric steel shells tied together. Annulus filled with concrete Shield Wall, Two Cylindrical Steel Structures filled with Concrete	1. Composite Construction Steel and Concrete although only steel appears to be used for resisting forces and moments	N/I[1]	1. Elimination of the OBE 2. Composite Design Criteria 3. Use of ASIC N690 4. Seismic Hydrodynamic loadings 5. Hydrodynamic loadings 6. Use of ACI-301 for Concrete Design	N/I[1]	1. Appropriate Construction Procedures for Concrete for Pouring 2. Use of ACI-301 for Concrete Fabrication	N/I[1]
Foundation Work	2 Basemats - (1) Reactor Building and Containment and (2) Control Building	N/I[1]	N/I[1]	1.Elimination of the OBE 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	N/I[1]	N/I[1]	N/I[1]

Table 3.3.3 - Unique Features and Attributes of the GE ABWR (continued)							
Structure	**Description**	**Unique Features and Attributes**					
		Physical	**Application**	**Design (Inc. Loads)**	**Analysis**	**Fabrication / Construction**	**Testing**
Reactor Building	Similar to Current Layouts. Reinforced Concrete Structure except that it is structurally interconnected with reinforced concrete containment structure	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Fatigue & Ratchet design criteria for SSE for components designed to NE 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	Both ACI-349 and ACI 318 are listed. No specifics on ACI-318 are provided	N/I[1]	N/I[1]
Control Building	Reinforced Concrete Structure with Steel Roof	1. Possible use of Modular Construction	1. Possible use of Modular Construction	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings 5. Possible use of Modular Construction	1. Possible use of Modular Construction	N/I[1]	N/I[1]

Table 3.3.3 - Unique Features and Attributes of the GE ABWR (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Radwaste Building	Reinforced Concrete Structure	1. Possible use of Modular Construction	1. Possible use of Modular Construction	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings 5. Possible use of Modular Construction	N/I[1]	1. Possible use of Modular Construction	N/I[1]
Containment Internal Steel	Structural Steel	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	N/I[1]	N/I[1]	N/I[1]
DEPSS	Drywell Equipment and Pipe Support Structure	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic loadings 4. Hydrodynamic loadings	N/I[1]	N/I[1]	N/I[1]

Table 3.3.3 - Unique Features and Attributes of the GE ABWR (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Containment Internal Concrete	Reinforced Concrete	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Seismic Hydrodynamic loadings 3. Hydrodynamic loadings	N/I[1]	N/I[1]	N/I[1]
Safety Class Cable Tray Supports	Seismic Category I Distribution System Supports	N/I[1]	1. Use of NEMA Standards for Cable Tray Design	1. Elimination of the OBE 2. Use of AISI-CFSDM. 3. Use of AISC N690	N/I[1]	N/I[1]	N/I[1]
Piping Supports	Safety Class Distribution System Supports	N/I[1]	N/I[1]	1. Elimination of OBE 2. Use of AISC ASD 3. Use of CC N476 4. Use of IE 79-02 (Base Plate/Anchor Bolts)	N/I[1]	N/I[1]	N/I[1]
Safety Class HVAC Supports	Safety Class Distribution System Supports	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of ASME AG-1 3. Use of AISC N690	N/I[1]	N/I[1]	N/I[1]

Table 3.3.3 - Unique Features and Attributes of the GE ABWR (continued)							
Structure	**Description**	**Unique Features and Attributes**					
		Physical	**Application**	**Design (Inc. Loads)**	**Analysis**	**Fabrication / Construction**	**Testing**
Fire Protection	Supports within primary containment boundary	N/I[1]	N/I[1]	1. NFPA-13 2. NFPA-14 3. ANSI B31.1	1. NFPA-13 2. NFPA-14 3. ANSI B31.1	N/I[1]	N/I[1]
	Barriers	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
Fuel Storage Racks	Possible Free Standing Fuel Storage Racks	N/I[1]	No specifics. Will be procured to Specification to be developed later.	1. Uses SRP 3.8.4, App. D as reference.	1. Uses SRP 3.8.4, App. D as reference.	N/I[1]	N/I[1]
Containment Leak Testing	Leakage Rate Testing	N/I[1]	N/I[1]	1. Use of ANS 56.8	N/I[1]	N/I[1]	1. Use of ANS 56.8
Missile Barrier	Concrete/Steel	N/I[1]	N/I[1]	N/I[1]	For concrete either Modified Petry or the TM5-855-1 formulas	N/I[1]	N/I[1]

Footnotes for Table 3.3.3:

[1] N/I = None Identified

3.3.4 GE SBWR

The GE SBWR is a compact simplified nuclear power plant which has similarities to the BWR designs based on Mark I containment type. The majority of the unique features are operational methods, systems, and component related. Modular construction aspects of the design are very ill defined at this time. However, if extensive modular construction were to be applied the SBWR would have a significant number of unique features which could result in it being classified from an evolutionary to an advanced reactor design. Table 3.3.4 provides a tabular summary of the unique features and/or attributes associated with the Seismic Category I, Safety Class Structures. The most significant of these unique features and/or attributes are:

- Elimination of the Operating Basis Earthquake (OBE) from the design basis
- Use of modular construction (Pre Fabrication) in the various aspects of the reactor building and structures and buildings
- Alternative approaches for the control ferretic steel welding are proposed
- Concrete strength reductions are proposed for concrete temperatures between 70° and 150°F
- The use of ANSI/AISC N690 for structural steel design and construction and the use of ASCE 4-86 for structural seismic analysis
- Seismic hydrodynamic loadings
- Hydrodynamic loadings

The features listed above will be discussed in greater depth in the following subsections. The remaining items listed in Table 3.3.1 will not be explicitly discussed but were considered in the identification of the code and standard changes provided in Section 4.0 of this document.

3.3.4.1 Elimination of the OBE

The USNRC is in the process of issuing revisions 10CFR100.23 and issuing 10CFR50, Appendix S which will state that if the inspection level earthquake (OBE) is defined at less than 1/3 of Safe Shutdown Earthquake, no explicit analysis for the OBE level earthquake will be required. This change is tailored toward new reactor designs and has been incorporated in the SSAR (design basis) for the SBWR. This unique design feature will require changes to all the primary industry consensus codes and standards which are the subject of this study. This change will require consideration of code modifications to provide for control of primary plus secondary stress limits for thermal and SSE loading conditions when placed in ASME Service Levels C and D for the metal containment structure. This will include the need to evaluate fatigue control for such loadings. The code required changes are discussed in more detail in Section 4.

3.3.4.2 Use of Modular Construction

Upon reviewing the SBWR Standard Safety Analysis Report no description is provided regarding the use of modular construction. Section 3.8 of the SSAR describes the reinforced concrete containment, containment internal structures and other Seismic Category I Structures. However, no description of the use of structural modules or analysis and design criteria is included in the SSAR.

Letters from GE to USNRC provide the majority of the input on the possible modular construction applications in the SBWR. Specific structural components proposed for modular construction approaches include:

(a) Reinforcing bar assemblies for the basemat, building and containment walls, drywell and suppression chamber slabs, containment top slab, columns, floor slabs and beams.

(b) Structural steel assemblies for the Reactor Building and Turbine Building superstructures. These modules will include roof trusses and siding.

(c) Steel structures that will also serve as forms for the turbine pedestal, drywell vent wall and RPV vessel.

(d) Equipment assemblies containing components such as piping, condensers, cranes, diesel generators, HVAC units and numerous other equipment. These modules will be applied in the Reactor, Turbine and Radwaste Buildings.

(e) Precast walls in the Reactor, Turbine, and Radwaste Buildings.

Reinforcing bar modules for the basemat, columns, walls, and beams will be prefabricated and lifted into position with cranes. Structural steel modules will be lifted above the operating floor to construct the steel superstructure. The containment wall and pool liners will be prefabricated and installed as modules. Numerous steel structures inside containment will be placed into position and later filled with concrete. This type of modularization will be used for the reactor pedestal, diaphragm floor, wall between drywell and suppression chamber and the GDCS pool walls.

Large composite modules will be used for the superstructure in the region above the grade clean area of the Reactor Building which houses the electrical and HVAC rooms. The large composite modules will contain a structural steel frame, precast siding panels, equipment and connecting piping, ducts and cabling. These modules will be assembled in a site fabrication area from smaller modules and components fabricated locally.

At this point in time the entire scope and content of the modularization effort is very preliminary. Recent indications from GE are that scope and scale of the modules may be reduced and possibly significantly reduced. The suggested code and standard changes required to address the use of modularization in the Westinghouse AP600 should be sufficient to address the majority of modular construction applications in the SBWR. The one exception is that for the SBWR, changes will be required in ASME BPVC Section III, Subsection CC in addition to ANSI/AISC N690 and ACI-349. However, there is insufficient detail currently available to develop suggested standards changes.

3.3.4.3 Ferritic Steel Welding Control

NUREG-0800 Section 5.2.3 presents a criterion (criterion 3.b(3) of Subsection II) for control of ferritic steel welding based on conformance to Reg. Guide 1.71 "Welding Qualification for Areas of Limited Accessibility," The SBWR proposes to use an alternative approach which meets the intent of Reg. Guide 1.71. Modifications should be made to the applicable welding code: AWS D1.1 to insure it is consistent with Reg. Guide 1.71 and allow the use of the SBWR criteria. Similar changes are also be suggested for to ASME IX.

3.3.4.4 Concrete Strength Reductions

The SBWR SSAR provides a criteria to derate the design allowable yield stress and the allowable design strength on concrete structures for temperature increases between 70°F and 150°F. It has not been common practice to derate concrete strength up to temperature limits defined by the Subsection CC Standard since capacity reduction of this type have been included in the 0.9 factor applied to the specified minimum yield strength of the reinforcing steel. Similarity ACI-349 strength deration are generally not used below 150°F. It is the authors' belief this derating is not necessary and no recommended changes are developed for the applicable standards.

3.3.4.5 Use of AISC N690

This specification has not been generically accepted by the USNRC. Further as will be discussed in Section 3.4, the USNRC has several concerns with the application of the current version. Therefore, changes of the specification are suggested to address these USNRC concerns.

3.3.4.6 Use of ASCE 4-86

ASCE 4-86 has not been generically accepted by the USNRC for use in the seismic analysis of nuclear power plant structures. It does in general agree with the seismic analysis requirements of the SRP and is referenced in several instances by Section 3.7.2 of the SRP. Minor changes to ASCE 4-86 are suggested to make it consistent with the SRP requirements.

3.3.4.7 Seismic Hydrodynamic Loadings

The SBWR uses a pressure suppression type of containment which includes a suppression pool inside the reactor building. During an earthquake there will be significant seismic excitation of the reactor building which will cause excitation of the water in the suppression pool resulting in direct and building filtered hydrodynamic loadings being applied to the reactor building and internal structures. Analysis techniques for these types of loadings currently do not existing in ASCE 4-86. Criteria for this type of analysis is currently being developed by the ASME (Appendix N), ASCE (Dynamic Analysis Committee) and the Department of Energy (Tank Seismic Experts Panel). Changes to ASCE 4-86 are proposed based on reviewing these three documents. Further it is suggested these loadings be incorporated in the applicable design loadings for the reactor building, containment and containment internal structures as part of the seismic input loadings.

3.3.4.8 Hydrodynamic Loads

The evaluation requirements for hydrodynamic loads resulting from the use of a pressure suppression containment is discussed in Section 3.5.2 and as shown in Table 3.5.2.1.

Table 3.3.4 - Unique Features and Attributes of the GE SBWR

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
RCCV	Reactor Building Containment Vessel (Reinforced Concrete with Steel Liner)	N/I[1]	N/I[1]	1.Elimination of OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	1. Use of ASCE 4-86	N/I[1]	N/I[1]
Reactor Building Structure	Reinforced Concrete and Steel Structure	1. Modular Construction	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Concrete Strength Reduction between 70° and 150°F 4. Seismic Hydrodynamic Loadings 5. Hydrodynamic Loadings	1. Use of ASCE 4-86	1. Special Ferritic Steel Welding 2. Modular Construction	N/I[1]

Table 3.3.4 - Unique Features and Attributes of the GE SBWR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Reactor Pedestal	Reinforced Concrete	1. Modular Construction	N/I(1)	1.Elimination of the OBE 2. Concrete Strength Reduction Between 70° and 150°F 3. Seismic Hydrodynamic Loadings 4. Hydrodynamic Loadings	1. Use of ASCE 4-86	1. Modular Construction	N/I(1)
Reactor Shield Wall	Structural Steel	1. Modular Construction	N/I(1)	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic Loadings 4. Hydrodynamic Loadings	1. Use of ASCE 4-86	1. Modular Construction	N/I(1)

Table 3.3.4 - Unique Features and Attributes of the GE SBWR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Basemat	Under RPV Pedestal Supports Entire Reactor Building	N/I[1]	N/I[1]	1. Elimination of OBE 2. Concrete Strength Reduction between 70° and 150°F 4. Seismic Hydrodynamic Loadings 5. Hydrodynamic Loadings	1. Use of ASCE 4-86	N/I[1]	N/I[1]
RVST	Reactor Vessel Stabilizer Truss	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic Loadings 4. Hydrodynamic Loadings	1. Use of ASCE 4-86	N/I[1]	N/I[1]
DGPSS	Support Platforms/Steel for Piping, Equipment, etc.	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Seismic Hydrodynamic Loadings 4. Hydrodynamic Loadings	1. Use of ASCE 4-86	N/I[1]	N/I[1]

Table 3.3.4 - Unique Features and Attributes of the GE SBWR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Drywell Airlocks	Upper and Lower	N/I(1)	N/I(1)	1. Elimination of the OBE 2. Level D Fatigue and Fatigue Ratchet Requirements for SSE 3. Seismic Hydrodynamic Loadings 4. Hydrodynamic Loadings	Use of ASCE 4-86	N/I(1)	N/I(1)
Drywell Head	Steel Structure	N/I(1)	N/I(1)	1. Elimination of OBE 2. Level D Fatigue and Fatigue Ratchet Requirements for SSE 3. Seismic Hydrodynamic Loadings 4. Hydrodynamic Loadings	1. Use of ASCE 4-86	N/I(1)	N/I(1)

Table 3.3.4 - Unique Features and Attributes of the GE SBWR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Diaphragm Floor	Barrier Between Drywell and Suppression Chamber	1. Modular Construction	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Concrete Strength Reduction between 70° and 150°F 4. Seismic Hydrodynamic Loadings 5. Hydrodynamic Loadings	1. Use of ASCE 4-86	1. Modular Construction	N/I[1]
GDCS Pool	Steel Lined Reinforced Concrete Structure	1. Modular Construction	N/I[1]	1. Elimination of the OBE 2. Use of AISC N690 3. Concrete Strength Reduction Between 70° and 150°F 4. Seismic Hydrodynamic Loadings 5. Hydrodynamic Loadings	1. Use of ASCE 4-86	1. Modular Construction	N/I[1]

Table 3.3.4 - Unique Features and Attributes of the GE SBWR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
PCCS	Passive Containment Cooling System	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]
Fuel Racks	N/A[2]	N/I[1]	N/I[1]	1. Use of ASME-III-NF for Fuel Racks	N/I[1]	N/I[1]	N/I[1]
HVAC Supports	Category I, Safety Related Distribution System Supports	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]
Cable Tray Supports	Category I, Safety Related Distribution System Supports	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]

Table 3.3.4 - Unique Features and Attributes of the GE SBWR (continued)

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication / Construction	Testing
Fire Protection	Barriers	N/I[1]	N/I[1]	1.Elimination of the OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]
	Safety Related Piping System Supports	N/I[1]	N/I[1]	1. Elimination of the OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]
Piping System Supports	Category I, Safety Related Distribution System Supports	N/I[1]	N/I[1]	1 Elimination of the OBE 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings 4. Use of AISC N690	N/I[1]	N/I[1]	N/I[1]
Containment Leak Testing	Periodic Leak Testing	N/I[1]	N/I[1]	1. Use of ANS 56.8 2. Use of ANSI N45.5	N/I[1]	N/I[1]	1. Use of ANS 56.8 2. Use of ANSI N45.4

Footnotes for Table 3.3.4:

(1) N/I = None Indicated
(2) N/A = Not Applicable

3.3.5 DOE/GA MHTGR

The MHTGR is the advanced reactor design which exhibits some of the most unique features and attributes of any of the advanced reactors. First the reactor is a gas cooled design using helium as the operating heat transfer medium. The reactor operates at high temperatures (approximately 650°C - 700°C) when compared to the current and advanced PWR and BWR reactor designs. Further the gas to steam conversion cycle currently discussed in the existing licensing documentation and information may be replaced with a direct gas turbine cycle. This change will have significant impact on the systems, components, and operational design but should have minimal impact on the structural design and structural design codes.

