Chapter 1

VCS UFSAR Table of Contents

VCS UFSAR Formatting Legend

Color	Description
Black	Original Westinghouse AP1000 DCD Tier 2 & Tier 2*, Revision 19 content
Purple	Departures from AP1000 DCD Tier 2 & Tier 2*, Revision 19 content
Blue	Standard FSAR content
Green	Site-specific FSAR content
Red	Linked cross-references (chapters, appendices, sections, subsections, tables, figures, and references)

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Chapter 1 Introduction and General Description of the Plant

1.1 Introduction

This Design Control Document (DCD) for the Westinghouse AP1000 simplified passive advanced light water reactor plant is incorporated by reference into the Design Certification Rule for the AP1000 design (Section II.A) of Appendix D to 10 CFR Part 52. The DCD is also submitted to the NRC for review and approval of an application for an amendment to the Design Certification Rule for the AP1000.

This Final Safety Analysis Report (FSAR) incorporates the Design Control Document (DCD) (as identified in Table 1.6-201) for a simplified passive advanced light water reactor plant provided by Westinghouse Electric Company, the entity originally sponsoring and obtaining the AP1000 design certification documented in 10 CFR Part 52, Appendix D. Throughout this FSAR, the “referenced DCD” is the AP1000 DCD submitted by Westinghouse as Revision 19 including any supplemental material as identified in Table 1.6-201. Unless otherwise specified, reference to the DCD refers to Tier 2 information, including references to the sensitive unclassified non-safeguards information (including proprietary information) and safeguards information, contained in the AP1000 DCD. Such DCD information is included in this combined license application in the same manner as it is included in the AP1000 DCD, i.e., references in the DCD are included as references in the FSAR, and material incorporated by reference into the DCD is incorporated by reference into the FSAR. Appropriate agreements are in place to provide for the licensee’s rights to possession (including constructive possession) and use of the withheld sensitive unclassified non-safeguards information (including proprietary information) and safeguards information referenced in the AP1000 DCD for the life of the project.

Appendix D to 10 CFR Part 52 is hereby incorporated by reference into the COL application.

This FSAR is hereby submitted under Section 103 of the Atomic Energy Act by SCE&G to the NRC as part of the application for two Class 103 combined licenses (COLs) to construct and operate two nuclear power plants under the provisions of 10 CFR Part 52 Subpart C.

1.1.1 Plant Location

The site for VCSNS Units 2 and 3 is located approximately 1 mile from the center of Unit 1 in western Fairfield County. The Monticello Reservoir provides the water requirements of Units 2 and 3. The reservoir is located east of the Broad River and west of South Carolina State Highway 215.

Units 2 and 3 will be located approximately 15 miles west of Winnsboro, the county seat. Newberry, the county seat of Newberry County, is about 17 miles away in a westerly direction. Columbia, the South Carolina state capital, is located about 26 miles to the southeast.

Figure 1.1-201 identifies the site location. Figure 1.1-202 shows the plant arrangement within the site.

The AP1000 is a standardized plant that is to be placed on a site with parameters described in Chapter 2, "Site Characteristics." The site parameters relate to the seismology, hydrology, meteorology, geology, heat sink, and other site-related aspects.

1.1.2 Containment Type

The containment building is a freestanding, cylindrical, steel containment vessel with elliptical upper and lower heads. It is surrounded by a seismic Category I shield building constructed of reinforced concrete and steel concrete composite modules. The containment vessel is an integral part of the passive containment cooling system. The vessel provides the safety-related interface with the ultimate heat sink, which is the surrounding atmosphere. Westinghouse is responsible, along with their contractor team members, for the design of the containment.

1.1.3 Reactor Type

The nuclear steam supply system (NSSS) for the AP1000 is a Westinghouse-designed pressurized water reactor.

1.1.4 Power Output

The plant's net producible electrical power to the grid is at least 1000 MWe, with a core power rating of 3400 MWt. In some safety evaluations a power level higher than the rated power level is employed.

1.1.5 Schedule

The scheduled completion date and estimated commercial operation date of nuclear power plants referencing the AP1000 design certification are provided as discussed in Subsection 1.1.7.

