GE Nuclear Energy

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Design and Analysis Similarities Between the ABWR and SBWR

Simplified Boiling Water Reactor

SBWR

9401060270 931229
PDR ADOCK 05200004
A PDR

GE Nuclear Energy

175 Curtner Avenue
San Jose, CA 95125

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DESIGN AND ANALYSIS SIMILARITIES BETWEEN THE ABWR AND SBWR

R. C. Challberg

Reviewed: _____
J. E. Leatherman, Manager
Licensing
SBWR Project

Approved: _____
P. W. Marriott, Manager
SBWR Project

DESIGN AND ANALYSIS SIMILARITIES

BETWEEN THE ABWR AND SBWR

Abstract

This report compares the Advanced Boiling Water Reactor (ABWR) and the Simplified Boiling Water Reactor (SBWR) Standard Safety Analysis Reports (SSAR) to determine where the design, philosophy or analysis methods are identical or similar. The purpose of this study is to assist the review of the SBWR Standard Safety Analysis Report and the Certified Design Material (Tier 1) by identifying similarities, thus eliminating the need for additional NRC review. Where there are minor differences, they are explained. The report indicates which sections contain significant differences, requiring normal review time and resources.

The study has found 181 sections out of a total of 380 SBWR SSAR sections (48%) to be either technically identical or similar with only minor differences (Category A or B) to the ABWR SSAR.

Category	Description	Number of Sections
A	Technically identical	114
B	Technically similar with minor differences	67
C	Significant differences	162
D	No comparison, technically different	37
	Total	380

At this time, significant differences exist between the certified design material (Tier 1) of both designs, primarily due to the changes to the ABWR Certified Design Material document after submittal of the SBWR Tier 1 document. The study also indicates which sections of the Tier 1 document are anticipated to be identical or similar to the ABWR finalized version, once the SBWR Tier 1 document is revised.

DESIGN AND ANALYSIS SIMILARITIES

BETWEEN THE ABWR AND SBWR

Purpose

The purpose of this report is to document at this point in time the identities and similarities in design, analysis methods and design philosophy between the Advanced Boiling Water Reactor (ABWR) and the Simplified Boiling Water Reactor (SBWR). The documentation of those identities and similarities is performed to assist in the review of the SBWR Standard Safety Analysis Report (SSAR) and the Certified Design Material document (Tier 1).

Scope

This study was made comparing the various sections of both SSARs and Tier 1 documents, looking for components, systems, structures, analysis methods or design philosophy which may be technically identical or similar with only minor differences. Since the phrase "similar with minor differences" is a relative term and open to interpretation, the minor differences noted between the two SSARs have been documented. No such tabulation of minor differences between the Tier 1 document sections has been made at this time, due to the current disparities. There are a number of sections in both Tier 1 documents which are anticipated to result in identical or technically similar sections once the SBWR Tier 1 document has been re-written to reflect the recent changes resulting from negotiations and discussions between the NRC, GE, NUMARC, DOE, EPRI and others.

The study categorized the various sections of the SSAR and Tier 1 document into the following four categories:

Category A	Technically identical
Category B	Technically similar with minor differences
Category C	Significant differences
Category D	No comparison, technically different

Refer to Table 1 for the complete listing of the SSAR sections. Refer to Table 2 for a listing of those SSAR minor differences. Refer to Table 3 for a listing of the Tier 1 document comparison.

Study Details

The study compared Amendment 31 and 32 of the ABWR SSAR (23A6100, Rev. 1) and Revision A (25A5113) of the SBWR SSAR.

The ABWR Design Certification Material (Tier 1) (25A5447, Revision 0) was compared to the SBWR Tier 1 Design Certification Document (25A5354, Revision A). Anticipated similarities are documented in Table 3. A future revision of this study will reflect the final comparison of these two documents.

The next revision of the SBWR SSAR and Tier 1 document will take advantage of the many discussions conducted during review of the ABWR documents. The SBWR documents will be revised to be identical or very similar, where applicable, to its predecessor. Notes 1 and 7 in Table 1 indicate where specific cases of similarity are known.

The study does not attempt to explain the significant differences found in Category C, but assumes that, because of their significance, the NRC reviewer will perform a complete review. The results of this study will change as the interaction between reviewer and designer progresses. It is the designer's intent to maintain the comparison current as the review status develops.

SSAR Section	Section Title	Category	Notes
Chapter 1	**Introduction and General Description of Plant**		
1.1	Introduction	C	
1.2	General Plant Description	C	
1.3	Comparison Tables	C	
1.4	Identification of Agents and Contractors	B	Some uncompleted plants omitted
1.5	Requirements for Further Technical Information	D	
1.6	Material Incorporated by Reference	C	
1.7	Engineering Drawings	D	
1.8	Interfaces for Standard Design	C	
1.9	Conformance with Standard Review Plan and Applicability of Codes and Standards	C	
App. 1A	Response to TMI Related Matters	C	
App. 1B	Failure Modes and Effects Analysis	D	
App. 1C	Conformance Assessment of the SBWR Design with the ALWR URD	D	
Chapter 2	**Site Characteristics**		
2.0	Site Characteristics	A	Note 3, page 31
2.1	Geography and Demography	A	Note 3, page 31
2.2	Nearby Industrial, Transportation, and Military Facilities	A	Note 3, page 31
2.3	Meteorology	A	Note 3, page 31
2.4	Hydrologic Engineering	A	Note 3, page 31
2.5	Geology, Seismology, and Geotechnical Engineering	A	Note 3, page 31
2.6	Requirements for Determination of Site Acceptability	B	Severe accidents use MACCS code
2.7	COL License Information	B	Severe accidents use MACCS code
App. 2A	Input to the MACCS Computer Code	C	
Chapter 3	**Design of Structures, Components, Equipment, and Systems**		
3.1	Conformance with NRC General Design Criteria	C	
3.2	Classification of Structures, Systems, and Components	C	
3.3	Wind and Tornado Loadings	B	Note 1, page 31
3.4	Water Level (Flood) Design	C	