The unique design aspects which has the most "impact" on the structural design area is the Safety Classification approach for structures, systems, and components. The MHTGR design approach uses a probabilistic approach to establish the safety classification of structures, systems and components as opposed to the more deterministic approach used in current operating nuclear plants. The significance of this approach is discussed in depth in subsequent subsections.

Table 3.3.5 provides a tabular summary of the unique features and/or attributes associated with the Seismic Category, Safety Class Structures. The most significant of the unique features and/or attributes are:

- Use of a unique safety classification approach for systems, structures, and components.
- Lack of a pressure retaining primary containment structure and the use of ACI-349 for design of a confinement structure.
- Potential high (>150°F) concrete temperatures
- Potential high (800°F) distribution support temperatures
- Deeply soil embedded reactor and other buildings
- Maintenance structure and its design philosophy.

The features listed above will be discussed in greater depth in the following subsections. The remaining items listed in Table 3.3.5 will not be explicitly discussed but were identified for possible future reference.

3.3.5.1 Safety Classification Approach

The MHTGR does not approach safety classification and Seismic categorization using a deterministic multi-level (SC-1, SC-2, SC-3) approach as put forth in Regulatory Guideline 1.26, appropriate sections of NUREG-0800 (SRP) and the ANS Standards (ANS 51.1, ANS 52.1). These criteria and standards classify plant events into 5 Plant Conditions (PC's) based on the "best estimate frequency of occurrence" with the objective that the more likely the event, the lower should be the resulting consequences of the event. For each plant condition offsite radiological dose criteria are established using the requirements of 10CFR50, Appendix I and factored values of the requirements of 10CFR100. Plant equipment must be assigned to one of four safety classes based on clearly specified definitions and classification requirements. [For example, it is required that all portions of the RCS pressure boundary be assigned Safety Class 1 (Quality Class A), irregardless of size, consequences of failure, or function.] Further in assigning these safety classes and the associated safety class boundaries a single failure criteria (as discussed in ANS 51.1 and ANS 52.2) and redundancy must be considered in the establishment of safety classification. Each plant is assumed to have a pressure retaining secondary containment structure which is assigned Safety Class 2. These documents also prescribe the industry codes and standards applicable to the design of each safety class item.

For the MHTGR Structures, Systems, and Components (SSC) are classified as either safety (only one Safety Class) or non safety with some consideration given to preventing non-safety SSC from failing safety class systems, structures and components. The methodology used to determine the safety class items can be summarized in a simplified manner as follows:

(1) Identification of three (3) categories of plant design basis events. (using probablistic risk assessment)

(2) Determination of radionuclide release rates for each of the events in each of the categories of (1).

(3) Establishment of acceptable radionuclide exposure levels for each of the events in each of the categories of (1). (Using 10CFR100 dosage guidelines)

(4) Defining as Safety Class those SSC required to insure the radionuclide

releases for a given event are less than those defined in (3).

When items are identified as Safety Class the "prescriptive set of industry codes and standards associated with LWR and ALWR safety-related equipment is not automatically applied to safety-related MHTGR SSCs. Rather, appropriate analyses and trade studies, including relevant probabilistic risk assessment, are utilized to determine appropriate industry codes, regulatory guidance and quality assurance (QA) programs for MHTGR safety related equipment." Also the MHTGR does not contain a pressure retaining secondary containment as do the ALWR designs. This item is discussed in Section 3.3.5.2.

It is also an unique feature that core damage frequency does not appear to be a fundamental evaluation parameter. This is because a fundamental design criterion is that the fuel temperature will not exceed acceptable values. Further the MHTGR criterion does not appear to consider single failures nor apply any requirement for redundant functionality or safeguards.

DOE has also provided several reports which "bridge" or demonstrate how this approach provides the same level of safety provided by the more traditional approaches currently used in SSC Safety Classification. This approach, if accepted by the USNRC, would result in the need for changes to many of the industry codes and standards currently associated with nuclear plant design. These changes would be required to relate the DOE/GA Safety Classification approach to the appropriate design levels and design requirements. Further load combinations and load concurrence in the subject codes and standards would require modification to be consistent with DOE/GA load concurrence assumed in the MHTGR event analysis.

For seismic categorization all SC items are identified as Seismic Design Required. They are not explicitly identified as seismic Category I, but the seismic design requirements currently discussed in the MHTGR PSID are consistent with USNRC requirements for Seismic Category I items (Regulatory Guideline 1.60 Response Spectra, etc.) For non safety items they are seismically designed to (UBC) Zone 2B. If however a non safety related item could be postulated to cause the failure of a safety class SSC then it will be designated as Safety Impact and a seismic interaction design and evaluation would be required.

3.3.5.2 Containment and Confinement Structures

As a result of the safety classification approach discussed in Section 3.3.5.1, a "Classical" pressure retaining containment structure built to the requirements of ASME III, Div. 1 Subsection NE or ASME III, Div. 2 Subsection CC is not used. This results in the ACI-349 Code being used for design of a "confinement" structure. Based on Stevenson and Associates experience with similar applications of ACI-349 in the DOE "Defense Weapons Complex" changes are required to ACI-349 to insure it is adequate as a "confinement" design code. Further the leak monitoring and testing standards which are currently applied in conjunction with a pressure retaining containment structure would require modification or new standards would have to be written for this DOE confinement approach. This item has further significance, in that, the control and confinement of the helium is much more difficult than control or confinement of steam used in the typical steam conversion cycles of the ALWR reactor designs.

3.3.5.3 Comment on Safety Classification Approach

It is not the intent of Section 3.3.5.1 and 3.3.5.2 to imply there is anything technically incorrect with the approach used by DOE/GA in the establishment of safety classification for SSC in the MHTGR. It is merely to point out that it is significantly different from current nuclear plant SSC safety classification. If it is accepted by the USNRC it will require numerous changes to safety related design codes, standards and specifications.

3.3.5.4 Elimination of the OBE

The USNRC is in the process of issuing revisions to 10CFR100.23 and issuing a 10CFR50, Appendix S which will state that if the review level earthquake (OBE) is defined as less than 1/3 of safe-shutdown no explicit design analysis for the OBE level earthquake will be required. This change is tailored toward <u>new</u> reactor designs however it has not yet been incorporated in the SSAR (design basis) for the MHTGR. This unique design feature if implemented will require changes to all the primary industry consensus codes and standards which are the subject of this study. The changes are wide ranging but include the elimination of the OBE from load combinations, the potential need to address seismic anchor motions effects for the SSE event, etc. The code required changes are discussed in more detail in Section 4.0.

3.3.5.5 High Temperature Issues

The operating temperature of the primary reactor vessel is 650°C - 700°C [1200°F - 1300°F]. This vessel is contained in a concrete silo embedded in the ground. The annular space between the vessel and silo appears to be approximately 6 feet. Therefore depending on the effectiveness of the reactor vessel insulation and cooling system the concrete reactor building could experience a long term high temperature environment. Changes to the design, fabrication (mixtures), and other associated codes used in the reactor building construction are required to insure these structures can withstand these potential long term high temperature effects. Further, distribution system supports could also be exposed to temperatures in excess of 480°C. This 480°C is the limit of many of the existing structural design codes (AISC-ASD, AISC-N690, ASME III). Therefore these codes will require modification should distribution supports be subjected to this high temperature environment.

3.3.5.6 Deeply Embedded Structures - Soil Structure Interaction Considerations for Seismic Loadings

The reactor silo, and a significant portion of the entire reactor building will be below grade. During seismic events the soil structure interaction for these types of deeply embedded structures will be significant. This will require complex and extensive consideration of soil structure interaction in the seismic analysis. While no specific code and standard has been specified for seismic analysis of the MHTGR it is very likely the ASCE 4-86 would be applied. The ASCE 4-86 Soil Structure Interaction (SSI) Analysis Methodology is primarily based on structures which have only embedded foundations. Therefore this section of ASCE 4-86 should be reviewed for applicability to the SSI analysis of deeply embedded structures and changes may be required. In addition the Reactor Silo flexibility must be addressed in this SSI analysis.

3.3.5.7 Maintenance Structure

The maintenance structure is a structural steel building (steel frame and panels) which encloses the Reactor Building, Reactor Service Building and Reactor Auxiliary Building above grade. It is stated that the maintenance enclosure is designed not to collapse under design basis conditions. Collapse is not clearly defined. Further design basis conditions are not clearly defined. Finally the Design Code is ANSI 5326 (AISC-SCM) which is not a nuclear related structural design code and standard. This feature while unique and of concern will not affect the codes and standards associated with this program unless component collapse and impact loads need to be considered on the safety related buildings within the maintenance structure.

Table 3.3.5 - Unique Features and Attributes of the DOE/GA MHTGR							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Vessel Support	Reactor Pressure Vessel Support Structure	1. Potential High Support Temperature	N/I(1)	1. High Support Operating Temperature	N/I(1)	N/I(1)	N/I(1)
Reactor Building	Reinforced Concrete Cylinder with a Flat Concrete Slab Base and Top	1. No Containment Structure 2. Confinement versus Containment Structure 3. High Concrete Temperatures	N/I(1)	1. Unique Safety Classification Bases versus Traditional Deterministic Approach 2. Use of ACI 349 as Confinement Design Code 3. Use of AISC-ASD for safety related structures 4. High Concrete and Steel Temperatures	1. Deeply embedded structure SSI analysis issues.	1. Use of AISC-ASD for Safety Related Steel Structures	1. Leakage monitoring and detection will be significantly different from current reactors.
Reactor Service Building	Reinforced Concrete and Structural Steel	N/I(1)	N/I(1)	1. Unique Safety Classification Approach versus Traditional Deterministic Approach 2. Use of AISC-ASD for safety related nuclear structures	N/I(1)	1. Use of AISC-ASD for Safety Related Steel Structures	N/I(1)

Table 3.3.5 - Unique Features and Attributes of the DOE/GA MHTGR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Auxiliary Building	Reinforced Concrete and Structural Steel	N/I[1]	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach 2. Use of AISC-ASD for safety related nuclear structures	N/I[1]	1. Use of AISC-ASD for Safety Related Steel Structures	N/I[1]
Reactor Cavity Cooling Panels	Intake/Exhaust Structures	1. Potential high temperature application		1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]
Reactor Cavity Cooling Panels	Plenum Structures	1. Potential high temperature application		1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]

Table 3.3.5 - Unique Features and Attributes of the DOE/GA MHTGR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Cavity Cooling Panels	Ducting Supports	1. Potential high temperature application	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]
Essential Uninterrupted Power System Supply	Cables/Trays/Conduit Supports	N/I[1]	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]
Essential DC Power System	Cable Trays & Conduit Supports	N/I[1]	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]

Table 3.3.5 - Unique Features and Attributes of the DOE/GA MHTGR (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Fire Protection System	Piping Supports	N/I[1]	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]
	Fire Barriers	N/I[1]	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/I[1]	N/I[1]	N/I[1]
Pipe Supports	Structural Steel and Component Standard	1. Potential High Temperature Application.	N/I[1]	1. Unique Safety Classification Approach versus Traditional Deterministic Approach	N/A	N/A	N/A

Footnotes for Table 3.3.5:

[1] N/I = None Identified

3.3.6 ABB PIUS

The PIUS Reactor design contains many innovative features in terms of passive safety systems and the overall reactor design and operation. As with the evolutionary reactors the majority of the innovative features are related to systems, components, and operation features. Unfortunately the available design basis information provides very few specific details on the structural design aspects or requirements for this facility. The safety classification system implies the use of a standard similar to ANS 51.1 and appears consistent with that standard. The seismic classification for the PIUS reactor uses the basic two seismic categories put forth in Regulatory Guideline 1.29, I or N. However the PIUS design adds a third class which is a subset of Seismic Category I which is a seismic Category I-P or P. This system appears consistent with domestic United States Nuclear Plant seismic classification and the classes are as defined below.

Seismic Category I is the highest class with the highest seismic demands. It is normally applied to systems, for which proper function is important for the safe shutdown and cooling of the reactor and/or for the containment integrity.

Seismic Category I-P (or P) is the next class and is a subset of the I class. It is applied to systems or parts of systems, for which only passive functions or structural integrity must be ensured in earthquake situations but there is no requirement for an active function.

Seismic Category N is the third class (Non-Seismic Category) for which no specific requirements on seismic capability are imposed as regards safety as is typical with current practice. Category N equipment must not jeopardize the function or integrity of any Category I or I-P equipment.

Based on the available data the following structural items are identified as unique features or attributes of this reactor.

- The use of Swedish National Standards for the reactor design
- Concrete Reactor Vessel integrated into the reactor containment building
- Containment Structure Design
- Seismic Hydrodynamic Loadings

3.3.6.1 Use of Swedish National Codes and Standards

For design, qualification and construction of the PIUS reactor ABB normally uses Swedish or European codes and standards. The design information supplied in the referenced PSID is based on the Swedish (European) standards. This includes design and construction in the following major areas:

- Cement quality
- Aggregate composition, mixing, transport and pouring
- Reinforcement steel
- Structural steel
- Form work and finish
- Tolerances, etc.

However, the prestressed concrete vessel is designed according to ASME III, Division 2 and the concrete containment is also based on ASME III, Division 2.

Several times in the PSID it is stated that the PIUS design will comply with the appropriate US industry codes and standards, but no US industry codes and standards are explicitly specified. It is the stated intent of the ABB to develop a comparison with applicable US Industry Codes and Standards and to provide this comparison during the detail design stage. However, this comparison was not available for review for this program. An in depth comparison of these codes and standards to U.S. industry consensus codes and standards is beyond the scope of this program for the advanced reactor designs. For now, it is simply identified as an unique design feature. However, should ABB decide to pursue an active licensing status with this reactor design, an indepth comparison of the Swedish/European Codes and Standards to the U.S. Industry Codes and Standards will be required.

3.3.6.2 Concrete Reactor Vessel and Integrated Reactor Building

There are currently no operating commercial power reactors with concrete vessels. A small test power reactor, Ft. St. Vrain, did have a concrete vessel but it is now permanently shutdown. Therefore while a concrete vessel is not unique it is certainly a novel approach when compared to current commercial power reactor designs. Further, since ASME III Division 2, Subsection CB has never been applied to a commercial power reactor design, it is probable that changes will be required to this code for this application. The identification of code changes specifically for this advanced reactor is (1) beyond the scope of this program and is (2) not possible due to the lack of definitive design information provided. Such a review should be conducted if ABB decides to actively pursue licensing this reactor design.

The "Reactor Building" is integrally attached to the Reactor Vessel and is identified as a pressure suppression containment. It is appears that the building design will be ASME III, Division 2, Subsection CC. Therefore, both ASME III, Division 2, Subsection CB and ASME III, Division 2, Subsection CC may require changes to address this interface and interaction. Further, the outer wall of the concrete reactor vessel may be the part of the pressure suppression containment which further complicates the design interface issue.

3.3.6.3 Containment Structure Design

The "Reactor building" is identified as housing the pressure suppression containment. However portions of the building which are not part of this pressure suppression containment may be designed to structural concrete design codes versus pressure retaining containment design codes. The containment appears similar to a typical BWR Mark III pressure suppression containment design. It is implied that this containment may be designed to ASME III Subsection CC but since Swedish codes and standards were used for design, it is not clear this will occur. If portions of the containment are designed to structural concrete codes such as ACI-349 then these structural concrete codes may require modification to address "containment or confinement type" functions and loadings that the buildings are required to perform. Further new radiation leakage monitoring standards may be required for these structures.