Table 1.1-203 displays the anticipated schedule for construction and operation of two AP1000 units at the VCSNS site. A site-specific construction plan and startup schedule will be provided to the NRC after issuance of the COL and when a final decision has been made to construct the plant.

1.1.6 Format and Content

1.1.6.1 Regulatory Guide 1.70

To the extent practical, the AP1000 DCD has used as a guide the format and content recommendations of Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," November 1978.

The DCD generally uses the same chapter, section, subsection, and paragraph headings used in the standard format. Where appropriate, the DCD is subdivided beyond the extent of the standard format to provide additional information specifically required for that area. Similarly, some of the passive features of the AP1000 require modification of the standard format and content either in terms of placement or type of material presented.

This FSAR generally follows the AP1000 DCD organization and numbering. Some organization and numbering differences are adopted where necessary to include additional material, such as additional content identified in Regulatory Guide 1.206.

1.1.6.2 Standard Review Plan

The technical guidance provided in NUREG-0800, is followed in the preparation of the AP1000 DCD. Standard Review Plan conformance is also determined in accordance with 10 CFR 50.34 to identify the deviations of the AP1000 DCD from the Standard Review Plan. See Subsection 1.9.2 for additional details on Standard Review Plan conformance.

1.1.6.3 Text, Tables, and Figures

AP1000 DCD tables of data are identified by the section or subsection number followed by a sequential number (for example, Table 3.2-3 is the third table of Section 3.2). Tables are located at the end of the section immediately following the text. Drawings, pictures, sketches, curves, graphs, plots, and engineering diagrams are identified as figures and are numbered sequentially by section or subsection similar to tables, and follow at the end of the applicable section or subsection.

FSAR tables, figures, and references are numbered in the same manner as the DCD, but the first new FSAR item is numbered as 201, the second 202, the third 203, and consecutively thereafter.

New appendices are included in the FSAR with double letter designations following the pertinent chapter (e.g., 12AA).

When it provides greater contextual clarity, an existing DCD table or figure is revised by adding new information to the table or figure and replacing the DCD table or figure with a new one in the FSAR. In this instance, the revised table or figure clearly identifies the information being added, and retains the same numbering as in the DCD. RN-13-004

1.1.6.4 Numbering of Pages

Text pages are numbered sequentially within each section or subsection.

1.1.6.5 Proprietary Information

The AP1000 DCD contains no proprietary information.

Some portions of this FSAR may be considered as proprietary, personal, or sensitive and withheld from public disclosure pursuant to 10 CFR 2.390 and Regulatory Issue Summary (RIS) 2005-026. Such material is clearly marked and the withheld material is separately provided for NRC review.

1.1.6.6 Acronyms

Table 1.1-1 provides a list of acronyms used in the AP1000 DCD and VCS Units 2 and 3 FSAR. Acronyms for systems are defined in the section in which they are used. Other acronyms may be defined in the section in which they are used. Table 1.7-2 provides a list of AP1000 **and** VCS Units 2 and 3 system designators.

1.1.7 Combined License Information

The construction and startup schedule information is addressed in Subsection 1.1.5.

Table 1.1-1 (Sheet 1 of 7)
AP1000 Acronyms

27J	Newberry Airport
ac	Alternating Current
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AFB	Air Force Base
AFCCC	Air Force Combat Climatology Center
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	As-Low-As-Reasonably Achievable
ALOHA	Areal Location of Hazardous Atmosphere
ALWR	Advanced Light Water Reactor
AMCA	Air Movement and Control Association
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ANSS	Advanced National Seismic System
API	American Petroleum Institute
ARCON96	X/Qs computer model described in NUREG/CR-6331
ARI	Air Conditioning and Refrigeration Institute
ASCE	American Society of Civil Engineers
ASHRAE	American Society of Heating, Refrigeration and Air-Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AWS	American Welding Society
BEACON	Best Estimate Analyzer for Core Operations - Nuclear
BOL	Beginning of Life
BOP	Balance of Plant
BTP	Branch Technical Position
CAE	Columbia Metropolitan Airport
CAV	Cumulative Absolute Velocity
CEM	U.S. Army Corps of Engineers, Coastal Engineering Manual
CEUS	Central and Eastern United States
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CMAA	Crane Manufacturers Association of American
CMT	Core Makeup Tank
COL	Combined Operating License/Combined License
COLA	Combined License Application
CPT	Cone Penetrometer Test