SSAR Section	Section Title	Category	Notes
3.5	Missile Protection		
3.5.1	Missile Selection and Description	B	Note 1, page 31
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	A	
3.5.3	Barrier Design Procedures	A	
3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping	B	Methodology same; Note 1, page 31
3.7	Seismic Design	B	Methodology same, Seismic Category II used
3.8	Design of Seismic Category I Structures	C	
3.9	Mechanical Systems and Components		
3.9.1	Special Topics for Mechanical Components	B	
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	C	
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports and Core Support Structures	C	
3.9.4	Control Rod Drive Systems	A	
3.9.5	Reactor Pressure Vessel Internals	C	
3.9.6	Inservice Testing of Pumps and Valves	C	
3.9.7	COL License Information	A	Note 1, page 31
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment	A	Note 1, page 31
3.11	Environmental Qualification Of Safety-Related Mechanical and Electrical Equipment	A	Note 1, page 31
3.12	Tunnels	B	Note 7, page 32 - Section to be added
App. 3A	Seismic Soil-Structure Interaction Analysis	A*/C	*Subsection 3A.5 only - Analysis Method
App. 3B	Computer Programs Used	A	Note 1, page 31
App. 3C	Guidelines For LBB Application	B	Examples slightly different
App. 3D	Equipment Qualification Design Environmental Conditions	C	
App. 3E	Evaluation of Results of Seismic Category I Structures	C	
Chapter 4	**Reactor**		
4.1	Summary Description	C	
4.1.1	Reactor Pressure Vessel	C	

SSAR Section	Section Title	Category	Notes
4.1.2	Reactor Internal Components	C	
4.1.3	Reactivity Control Systems	B	Supplemental reactivity control
4.1.4	Analysis Techniques	B	Neutron fluence code
4.1.5	COL License Information	A	
4.1.6	References	D	
4.2	Fuel System Design	A	
4.2.1	Design Basis	A	Note 7, page 32
4.2.2	Description and Design Drawings	B	Shorter active fuel length (Note 7)
4.2.3	Design Evaluation	A	Note 7, page 32
4.2.4	Testing, Inspection and Surveillance Plans	A	Note 7, page 32
4.3	Nuclear Design	A	Note 7, page 32
4.3.1	Design Basis	A	Note 7, page 32
4.3.2	Nuclear Design Description	B	732 bundles vs. 872 bundles (Note 7)
4.3.3	Analytical Methods	A	Note 7, page 32
4.4	Thermal and Hydraulic Design	C	
4.4.1	Description of Thermal and Hydraulic Design of the Reactor Coolant System	C	
4.4.2	Loose Parts Monitoring System	B	Mounting locations of sensors (Note 1)
4.4.3	COL License Information	A	
4.5	Reactor Materials		
4.5.1	Control Rod Drive System Structural Materials	A	
4.5.2	Reactor Internal Materials	B	Differences based on new components
4.5.3	COL License Information	A	Note 1, page 31
4.6	Functional Design of Fine Motion Control Rod Drive Systems		
4.6.1	Information for Control Rod Drive Systems	B	High pressure makeup capability
4.6.2	Evaluation of the CRDS	A	
4.6.3	Testing and Verification of the CRDS	A	Note 1, page 31
4.6.4	Information for Combined Performance of Reactivity Control Systems	A	
4.6.5	Evaluation of Combined Performance	A	
4.6.6	COL License Information	A	
4.6.7	References	A	

SSAR Section	Section Title	Category	Notes
App. 4A	Typical Control Rod Patterns and Associated Power Distribution for SBWR	C	
App. 4B	Fuel Licensing Acceptance Criteria	A	Note 1, page 31
App. 4C	Control Rod Licensing Acceptance Criteria	A	Note 1, page 31
App. 4D	Stability Evaluation	D	
Chapter 5	**Reactor Coolant System and Connected Systems**		
5.1	Summary Description	C	
5.2	Integrity of Reactor Coolant Pressure Boundary		
5.2.1	Compliance with Codes and Code Cases	A	
5.2.2	Overpressure Protection	C	
5.2.3	Reactor Coolant Pressure Boundary Materials	A	Note 1, page 31
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	B	Support skirt weld different (Note 1)
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	C	
5.2.6	COL License Information	C	
5.2.7	References	C	
5.3	Reactor Vessel		
5.3.1	Reactor Vessel Materials	B	Higher fluence to vessel wall (Note 1)
5.3.2	Pressure-Temperature Limits	B	Higher fluence to vessel wall (Note 1)
5.3.3	Reactor Vessel Integrity	B	Shroud support different (Note 1)
5.3.4	COL License Information	A	Note 1, page 31
5.3.5	References	B	Second reference not applicable to SBWR
5.4	Component and Subsystem Design		
5.4.1	Reactor Recirculation System	D	
5.4.2	Steam Generators (PWR)	A	
5.4.3	Reactor Coolant Piping	A	
5.4.4	Main Steamline Flow Restrictors	B	Two instead of four
5.4.5	Main Steamline Isolation System	B	Two steamlines, slightly higher flow
5.4.6	Isolation Condenser System (ICS)	D	
5.4.7	Residual Heat Removal (RHR) System	C	
5.4.8	Reactor Water Cleanup /Shutdown Cooling System	C	
5.4.9	Main Steamlines and Feedwater Piping	B	Differs by number and size of feedwater

SSAR Section	Section Title	Category	Notes
5.4.10	Pressurizer	A	
5.4.11	Pressurizer Relief Discharge System	A	
5.4.12	Reactor Coolant System High Point Vents	D	
5.4.13	Safety/Relief Valves	D	
5.4.14	Component Supports	A	
5.4.15	COL License Information	A	Note 1, page 31
5.4.16	References	A	
Chapter 6	**Engineered Safety Features**		
6.0	Engineered Safety Features - General	C	
6.1	Engineered Safety Feature Materials	A	
6.1.1	Metallic Materials	C	
6.1.2	Organic Materials	B	Environmental conditions different
6.2	Containment Systems		
6.2.1	Containment Functional Design	C	
6.2.2	Passive Containment Cooling System	D	
6.2.3	Safety Envelope Functional Design	C	
6.2.4	Containment Isolation System	C	
6.2.5	Flammability Control System	D	
6.2.6	Containment Leakage Testing	A	
6.2.7	COL License Information	C	
6.2.8	References	C	
6.3	Emergency Core Cooling System		
6.3.1	Design Basis and Summary Description	C	
6.3.2	Gravity Driven Cooling System	D	
6.3.3	Automatic Depressurization Subsystem	C	
6.3.4	ECCS Performance Evaluation	C	
6.3.5	COL License Information	C	
6.3.6	References	C	
6.4	Control Room Habitability Systems	C	
6.5	Fission Product Removal		