3.3.6.4 Seismic Hydrodynamic Loadings

With the use of pressure suppression containment design effects of Seismic Excitation of the suppression pool (as required for the AP600, ABWR, and SBWR) will also be required for this reaction design. The code and standard changes required for the AP600, ABWR, and SBWR should be sufficient to adequately address the demand and capacity evaluation of these loadings on this reactor design.

3.3.6.5 Hydrodynamic Loadings

The design for hydrodynamic loads resulting from the use of a pressure suppression containment is discussed in Section 3.5.2 and shown in Table 3.5.2.1.

Table 3.3.6 - Unique Features and Attributes of the ABB PIUS							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Building	Contains Safety Related Structures (Prestressed Concrete Cylinder with Concrete Flat Bottom Top)	N/I[1]	1. May be using a mixture of "containment " design codes and standard concrete design codes	1. Use of Swedish Codes and Standards 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Control SVCS Building	Central Control Room	N/I[1]	N/I[1]	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Concrete Vessel	Monolith	N/I[1]	N/I[1]	1. Seismic Hydrodynamic Loadings 2. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]
Concrete Vessel Liner	Steel	N/I[1]	N/I[1]	1. Seismic Hydrodynamic Loadings 2. Hydrodynamic Loadings	N/I[1]	N/I[1]	N/I[1]

Table 3.3.6 - Unique Features and Attributes of the ABB PIUS (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Concrete Containment	Pressure Suppression Containment which is part of the Reactor Building	1. Not clear (from a construction standpoint) how this interfaces with Reactor Building	N/I[1]	1. Use of Swedish Codes and Standards 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Reactor Supports	Safety Related RPV Supports	N/I[1]	N/I[1]	1. Use of Swedish Codes and Standards 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Reactor Pool	Fills Concrete Vessel Cavity between Reactor Internal Assembly and Concrete Vessel.	N/I[1]	N/I[1]	1. Use of Swedish Codes and Standards 2. Seismic Hydrodynamic Loadings 3. Hydrodynamic Loadings	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Cooling Towers for Reactor Pool Coolers	Natural Draft Cooling Towers on Top of Reactor Building.	N/I[1]	N/I[1]	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	1. Use Swedish Codes and Standards	N/I[1]

Table 3.3.6 - Unique Features and Attributes of the ABB PIUS (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Piping System Supports	Seismic Category I Supports	N/I[1]	N/I[1]	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Cable Tray Supports	Seismic Category I	N/I[1]	N/I[1]	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	1. Use of Swedish Codes and Standards	N/I[1]
Containment Leak Testing	Leaking Testing shall be per Appendix J to US 10CFR Part 50	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]

Footnotes for Table 3.3.6:

[1] N/I = None Identified

3.3.7 DOE/GE PRISM

The available design basis documentation on the PRISM Reactor is somewhat dated in that it was issued in December 1987. The reactor is unique in that it uses liquid sodium as the reactor coolant loop working median versus steam conversion cycles used in most of the evolutionary and advanced reactors. While the reference documentation is somewhat vague it appears that a majority of the buildings and structures will follow standard construction practice, the intent is to use ACI-349 for concrete construction but to use the AISC Steel Construction Manual for Steel Design (AISC-ASD) in lieu of AISC N690. Also the current Seismic Design Basis will be for both a Level B Operating Basis Earthquake and a Level D Safe Shutdown Earthquake. The Seismic Categorization and Safety Classification appears consistent with current industry procedures and uses 10CFR50.55(a) and Regulatory Guide 1.26. It is stated that some 'interpretations" in this criteria are required due to the differences in Liquid Metal Reactors and Light Water Reactors. However, there was no explicit discussions of these interpretations.

Table 3.3.7 provides a summary of these structural design and construction which are unique to this reactor design. The following summarizes the most significant of these features:

- Use of Modular Construction
- Deeply Embedded Buildings
- Use of Seismic Isolators
- Use of AISC-ASD for Design of Safety Related Steel Structures
- Containment Vessel Design Standards

The following sections provide more detailed information on these items.

3.3.7.1 Use of Modular Construction

While details are limited in the available documentation it appears that significant portions of the PRISM Design will be based on the use of true modular construction. However, the actual use of this modular construction is not discussed in any detail in the available Design Basis information. The reactor modules will be a standard design that would be built at a fabricator(s) shop and could be shippable by rail. Once more information of these modular construction details is available, the impact on the applicable industry codes and standards should be reviewed.

3.3.7.2 Deeply Embedded Buildings

The Reactor Silo, essentially the entire reactor building, and a significant portion of the Steam Generator Building will be below grade. For the analysis of seismic events the soil structure interaction effects for these types of deeply embedded structures will have a significant influence on the analytically predicted seismic response. The current Seismic Structural Analysis Criteria is per Bechtel Power Corporation Topical Report BC-TOP-4A, Revision 3, 10/31/74 versus the ASCE 4-86 Standard. This document was not available for review as part of this program. However, the Soil Structure Interaction analysis techniques in this standard as well as ASCE 4-86 which is the industry consensus standard for seismic structural analysis, should be evaluated for applicability to the the analysis of this structure. The ASCE 4-86 Soil Structure Interaction methodology is primarily for structures which have only embedded foundations. Therefore this section of ASCE 4-86 should be reviewed for applicability to deeply embedded structures and some changes may be requested to this standard. In addition the Reactor Silo flexibility must be addressed in the SSI analysis and this requirement should also be addressed in ASCE 4-86.

3.3.7.3 Seismic Isolators

The PRISM Design uses large high damping non-linear natural rubber bearings as seismic isolators to isolate the PRISM Reactor Module from the reactor building. There are currently no industry consensus codes and standards which cover the design of such isolators. Either existing standards must be modified or new standards developed to address design and fabrication of this support.

3.3.7.4 Use of AISC-ASD

The AISC-ASD (Steel Construction Manual) is a commercial design code. It was used for the design of structural steel in many of the currently operating domestic United States Nuclear Power Plants. Its use, however, requires additional stipulations on design criteria, materials, weld inspections, quality assurance, etc. Most ALWR and advanced reactor designs specify the use of AISC N690 for the design and construction of safety related steel structures. Therefore, the use of the AISC-ASD is identified as an unique feature for this design.

3.3.7.5 Containment Vessel Design Standards

The Reactor Containment Vessel and Closure Head are being designed to ASME BPVC Section III, Division 1, Subsection NB. This is an upgrade in design criteria from Subsection NE (Class MC) which is normally used in the design of reactor Containment Vessels and closures. Also the containment vessel is Safety Class 1 versus Safety Class 2 which is traditionally used for pressure retaining containment vessels. While these features are unique aspects of this design, they will not require any additional changes to the codes and standards being used in the design and construction of these items.

Table 3.3.7 - Unique Features and Attributes of the DOE/GE PRISM Reactor

Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Vessel		1. Modular Construction 2. Reactor Vessel and Containment Vessel Appear to be an Integral Structure	N/I[1]	N/I[1]	N/I[1]	1. Modular Construction	N/I[1]
Containment Vessel		1. Use of Seismic Isolators 2. Modular Design and Construction 3. Reactor Vessel and Containment Vessel Appear to be an Integral Structure	1. Use of Section NB for Design and Fabrication	1. Use of Section NB for Design and Fabrication	1. Use of Section NB for Design and Fabrication	1. Modular Construction 2. Use of Section NB for Design and Fabrication	N/I[1]
Reactor Containment Vessel Closure Head		1. Use of Seismic Isolators 2. Modular Construction 3. Reactor Vessel and Containment Vessel Appear to be an Integral Structure	1. Use of Section NB for Design and Fabrication	1. Use of Section NB for Design and Fabrication	1. Use of Section NB for Design and Fabrication	1. Modular Construction 2. Use of Section NB for Design and Fabrication	N/I[1]
<u>Reactor Building</u> HAAC	Head Access Area Closure	1. Use of Seismic Isolators 2. Modular Construction	N/I[1]	N/I[1]	N/I[1]	1. Modular Construction	N/I[1]

Table 3.3.7 - Unique Features and Attributes of the DOE/GE PRISM Reactor (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Building Reactor Silo	Reactor Silo, Reinforced Concrete Structure	1. Use of Seismic Isolators 2. Modular Construction	N/I[1]	1. Use of AISC-ASD for Steel Design	N/I[1]	N/I[1]	N/I[1]
Reactor Building E&IV	Electrical & Instrument Vaults	1. Use of Seismic Isolators 2. Modular Construction	N/I[1]	N/I[1]	N/I[1]	1. Use of Modular Construction	N/I[1]
Reactor Building (SDTV)	Primary sodium Processing and Sodium Drain Tank Vaults	1. Use of Seismic Isolators 2. Modular Construction	N/I[1]	N/I[1]	N/I[1]	1. Use of Modular Construction	N/I[1]
Reactor Building RVACS	Inlet & Outlet Duct Supports	1. Use of Seismic Isolators 2. Modular Construction	N/I[1]	Use of AISC-ASD for Steel Design	N/I[1]	1. Use of Modular Construction	N/I[1]
Reactor Building RVACS	Horizontal Plenums	1. Use of Seismic Isolators	N/I[1]	N/I[1]	N/I[1]	1. Use of Modular Construction	N/I[1]
Reactor Building RVACS	Collection Cylinder	1. Use of Seismic Isolators	N/I[1]	N/I[1]	N/I[1]	1. Use of Modular Construction	N/I[1]
Reactor Building RVACS	Shielding Concrete	1. Use of Seismic Isolators	N/I[1]	N/I[1]	N/I[1]	1. Use of Modular Construction	N/I[1]

Table 3.3.7 - Unique Features and Attributes of the DOE/GE PRISM Reactor (continued)							
Structure	Description	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Reactor Building Seismic Isolators	Seismic Isolators	1. Use of Seismic Isolators	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
Radioactive Waste Buildings	Ground Floor & Curbs	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
Mobile Refueling Enclosure	Wall & Roof Steel Frame	N/I[1]	N/I[1]	1. Use of AISC - ASD for Design of a Safety Class Structure	N/I[1]	N/I[1]	N/I[1]
Mobile Refueling Enclosure	Bridge Crane	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
SC Cable Trays	Cable Tray Supports	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
SC Piping Systems	Piping Supports	N/I[1]	N/I[1]	1. ASME CC N47	1. ASME CC N47	N/I[1]	N/I[1]
Fire Protection Systems	Piping Supports	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]
	Fire Barriers	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]	N/I[1]

Footnotes for Table 3.3.7:

(1) N/I = None Identified

3.3.8 AECL CANDU-3U

The design documentation for CANDU-3U was very preliminary, lacking in detail, and incomplete. This limited the depth at which the unique structural features associated with this design could be reviewed. Table 3.3.8 provides a tabular summary of the unique features for this reactor design which could be identified from the data available. The most significant of these attributes are:

- The design is based exclusively on Canadian National Codes and Standards.
- A unique Seismic Design Criteria
- Potential Extensive use of Modular Construction (pre-fabrication)
- Unique Philosophy on Containment Leak Rate Monitoring

The features listed above will be discussed in greater depth in the following subsections. It should be further noted that by reading the available information, it is the appears that there are philosophical differences between CANDU-3U design basis and typical domestic United States Reactor design bases. There is not, however, sufficient data available to quantify this observation.

CANDU-3U systems are separated into two groups, Group 1 and Group 2. Group 1 systems are those used during normal operation of the plant. Group 2 systems are standby systems that operate to perform accident mitigating functions. The systems in each group are capable of shutting the reactor down, cooling the reactor, and monitoring plant conditions independent of the other group. The grouping and separation philosophy is described in detail in SSAR Section 3.1.2.

The Group 1 systems are designed and operated to prevent the occurrence of accidents and transients. Group 2 systems are designed to mitigate the effects of design basis accidents and the effects of severe widespread external events. The safety function of Group 1 systems comes from reliability considerations, that is, to assure the target frequencies of severe accidents is sufficiently low. The Group 1 systems may provide selected backup mitigating functions for some design basis events. These functions are defined and confirmed during the design by means of a probabilistic risk assessment. In such cases, Group 1 systems, and associated structures, are selectively qualified to assure their operability for an event in which their function is credited. Unlike the Group 1 systems, all Group 2 systems and structures are qualified for site hazards such as earthquakes and tornadoes.

3.3.8.1 Use of Canadian National Standards

All codes and standards discussed and presented in the reference CANDU-3U information are Canadian National Codes and Standards. In many cases these standards would be similar to the standards which are the subject of the review effort but it would require an in depth comparative study to establish the similarities and differences. Recent communications between the AECL and the USNRC implies the CANDU-3U intends to meet the intent of US Industry Codes and Standards for Group 2 SSC. The actual mechanism which this will be accomplished is unclear at the time of this report. However there does not currently appear to be Canadian National Standards which are comparable to ASCE 4-86 or AISC N690.

3.3.8.2 Unique Safety Classification System

Based on the currently available information structures, systems, and components are only classified as safety or non safety. No detailed subclassification (such as SC-1, SC-2, SC-3) appears to have been applied to structures, systems, and components at this time. It is not clear from the available documentation but it appears possible that a probabilistic approach versus deterministic approach was used in establishing these safety structures, systems, and components. However, some Canadian National Codes and Standards appear to reference the ASME BPVC Section III and therefore it is possible a further subclassification will be done at a later date.

3.3.8.3 Unique Seismic Design Criteria

The philosophy adopted in the CANDU-3U to satisfy seismic design requirements has the following features as described below:

Two earthquake levels are defined as envelopes for the design, in order to achieve the safety objective. These are:

> <u>The Design Basis Earthquake (DBE)</u>
>
> The Design Basis Earthquake means an engineering representation of the potentially severe effects of earthquakes applicable to the site that have sufficiently low probability of being exceeded during the lifetime of the plant. The DBE effects on the site are described by the DBE Ground Response Spectra (GRS). Its effects within structures at the site are described by Floor Response Spectra (FRS) which are

developed for selected locations in each structure.

Site Design Earthquake (SDE)

The Site Design Earthquake (SDE) means an engineering representation of the effects at the site of a set of possible earthquakes with an occurrence rate, based on historical records, not greater than 0.01 per year. The SDE effects on the site and within structures at the site are described by Ground Response Spectra and Floor Response Spectra.

The safety objectives of seismic design of a plant are two fold to ensure the following occurs:

Objective 1: Following a DBE:

•The reactor can be shutdown and maintained in the shutdown state.

•The fuel in the reactor can be cooled.

•The heat transport system integrity can be maintained for fuel cooling. (i.e., no Loss of Coolant Accident as a result of earthquake). Therefore no combination of LOCA and earthquake loads is considered in the CANDU-3U design basis.

•The containment boundary can be maintained and the associated systems remain operational.

•The plant can be controlled and monitored from a qualified area, the Secondary Control Area (SCA).

•The main control room (MCR) remains available to the extent necessary to protect the operator and a qualified route is provided for safe access to the secondary control area (SCA).

•Critical structures and systems outside containment are maintained so as not to cause radioactivity releases beyond allowable accident limits.

Objective 2: Following an SDE occurring 24 hours or more after a LOCA:

•The reactor fuel can continue to be cooled

•Essential variables can continue to be monitored from the SCA.

It appears that all structures, systems, and components within the reactor building require seismic qualification. The safety related structures, systems, and components outside the reactor building, which requires seismic qualification, are located in the physically distinct Group 2 area, which is separated from the Group 1 area which also includes the structures, systems and components directly associated with power production.

The structures, systems, and components in the Group 2 area of the plant are seismically qualified. Qualification complies with Canadian National Standards on Seismic Design.

A seismic survey of the plant is performed to establish that the as-built, as installed condition of the facility and its equipment satisfied the seismic qualification requirements. This is usually in the form of an in site visual inspection.

3.3.8.4 Modular Construction

The information discussed in meetings with members of the USNRC indicate that extensive modular construction may be used in the construction of the CANDU-3U. There is, however, insufficient information available to determine any specific details on the type and amount of modular construction which will be used. Further since the design codes being specified are Canadian National Codes and Standards is not certain that the changes proposed for AISC N690 and ACI-349, to address modular construction in the AP600 design, would be implemented on this reactor type design.