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Table 1.1-1 (Sheet 2 of 7)
AP1000 Acronyms

CPW	Construction Potable Water
CR	Compression Ratio
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
CRR	Cyclic Resistance Ratio
CSA	Control Support Area
CSDRS	Certified Seismic Design Response Spectrum
CSR	Cyclic Stress Ratio
CVS	Chemical and Volume Control System
CWS	Circulating Water System
DAC	Design Acceptance Criteria
DB	Dry Bulb
DBA	Design Basis Accident
DBE	Design Basis Event
dc	Direct Current
DCD	Design Control Document
D-EHC	Digital Electrohydraulic Control
DEGADIS	Dense Gas Dispersion Computer Model
DEMA	Diesel Engine Manufacturers Association
DNAG	Geological Society of America, Decade of North American Geology (project)
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	U.S. Department of Energy
DS	Direct Shear
EAB	Exclusion Area Boundary
ECFS	East Coast Fault System
ECMA	East Coast Magnetic Anomaly
EMI	Electromagnetic Interference
EOF	Emergency Offsite Facility
EOF	Emergency Operations Facility
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EQPARAM	Electric Power Research Institute Computer Code
ER	Environmental Report
ERF	Emergency Response Facility
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
EST	Earth Science Team
ETR	Energy Transfer Ratio
ETSZ	Eastern Tennessee Seismic Zone
FAA	Federal Aviation Administration

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AP1000 Acronyms

FDW	Fairfield County Airport	
FERC	Federal Energy Regulatory Commission	
FID	Fixed Incore Detector	
FM	Factory Mutual Engineering and Research Corporation	
FMEA	Failure Modes and Effects Analysis	
FRISK88	Risk Engineering, Inc. computer model software	
FS	Factor of Safety	
FSAR	Final Safety Analysis Report	RN-12-044
GDC	General Design Criteria	
GMRS	Ground Motion Response Spectrum	
GSI	Generic Safety Issues	
HEC-RAS	Hydraulic Compacted Model Developed by the USAEC	
HEPA	High Efficiency Particulate Air	
HFE	Human Factors Engineering	RN-12-044
HVAC	Heating, Ventilation and Air Conditioning	
I&C	Instrumentation and Control	
IBC	International Building Code	RN-13-003
ICEA	Insulated Cable Engineers Association	
ICRP	International Commission on Radiation Protection	RN-12-044
IDLH	Immediately Dangerous to Life and Health	
IEEE	Institute of Electrical and Electronics Engineers	
IES	Illumination Engineering Society	
ILRT	Integrated Leak Rate Test	
INEL	Idaho National Engineering Laboratory	
I/O	Input/Output	
IPCS	Integrated Plant Computer System	
IRWST	In Containment Refueling Water Storage Tank	
ISA	Instrument Society of America	
ISI	Inservice Inspection	
IST	Inservice Testing	
ITAAC	Inspections, Tests, Analyses and Acceptance Criteria	RN-12-044
LBB	Leak-Before-Break	
LCD	Local Climatological Data	
LFL	Lower Flammability Limit	
LOCA	Loss of Coolant Accident	
LOFT	Loss of Flow Test	RN-12-044
LOOP	Loss of Offsite Power	
LOSP	Loss of System Pressure with Degraded ECCS Operation	
LPZ	Low Population Zone	
LSB	Last Stage Blade	
LST	Local Standard Time	