SSAR Section	Section Title	Category	Notes
6.5.1	ESF Atmosphere Cleanup Systems	D	
6.5.2	Containment Spray as a Fission Product Cleanup System	A	
6.5.3	Fission Product Control Systems and Structures	C	
6.5.4	Ice Condenser as a Fission Product Cleanup System	C	
6.5.5	Suppression Pool as a Fission Product Cleanup System	D	
6.5.6	COL License Information	C	
6.5.7	References	D	
6.6	Inservice Inspection of Class 2 and 3 Components	B	Note 1, page 31
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	D	Note 1 and note 4, page 31
App. 6A	Containment Loads	C	
Chapter 7	**Instrumentation and Control Systems**		
7.1	Instrumentation and Controls - Introduction	C	
7.2	Reactor Trip System		
7.2.1	Reactor Protection System	B	No MS high radiation trip (Note 1)
7.2.2	Neutron Monitoring System	B	Eight versus ten SRNM channels (Note 1)
7.2.3	Suppression Pool Temperature Monitoring System	A	Note 1, page 31
7.2.4	COL License Information	C	
7.3	Engineered Safety Features Systems		
7.3.1	Emergency Core Cooling System	D	
7.3.2	Passive Containment Cooling System	D	
7.3.3	Leak Detection and Isolation System	C	
7.3.4	Safety System Logic and Control	C	
7.3.5	Essential Multiplexing System	B	Simplified system versus ABWR
7.3.6	Flammability Control System	D	
7.3.7	COL License Information	C	
7.4	Shutdown Systems		
7.4.1	Standby Liquid Control	C	
7.4.2	Remote Shutdown System	C	
7.4.3	Reactor Water Cleanup/Shutdown Cooling System	C	
7.4.4	Isolation Condenser System	D	

SSAR Section	Section Title	Category	Notes
7.4.5	Alternate Rod Insertion	A	
7.4.6	COL License Information	C	
7.5	Information Systems Important to Safety		
7.5.1	General I&C Conformance to Regulatory Guide 1.97	C	
7.5.2	Containment Atmospheric Monitoring System	B	Same functions, SBWR simpler system
7.5.3	Process Radiation Monitoring System	C	
7.5.4	Area Radiation Monitoring System	A	
7.5.5	Other Information Systems	C	
7.5.6	COL License Information	C	
7.6	Interlock Systems Important to Safety		
7.6.1	HP/LP System Interlock Function	C	
7.6.2	Safety/Non-Safety System Interlocks	C	
7.6.3	COL License Information	C	
7.7	Control Systems		
7.7.1	Nuclear Boiler System	B	No core plate or pump deck dP
7.7.2	Rod Control and Information System	B	Number of rods per gang different
7.7.3	Feedwater Control System	B	High water level trips (Note 1)
7.7.4	Automatic Power Regulator System	B	No output demand to recirc. control
7.7.5	Steam Bypass and Pressure Control System	B	No output demand to recirc. control
7.7.6	Process Computer System	B	Inputs and processes different
7.7.7	Non-Essential Multiplexing System	B	Slightly different input and outputs
7.7.8	Neutron Monitoring System - Non-Safety-Related Subsystem	C	
7.7.9	Containment Atmospheric Control System	B	No containment vent capability
7.7.10	Other Control Systems	C	
7.7.11	COL License Systems	C	
App. 7A	A Fixed In-Core Calibration System for the Neutron Monitoring System	D	
Chapter 8	**Electric Power**		
8.1	Electric Power-Introduction	C	
8.2	Offsite Power System	C	
8.3	Onsite Power Systems	C	

SSAR Section	Section Title	Category	Notes
Chapter 9	**Auxiliary Systems**		
9.1	Fuel Storage and Handling		
9.1.1	New Fuel Storage	C	
9.1.2	Spent Fuel Storage	B	Location and number different (Note 1)
9.1.3	Fuel and Auxiliary Pools Cooling System	C	
9.1.4	Light Load Handling System (Related to Refueling)	C	
9.1.5	Overhead Heavy Load Handling	C	
9.2	Water Systems		
9.2.1	Plant Service Water System	C	
9.2.2	Reactor Component Cooling Water System	C	
9.2.3	Make-up Water System	C	
9.2.4	Potable and Sanitary Water Systems		Not in the SBWR scope
9.2.5	Ultimate Heat Sink	D	
9.2.6	Condensate Storage and Transfer System	C	
9.2.7	Chilled Water System	C	
9.2.8	Turbine Component Cooling Water System	C	
9.2.9	COL License Information	C	
9.3	Process Auxiliaries		
9.3.1	Compressed Air System	A	See 9.3.6 and 9.3.7
9.3.2	Process and Post-Accident Sampling Systems	B	Different sample points
9.3.3	Equipment and Floor Drainage Systems	C	
9.3.4	Chemical and Volume Control System (PWR)	A	Not applicable to BWRs
9.3.5	Standby Liquid Control System (BWR)	C	
9.3.6	Instrument Air System	B	Different cooling system
9.3.7	Service Air System	B	Breathing air system included
9.3.8	High Pressure Nitrogen Supply System	C	
9.3.9	Hydrogen Water Chemistry System	B	Note 1, page 31
9.3.10	Oxygen Injection System	A	Note 1, page 31
9.3.11	COL License Information	D	
9.4	Air Conditioning, Heating, Cooling, and Ventilation		

SSAR Section	Section Title	Category	Notes
9.4.1	Control Room Area Ventilation System	C	
9.4.2	Spent Fuel Pool Area Ventilation System	C	
9.4.3	Radwaste HVAC System	A	Note 8, page 32
9.4.4	Turbine Building HVAC System	C	
9.4.5	Engineered Safety Feature Ventilation System		Not applicable to the SBWR
9.4.6	Reactor Building HVAC System	C	
9.4.7	Drywell Cooling System	C	
9.4.8	Containment Atmospheric Control System	B	No containment vent, oxygen less than 4%
9.4.9	COL License Information	D	
9.5	Other Auxiliary Systems		
9.5.1	Fire Protection System	C	
9.5.2	Communications Systems	B	Parts of system left up to COL
9.5.3	Lighting System	C	
9.5.4	Diesel-Generator Fuel Oil Storage and Transfer System	C	
9.5.5	Diesel-Generator Jacket Cooling Water System	C	
9.5.6	Diesel-Generator Starting Air System	C	
9.5.7	Diesel-Generator Lubrication System	C	
9.5.8	Diesel-Generator Combustion Air Intake and Exhaust System	C	
9.5.9	COL License Information	C	
Chapter 10	**Steam and Power Conversion System**		
10.1	Summary Description	C	
10.1.1	Protective Features	B	No reheat, high velocity moisture separation
10.1.2	COL License Information	A	
10.2	Turbine Generator		
10.2.1	Design Basis	C	
10.2.2	Description	C	
10.2.3	Turbine Disk Integrity	A	
10.2.4	Evaluation	A	
10.2.5	COL License Information	A	Note 1, page 31
10.2.6	References	A	