3.3.8.5 Unique Philosophy on Containment Leak Rate Monitoring

In this area the available documentation references a Canadian document CAN3-N287.6-M80 "Pre-Operational and Proof of Leakage Rate Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants" which was not available for review. From the text available, it appears that there are a number of substantial differences in philosophy between the CANDU-3U design basis and current US regulations leak rate testing and monitoring area.

Table 3.3.8 - Unique Features and Attributes of the AECL CANDU-3							
Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Special internals	20800	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	1. Unique and Different Philosophy on Leak Rate Monitoring
Reactor Building	21100	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	1. Unique and Different Philosophy on Leak Rate Monitoring
Reactor Auxiliary Building	21200	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	1.Unique and Different Philosophy on Leak Rate Monitoring

Table 3.3.8 - Unique Features and Attributes of the AECL CANDU-3 (continued)							
Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Group 2 Pumphouse	This includes Pumphouse (23500), Intake Channel & Structures (23600) and Outfill Channel and Structures (23700) Intake & Discharge Ducts, (23800) and Recirculation Structure (23900)	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Group 2 Building	24200	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Maintenance Building	25000	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]

Table 3.3.8 - Unique Features and Attributes of the AECL CANDU-3 (continued)							
Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Fuel Storage Structures	35200	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Safety Class Cable Trays Supports	57400	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Safety Class Piping System Supports	Safety Class Piping Distribution Supports	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Reactor Building Ventilation System Supports	HVAC Supports 73120	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]

Table 3.3.8 - Unique Features and Attributes of the AECL CANDU-3 (continued)

Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Irradiated Fuel Storage Bay Ventilation System - Supports	HVAC Supports 73160	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Group 2 Service Building	HVAC Supports 73310 73320 73330	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
Fire Protection Systems	Barriers	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]
	Supports 74200	1. Potential Application of Prefabrication or Modular Construction	N/I[2]	1. Use of Canadian Codes and Standards 2. Unique Safety Classification Philosophy 3. Different Seismic Design Criteria	N/I[2]	1. Use of Prefabrication/ Modular Construction	N/I[2]

Table 3.3.8 - Unique Features and Attributes of the AECL CANDU-3 (continued)							
Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Containment Leak Testing	Periodical Leak Testing	N/I[2]	N/I[2]	1. Use of Canadian Codes and Standards	N/I[2]	N/I[2]	1. Use of Canadian Codes and Standards

Footnotes for Table 3.3.8:

[1] These numbers represent the GSI numbers assigned in the Conceptual Safety Report, Appendix D1 of Volume I.
[2] N/I = None Identified

3.3.9 EPRI-URD

The EPRI -URD was reviewed in a cursory manor to identify any potentially highly unique suggested design approaches. This document provides its requirements in a general functional format versus actual implicit design requires. The majority of the unique aspects of the document, as with the ALWR reactor designs, are in the systems, components, and operational areas. The majority of the significant aspects of the structural design basis have been previously discussed in one or more of the ALWR reactor design sections. Table 3.3.9 provides a summary of some of the most relevant unique design features and attributes.

The EPRI-URD suggests the use of experience based seismic qualification for many systems, structures, and components. This suggestion is similar to the approach used for resolution of Unresolved Safety Issue A-46 (SQUG). This includes distribution systems and distribution system supports for piping system, supports for piping system, cable tray systems, HVAC systems, and equipment supports. This aspect is a unique feature for the EPRI-URD which will require modification to all the associated codes and standards.

Table 3.3.9 - Unique Features and Attributes of the EPRI-URD							
Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Steel Containment	N/R	1. Use of Modular and Other Advanced Construction Techniques	1. 60 Year Operational Life	1. Elimination of the OBE	1. Use of ASCE 4-86	N/I[2]	N/I[2]
Concrete Containment	N/R	1. Use of Modular and Other Advanced Construction Techniques	1. 60 Year Operational Life	1. Elimination of the OBE	1. Use of ASCE 4-86	N/I[2]	N/I[2]
Building Foundations	N/R	1. Use of Modular and Other Advanced Construction Techniques	1. 60 Year Operational Life	1. Elimination of the OBE	1. Use of ASCE 4-86	N/I[2]	N/I[2]
Steel Super Structures	N/R	1. Use of Modular and Other Advanced Construction Techniques	1. 60 Year Operational Life 2. Use of AISC N690	1. Use of AISC N690 2. Elimination of the OBE 3. Use of Probabilistic Methods for Load Combinations 4. Use of Factored Unidirectional Loads and Load Combinations	1. Use of ASCE 4-86	N/I[2]	N/I[2]
Concrete Super Structures	N/R	1. Use of Modular and Other Advanced Construction Techniques	1. 60 Year Operability Life 2. Use of AISC N690	1. Elimination of the OBE	1. Use of ASCE 4-86	N/I[2]	N/I[2]

Table 3.3.9 - Unique Features and Attributes of the EPRI-URD (continued)							
Structure	Description[1]	Unique Features and Attributes					
		Physical	Application	Design (Inc. Loads)	Analysis	Fabrication Construction	Testing
Piping Supports	N/R	1. Use of Modular and Other Advanced Construction Techniques	N/I[2]	1. Elimination of the OBE 2. Use of AISC N690	1. Use of ASCE 4-86	N/I[2]	N/I[2]
Cable Tray Supports	Safety Class/Seismic Category Tray Structures	N/I[2]	N/I[2]	1. Elimination of the OBE 2. Use of Experience Based Seismic Qualification	1. Use of ASCE 4-86	N/I[2]	N/I[2]
HVAC Supports	Safety Class/Seismic Category Duct Structures	N/I[2]	N/I[2]	1. Use of Experience Based Seismic Qualification 2. Use of ASME AG-1	1. Use of ASCE 4-86	N/I[2]	N/I[2]
Fire Protection Systems	Supports	N/I[2]	N/I[2]	1. Use of Experience Based Seismic Qualification	1. Use of ASCE 4-86	N/I[2]	N/I[2]
	Fire Barriers	N/I[2]	N/I[2]	N/I[2]	N/I[2]	N/I[2]	N/I[2]
Containment Leak Testing	Periodic Leak Testing	N/I[2]	N/I[2]	1. Use of ANS 56.8	N/I[2]	N/I[2]	1. Use of ANS 56.8 2. Change of Airlock Test Internal

Footnotes for Table 3.3.9:

(1) Structure Names are self explanatory and therefore since this is a generic criteria (design) specific descriptions are not required, N/R.
(2) N/I = None Identified

3.4 Identification of Regulatory (USNRC) Requirements and Concerns

In addition to the standard documents used to identify USNRC nuclear plant design criteria (CFR, NUREG-0800, Regulatory Guidelines, etc.) the main document used to define the USNRC design requirements for the Evolutionary Reactor designs was letter SECY-93-087. The majority of the features defined in this section are taken from that document. To a lesser extent some features and suggested changes not otherwise addressed in SECY-93-087 were taken from the other referenced USNRC documents.

3.4.1 SECY-93-087 and Related Documents Requirements

This section presents USNRC design requirements for the evolutionary reactor designs as put forth in USNRC letter SECY-93-087. The letter and the reference and attachments to the letter were used as a basis for the USNRC requirements for the evolutionary reactor designs.

3.4.1.1 Elimination of the OBE

The item has been discussed extensively in the unique feature review for the evolutionary reactor designs. This criteria change will require changes to all codes and standards which are the subject of this review program.

3.4.1.2 Additional Containment Design Requirements

The SECY-93-087 letter identifies additional severe accident design requirements for the containment systems. The accidents which have significance from a structural design stand point include:

- Hydrogen Deflagration
- Core Debris Coolability - Steam Explosion
- High Pressure Core Melt Injection
- Containment Bypass Pressure Suppression - Detailed Containment Vent

The severe accident evaluation requirements will require changes to the containment design codes. The changes will be primarily a definition of the appropriate capacity criteria which should be used in the evaluation of severe accidents to the extent that severe accidents are identified as a design basis. For example SECY-93-087 has specified a probability of failure limit of 0.1 for the containment under severe accident conditions. For steel containments (ASME-III, Class MC) it has been determined that Level C Service Limits would meet this required probability of failure. Similar criteria is needed in ASME III, Div. 2, Subsection CC (ACI-359) for concrete containments. In addition the probablity criteria will also necessitate that analysis criteria be added to Appendix N of ASME III to provide methodology to calculate applied loadings.

3.4.1.3 Tornado Design

The USNRC intends to accept a reduced tornado design criteria for the Advanced Light Water Reactor and Advanced Reactor Design namely:

300 mph vs 360 mph Wind Speed
2.0 psi vs 3.0 psi Pressure Drop

In current domestic United States operating nuclear power plants the tornado design-basis requirements have been used to establish structural requirements (such as minimum concrete wall thicknesses) to protect nuclear plant safety-related SSC against effects not explicitly addressed in regulatory guidance (such as RG or the SRP). Specifically, the staff has routinely reviewed and evaluated aviation crashes (involving general aviation light aircraft), nearby explosions, and explosion debris or missiles, taking into account the tornado protection requirements. Depending on how the design basis is established for these man-made hazard phenomena, the staff's acceptance of these reduced tornado criteria may now require explicit consideration of some external impact hazards such as small airplane crash and explosion (malevolent vehicle). Therefore essentially all the standards which are the subject of the program will require some modifications to address these additional loadings (aircraft crash, malevolent vehicle). Further standards similar to these associated with natural phenomenon hazards should be developed to define the applied loading from these events.

3.4.1.4 Containment Leak Testing

The applicable containment leak testing industry consensus codes and standards should be updated to conform with the requirements of SECY-93-087 and the guidelines of Draft Regulatory Guideline MS 021-5.

3.4.1.5 Shell Buckling

The USNRC has accepted ASME BPVC Code Case N-284 for the evaluation of containment shell buckling. This Code Case should be incorporated into ASME III, Subsection NE.

3.4.1.6 ACI-349 Appendix B

In Appendix F of NUREG-1503 the USNRC has presented several concerns with Appendix B of ACI-349. Changes are proposed to the Appendix B of ACI-349 to address the concerns put forth by the USNRC and to provide a generic anchorage qualification approach including use of manufacturer's standard anchorage for distribution system supports.

3.4.1.7 ANSI/AISC N690

The USNRC, in NUREG-1503, accepted AISC N690 for use on the ABWR but with significant exceptions as outlined in Appendix G of that document. In addition in NUREG-1462 it was also accepted for use on the ABB/CE System 80+ with similar exceptions. In both cases the NSSS vendors committed to comply with these exceptions. Changes to AISC N690 should be developed to address these exceptions and concerns or the exceptions published in a Regulatory Guideline similar to that which was done for ACI-349.

3.4.2 Other Sources of USNRC Guidance

This section provides other sources of possible code changes which should be made to increase the applicability of the subject codes and standards to ALWR and Advanced Reactor designs.

3.4.2.1 Regulatory Guide 1.26

This Regulatory Guideline provides guidance for classification of systems, structures, and components for Light Water Reactors. ANS 58.13-1993 should be reviewed as a replacement for Reg. Guide 1.26. Further similar ANS Standards should be written to address liquid metal, heavy water, and gas-cooled reactors. The recommendation of changes to these standards is outside the scope of this review but the suggestion is provided for consideration.

3.4.2.2 Regulatory Guide 1.57

Regulatory Guide 1.57 states: "neither Section III nor any other published code or national standard provides adequate guidance for safety combinations of loading for design or for identifying Seismic Category I components....". This statement applies to Metal Primary Containment System Components. For concrete containments such guidance is provided in ACI-359. Guidance for selecting combinations of loading for design of Seismic Category I metal containment system components should be considered similar to what is done for industry standards governing steel and concrete structures and concrete pressure retaining components. However, it has long been ASME BPVC policy to have such loads defined in the design specification by the owner. It may be possible to provide guidance on these load combinations in an existing or new non-mandatory appendix to the ASME BPVC, Section III. Alternatively such guidance could be provided in a revision to ANS standards 58.14 or 50.1 which are intended to replace both ANS 51.1 and 51.2.

3.4.2.3 Regulatory Guide 1.59

Regulatory Guide 1.59 states "techniques for evaluating the effects of tsunami will be presented in a future appendix." Techniques for evaluating effects of tsunami should be placed in an appropriate analysis code and the loads identified for evaluation in all the codes which are the subject of this review. An effort to draft an ANS Standard (ANS 3.4) on tsunami was previously initiated but was abandoned in 1984.

3.4.2.4 Regulatory Guide 1.117

This regulatory guideline provides guidance for tornado events for Light Water Reactors. This guidance should be incorporated into the industry consensus tornado design standard and may require some modifications to expand it to the Non Light Water Advanced Reactor designs.

3.4.2.5 SRP Section 3.5.3

SRP Subsection II.B.1 of Section 3.5.3 provides acceptance criteria for Local Damage Prediction resulting from internal and external missiles. The acceptable methods identified are:

Concrete:	Modified NDRC formula
Steel:	Stanford Tests
Composite Sections:	Recht and Ipson

SRP Subsection II.B.2 of Section 3.5.3 provides acceptance criteria for overall Damage Prediction.

These methods and criteria should be included in the appropriate industry consensus code and standard on missile design. Currently no such standard exists but ASCE manuals and publications are available which could be used to develop such a standard.

3.4.2.6 SRP Section 3.6.2

Consideration should be given to incorporating current pipe whip analysis and design criteria into the appropriate code. Recommendations to that effect are provided in Section 4.0.

3.4.2.7 SRP Section 3.8.4

SRP Section 3.8.4, Appendix D provides significant guidance for the design and analysis of spent fuel storage racks. This information should be incorporated in the industry code and standard judged applicable for the design of Fuel Storage Racks. There are two possible Standards in which this could be incorporated either the ASME BPVC, Section III, Division 1, Subsection NF or AISC N690. It is the author's reccomendation that they be incorporated into AISC N690.

3.4.2.8 I&E Bulletin 79-02

The baseplate issues presented in the I&E Bulletin are currently not addressed in industry consensus codes and standards. Therefore standard changes should be made to the applicable distribution support design standards.

3.5 Review of Applicable Codes and Standards and Identification of Code Deficiencies

3.5.1 Review Process

The review of the industry standards which were the subject of the program was conducted so as to identify necessary standard changes in two areas. The first of these areas are changes which were required to provide adequacy of a given industry consensus standard for application to new or unique features of ALWR or Advanced Reactor designs. The second of these areas which the identification of "generic" existing deficiencies which should be addressed so the subject industry consensus standards are more directly applicable to the design and construction of ALWR or Advanced Reactors.

While the original program plan identified the "generic" applicability review as a unique activity it was actually conducted together with the reactor design review effort. This was necessary to compare ALWR and advanced reactor features and design requirements directly to appropriate or applicable industry consensus standards. The areas where the subject industry consensus standards require modification to make them applicable to unique ALWR or advanced reactor design features were identified and summarized in the Section 3.3.

In conducting the reviews mentioned in the above paragraph there were aspects of the ALWR and Advanced Reactor Designs, which while they were not unique features or attributes of the structural design or construction of these reactors, were not adequately addressed in the existing language and guidance of the industry consensus standards which were the subject of this program. These type of issues were considered "generic" deficiencies in the subject industry consensus standards. In some cases resolution of these generic items should be addressed by modification to the existing industry consensus standards which are the subject of this review. In some instances development of new industry consensus standards should be considered. Finally consolidation and simplification of some industry consensus standards would be the most appropriate course of action. These generic deficiencies are discussed in detail in Section 3.5.2.

It is important to note that the vast majority of the industry standards, guidance, and criteria reviewed under the program is acceptable for use in the design and construction of ALWR's or advanced reactors. The suggested changes are limited to isolation design aspects or issues. Part of the reason, is, as previously discussed, that in the civil-structural area the evolutionary or advanced reactors have very few unique design features or attributes. The majority of unique or evolutionary aspects of the design are in the systems, components, or operational areas with little direct impact on the structural design and construction.

3.5.2 Existing Code or Standard Deficiencies and Code Case Review

This section summarizes the necessary code changes required to address generic deficiencies or missing design and construction criteria that should be incorporated into the referenced codes and standards to increase their applicability and use in the design of ALWR and Advanced Reactors. Also, reviews are applicable to ASME BPVC Code Cases for possible incorporation into the applicable sections of ASME III.

3.5.2.1 Existing Code or Standard Deficiencies

Table 3.5.2.1 provides detailed summaries of the identified deficiencies in the subject industry consensus standards, the necessary changes, reason or subject for which change is required, the affected reactors and the affected industry consensus standards. In several cases the changes could be put in one or more standards. The final location of these changes is provided in Section 4 along with the changes required as a result of the review and investigation results summarized in Sections 3.3, 3.4 and 3.5.

Table 3.5.2.1 Changes Required to Address Existing Deficiencies in the Subject Industry Consensus Codes and Standards

Applicable Reactors	Item	Seismic Cat.	Safety Class	Applicable Code/STD.	Description of Needed Change
AP600 Sys 80+ ABWR SBWR	Containment (Metal) Design Stress Levels	I	SC-1	ASME-III-NE	Currently Subsection NE has more restrictive Level D allowable stress limits than it has Level C stress limits. This is inconsistent with the ASME Design Philosophy and should be changed. Currently there are proposed changes to these sections in progress within the ASME BPVC organization and these proposed changes will be reviewed for acceptability before proposing any final changes to this subsection of the code.
AP600 Sys 80+	Containment Buckling Criteria	I	SC-1	ASME III-NE	Code Case N-284 should be incorporated into Subsection NE for containment buckling evaluations. The code case should be reviewed for any necessary technical changes especially in relation to the buckling of spherical shells and stresses induced by differential heating.
ALL	Tornado Design Criteria and Methods	I	SC-1 SC-2 SC-3	Multiple	Current tornado design criteria are provided in several publications including ASCE Papers #3269 and #4933, ANS Standard 2.3, Reg. Guides and NUREGs. This data should be combined into one specific code to provide definite tornado demand design criteria. Further, tornado capacity criteria should be explicitly discussed in the subject design codes and standards and reference the single demand definition code or standard. This design criteria should also clearly identify external missiles. This includes consideration on the study results put forward in NUREG/CR-4461.
ALL	Wind Design Criteria and Methods	I	SC-1 SC-2 SC-3	Multiple	Currently wind design criteria is provided in several areas including the ASCE 7-93 standard and, ASCE Papers #3269 and #4933. These criteria should be consolidated into one standard.

Table 3.5.2.1 Changes Required to Address Existing Deficiencies in the Subject Industry Consensus Codes and Standards (continued)					
Applicable Reactors	Item	Seismic Cat.	Safety Class	Applicable Code/STD.	Description of Needed Change
ALL	Design Criteria and Methods 1. Small Aircraft Crash 2. Accidental Explosion 3. Malevolent Vehicle	I	SC-1 SC-2 SC-3	Multiple	With the potential reduction in the tornado design basis the existing tornado design criteria may no longer be able to serve as a surrogate for this item. Therefore standard(s) need to be developed to provide explicitly demand criteria definition methods for these events. Further the subject designed standards need to modified to provide explicit capacity for these events and reference the appropriate demand definition standard.
ALL	Missile Design Criteria 1. Exterior Missiles 2. Turbine Missiles 3. Pipe Break & Impact	I	SC-1 SC-2 SC-3	Multiple-TBD(1)	Currently missile design criteria is provided in various areas including ANS 56.1 (Turbine Missiles), ASCE Papers and References, several formulas and papers by NDRC, BRL, Stanford, etc. Four specific needs are identified for this item. 1. The need to consolidate missile demand criteria in one standard. 2. The need to consolidate impact and damage demand criteria definition into the standard. 3. The need to modify the appropriate subject codes and standards to provide missile capacity criteria. 4. The effect of missiles on composite structure design is also required.
ALL	Missile Shield and Barrier Design Criteria	I	SC-1 SC-2 SC-3	ACI-349 AISC N690 ANS 58.2 ANS 58.3	Subject codes and standards do not provide adequate guidance on missile barrier design. Using the demand prediction methods discussed in the previous item, the codes should be modified to provide such design guidance. This should include consideration of highly inelastic behavior.
ALL	Consideration of Severe Accidents in Containment Design	II	SC-2	ASME III-NE ASME III-App. N	The USNRC has defined several severe accidents which must be considered in the ALWR and advanced reactor design. While these are not identified as design basis events, it appears that some reactor designs may treat them as such. It may be appropriate to include the demand evaluation criteria for these events (because of their dynamic nature) in Appendix N.

Table 3.5.2.1 Changes Required to Address Existing Deficiencies in the Subject Industry Consensus Codes and Standards (continued)					
Applicable Reactors	Item	Seismic Cat.	Safety Class	Applicable Code/STD.	Description of Needed Change
AP600 ABWR Sys 80+ SBWR	New & Spent Fuel Storage Rack Design, Fabrication, etc.	I		ASCE 4-86 ASME III-App. N AISC-N690	1. While ASME BPVC Section III, Subsection NF has in general been used for fuel rack design, it is not directly applicable since fuel elements and racks are not pressure retaining components and Subsection NF does not have specific design requirements applicable to stainless steel in compression. It is recommended that AISC N690, with any necessary modifications, be used for fuel rack design. 2. Currently no standard provides a procedure for the seismic analysis of free standing fuel racks in fuel storage pools. This should be incorporated into Appendix N of ASME BPVC and/or ASCE 4-86 for the actual fuel rack design and analysis. In addition for the design and analysis of the pool building structure changes should be made to ASCE 4-86. 3. The design requirements of SRP-3.8.4 Appendix D should be considered in this effort.
EPRI/URD	Masonry Wall Designs	I		ACI-530	The EPRI-URD specifies the acceptability of masonry walls in Category I buildings. A specific set of design and fabrication criteria should be developed for Category I masonry wall designs.
AP600 (ALL)	Weld Inspection Criteria	I	SC-1	AWS D1.1	The NCIG-01 Welding Standards for visual inspection should be incorporated into AWS D1.1 for application in inspection of ALWR and Advanced Reactors designs.
ALL	Minimum Design Load Requirements for Nuclear Power Plant Structures (Snow, Rain, Wind, Tornado ,Tsunami, Basic Design Loads)	I	SC-1 SC-2 SC-3	TBD(1)	For Seismic Category I structures minimum based design loads for normal events such as live loads, etc., should be specified in a code or standard. This could be a section of ASCE 7-93 or in a new ANS or ASCE Standard.

Table 3.5.2.1 Changes Required to Address Existing Deficiencies in the Subject Industry Consensus Codes and Standards (continued)					
Applicable Reactors	Item	Seismic Cat.	Safety Class	Applicable Code/STD.	Description of Needed Change
ALL	Minimum Design Load Requirements for Nuclear Power Plant Structures (Snow, Rain, Wind, Tornado, Tsunami, Basic Design Loads)	II	NNS	ASCE 7-93	For Seismic Category II or Seismic II/I structures a separate section should be added to ASCE 7-93 to provide minimum design loads for Power Reactor Seismic Category II structures for external events and basic design loads such as live load, etc.
Sys 80* AP600 ABWR CANDU-3 SBWR PIUS PRISM	Containment Leak Testing	I	SC-2	ANS 56.8 ASME III-IWE/IWL	These standards should be updated to include the suggested and necessary changes given in Regulatory Guideline MS-021-5.
AP600 SBWR CANDU-3	Modular Construction Issues	I	SC-1 SC-2 SC-3	Multiple ASME BPVC Subsections	For Modular Construction of ASME Structures, Systems and Components, construction will have two phases: at the fabrication shop and "on site". This raises issue in terms of N stamping primary pressure retaining boundaries. The capability to have a system or component N-stamped by both the fabricator and the constructor should be reviewed by the ASME BPVC. While this is primarily for equipment which is beyond the scope of this program it may have applicability for ASME III-NE/NF components. This concern was raised by M.K. Ferguson in the Construction Plan for the AP600.
AP600 SBWR CANDU-3	Modular Construction Issues	I	SC-1 SC-2 SC-3	Multiple	Applicable standards need to be modified to incorporate construction loads and transportation loads as normal design loads for modular construction.

Table 3.5.2.1 Changes Required to Address Existing Deficiencies in the Subject Industry Consensus Codes and Standards (continued)					
Applicable Reactors	Item	Seismic Cat.	Safety Class	Applicable Code/STD.	Description of Needed Change
SBWR ABWR PIUS	Hydrodynamic Loadings in Pressure Suppression Containments	I	SC-1 SC-2 SC-3	ASCE 4-86 ASME III App. N. Multiple	These designs use pressure suppression containments which will result in direct and building filtered hydrodynamic loadings resulting from SRV operation and accident loads. ASCE 4-86 or ASME III App. N should be modified or a new standard written to provide demand prediction criteria for these loadings. The other subject standards should be modified to include appropriate capacity criteria for these loadings.
ALL	Distribution System Support	I		ASME III-NF AISC N690 IEEE-628 AISI-CFSDM NFPA-13 AISC-ASD ASME AG-1 ACI-349, Appendix B	Currently a multitude of codes and standards are being used for distribution system support designs including: ASME III-NF, ANSI/AISC N690, IEEE-628, AISI-CFSDM, ASME AG-1, AISC-ASD, NFPA-13, etc., these should be consolidated into one set of design standards for all safety related distribution system supports.
ALL	Fire Barrier Seismic Design Criteria	I		NFPA-80 NFPA-80A NFPA-803	Currently no standard provides adequate seismic and extreme load design criteria for Seismic Category I or II fire barriers. Further most SSAR criteria is very indefinite on applicable design criteria. These NFPA Standards should be modified to incorporate extreme load design of fire protection barriers. Specifically NFDA 803 should be appropriately modified.

Table 3.5.2.1 Changes Required to Address Existing Deficiencies in the Subject Industry Consensus Codes and Standards (continued)					
Applicable Reactors	Item	Seismic Cat.	Safety Class	Applicable Code/STD.	Description of Needed Change
MHTGR PRISM	Code Case N-47 (incorporation in the Code)	I	SC-1	ASME III	This code case will have applicability for components used in the MHTGR and PRISM reactor designs due to the potentially higher operating temperatures of these designs. The applicability of this case is to components which are primarily out of scope of this review program. However it may have some applicability to component supports attached to high temperature distribution systems. Therefore this code case should be considered for incorporation into the ASME BPVC as a mandatory appendix and referenced in the appropriate design sections. The code case should also be expanded to Class 2/3 components.
AP600 SBWR CANDU-3U	Modular Construction Issues	I	SC-2	AISC N690 ACI 349	ACI and AISC Standards do not cover configurations such as composite wall modules being used in the AP600. A special concern with the concrete filled modules is the design equations and criteria required to address buckling and shear transfer of these type of structures. These items must be considered in the changes to these standards for modular construction.

Footnote for Table 3.5.2.1:

(1) TBD means "To Be Defined" in Section 4.0.

3.5.2.2 Code Cases

Table 3.5.2.2 and Table 3.5.2.3 lists ASME Boiler and Pressure Vessel Code Cases which have applicability to the industry consensus codes and standards associated with this program. Two of these code cases N-284 and N-47 should be incorporated into ASME III as discussed in Section 3.5.2.1. The balance of the code cases are related to the fabrication and inspection of supports using Subsection NF as a design basis. They need to be considered when developing a consolidated distribution system support design criteria. However, Section 4 will define which if any of these code cases should be incorporated into ASME-III and/or ASME IX.

Table 3.5.2.2 Section III, Division 1 Code Cases	
Case	Subject
N-47	Class 1 Components in Elevated Temperature Service
N-71[2]	Additional Materials for Subsection NF Class 1, 2, 3 and MC Component Supports Fabricated by Welding
N-249[2]	Additional Materials for Subsection NF Class 1, 2, 3 and MC Component Supports Fabricated without Welding
N-284[1]	Metal Containment Shell Buckling
N-309[1]	Identification of Material for Component Supports
N-337[2]	Use of ASTM B525-70 Grade II, Type II, Sintered Austenitic Stainless Steel for Class 1, 2, 3, and MC Component Standard Supports
N-393	Repair Welding Structural Steel Rolled Shapes and Plates for Component Supports
N-403	Reassembly of Subsection NF Component and Piping Supports
N-420[1]	Linear Energy Absorbing Supports for Subsection NF, Class 1, 2, and 3 Component and Piping Supports
N-433[1]	Non-Threaded Fasteners for Section III, Division I, Class 1, 2, and 3 Component and Piping Supports
N-476[1]	Class 1, 2, 3 and MC Linear Component Supports - Design Criteria for Single Angle Members
N-500	Alternative Rules for Standard Supports
N-510	Borated Stainless Steel for Class CS Core Support Structures and Class 1 Supports

Footnotes for Table 3.5.2.2:

(1) Endorsed by USNRC Staff in Regulatory Guideline 1.84.
(2) Endorsed by USNRC Staff in Regulatory Guideline 1.85.

Table 3.5.2.3 Section XI Code Cases	
Case	Subject
N-491	Rules for Examination of Class 1, 2, 3, and MC Component Supports of Light Water Cooled Reactors

3.6 Consideration of Actual Earthquake Experience in Seismic Design

3.6.1 Description of the Investigative Effort

In the development of seismic design criteria standards for SSC an important input consideration is the performance of SSC in actual strong motion earthquakes. Therefore as part of this program to evaluate the adequacy of United States Industry Codes and Standards, Stevenson and Associates undertook an investigation to determine changes which should be made to address observations of the response of actual SSC to strong motion earthquakes. This investigation was conducted in 3 parts and is contained in Appendix A to this report. The first part of this study involved a summary and an overview of the performance of distribution systems subjected to strong motion earthquakes. The second part consisted of a summary of an "on-site" investigation by Stevenson and Associates of the response of distribution systems and system supports subjected to the 1994 Northridge California earthquake. Finally, the third is a summary of an "on-site" investigation of the response of SSC to the 1995 Kobe, Japan earthquake. The resulting conclusion and observations from this experience data are provided in Appendix A and summarized below.

3.6.2 Observations/Suggested Changes

This Section provides suggested changes to industry codes and standards resulting from investigation of the response and performance of industry and power plant facilities subjected to strong motion earthquakes.

3.6.2.1 Distribution Systems

The commercial design codes and practice for electric cable systems and HVAC ducting systems appear to be adequate to insure these systems can withstand strong motion earthquakes up to at least 0.5g. This capacity exists provided the supports and support anchorage behave in a ductile manner and the supports have vertical load carrying capacity significantly greater than that required to carry dead loads. It is however imperative that adequate attention is provided to the anchorage of equipment to which these systems are attached, so as to limit seismic anchor motions. For seismic Category I systems there are several sets of standards which could be applied to the design of supports for these distribution systems including:

•AISI-CFSDM
•AISC-N690
•IEEE-628
•ASME AG-1
•SMACNA Standards

To avoid confusion and promote standardization for Advanced Reactors, it is suggested that the USNRC should establish via a Regulatory Guidance document the preferred application of these standards to the design of HVAC and raceway systems. Further it is suggested for these applications the standards and overall design distribution system designs should consider the provision of "design by rule" criteria which are not based on frequency response characteristics and are based on actual earthquake experience and which promotes the use of ductile design concepts to reduce the size and costs associated with HVAC and raceway system supports. The raceway system evaluation criteria developed by the SQUG Program provides a good basis for the development of such an experienced based "design by rule" criteria.

The poor performance exhibited by fire protection piping and sprinkler systems would indicate that changes in the commercial design codes and industry practice are warranted for application to Seismic Category I/II fire protection piping. It is suggested an appendix to NFPA-13 or a new NFPA standard be developed for Seismic Category I fire protection systems. Items which should be considered in the standard include:

•Elimination of the use of cast iron, malleable iron, and friction fittings and connections

•More restrictive lateral and vertical span limitation for systems containing threaded fittings.

•Expanded guidance on spatial interaction issues

•More design guidance for seismic anchor motion

•Provision of support and support welding details which insure a ductile failure mode.

These considerations plus the application of an experienced based "design by rule" approach could significantly enhance the seismic capacity of these systems. Also these standards should promote the use of ductile support anchorage design concepts to increase reliability and reduce the costs associated with piping system supports.