SSAR Section	Section Title	Category	Notes
10.3	Main Steam Supply System		
10.3.1	Design Bases	A	Note 1, page 31
10.3.2	Description	B	Number of steam lines, velocities
10.3.3	Evaluation	A	Note 1, page 31
10.3.4	Inspection and Testing	A	
10.3.5	Water Chemistry (PWR)	A	Not applicable to BWRs
10.3.6	Steam and Feedwater System Materials	A	
10.3.7	COL License Information	A	
10.3.8	References	A	
10.4	Other Features of Steam and Power Conversion Systems		
10.4.1	Main Condenser	C	
10.4.2	Main Condenser Evacuation System	A	
10.4.3	Turbine Gland Sealing System	B	Only one low pressure turbine
10.4.4	Turbine Bypass System	C	
10.4.5	Circulating Water System	B	SBWR discusses a cooling tower as heat sink
10.4.6	Condensate Purification System	B	SBWR has fewer demineralizer vessels
10.4.7	Condensate and Feedwater System	C	
10.4.8	Steam Generator Blowdown System (PWR)	A	Not applicable to BWRs
10.4.9	Auxiliary Feedwater System (PWR)	A	Not applicable to BWRs
10.4.10	COL License Information	A	
Chapter 11	**Radioactive Waste Management**		
11.1	Source Terms	B	Noble gas & iodine one-half of ABWR
11.2	Liquid Waste Management System	C	
11.3	Gaseous Waste Management System	B	One half condenser in-leakage
11.4	Solid Waste Management System	C	
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	C	
Chapter 12	**Radiation Protection**		
12.1	As Low As Reasonably Achievable (ALARA)	A	

SSAR Section	Section Title	Category	Notes
12.2	Plant Sources	C	
12.3	Radiation Protection	C	
12.4	Dose Assessment	C	
12.5	Operational Radiation Protection Program	A	
App. 12A	Calculation of Airborne Radionuclides	A	
Chapter 13	**Conduct of Operations**		
13.1	Organizational Structure of Applicant	A	
13.2	Training	A	
13.3	Emergency Planning	A	Note 1, page 31
13.4	Review and Audit	A	
13.5	Plant Procedures	A	Note 1, page 31
13.6	Physical Security	A	Note 1, page 31
Chapter 14	**Initial Test Program**		
14.1	Initial Plant Test Program - PSAR	A	
14.2	Initial Plant Test Programs - FSAR		
14.2.1	Summary of Test Program and Objectives	A	Note 1, page 31
14.2.2	Test Procedures	A	Note 1, page 31
14.2.3	Test Program's Conformance with Regulatory Guides	C	
14.2.4	Utilization of Reactor Operating and Testing Experience in the Development of Test Program	C	Note 6, page 31
14.2.5	Trial Use of Plant Operating and Emergency Procedures	A	
14.2.6	Initial Fuel Loading and Initial Criticality	B	Post fuel load vibration test not performed
14.2.7	Test Program Schedule and Sequence	C	
14.2.8	Individual Test Descriptions	C	
14.2.9	COL License Information	A	Note 1, page 31
14.2.10	References	C	
14.3	Design Certification Material	A	Note 1, page 31

SSAR Section	Section Title	Category	Notes
Chapter 15	**Accident Analyses**		
15.0	Accident Analyses	C	
15.0.1	Event Analytical Objective	C	
15.0.2	Analytical Categories	A	
15.0.3	Event Evaluation	C	
15.0.4	COL License Information	A	
15.0.5	References	C	
15.1	Decrease In Reactor Coolant Temperature		
15.1.1	Loss of Feedwater Heating	B	Methodology similar, results differ slightly
15.1.2	Feedwater Controller Failure - Maximum Demand	C	
15.1.3	Pressure Regulator Failure - Open	C	
15.1.4	Inadvertent Safety/Relief Valve Opening	C	
15.1.5	Spectrum of Steam System Piping Failures Inside and Outside Containment in a PWR	A	
15.1.6	Inadvertent RWCU/SDC Shutdown Cooling Operation	C	
15.1.7	COL License Information	A	
15.1.8	References	A	
15.2	Increase in Reactor Pressure		
15.2.1	Pressure Regulator Failure - Closure	C	
15.2.2	Generator Load Rejection	C	
15.2.3	Turbine Trip	C	
15.2.4	MSIV Closures	C	
15.2.5	Loss of Condenser Vacuum	C	
15.2.6	Loss of Non-Emergency AC Power to Station Auxiliaries	C	
15.2.7	Loss of Feedwater Flow	C	
15.2.8	Feedwater Line Break	C	
15.2.9	Failure of RWCU/SDC Shutdown Cooling	C	
15.2.10	COL License Information	C	
15.2.11	References	C	
15.3	Decrease In Reactor Coolant System Flow Rate		
15.3.1	Loss of Forced Reactor Coolant Flow - Trip of Pump	D	

SSAR Section	Section Title	Category	Notes
15.3.2	Loss of Forced Reactor Coolant Flow - Controller Malfunctions	D	
15.3.3	Reactor Coolant Pump Rotor Seizure	D	
15.3.4	Reactor Coolant Pump Shaft Break	D	
15.3.5	COL License Information	D	
15.3.6	References	D	
15.4	Reactivity And Power Distribution Anomalies		
15.4.1	Rod Withdrawal Error - Low Power	C	
15.4.2	Rod Withdrawal Error at Power	A	
15.4.3	Control Rod Maloperation (System Malfunction or Operator Error)	A	
15.4.4	Abnormal Startup of Idle Recirculation Pump	D	
15.4.5	Recirculation Flow Control Failure With Increasing Flow	D	
15.4.6	Chemical and Volume Control System Malfunctions	A	
15.4.7	Misplaced Bundle Accident	A	Note 1, page 31
15.4.8	Rod Ejection Accident	A	
15.4.9	Control Rod Drop Accident	A	
15.4.10	COL License Information	A	Note 1, page 31
15.4.11	References	C	
15.5	Increase in Reactor Coolant Inventory		
15.5.1	Inadvertent Startup of Isolation Condenser	D	
15.5.2	Chemical Volume Control System Malfunction (or Operator Error)	A	
15.5.3	BWR Transients Which Increase Reactor Coolant Inventory	A	
15.5.4	COL License Information	A	
15.5.5	References	A	
15.6	Decrease In Reactor Coolant Inventory		
15.6.1	Inadvertent Safety/Relief Valve Opening	A	
15.6.2	Failure of Small Line Carrying Primary Coolant Outside Containment	B	Methodology same, results slightly different
15.6.3	Steam Generator Tube Failure	A	
15.6.4	Steam System Piping Break Outside Containment	B	Methodology same, results different
15.6.5	Loss-of-Coolant Accident - Inside Containment	D	
15.6.6	Feedwater Line Break - Outside Containment	D	
15.6.7	COL License Information	C	