For piping system supports the experience data suggests the design rules of ASME BPVC Section III, Division 1, Subsection NF and AISC N690 if appropriately applied should provide adequate margin. The major issues with piping system and piping system supports would appear to be with the design practice and analytical methods currently applied to these systems. This current practice which results in high frequency stiff piping systems would appear to be overly conservative from a seismic inertial load standpoint and less conservative than conventional construction for seismic anchor motions and thermal expansion loadings. Consideration should be given to modify the design practice to promote low frequency, flexible piping systems with appropriate control of spatial interaction issues and large piping deflections. Also these codes should promote the use of ductile support and anchorage design concepts to balance the margins between seismic inertial loads and seismic anchor motions.

3.6.2.3 Buildings/Structures

The recent earthquake experience supports the need to consider displacement and story drift limits as well as stress limits in structural members responding to earthquake ground motions. This is particularly true for facilities near (within 10 km) the epicenter or fault rupture lines of thrust type faults from a damaging earthquake. Changes to ACI-349 and AISC N690 should be considered to address this issue.

A second concern resulting from the earthquake experience investigations is the potential of brittle fracture of carbon steel members. Current material specification or selection and post weld heat treatment requirements contained in AISC Specification N690 and AWS D1.1 should be reviewed for possible modification to address this issue.

3.6.3 Further Discussion of Distribution System Analysis Issues

Distribution systems different significantly in seismic response and behavior from buildings, tanks, and other types of structures. In general they are exposed to building filtered seismic loads and typically exhibit significant non-linear response and ductile behavior. The design standards associated with distribution system supports must consider these unique features to provide adequate and appropriate design criteria for distribution system supports.

It is very difficult because of the non linear geometric behavior to rigorously analyze and accurately predict the response of safety related distribution systems. Such analyses would need to consider time dependant support gaps and impacts, inelastic material response, etc. which would require time-history seismic input and non-linear geometric and material response analysis capability. It should also be noted that in a typical nuclear plant design that there are several hundred thousand feet of such distribution systems typically divided into 1000 or more problems of 100 to 200 feet in length each. This compares to only 6-8 building analyses with approximately the same complexity. This amount of rigorous analysis can add significantly to design and analysis costs of a standard Nuclear Plant Design. These increased costs can effect the feasibility of construction of the Advanced Reactor plants when compared with other cycles (coal, Gas, and Oil, etc.) where no such rigorous distribution system analyses are required. Both USNRC Section SRP 3.7.3 and ASME-III Appendix N have provided simplified approaches to seismic design of distribution systems. However, even these simplified approaches add over 10 percent to the total cost of a nuclear power plant as would be required for an equivalent size conventional fuel power plant built in a high seismic zone.

Therefore the use of earthquake experience data coupled with the review of available test and analysis data was considered in conjunction with the previous knowledge and data obtained in Section 3.2, 3.3, 3.4 and 3.5 in developing the recommended code changes for distribution system supports.

3.7 Other Outputs From This Program

In conducting this review effort a significant amount of data related to the ALWR and Advanced Reactors and related subject Industry Codes and Standards was generated. At the request of the USNRC specific portions of this information was compiled for use in other ongoing USNRC research activities. Appendix B provides a comparative study for four Civil Structural Design Codes. The purpose of this study was to compare the most current revision of the industry code and standard to that revision cited in the Standard Review Plan (NUREG-0800). Appendix C presents a comparison of the seismic, wind, and tornado design basis for the reactor designs which were the subject of this review. As previously discussed Appendix A provides an overview of investigations of distribution systems support performance in recent strong motion damaging earthquakes.

4.0 Phase II Review Effort

This phase of the program developed reccommended industry standard changes resulting from the Phase I effort for application of the codes and standards to ALWR and Advanced Reactor Plant Designs. These recommendations are presented in this section and are based on the observations and the review effort described in Section 2.0 and Section 3.0. In addition, a recommended plan of action for securing the necessaary standard changes is provided.

4.1 Evaluation of Recommended Changes to Industry Consensus Codes and Standards

Due to the focus of this review on the ALWR designs and the limited availability of design data for advanced reactors the recommended industry standard changes are targeted toward the ALWR's. However, when possible changes applicable to major design aspects of the advanced reactors are also provided. The recommended changes are general in nature as specific code textual changes and modifications are beyond the scope of this program and are the responsibility of the various committees having cognizance for the subject standards.

Most of the recommended standard changes are grouped by jurisdictional body (ASME, ASCE, etc.). Some general or cross jurisdictional changes are provided within the various subsections.

4.1.1 ASME Boiler and Pressure Vessel Code

This subsection provides the suggested changes to the ASME Boiler and Pressure Code. The suggested changes are provided in the Tables outlined below.

Table 4.1.1.1 -	Recommended Changes to Section III, Division 1, Subsection NE
Table 4.1.1.2 -	Recommended Changes to Section III, Division 2, Subsection CB (ACI-359)
Table 4.1.1.3 -	Recommended Changes to Section III, Division 2, Subsection CC (ACI-359)
Table 4.1.1.4 -	Recommended Changes to Section III, Division 1, Subsection NF
Table 4.1.1.5 -	Recommended Changes to Section III, Division 1, Appendix N
Table 4.1.1.6 -	Recommended Changes to Section XI

The reader if also referred to section 4.1.7 for a discussion on standard changes recommended in relation to ASME BPVC, Sec. III, Div. 1, Class MC Components.

4.1.2 ACI Standards

The majority of the changes recommended to the ACI Standards are in ACI-349, which is the primary standard for the design of Nuclear Safety Related Reinforced Concrete Structures. The recommended changes are summarized in Table 4.1.2.1. Depending on the actual code language there could be additional changes required to several of the supporting ACI Standards but identification of possible subordinate changes is beyond the scope of this program.

The EPRI-URD states the Seismic Category I blockwalls should be permitted in ALWR's. While ACI-530 and ACI-530.1 provide general design guidance for masonry structures it is suggested that a new standard ACI-530.2 be developed to provide the "Specifications for Design of Seismic Category I Masonry Structures in Nuclear Power Plants." It is further suggested it be developed in conjunction with the ASCE, as was ACI-530 and ACI-530.1. In developing the standard, advantage should be taken of recent work done at Oak Ridge National Laboratory on seismic testing of block walls and other recent DOE design and detailing guidance to insure block wall structures are seismic resistant.

4.1.3 ASCE Standards

The first ASCE Standard for which changes are recommended is the ASCE 4 Standard (current revision 4-86). These recommended changes are provided in Table 4.1.3.1.

The second area for which modification to ASCE codes and standards is required is in the area of minimum design loads for nuclear power plants. Currently several standards and regulatory guidelines are discussed for use with the ALWR and Advanced Reactor designs, including ASCE-7 and various building codes. It is recommended that a new ASCE standard be developed for nuclear power plant structures. This standard should address the following areas:

(a) Minimum design loads for nuclear safety related structures in nuclear power facilities.

(b) Minimum design loads for non-safety related structures in nuclear power facilities.

(c) Man-made hazard phenomenon design requirements for safety related and non-safety related facilities in nuclear power plants.

While one standard is recommended if deemed appropriate by the jurisdictional bodies, three standards could be developed. The first item [(a) above] would provide minimum design loads such as live load, deadload, snow, etc. for safety related structures in nuclear power plants. This standard should also include minimum design loads for Tsunami. Where appropriate there would be reference to other applicable standards such as ANS 2.3 (tornado and wind load). The second item [(b) above] would provide minimum design load criteria for NNS and non-safety related structures in nuclear power plants. The third item [(c) above] would provide nuclear power plant design load requirements for such man-made hazard phenomena as malevolent vehicle design, aircraft crash, etc. This would consolidate design guidance and criteria for these items which is currently provided in ASCE Papers, NUREG's, and Regulatory Guidelines. These standards would establish the appropriate demand criteria. The capacity criteria would be the responsibility of the governing design code.

Finally it is recommended that a new standard be developed which provides capacity criteria for the design of missile barriers and shields and pipewhip restraints in nuclear power plants. This standard would consolidate data currently contained in ANS standards, ASCE committee reports, various technical papers and documents (NDRC, Stanford, etc.) and provide a single consistent design document. Further it should incorporate inelastic and high ductility design and analysis techniques. It is suggested that the demand criteria for these events be provided in an updated ANS Standard as discussed in Section 4.1.7.

4.1.4 AISC Specifications

The AISC Specification for which changes are recommended is the N690 Specification. The recommended changes are provided in Table 4.1.4.1. At least one advanced reactor states an intention to use the AISC Steel Construction Manual for Safety Related Structures. If this is in fact done significant changes would be required in this specification to make it (1) consistent with N690 and (2) to address quality assurance requirements (Quality Classes).

4.1.5 AISI Cold Formed Steel Design Manual

Several reactor designs intend to use the AISI Cold Formed Steel Design Manual for design of Cable Trays and supporting members for HVAC, cable trays, and conduits. This specification should be modified to provide higher allowable stress levels for emergency and faulted [or abnormal and extreme] (including the SSE) events. This standard should also incorporate the use of ductile design details which permit high levels of ductility (which have been shown beneficial by actual earthquake experience) for the extreme events. Anchorage and base plate designs should reference ACI-349 Appendix B (modified as discussed in Section 4.1.2) for concrete anchorage design and steel embedments. Appropriate Quality Assurance (Quality Classes) and material traceability requirements for nuclear safety related design should be added via an appendix.

4.1.6 NFPA Standards

NFPA-803 should be modified as shown in Table 4.1.6.1 to provide appropriate restrictions on NFPA-13 and NFPA-14 for use with Seismic Category I, Safety Class Fire protection systems and provide design guidance for Seismic Category I Fire Protection Barriers. The seismic design criteria in NFPA-13 should be modified to better address seismic interaction criteria, seismic anchor motions, and support details and fabrication.

If cold formed steel members are to be used for support of fire protection systems, the AISI-CFSDM (modified as suggested in Section 4.1.5) should be referenced. For hot rolled steel supporting members the AISC N690 Specification (modified as suggested in Section 4.1.4) should be referenced. Concrete anchorage and embedment details should reference ACI-349, Appendix B. Finally the standards (NFPA-13, NFPA-14) should be modified to provide higher stress allowables for emergency and faulted [or abnormal and extreme] conditions. The latter three suggestions could best be accomplished by the addition of an appendix for Nuclear Power Plant Safety Related, Fire Protection Systems.

4.1.7 ANS Standards

The ANS Standard 2.3 for wind and tornado design (demand) criteria should be modified to be consistent with current Advanced Reactor design criteria and should also consider recent research in this area such as NUREG/CR-4461) and the work conducted by several national labs for the DOE weapons complex. This standard should be expanded to also include strong wind and hurricane design (demand) criteria. It is also noted that an industry standard is required to provide design limits and loading combinations for metal reactor containment system (ASME BPVC, Sec. III., Div. 1, Class MC) components such as currently provided in Regulatory Guideline 1.57. As discussed in Section 3.4.2.2 the ASME BPVC currently defers this responsibility to the owner via the Design Specification. It is recommended this guidance be provided in either the ANS 58.14 or ANS 50.1 Standards currently under development by the ANS. Suggested changes to other ANS Standards are provided in Table 4.1.7.1.

4.1.8 HVAC and Cable Tray Supports

There are currently a significant number of standards and committee reports being referenced and used for the construction safety class, seismic category I/II HVAC and raceways systems. It is recommended that these be consolidated into two design standards.

(a) IEEE-628 for Conduit and Cable Trays
(b) ASME AG-1 for HVAC Systems

To accomplish this the standards may require modification and enhancements to provide all necessary design data. Specification of these type of changes, except as they may effect support design and construction, is out of the scope of this program. However they should be modified to reference AISC N690, AISC-CFSDM, and ACI-349 Appendix B (with the suggested modifications contained in this report) for the design of structural supporting members and concrete anchorage embedments. Specifically for ASME AG-1 support design criteria should be eliminated in the standard.

4.1.9 AWS Standards

Suggested changes to the AWS D1.1 and D1.3 Structural Steel Welding Standards are provided in Table 4.1.9.1.

Table 4.1.1.1 Recommended Changes to ASME BPVC, Section III, Div. 1, Subsection NE		
Needed Change	Reason or Basis for the Change	Discussion of the Change
Earthquake Design Basis	The proposed Appendix S to 10CFR50 will eliminate the OBE from the design basis. (Level B events)	1. This change may require code modifications to provide for control of primary plus secondary stress limits for thermal and SSE loading conditions for ASME Service Levels C and D for the metal containment structure. This may include the need to provide for a fatigue control for loadings (including SSE) which generate primary plus secondary stress range conditions. 2 Overall the code should provide two design paths; one in which both an OBE and SSE exist and one in which only a SSE exists. This will require review and changes to related sections such as NCA, mandatory appendices and non-mandatory ASME BPVC appendices.
Level C and Level D Stress Limits	1. Currently the Level D stress limits are more restrictive than the Level C limits which is inconsistent with ASME BPVC philosophy.	1. The Service Level C and Level D stress limits should be modified such that the Level C stress limits are more restrictive than the Level D stress limits. Efforts are currently on going in the ASME BPVC committees to accomplish this goal.
Incorporate Buckling Criteria of CC N284 into the Code	1. Subsection NE requires the consideration and evaluation of shell buckling for containment vessels but provides no specific criteria to accomplish this. For many of the load cases of interest CC N284 provides the necessary guidance.	1. Incorporate Code CC N 284 into the code to provide the necessary buckling criteria, including consideration of differential thermal loadings.
Non-integral interface criteria	1. For the AP600 the containment vessel simply rests on the foundation and basemat of the reactor building (nuclear island). There is no internal attachment of these structures.	1. Define the design methods and criteria required to evaluate and design this interface. 2. Develop the design criteria for any additional loads such as impact, wedging or friction loads, etc. 3. Determine what potential post fabrication settlement loads could exist, how to evaluated them and any construction inspection criteria which is required. 4. Stability and potential overturning criteria are required.

Table 4.1.1.1 Recommended Changes to ASME BPVC, Section III, Div. 1, Subsection NE (continued)		
Needed Change	Reason or Basis for the Change	Discussion of the Change
Weir System Design Criteria	1. Use of Weir cooling system with flow fins, etc.	1. Establish clear jurisdictional boundaries and design and fabrication criteria for interface, including loads, welding requirements, inspection requirements, and any stress relieving requirements. 2. Review the current Level D stress criteria for post LOCA secondary stresses induced in the vessel by cooling water flow and modify as required.
Modular Construction Issues	1. Modular construction and prefabrication of containment vessels which may be used on the AP600 and some advanced reactors may blur definition of N-certificate holder and N-Certification (N-Stamp) jurisdiction. 2. Transportation and prefabrication loadings could exist.	1. Modify NE and NCA to provide a N-certification (N-stamp) program consistent with the needs of partial prefabrication and modular construction. This could also involve changes to Section IX and V with regard to weld inspections. 2. Provide appropriate stress criteria and load combination considerations for transportation and prefabrication loadings.
Severe Accident Criteria	1. The USNRC has defined several extreme accidents for ALWR and advanced reactor design. While these are not design basis events some reactor designs may treat them as such.	1. Review existing capacity criteria for these potentially new limiting design basis events.
Earthquake Experience Related Changes	To modify the specification to address observations from the behavior of structures subjected to recent strong motion earthquakes.	For Earthquake Design (SSE) the Code should be modified to require displacement limiting acceptable criteria in addition to the stress limiting criteria currently in the Code.
Earthquake Stability	To address USNRC concerns raised in the review of AP600.	Provide criteria to evaluate the potential for "lift-off" and overturning during a SSE Event.

Table 4.1.1.2 Recommended Changes to ASME BPVC, Section III, Div. 2, Subsection CB		
Needed Change	**Reason or Basis for the Change**	**Discussion of the Change**
Earthquake Design Basis	The proposed Appendix S to 10CFR50 will eliminate the OBE from the design basis.	This change may require code modification to provide for control of primary plus secondary stress limits for thermal and SSE loading conditions for ASME Service Levels C and D for the steel liner. This could include the need to provide for a fatigue control for loadings (including SSE) which generate primary plus secondary stress range conditions.
Severe Accident Criteria	The USNRC has defined several severe accidents for the ALWR and advanced reactor design. While these are not design basis events some reactors may meet them as such.	Review existing capacity criteria for these potentially new limiting design basis events.

Table 4.1.1.3 Recommended Changes to ASME BPVC, Section III, Div. 2, Subsection CC (ACI-359)

Needed Change	Reason or Basis for the Change	Discussion of the Change
Jurisdictional Boundaries	The ABWR Reactor Building is a structure monolithically attached to the containment. The containment is constructed to ASME BPVC, Division 2, Subsection CC, while the reactor building is constructed to ACI-349.	Changes are required to address the code jurisdictional boundaries, interconnection joints, and applied load interface and load transfer. Also construction interface boundaries and jurisdictional definitions should be reviewed.
Severe Accident Criteria	The USNRC had defined several severe accidents for ALWR and advanced reactor design. While these are not design basis events some reactor designs may treat them as such.	Review existing capacity criteria these potentially new limiting design basis events.
Earthquake Experience Related Changes	To modify the specification to address observations from the behavior of structures subjected to recent strong motion earthquakes.	For earthquake design (SSE) the specification should be modified to require displacement and story drift limiting acceptance criteria in addition to the strength criteria currently in the code. Additional criteria related to brittle fracture may also be required.
Man-made Hazard Criteria	Address man-made phenomenon hazards.	Loadings, load combinations and acceptance criteria should be provided for man-made phenomenon hazards in a manner similar to that provided for natural hazards.
Elimination of the OBE	The proposed Appendix S to 10CFR50 will eliminate the OBE from the design basis.	Changes should be made to eliminate or provide the option to eliminate the OBE from the design basis equations, load combinations, and behavior criteria.
Earthquake Stability	To address USNRC concerns raised in the review of the AP600.	Provide criteria to evaluate the potential for "uplift" and overturning during a SSE Event.

Table 4.1.1.4 Suggested changes to ASME BPVC, Section III, Div. 1, Subsection NF		
Needed Change	Reason or Basis for the Change	Discussion of the Change
Simplify Pipe Support Design and Construction	1. Use STD MSS-SP-58 standard supports for piping. 2. Simplify complexity of structural support designs. 3. Reduce material traceability requirements. 4. Reduce weld inspection requirements. These changes are necessary to reduce costs and accelerate construction schedules consistent with the need to make ALWR's economically viable.	1. The design sections of NF for piping supports need simplification to reduce the complexity and design efforts associated with piping supports. Needed changes include: (a) Direct reference use of MSS-SP-58 standard supports including the incorporation of Code Case N500 into the code. This includes a reduction of material traceability requirements. (b) Direct reference and simplified application of AISC N690 for structural steel supports includes use of the reduced weld inspection criteria of N690, use of AWS D1.1 for welding, simplification of the N690 criteria consistent with the simple nature of pipe support structures and reduce material traceability requirements consistent with N690. (Effort in this area is currently ongoing in ASME BPVC Subgroup on Design.) 2. Material traceability and weld inspection requirements should be more consistent with non-nuclear power plant requirements. This is based in part on the actual seismic performance of MSS-SP-58 component standard supports when subjected to strong motion earthquakes.
Seismic Isolator Design Section	The Prism reactor design intends to use seismic isolators for support of Reactor Building. These will be Seismic Category I, Safety Class 3 Devices.	New design and fabrication subsections will be required to adequately address the design of these components. It is also possible that new a Section II Materials Specification will be required for these isolators.
Support Base Plates and Anchorage	Base plates and concrete anchors are an integral part of pipe support design criteria. As such they should be addressed in NF.	Incorporate concrete anchorage, embedment design and selection criteria, and flexible/rigid base plate design criteria. This can be done by permitting the use of manufacturer's allowable loads and the specification of ACI-349, Appendix B for such designs. Also appropriate factors of safety considering the redundancy of piping supports and actual earthquake experience should be developed.

Table 4.1.1.4 Suggested changes to ASME BPVC, Section III, Div. 1, Subsection NF (continued)

Needed Change	Reason or Basis for the Change	Discussion of the Change
Incorporation of Code Cases	Incorporation of Code Cases makes the code more complete and reduces design efforts by eliminating the tracking and referencing of the use of these code cases.	The following codes cases should be incorporated into Subsection NF and/or the applicable section of BPVC: N-71 N-403 N-249 N-433 N-309 N-476 N-337 N-500 N-393 N-510
Modular Construction Issues	1. Several reactors intend to use modular construction for Mechanical Packages (which contain NF scope piping supports) the Code N-Certification (N-Stamp) responsibility is blurred, especially when these mechanical packages are assembled in vendors facilities and connected to field constructed systems.	1. The N-Certification (N-Stamp) Process needs to be reviewed relative to the use of multi-vendor, modular construction and field assembly. This could also affect NCA Quality Assurance and material traceability issues. 2. Provide appropriate stress criteria and load combinations for consideration of transportation and prefabrication loadings.
Torsion Design Issues on Open Sections	Add necessary torsional design criteria to support AP600 designs and improve the code.	In conjunction with incorporation of CC N476, appropriate ASCE papers, etc., should be used to develop a simple and effective torsional design and evaluation criteria. In addition the code should provide a table of Principal Moments of Inertia and Principal Section Modulii for most common open sections, including angles, channels, tees, back to back angles, and back to back channels, etc.

Table 4.1.1.5 Recommended Changes to ASME BPVC, Section III, Div. 1, Appendix N

Needed Change	Reason or Basis for the Change	Discussion of the Change
Seismic Hydrodynamic Analysis Techniques	Due to the use of water filled structures such as the AP600 concrete and steel tanks, BWR pressure suppression containment vessels, etc.	Subsections should be added to this appendix to provide rules for determination of seismic demand predictions for: (a) Fluids contained in basins (b) Fluid-fill tanks and vessels This should include impulse and impact load effects.
LOCA Hydrodynamic Load Definition (Building Filtered Loads)	GE BWR and reactor designs will use pressure suppression containments. ABB/CE and W AP600 have IRWST discharge loads.	Subsections should be added to provide rules for determination of pressure suppression containment and larger storage tank hydrodynamic loadings and the associated building filtered hydrodynamic loads. This is for both Level D LOCA events and Level B SRV discharge loadings.
Free Standing Fuel Rack Seismic Analysis Methodology	No standard analysis methodology currently exists.	The code should be updated to provide seismic analysis methodology for Free Standing Spent Nuclear Fuel Storage Racks. This should include hydrodynamic storage rack interaction effects.
Severe Accident Mechanical Load Definition	If severe accidents are to be used as design basis events no standard analysis methods currently exists.	Rules and analysis methods to determine the appropriate applied mechanical loadings for severe accident events.

Table 4.1.1.6 Recommended Changes to ASME BPVC, Section XI

Needed Change	Reason or Basis for the Change	Discussion of the Change
IWE, IWL Changes	Multiple Guidance Documents exist.	The standard should be modified to provide consistency with the ANS 56.8 Standard and Draft Regulatory Guideline MS-021-5.

Table 4.1.2.1 Recommended Changes to ACI-349		
Needed Change	**Reason or Basis for the Change**	**Discussion of the Change**
Earthquake Design Basis	The proposed Appendix S to 10CFR50 will eliminate the OBE from the design basis.	Changes should be made to eliminate or provide the option to eliminate the OBE from the design basis equations, load combinations and strength criteria.
Ductile Detailing	Improve the seismic resistance design of related reinforced concrete structures.	Add a chapter similar to Chapter 21 to ACI-318 to provide improved detailing of connections, etc., to improve the ductility and seismic capacity of nuclear safety related structures.
Earthquake Experience Related Changes	Modify specification to address observations from the behavior of structures subjected to recent strong motion for earthquakes.	The earthquake (SSE) design criteria should be evaluated for displacement story drift limiting acceptance criteria in addition to the current strength limiting criteria. This new criteria needs to be developed.

Table 4.1.2.1 Recommended Changes to ACI-349 (continued)

Needed Change	Reason or Basis for the Change	Discussion of the Change
Composite or Modular Construction	Westinghouse AP600, GE ABWR, GE SBWR, CANDU-3U modular or composite construction items.	1. Changes should be made to address composite steel and concrete members being used in the subject reactor designs. Specific areas where the code lacks or provides insufficient requirements include composite concrete steel in-filled walls and members subjected to out-of-plane flexural, in-plane shear as well as vertical compressive loads, and the use of partially encased steel beams without mechanical shear connectors for composite design of slabs. Changes should be made in the areas of design when the concrete is a primary load carrying member as well as considerations (e.g. displacement limits, etc.) for protection of the concrete subjected to loads assumed taken primarily or solely by steel elements. Appropriate design and evaluation criteria should be provided along with required detailing to insure that the assumed composite actions occurs. Such changes should be co-ordinated with applicable sections to AISC N690. 2. For Modules prefabricated off site (at vendor facilities) prefabrication and transportation load evaluation criteria should be added to the code. 3. Concrete placement criteria and requirements for the prefabricated modules as being used in the Westinghouse AP600 should be provided by an Appendix to this standard. This appendix can directly provide the criteria or it can be provided by reference to other existing ACI standards. This appendix must provide necessary testing and quality assurance requirements.
Concrete Water Holding Facility Design Criteria	Several advanced reactors contain concrete water storage tanks and spent fuel pools	Enhance and specify guidelines for the design of concrete water storage facilities and spent fuel pools. This should include consideration of hydrodynamic loads and steel (stainless steel) liner design. This liner design should address leakage protection, concrete interaction, and any potential fatigue design requirements. This also includes possible modifications to Appendix C for impulse and impact loads.

Table 4.1.2.1 Recommended Changes to ACI-349 (continued)

Needed Change	Reason or Basis for the Change	Discussion of the Change
High Temperature Concrete Design	At least one reactor, MHTGR, could have concrete design and operating temperatures in excess of the 150°F limit currently in ACI-349. In addition the GE SBWR has committed to use reduced concrete strength at the 150°F Level.	Provide more rigorous elevated temperature reinforced concrete design criteria and requirements.
New Commodity Loads	The use of large in plant water storage facilities in some advanced reactors and the deep embedment of the other facilities.	Add a new commodity load to the load definition which is different from dead and live loads and can be used to address these items. This is done partially in recognition that commodities respond dynamically at their frequencies and not the frequency of their supporting member.
Appendix B Modifications	Consider USNRC concerns in Appendix F or NUREG-1503, IE Bulletin 79-02 and use of manufacturer's standard anchorage.	Changes should be considered to Appendix B to address USNRC concerns, IE Bulletin 79-02 and to provide a standard anchorage and embedment design and qualification approach including the use of manufacturer's standard anchorage allowables for distribution system supports. This includes enhanced design criteria for expansion anchors considering recent earthquake experience and the criteria developed by the SQUG organization in support of the resolution of USI A-46. In addition IE 79-02 baseplate flexibility design considerations should also be incorporated into Appendix B.
Jurisdictial Boundaries	The ABWR Reactor Building is a structure monolithically attached to the containment. The containment is constructed to ASME BPVC, Division 2, Subsection CC (ACI-359) while the reactor building is designed to ACI-349.	Changes are required to address the code jurisdictional boundaries, inter-connection joints and applied loading, interface and load transfer. All construction interface boundaries and jurisdictional definition should be reviewed.
Increase Flexural Steel Quantities	Avoidance of brittle response due to earthquake loadings.	In deep members the design moment may be greater than the cracking moment. If the applied moment exceeds the cracking moment the flexural reinforcement will be suddenly loaded and could fail. Therefore for these types of members the minimum required flexural reinforcing steel (rebar) should be increased, but these changes must be developed such that the balance of reinforcement limitations are maintained.

Table 4.1.2.1 Recommended Changes to ACI-349 (continued)		
Needed Change	**Reason or Basis for the Change**	**Discussion of the Change**
Confinement Design Criteria	Some advanced reactors (MHTGR) intend to used ACI-349 for design of a "confinement" structure.	Modifications to this code should be made by the addition of a new chapter or appendix which provides the additional requirements which will be necessary to insure the confinement function. This would include allowable deformation, cracking, and leakage permitted to provide this confinement function.
Man-made Hazard Criteria	Address man-made phenomenon hazards.	Loadings, load combinations and acceptance criteria should be provided for man-made phenomena hazards.
Earthquake Stability	To address USNRC concerns raised during review of the AP600.	Provide criteria to evaluate the potential for "uplift" and overturning during an SSE Event.

Table 4.1.3.1 Recommend Changes to ASCE 4-86

Needed Change	Reason or Basis for the Change	Discussion of the Change
Seismic Hydrodynamic Analysis Techniques	Due to water-filled structures such as Westinghouse AP600 concrete and steel tanks, pressure suppression containment vessels, etc.	Subsections should be added to this appendix to provide rules for determination of seismic demand predictions for: (a) Fluids contained in basins (b) Fluid-filled tanks and vessels This should include impulse and impact load effects. (This item is currently under consideration by the responsible committee.)
LOCA and SRV Hydrodynamic Load Definition (Building Filtered Loads)	GE BWR and other reactor designs will use pressure suppression containments Westinghouse and ABB/CE have IRWST discharge loads.	Subsections should be added to provide rules for determination of pressure suppression containment and larger storage tanks hydrodynamic loadings and the associated building filtered hydrodynamic loads. This is for both Level D LOCA events and Level B SRV discharge loadings.
Soil Structure Interaction Effects	Several reactor designs are essentially "buried" in soil. (Deeply embedded)	Soil structure analysis techniques need to be reviewed to insure they are adequate for deeply embedded structures.
Analysis for large input Velocity & Displacement Demands	Actual Experience and observations in recent near-field strong motion earthquakes	Appropriate Analysis methods and requirements should be added to predict the displacement and story drift response caused by large input Velocity and Displacement based demands in addition to input acceleration based demands currently used in design.
Free Standing Fuel Rack Seismic Analysis Methodology	No analysis methodology currently exists.	The code should be updated to provide seismic analysis methodology for free standing spent nuclear fuel storage racks. This should include hydrodynamic storage rank interaction effects.
Modular Construction Analysis	Provided appropriate dynamic modeling and analysis of composite steel and concrete walls and slabs.	Modeling techniques should be enhanced to provide guidance on determination of appropriate stiffness, particularly for concrete in-filled steel structures, to obtain proper dynamic response of composite members and elements. In conjunction, appropriate damping factors should be provided for use with current dynamic analysis methodologies for such composite structural elements.

Table 4.1.4.1 Recommended Changes to AISC N690		
Needed Change	**Reason or Basis for the Change**	**Discussion of the Change**
Fuel Racks Design Criteria	There is currently not a definitive standard for fuel rack design, therefore such a standard is required.	A new section or appendix to N690 should be developed specifically for nuclear fuel storage rack design. This should incorporate the requirements current put forth in the SRP Section 3.8.4 Appendix D. It must also consider fixed and free standing fuel storage racks [optionally this could be added to ASME BPVC, Section III, Subsection NF].
Various Technical Items	To address USNRC concerns with the N690 specification.	The cognizant specification committee should address the concerns put forward by the USNRC in Appendix G of NUREG-1503. While most of these suggestions put forth in NUREG-1503 should probably be incorporated in AISC N690, the authors cannot recommend the proposed stress limit coefficient reduction. However the cognizant committee should review and consider it.
Earthquake Experience Related Changes	To modify the specification to address observations from the behavior of structures subjected to recent near field strong motion earthquakes.	1. For earthquake design (SSE) the specification should be modified to required displacement and Story Drift limiting acceptance criteria in addition to the stress limiting criteria currently in the specification. 2. Materials and post weld heat treatment procedures require review to insure they eliminate the potential for brittle fracture during seismic events.
Earthquake Design Basis	The proposed Appendix S to 10CFR50 will eliminate the OBE from the design basis.	Changes should be made to eliminate or provide the option to eliminate the OBE from the design basis equations, load combinations and stress criteria.
Clarify Restraint of Free and Displacement Loads	Provide needed standard clarification on these issues.	The standard should be modified to clearly define the restraint of free end displacement load. Also it should clearly differentiate between primary applied loads which result from the free end displacement restraint of other members and actual member free end displacement restraint loadings.