SSAR Section	Section Title	Category	Notes
15.6.8	References	C	
15.7	Radioactive Release		
15.7.1	Waste Gas System Failure	A	
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	A	
15.7.3	Postulated Radioactive Release Due to Liquid Radwaste Tank Failure	B	Methodology same, results different
15.7.4	Fuel Handling Accidents	B	Methodology same, results different
15.7.5	Spent Fuel Cask Drop Accident	B	Methodology same, results different
15.7.6	COL License Information	C	
15.7.7	References	C	
15.8	Anticipated Transients Without Scram		
15.8.1	Requirements	A	
15.8.2	Plant Capabilities	C	
15.8.3	Performance Evaluation	C	
15.8.4	COL License Information	A	
15.8.5	References	B	Second reference not used
Chapter 16	**Technical Specifications**	Note 2	Currently under development
Chapter 17	**Quality Assurance**		
17.0	Introduction	B	Table 17.0-1 different; Note 1, page 31
17.1	Quality Assurance During Design And Construction	B	Design approval different
17.2	Quality Assurance During the Operations Phase	A	
17.3	Reliability Assurance Program During Design Phase	B	Different example; Note 1, page 31
Chapter 18	**Human Factors Engineering**		
18.1	Introduction	B	App. 18G not included; Note 1, page 31
18.2	Design Goals and Design Bases	A	
18.3	Planning, Development and Design	A	Note 1, page 31
18.4	Control Room Standard Design Features	A	Note 1, page 31
18.5	Remote Shutdown System	A	Note 1, page 31
18.6	Systems Integration	A	Note 1, page 31

SSAR Section	Section Title	Category	Notes
18.7	Detailed Design of the Operator Interface System	A	Note 1, page 31
App. 18A	Emergency Procedures Guidelines	C	
App. 18B	SBWR EPG Compared To Generic EPG	C	
App. 18C	EPG Input Data and Calculation Results	C	
App. 18D	Operator Interface Equipment Characterization	B	Arrangement of control room
App. 18E	SBWR Man-Machine Interface Systems Design and Implementation	A	Note 1, page 31
	Process		
App. 18F	Emergency Operation and Controls	C	
Chapter 19	**Response to Severe Accident Policy Statement**		
19.1	Introduction	C	
19.2	Plant Description, Assumptions, and Methodology	C	
19.3	Internal Event Accident Sequence Analysis	C	
19.4	Containment Performance	C	
19.5	External Events Analysis	C	
19.6	Consequence Analysis	C	
19.7	Sensitivity Studies	C	
19.8	Conclusions	C	
App. 19A	Internal Events	C	
App. 19B	Containment Performance	C	
App. 19C	External Events	C	
App. 19D	External Events (Seismic)	C	
App. 19E	Consequence Analysis	C	
App. 19F	Miscellaneous	C	
App. 19G	Response to CP/ML Rule	B	
App. 19H	USI/GSI Applicability	D	
Chapter 20	**Question and Response Guide**	C	
Chapter 21	**Engineering Drawings**	C	

Table 2

SSAR Sections - Category B

Technically Similar With Minor Differences

Chapter 1 Introduction and General Description of Plant

1.4 Identification of Agents and Contractors

The SBWR Table 1.4-1 has omitted some of the GE BWR plants which were not completed or which did not reach low power conditions, such as Shoreham, Alto Lazio 1 and 2, and Hope Creek 2.

Chapter 2 Site Characteristics

2.6 Requirements for Determination of Site Acceptability

The SBWR site acceptability for severe accidents is based upon the use of the MACCS computer code, while ABWR design has utilized the CRAC 2 computer code. MACCS code is the successor to the CRAC 2 code.

2.7 COL License Information

The SBWR site acceptability for severe accidents is based upon the use of the MACCS computer code, while ABWR design has utilized the CRAC 2 computer code. MACCS code is the successor to the CRAC 2 code.

Chapter 3 Design of Structures, Components, Equipment, and Systems

3.3 Wind and Tornado Loadings

The methodology and input values are identical to ABWR. The one minor difference is that ABWR does not categorize structures as Seismic II, whereas SBWR does and therefore SBWR applies the listed wind and tornado loads to both Seismic Category I and II structures.

3.5.1 Missile Selection and Description

The equipment requiring protection will be somewhat different for each plant design based upon their unique features. In addition, the ABWR SSAR discusses the RCIC turbine as a potential missile source, which does not apply to the SBWR. The discussion in the ABWR SSAR regarding protection of the Standby Gas Treatment System does not apply to the SBWR.

3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

The methodology for analysis and break scenarios is identical for both designs. The tables listing the safety-related equipment and high energy piping are different based upon the unique features of each design.

3.7 Seismic Design

The methodology used for seismic design for both the ABWR and SBWR is the same. The SBWR utilizes Seismic Category II design for some structures, systems and components. This category applies where the structure, system or component performs no safety-related function, and whose continued function is not required, but whose structural failure or interaction could degrade the functioning of a Seismic Category I structure, system or component to an unacceptable safety level, or could result in incapacitating injury to occupants of the control room. In addition, the SBWR reactor building embedment is different by less than 3 meters.

3.9.1 Special Topics for Mechanical Components

The ABWR discusses the ECCS pumps, SLC pumps, RCIC turbine and RRS motors for faulted loading conditions. These active components do not exist on the SBWR design. Also, for the ABWR design, an inelastic analysis is used for the reactor internal pumps (RIP) motor casing restraints. The SBWR does not incorporate internal pumps and therefore does not have pump motor casing restraints.

3.12 Tunnels

This section will be added to the SBWR SSAR and it is anticipated that the description will be very similar to the comparable section in the ABWR SSAR.

Appendix 3C
Subsection 3C.6 Guidelines for Preparation of an LBB Report

The examples given for analysis of main steamline and feedwater piping are slightly different due to the differences in ABWR versus SBWR design (e.g., four steamlines, DPV's, etc.)

Chapter 4 Reactor

4.1.3 Reactivity Control Systems

This short section discusses the mechanisms which can be used to control reactivity. The SBWR cannot use core flow changes directly to change the

percent of voids in the core and thereby change the reactivity. Major power changes and control will be performed by control rod withdrawal or insertion.