Table 4.1.4.1 Recommended Changes to AISC N690 (continued)

Needed Change	Reason or Basis for the Change	Discussion of the Change
Composite or Modular Construction	Address modular construction and composite action issues of AP600, SBWR, ABWR and CANDU-3U	1. Changes should be made to address composite steel and concrete members as being used in the subject reactor designs. The changes should be made in the areas of design when the steel is primarily load carrying member. Particularly, changes should address concrete in-filled steel walls and members subjected to out-of-plane shear and bending, in-plane shear, as well as vertical compressive loads, and composite design of partially embedded steel sections without mechanical shear connectors. Requirements for determination of effective flange width, width-thickness ratio, and slenderness effects should be addressed for such structural members. Acceptance requirements are needed for composite concrete in-filled steel walls subjected to 2-D and 3-D stress states. Appropriate design and evaluation criteria should be provided along with the required detailing to insure composite action occurs. 2. For modules prefabricated off site (at vendor facilities) prefabrication and transportation load evaluation criteria should be added to the standard.
Man-made Hazard Criteria	Address man-made phenomenon hazards.	Loadings, load combinations and acceptance criteria should be provided for man-made phenomena hazards.

Table 4.1.6.1 Recommended Changes to NFPA-803		
Needed Change	**Reason or Basis for the Change**	**Discussion of the Change**
Seismic Category I Design Criteria	Enhance seismic capacity of Safety Class or Seismic Category Fire Protection Systems (which exist in some Advanced Reactors).	The standard should be modified to specify the use of NFPA-13 and NFPA-14 for sprinkler, standpipe, and hose systems but provide limitations which would enhance the seismic capacity of these systems that are designated Safety Related, Seismic Category I/II systems. Items that should be considered: (1) Elimination of the use of cast iron, malleable iron, and friction fittings and connections, (2) reduced lateral and vertical span limitations for systems containing threaded fittings, (3) limitations on the support types and materials and (4) apply appropriate Quality Assurance standards and materials traceability requirements.
Seismic Category Design Criteria	Fire barrier seismic design criteria currently does not exist.	Add seismic design and qualification criteria for Seismic Category I/II Fire Protection Barriers in an appendix.
Man-made Hazard Criteria	Address Man-made Phenomena Hazards	Add a section defining fire protection needs and design criteria in response to man-made phenomena hazards

Table 4.1.7.1 Recommended Changes to ANS Standards		
Needed Change	**Reason or Basis for the Change**	**Discussion of the Change**
ANS 51.1 and ANS 51.2 Update or New Standards as appropriate. [1]	Use of ANS 51.1 and ANS 51.2 in lieu of Regulatory Guidance 1.26.	ANS 51.1 and 51.2 should be modified to be consistent with Regulatory Guideline 1.26 so that they can be referenced in lieu of Regulatory Guideline 1.26.
New ANS Standard	Develop ANS Safety Criteria Standards for gas cooled, liquid metal, and heavy water reactors similar to those which exist for PWR's and BWR's.	Develop new ANS Standards for gas cooled, liquid metal, and heavy water reactors
ANS 56.1 Update	Consolidate missile design criteria.	This standard should be updated to provide demand criteria for all types of potential missiles including turbine, interior, and exterior. It should be developed considering the existing, standards, papers, and other references on this subject.
ANS 58.2 Update	Enhance pipewhip demand load prediction.	This standard should be updated to provide more definitive demand definition criteria for pipe whip loadings. It should be modified considering the existing standards, papers, and the references on the subject. Changes would also be required to address elevated temperature reactors and reactors using process fluids other than Light Water(Liquid Metal, Gas-cooled, Heavy Water reactors, etc.)

Footnotes for Table 4.1.7.1:

[1] It should be noted that the ANS is currently developing the ANS 58.14 and 50.1 standards as replacements for ANS 51.1 and 51.2. It may be more appropriate to incorporate these suggested changes into these under development standards.

Table 4.1.9.1 Recommended Changes to AWS D1.1 and D1.3

Needed Change	Reason or Basis for the Change	Discussion of the Change
Weld Inspection Criteria	Improved, more cost effective weld inspection standard.	The NCIG-01 welding standards for visual weld inspection should be incorporated in AWS D1.1.
Brittle Fracture Considerations	Earthquake Experience Observations	Post weld heat treatment requirements should be reviewed for possible modification to address potential brittle fracture during strong motion earthquakes.
Ferritic Steel Welding	Consistent with SBWR design criteria and Regulatory Guideline 1.71.	The code should be reviewed to insure it provides controls on ferritic steel welding consistent with Regulatory Guideline 1.71 and the SBWR criteria.

4.2 Suggest Course of Action for Implementation of the Recommended Changes to Industry Consensus Standards

There are two distinct courses of action which should be followed to obtain implementation of the recommended changes to subject standards. The first course of action would apply when there is a need for development of a new standard. For existing standards, the cognizant jurisdictional organization or committee should be identified and a presentation which discusses the recommended changes should be made at a regularly scheduled meeting of this organization. The organization could be questioned as the most appropriate course of action to implement the recommend changes. This usually involves (1) preparation of actual code language changes, (2) refer to these proposed changes to a Special Task Group, or a standing Subgroup or Working Group, and (3) following the changes through the established organizational process. An important part of this effort is the existence of a person or person(s) who are interested in "shepherding" the changes through the approval process and who can provide any necessary technical support and response to the various committee member questions and concerns.

For new standards the ruling jurisdictional body for the cognizant organization should be approached and a "need" presentation made. The ruling body will then refer the changes to an existing committee, form a new subgroup or working group or refer the changes to a special task group. Once this is done the process and efforts required to develop the new standard are essentially the same as for existing standards.

Considering the number of standards affected by the significance of the changes, and the interjurisdictional nature of the recommendations of this program it will require a significant effort to secure implementation of the recommended changes. It is suggested that the following steps should be followed in this effort.

(1) Establish the appropriate logic flow and priority for the recommended changes

(2) Establish a "shepherding" person or persons for each set of changes on a standard by standard or Jurisdictional by Jurisdictional basis.

(3) Insure that adequate technical resources are in place to provide the technical support in responding committee concerns, questions, and information requests.

5.0 Summary, Ancillary Data, Recommendations for Further Investigation

5.1 Summary

The changes recommended for subject Industry codes and standards focus on four major areas:

(a) Unique design features of the Advanced Reactor Designs

(b) The consolidation, clarification, and upgrading of existing standards to provide more concise design guidance.

(c) Addressing Existing Defiences and Shortcomings design guidance in existing standards

(d) Development of new standards to provide needed design guidance currently not supplied by industry consensus standards.

If the recommended changes were implemented in the subject standards it would provide for simplification and streamlining of the construction process for Advanced Reactors. It would also enhance the quality, technical design basis, and overall safety of these reactors. Finally it would help reduce the costs associated with both new construction and operation.

Considering these factors the authors suggest that the USNRC initiate the process of securing the incorporation of these suggested recommended modifications into the subject standards. This document and the knowledge gained through this program provides a strong basis for the presentations to the appropriate standards organizations concerning the need and basis of the recommended standard changes. It provides both a needs analysis and the recommended changes to address this need.

The consensus process associated with incorporation of these types of changes is deliberately slow and can easily spread over a 2 to 3 year time period. For that reason this effort should be initiated as soon as possible.

5.2 Ancillary Information

In addition to the recommended code changes, this program has resulted in a significant amount of valuable ancillary information including:

(a) Correlation of structures, Industry Standards, and Regulatory, Guidelines to the appropriate sections of the existing SRP

(b) Identification of the Safety Class, Seismic Category I Structures for the eight Advanced Reactor Designs. Also correlating this information to the applicable construction standard(s).

(c) Identification of the unique structural features for each of the eight subject Advanced Reactors Designs.

(d) Identification of the USNRC concerns with the subject codes and standards

(e) Identification of deficiencies in the subject codes and standards

(f) A prospective on "Lessons Learned" from structures subjected to strong motion earthquakes. This includes observations from on site investigations of the recent Northridge, California (1994) and Kobe, Japan (1995) Earthquakes.

(g) Detailed Comparisons of the current versions of selected standards to those versions currently cited in the Standard Review Plan

(h) Detailed comparison of the Wind and Tornado Design Basis for all eight of the subject Advanced Reactor Designs.

This report was tabulated and formatted such that this ancillary information is readily accessible and can be used by various groups within the USNRC for ongoing research programs and licensing efforts.

5.3 Suggestions for Further Investigation

This section provides some suggested areas for future investigation which would be complimentary to this program.

It is suggested that a Industry Standard applicability review effort similar to this program for structures should be considered for Electrical and Mechanical components associated with the Advanced Reactors Designs. This would include a special focus on the modular construction of electrical/mechanical component packages currently proposed for the Westinghouse AP600 and several advanced reactors.

It is suggested that a program which conducts a detailed regulatory impact evaluation for the standards compared in Appendix B. This would provide a detailed assessment of the differences in the SRP cited verses the current version, possible exceptions to the current versions which should be considered by the USNRC, and recommended changes to the SRP to implement the latest versions of these codes and standards.

It is also suggested that a follow on program be considered to upgrade this report to the later (ongoing) revisions of the Westinghouse AP600 SSAR.

6.0 References

Provided in this section is the complete listing of all reference documents reviewed as part of this program. These references are segmented by Reactor Design, USNRC Criteria, Industry Consensus Codes and Standards and Miscellaneous References. This was done to allow for the easy addition of reference documents as they became available during the course of the program.

6.1 Westinghouse Electric Corporation AP600 Related Reference Documents

6.1-[1] Westinghouse Electric Corporation, "AP600 Standard Safety Analysis Report," W Document #DEAC03905F18495, Vols. 1-9, Proprietary Vols. 1-3, Revision 2, March 31, 1995. Portions are proprietary information. Not publicly available.

6.1-[2] Westinghouse Electric Corporation, "Finned Floor Modules," Technical Description, July 16, 1993. Proprietary information. Not publicly available.

6.1-[3] Westinghouse Electric Corporation, "Overview of Status and Issues with the AP600 SSAR," Presentation Handouts, Received March 30, 1993. Proprietary information. Not publicly available.

6.1-[4] Westinghouse Electric Corporation, Johnson, et. al., "Modular Construction Approach for Advanced Nuclear Plants," presented at the ANS International Conference, October 30-November 4, 1988 in Washington, D.C.

6.1-[5] M.K. Ferguson, "MK Ferguson Construction Plan, Revision 2," November 1991. Proprietary information. Not publicly available.

6.1-[6] Stevenson and Associates, "Minutes/Handouts from AP600 Meeting between Westinghouse Electric Corporation and the USNRC," held February 10, 1993 at USNRC White Flint Offices, Washington, D.C. Proprietary information. Not publicly available.

6.1-[7] Brookhaven National Laboratory, "Reliability of Modular Construction," Program Description and Overview, January 25, 1993. USNRC Proprietary information. Not publicly available.

6.1-[8] Brookhaven National Laboratory, "Reliability of Modular Construction; Phase I Report - Identification of the Issues," Draft, August 1993. USNRC proprietary information. Not publicly available.

6.1-[9] United States Nuclear Regulatory Commission, "Summary of Meeting to Discuss Modular Construction," Meeting between USNRC and Avondale Shipyards, April 12, 1994.

6.2 ASEA Brown Boveri/Combustion Engineering System 80+ Related Reference Documents

6.2-[1] ASEA Brown Boveri/Combustion Engineering, "Combustion Engineering Standard Safety Analysis Report," Chapters 1-6, App. 7A, Chapters 9-11, Chapter 15, Chapter 19, Amendment W, June 17, 1994. Portions are proprietary information. Not publicly available.

6.2-[2] ASEA Brown Boveri/Combustion Engineering, Letter from C. Brinkman to USNRC, "Modifications to Topical Report CENPD-210, Revision 7," Letter No. LD-92-062, April 30, 1992.

6.2-[3] ASEA Brown Boveri/Combustion Engineering, Letter from C. Brinkman to USNRC, "System 80+ Structural Design Information," Letter No. LD-93-042, March 10, 1993. Proprietary information. Not publicly available.

6.2-[4] ASEA Brown Boveri/Combustion Engineering, Letter from V. Paggen to T. Adams, "System 80+ Industrial Codes and Standards," LD-93-101, June 25, 1993.

6.2-[5] United States Nuclear Regulatory Commission, "Draft Safety Evaluation Report Related to the Design Certification of CE System 80+," NUREG-1462, September 1992.

6.2-[6] United States Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Design Certification of CE System 80+," NUREG-1462, August 1994.

6.3 General Electric Corporation Advanced Boiling Water Reactor Related Reference Documents

6.3-[1] General Electric Corporation, "Advanced Boiling Water Reactor - Standard Safety Analysis Report," GE Document #23A6100, Chapters 1-3, App. 3A, Chapters 4-9, App. 9A, Chapters 10-20, Revision 5, Amendment 34, May 1994. Portions are proprietary information. Not publicly available.

6.3-[2] Bechtel Power Corporation, "Tornado and Extreme Wind Design for Nuclear Power Plants," Topical Report #BC-TOP-3A, Revision 3, August 1974. Proprietary information. Not publicly available.

6.3-[3] United States Nuclear Regulatory Commission, "Preliminary Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor," NUREG-1469, October, 1992.

6.3-[4] United States Nuclear Regulatory Commission, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor," NUREG-1503, July 1994.

6.3-[5] United States Nuclear Regulatory Commission, "Safety Evaluation of the ABWR Piping Design and Related ITAAC," June 30, 1992.

6.3-[6] United States Nuclear Regulatory Commission, "Letter from G. Bagchi to R. Pierson, "Geosciences and Structural Engineering Inputs to Final Safety Evaluation Report on GE Advanced Reactors Section 2.1, 2.2, 2.4 through 2.5.6, 3.3.1, 3.3.2 and 3.4.2," June 23, 1992.

6.4 General Electric Corporation Simplified Boiling Water Related Reference Documents

6.4-[1] General Electric Corporation, "Simplified Boiling Water Reactor Standard Safety Analysis Reports," GE Doc. No. 25A5113, Chapters 1-3, App. 3E, Chapter 5, 6, Chapter 9, Revision A, August 1992.. Portions are proprietary information. Not publicly available.

6.4-[2] General Electric Corporation, "Extract from SBWR Technical Description for NRC Staff," December 1989.

6.4-[3] General Electric Corporation, "Letter from P. Marriott to J. Wilson, "Use of Modular Construction in the SBWR," Letter No. MFN No. 008-93, January 18, 1993.

6.4-[4] Bechtel Power Corporation, M.P. Lagache, "Constructibility Enhancements for Advanced BWR's," presented at the 1992 American Nuclear Power Conference, October 1992.

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NRC FORM 335 (2-89) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) NUREG/CR-6358 Vol. 1
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J. Costello, NRC Project Manager

11. ABSTRACT *(200 words or less)*

Through out its history, the USNRC has been committed to the use of US industry consensus standards for the design, construction, and licensing of commercial nuclear power facilities. The existing industry standards are based on the current class of light water reactors and as such may not adequately address design and construction features of the next generation of Advanced Light Water Reactors and other types of Advanced Reactors. As part of their on-going commitment to industry standards, the USNRC commissioned this study to evaluate U.S. industry structural standards for application to Advanced Reactors. The initial review effort included: (1) the review and study of the relevant reactor design basis documentation for eight Advanced Reactor Designs, (2) the review of the USNRC's design requirements for advanced reactors, (3) the review of the latest revisions of the relevant industry structural standards, and (4) the identification of the need for changes to these standards. The results of these studies were used to develop recommended changes to industry consensus structural standards which will be used in the construction of Advanced Reactors. Over seventy sets of proposed standard changes were recommended and the need for the development of four new standards was identified. In addition to the recommended standard changes, several other sets of information were extracted for use by USNRC in other programs. This information included: (1) detailed observations on the response of structures and distribution system supports to the recent Northridge, CA (1994) and Kobe, Japan (1995) earthquakes, (2) comparison of versions of certain standards cited in the standard review plan to the most current versions, and (3) comparison of the seismic and wind design basis for the subject reactor designs. Finally provided is a suggested plan of action for implementation of the recommended industry standard changes.

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