4.1.4 Analysis Techniques

The computer code used to perform the neutron fluence calculations for the SBWR is DORT, which is the successor to DOT4.4, used in the ABWR analysis.

4.2.2 Description and Design Drawings

The SBWR fuel design has an active fuel length of 2743 mm (108 in.) versus 3708 mm (146 in.) for the ABWR design. The SBWR fuel assembly has four water rods versus two for ABWR. The control rod has a different lower end configuration shown in Figure 4.2-1.

4.3.2 Nuclear Design Description

The SBWR will have 732 fuel assemblies in the core versus 872 assemblies for the ABWR.

4.4.2 Loose Parts Monitoring System

The Loose Parts Monitoring System design requirements are identical for both the ABWR and SBWR. The only minor difference would be the potential mounting locations of the sensors. Due to differences in the external configuration of the two reactor vessels, the sensors may be located at different locations or on different vessel nozzles. The design will locate the sensors at strategic locations where potential loose parts may naturally collect.

4.5.2 Reactor Internal Materials

The minor differences are a result of the different internal components used for the SBWR. The list in Subsection 4.5.2.1 includes the chimney and chimney partitions which are unique to the natural circulation design of the SBWR. The chimney head serves the same function as the shroud head in the ABWR. The majority of the internal components are manufactured from either Type 304L or 316L stainless steel, with the exceptions as shown.

4.6.1 Information for Control Rod Drive System

The SBWR CRD System provides high pressure makeup water to the reactor when the normal supply (feedwater) is unable to maintain water level and to prevent the water level from falling below Level 2. In addition, the SBWR CRD System only provides high pressure purge flow to the Reactor Water Cleanup/Shutdown Cooling pumps (not the RIPs as with the ABWR).
The Fine Motion Control Rod Drive (FMCRD) design is identical for both the ABWR and SBWR except for the length of stroke. The ABWR stroke is 3658 mm (144 in.), while the SBWR stroke is 2616 mm (103 in.). Both drives are designed to provide the same 100% scram insertion time of ≤ 2.80 seconds,

which is used in all transient and accident analyses. Therefore, the SBWR control rod, with its shorter stroke, actually travels slower than the ABWR control rod, the end result being lower impact loads to the control rod and drive at the end of the scram stroke.

Chapter 5 Reactor Coolant System and Connected Systems

5.2.4 Preservice and Inservice Inspection and Testing of Reactor Coolant Pressure Boundary

The philosophy and methodology for preservice and inservice inspection and testing are the same for both the ABWR and SBWR. Section XI of the ASME Code provides the primary source of requirements for inspection. The reactor vessel support skirt weld (knuckle-to-skirt) of the SBWR is designed to be outside the boundary of Subsection IWB. Other differences between the two reactor designs are reflected in the Examination Categories Table (Tables 5.2-8 and 5.2-6, respectively).

5.3.1 Reactor Vessel Materials

The materials, fabrication processes and compliance with the applicable Regulatory Guides are the same for both the ABWR and SBWR. Minor differences exist in the radiation effects upon the beltline material due to the different neutron fluences experienced. In addition, the SBWR vessel beltline surveillance test material will have additional surveillance specimens included for determination of K_{IC} at "upper shelf" temperature.

5.3.2 Pressure Temperature Limits

The methodology used to determine the vessel pressure temperature limitations to prevent brittle fracture is identical for both plants. The predicted fluence for the SBWR vessel wall is higher than that of the ABWR.

5.3.3 Reactor Vessel Integrity

The shroud support of the ABWR is different than that of the SBWR. The ABWR uses a number of support legs welded to the bottom of the vessel, whereas the SBWR uses a number of brackets welded to the side wall of the vessel. The number of nozzles and the use for each nozzle will be different for each design. The expected end-of-life neutron fluence will be higher for the SBWR because of the narrower annulus between the fuel and the vessel wall.

5.3.5 References

The SBWR SSAR does not utilize the second reference quoted in the ABWR SSAR.

5.4.4 Main Steamline Flow Restrictors

The ABWR has four steamlines each using a flow restrictor. The SBWR only has two main steamlines each with a flow restrictor. All other design features are the same.

5.4.5 Main Steamline Isolation System

The ABWR design utilizes four main steamlines, while the SBWR has only two. The rated flow of the main steam isolation valves (MSIV) is about 1.5% higher for the SBWR.

5.4.9 Main Steamlines and Feedwater Piping

The SBWR design uses two main steamlines versus four for ABWR, both using 28-in. piping. In addition, the SBWR feedwater lines are smaller than those of the ABWR. The SBWR has four feedwater inlet nozzles at the vessel versus six for the ABWR.

Chapter 6 Engineered Safety Features

6.1.2 Organic Materials

The methodology for both designs is the same. The environmental conditions for any organic materials will be different due to the differences in both configurations and accident scenarios.

6.6 Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping

The philosophy and methodology for preservice and inservice inspection and testing are the same for both the ABWR and SBWR. Section XI of the ASME Code provides the primary source of requirements for inspection. Differences between the two reactor designs are reflected in the Examination Categories Table (Table 6.6-1).

Chapter 7 Instrumentation and Control Systems

7.2.1 Reactor Protection System

The SBWR design does not provide a scram upon rapid core flow decrease, nor on main steamline high radiation. In addition, the SBWR has added a reactor vessel high water level scram.

7.2.2 Neutron Monitoring System

The SBWR utilizes eight Source Range Neutron Monitoring (SRNM) channels versus ten channels for the ABWR. The SBWR utilizes 21 Local Power Range Monitoring (LPRM) strings, while the ABWR utilizes 52 LPRM strings. In addition, for the SBWR, each division of the Power Range Neutron Monitoring (PRNM) has its own independent Local Power Range Monitors (LPRM) but averages all LPRM values from all PRNM divisions and transmits through fiber optic cable to all PRNM divisions to form the final Average Power Range Monitor (APRM) values.

7.3.5 Essential Multiplexing System

For the SBWR design, all output trips are hard-wired. The SBWR has a reduced number of manual ECCS controls which are hard-wired to load-drivers provided primarily to explosive valves or solenoids.

7.5.2 Containment Atmospheric Monitoring System

The SBWR design uses a simpler, more universal design, with combined gas calibration functions. Both the SBWR and ABWR systems provide the same functions to satisfy Regulatory Guide 1.97.

7.7.1 Nuclear Boiler System

The SBWR does not measure the core plate differential pressure, nor the pump deck differential pressure. GE is currently working on a modification to the water level instrumentation to resolve the reference column non-condensable gas issue.

7.7.2 Rod Control and Information System

The number of rods per rod gang are different because of the difference in total number of control rods for each design. For the SBWR design, the non-safety Multi-channel Rod Block Monitor (MRBM) and the safety-related Automatic Thermal Limit Monitor are both part of the RC&IS. With the ABWR, the MRBM function is part of the non-safety portion of the Neutron Monitoring System.

7.7.3 Feedwater Control System

The vessel high water level functions are different for each plant. For the SBWR, upon reaching water Level 8, the protective action will trip the main turbine and reduce feedwater demand to zero. Upon reaching water Level 9, the control system will trip the feedwater pumps. For the ABWR, upon reaching Level 8 all of these protective functions occur. The ABWR does not have a Level 9 trip function. For the ABWR, the Feedwater Control System sends a signal to the Reactor Recirculation System to runback core flow upon reaching water Level 4 and to trip three of the internal pumps upon reaching Level 3. These functions do not exist for the SBWR.

7.7.4 Automatic Power Regulator System

The SBWR system does not provide an output demand signal to the Reactor Recirculation Control System.

7.7.5 Steam Bypass and Pressure Control System

The SBWR design does not provide an output demand signal to the Recirculation Control System.

7.7.6 Process Computer System

The SBWR design and design philosophy of the Process Computer System are the same as those of the ABWR. Different process parameters are provided as input and different calculational processes are performed for each plant, depending upon the particular design.

7.7.7 Non-Essential Multiplexing System

The SBWR design and design philosophy for the Non-Essential Multiplexing System are the same as those of the ABWR. Slightly different data as input and slightly different outputs may be provided, depending on the particular plant, but the design philosophy for the multiplexing system remains the same as the ABWR.

7.7.9 Containment Atmospheric Control System

The SBWR Containment Atmospheric Control System does not provide the capability for venting the containment during severe accident conditions.

Chapter 9 Auxiliary Systems

9.1.2 Spent Fuel Storage

The design requirements and philosophy are the same for both the ABWR and SBWR. The only minor differences are the location of the spent fuel storage pool and the total number of available spaces to store the spent fuel.

9.3.2 Process and Post-Accident Sampling System

There are minor variations in system details and sample points.

9.3.6 Instrument Air System

For the SBWR, the instrument air compressors are cooled by the Reactor Closed Cooling Water System instead of the Turbine Closed Cooling Water System, as on the ABWR. In addition, the Instrument Air System has no

containment penetrations but uses the High Pressure Nitrogen System piping to penetrate the containment. The SBWR compressors are the rotary screw type.

9.3.7 Service Air System

The Service Air System for the SBWR has incorporated the Breathing Air System into its design. The SBWR compressors are the rotary screw type.

9.3.9 Hydrogen Water Chemistry System

The HWC System is essentially the same for both the ABWR and SBWR but with slight variations due to differences in the design of both reactor systems.

9.4.3 Radwaste HVAC System

The SBWR general area exhaust pathway utilizes a HEPA filter.

9.4.8 Containment Atmospheric Control System

The SBWR containment is controlled to ≤ 4.0% O_2, rather than the ≤ 3.5% O_2 for the ABWR. In addition, the SBWR Containment Atmospheric Control System does not provide the capability for venting the containment during severe accident conditions. There are minor differences in the nitrogen delivery system.

9.5.2 Communications System

For the ABWR, the private automatic branch telephone exchange (PABX) and the in-plant radio system are left up to the COL applicant. These subsystems are briefly discussed in the SBWR SSAR.

Chapter 10 Steam and Power Conversion System

10.1.1 Protective Features

The SBWR turbine does not utilize moisture separator/reheaters, but utilizes high velocity moisture separators (HVS).

10.3.2 Description

The SBWR has only two main steamlines versus four on the ABWR. The ABWR main steamlines are designed for 150 ft/sec normal steam flow, while the SBWR steamlines are designed for 160 ft/sec normal steam flow.

10.4.3 Turbine Gland Sealing System

The SBWR main turbine has only one low pressure turbine versus three for the ABWR main turbine, which results in minor piping configuration differences.

10.4.5 Circulating Water System

The two plants have essentially identical circulating water systems, but the SBWR discusses a cooling tower as the heat sink, while the ABWR designates the type of heat sink as a COL license issue.

10.4.6 Condensate Purification System

The SBWR currently has four demineralizers, while the ABWR has six.

Chapter 11 Radioactive Waste Management

11.1 Source Terms

The SBWR source terms for noble gas fission products and iodine fission products are one-half the comparable ABWR values based upon the shorter active fuel length and lower power density, combined with fuel of current improved designs for the SBWR. The methodology for calculating the waste management source terms is the same; refer to the applicable tables for actual quantities. This section in the ABWR SSAR has been moved to Subsection 12.2.1 in the SBWR SSAR.

11.3 Gaseous Waste Management System

The Offgas System for the SBWR is designed to accommodate one half the condenser in-leakage as the ABWR; thus, the sizing of equipment is commensurate with this reduced design requirement (e.g., the charcoal storage requirement is reduced). In addition, the ABWR Offgas System contains a HEPA filter, which does not exist in the SBWR system.

Chapter 14 Initial Test Program

14.2.6 Initial Fuel Loading and Initial Criticality

For the SBWR, the pre-critical, post-fuel load flow test of the reactor internals for flow-induced vibration assessment cannot be performed.

Chapter 15 Accident Analyses

15.1.1 Loss of Feedwater Heating

The methodology used in the analysis for both reactor designs is the same. The only minor differences are in the final results. The ABWR experienced a Δ CPR of 0.07, while the SBWR experienced a Δ CPR of 0.12. The feedwater temperature drop for ABWR was 30°C, while for the SBWR the drop was 35°C.

15.6.2 Failure of Small Line Carrying Primary Coolant Outside Containment

The methodology is the same, but the values used in the methodology and the results are different.

15.6.4 Steam System Piping Break Outside Containment

The methodology is the same, but the values used in the methodology and the results are different.

15.7.3 Postulated Radioactive Release Due To Liquid Radwaste Tank Failure

The methodology used in the analysis for both reactor designs is the same. The only differences are in the final results.

15.7.4 Fuel Handling Accidents

The methodology is the same and varies by numeric plant detail. In addition, the ABWR analysis uses the SBGT System to filter the release to the environment, while the SBWR does not use such a system.

15.7.5 Spent Fuel Cask Drop Accident

The methodology used in the analysis for both reactor designs is the same. The only differences are in the final results.

15.8.5 References

The second reference quoted for Amendment 32 of the ABWR SSAR is not used for the SBWR SSAR.

Chapter 17 Quality Assurance

17.0 Introduction

Table 17.0-1 shows some minor differences. For the SBWR, there will be no exception taken to Regulatory Guide 1.28, Rev. 3. In addition, Regulatory Guides 1.58, 1.64, 1.174, 1.88, 1.123, 1.144, and 1.146 have been withdrawn since the ABWR SSAR was originally written.

17.1 Quality Assurance During Design and Construction

SBWR SSAR Subsection 17.1.3, "Design Control", specifies that GE exclusively will be the approval authority for all design documents. For the Kashiwazaki 6 and 7 ABWR, GE and the two technical associates have been the approval authority for the design documents. For the U.S. ABWR, GE exclusively is the design approval authority.

17.3 Reliability Assurance Program During Design Phase

The example used in the ABWR SSAR to demonstrate the implementation of the D-RAP program is different than the example used for the SBWR.

Chapter 18 Human Factors Engineering

18.1 Introduction - Human Factors Engineering

Appendix 18G, "Design Development and Validation Testing", of the ABWR SSAR is not included in the SBWR SSAR. This appendix is a summary of the design history of the ABWR control room design and is also applicable to the SBWR control room design, which follows the technological advances of the ABWR.

Appendix 18D Operator Interface Equipment Characterization

The arrangement of the SBWR control room, as shown in the example sketch, is slightly different than the example ABWR control room.

Chapter 19 Response to Severe Accident Policy Statement

Appendix 19G Response to CP/ML Rule

All subsections are technically identical except for the following subsections which would be considered in Category C:

19G.2.11
19G.2.12
19G.2.27
19G.2.36
19G.2.44
19G.2.45

Tier 1 Section	Section Title	Category	Notes
1.0	**Introduction**	A	Note 5, page 31
1.1	General Plant Description	D	Delete - Note 5, page 31
1.2	General Provisions	B	Note 5, page 31
1.3	Definitions	A	Note 5, page 31
2.0	**Design Descriptions and ITAACs**		
2.1	Nuclear Steam Supply	C	Note 5, page 31
2.2	Controls and Instrumentation	C	Note 5, page 31
2.3	Radiation Monitoring Systems	B	Note 5, page 31
2.4	Core Cooling Systems	C	Note 5, page 31
2.5	Reactor Servicing Equipment	C	Note 5, page 31
2.6	Reactor Auxiliary Systems	C	Note 5, page 31
2.7	Control Panels	B	Note 5, page 31
2.8	Nuclear Fuel	A	Note 5, page 31
2.9	Radioactive Waste Management System	C	Note 5, page 31
2.10	Power Cycle	B	Note 5, page 31
2.11	Station Auxiliaries	C	Note 5, page 31
2.12	Station Electrical System	C	Note 5, page 31
2.13	Power Transmission	A	Note 5, page 31
2.14	Containment and Environmental Control Systems	C	Note 5, page 31
2.15	Structures and Servicing Systems	C	Note 5, page 31
2.16	Intake Structure and Servicing Equipment	C	Note 5, page 31
2.17	Yard Structures and Equipment	C	Note 5, page 31
3.0	**Non-System Based Tier 1 Material**		
3.1	Piping Design	B	Note 5, page 31
3.2	Safety System Logic and Control	C	Note 5, page 31
3.3	Software Development	A	Incorporate into 3.2 - Note 5, page 31
3.4	Human Factors Engineering	A	Note 5, page 31
3.5	Radiation Protection	C	Note 5, page 31

Tier 1 Section	Section Title	Category	Notes
4.0	**Interface Tier Material**		
4.1	Ultimate Heat Sink	C	Note 5, page 31
4.2	Offsite Power System	A	Incorporate into 2.12 - Note 5, page 31
4.3	Potable and Sanitary Water System	A	Incorporate into 2.12 - Note 5, page 31
4.4	Plant Service Water System	C	Incorporate into 2.12 - Note 5, page 31
4.5	Circulating Water System	A	Incorporate into 2.12 - Note 5, page 31
4.6	Makeup Water System	C	Note 5, page 31
4.7	Communication System	A	Incorporate into 2.12 - Note 5, page 31
4.8	Airborne Particulate Radiation Monitoring System	D	To be deleted - Note 5, page 31
5.0	**Site Parameters**	A	Note 5, page 31
App. A	**Legend for Figures Included in Tier 1**	A	Note 5, page 31

Comparison Notes

(As found in the SSAR and Tier 1 Section Comparisons, Table 1 and Table 3)

Note 1: The ABWR SSAR has recently been submitted as Amendments 31, 32 and 33, incorporating many changes as a result of Tier 1, ITACCS discussions and review comments and resolutions. The next revision of the SBWR SSAR will take advantage of these many discussions during review of the ABWR document and the many changes incorporated. The SBWR SSAR will be updated to be more in line with its predecessor in this particular section. The comparison category has been selected in anticipation of this SBWR SSAR rewrite.

Note 2: Currently, Chapter 16, "Technical Specifications", is under development for the ABWR; therefore, no comparison has been attempted at this time.

Note 3: Chapter 2, "Site Characteristics", is written in a different format than the ABWR SSAR, but the envelope of site-related parameters used in the design of the SBWR and ABWR are technically identical.

Note 4: Section 6.7, "Main Steam Isolation Valve Leakage Control System (BWR)", does not discuss the MSIV Leakage Control System per se. The philosophy is the same for both plant designs. Credit is taken for the structural integrity of the main steamlines and the main condenser for holdup and plate-out of radionuclides. Section 6.7 of the ABWR SSAR discusses the High Pressure Nitrogen System.

Note 5: The Tier 1 documents as currently reviewed have major differences which, in some cases, will be changed to reflect the recent dynamic nature of the ABWR Tier 1 document. A later revision of this report will show a system-by-system comparison with technically identical or technically similar sections indicated in the same manner as the SSAR comparison.

Note 6: This section requires revision to more clearly identify the significant differences between SBWR and previous BWR designs.

Note 7: Sections with Note 7 either do not exist or are under different section title name in the current revision of the SBWR SSAR. They are expected to be either added or replaced in the future revisions of the SSAR. The new sections will incorporate the recent changes that have taken place in the ABWR SSAR.

Note 8: The SBWR Radwaste HVAC System design will have the HEPA filter removed based upon a recent design review. Following this change, the two systems will be the same.

GE Nuclear Energy

175 Curtner Avenue
San Jose, CA 95125