Table 1.1-1 (Sheet 4 of 7)
AP1000 Acronyms

LWR	Light Water Reactor
M	Moment Magnitude
MAAP	Modular Accident Analysis Programs
MACTEC	Mactec Engineering and Consulting of Charlotte, North Carolina
MCC	Motor Control Center
MCR	Main Control Room
MFCV	Main Feedwater Control Valve
MFIV	Main Feedwater Isolation Valve
ML	defined by Unified Soil System Classification
ML/MH	defined by Unified Soil System Classification
Mmax	Maximum Magnitude
MOV	Motor-operated Valves
MPC	Maximum Permissible Concentration
MSDS	Material Safety Data Sheet
MSF	Magnitude Scaling Factor
MSIV	Main Steam Isolation Valve
MSL	Mean Sea Level
MSLB	Main Steam Line Break
MW	Megawatt
MWe	Megawatts, electric
MWR	Moderately Weathered Rock
MWt	Megawatt, thermal
NAD 83	North American Datum, 1983
NAVD	North American Vertical Datum
NAVD88	North American Vertical Datum, 1988
NCDC	National Climatic Data Center
NEC	National Electrical Code
NEI	Nuclear Energy Institute
NEMA	National Electrical Manufacturers Association
NFPA	National Fire Protection Association
NGVD29	National Geodetic Vertical Datum, 1929
NOAA-CSC	National Oceanic and Atmospheric Administration Coastal Services Center
NPDES	National Pollutant Discharge Elimination System
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NUMARC	Nuclear Management and Resources Council (Superseded by NEI)
NUREG	Report designator for NRC reports
NWS	National Weather Service
ORE	Occupation Radiation Exposure
OSHA	Occupational Safety and Health Administration

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Table 1.1-1 (Sheet 5 of 7)
AP1000 Acronyms

PAVAN	Computer Code as described in NUREG/CR-2858
PBA	Power Block Area
PCS	Passive Containment Cooling System
PGA	Peak Ground Acceleration
pH	Potential of Hydrogen, measure of acid or base
P&ID	Piping and Instrumentation Diagram
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PRHR	Passive Residual Heat Removal
PRHR HX	Passive Residual Heat Removal Heat Exchanger
PSHA	Probabilistic Seismic Hazard Analysis
PSHAKE	Bechtel Computer Program for Equivalent Linear Seismic Response Analysis of Horizontally Layered Soil Deposits
PVC	Poly-Vinyl-Chloride
PWR	Pressurized Water Reactor
PWR	Partially Weathered Rock
PXS	Passive Core Cooling System
QA	Quality Assurance
QA / QC	Quality Assurance / Quality Control
RAP	Reliability Assurance Program
RCDT	Reactor Coolant Drain Tank
RCL	Reactor Coolant Loop
RCS	Reactor Coolant System
RCTS	Resonant Column Torsional Shear
RFI	Radio Frequency Interference
R.G.	Regulatory Guide
RIS	Regulatory Issue Summary
RMS	Radiation Monitoring System
RNS	Normal Residual Heat Removal
RQD	Rock Quality Designation
RR	Recompression Ratio
RSW	Remote Shutdown Workstation
RTD	Resistance Temperature Detector
RV	Reactor Vessel
SC14	Shealy Airport
SC63	Summer Nuclear Station; private unattended helipad
SCDHEC	South Carolina Department of Health and Environmental Control
SCDOT	South Carolina Department of Transportation
SCE&G	South Carolina Electric and Gas
SECPOP	Sandia Laboratories computer code used to calculate population by EPZs

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Table 1.1-1 (Sheet 6 of 7)
AP1000 Acronyms

SECY	Secretary of the Commission Letter
SEL	Site Elevation
SER	Safety Evaluation Report
SERCC	Southeast Regional Climate Center
SEUSSN	Southeastern United States Seismic Network
SM	Defined by Unified Soil System Classification
SMACNA	Sheet Metal and Air Conditioning Contractors National Association
SNM	Special Nuclear Material
SPT	Standard Penetration Test
SRP	Standard Review Plan
SSAR	Standard Safety Analysis Report
SSE	Safe Shutdown Earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	Soil Structure Interaction
STEL	Short Term Exposure Limit
SUNSI	Sensitive Unclassified Non-Safeguards Information
SWB	Service Water Building
SWS	Service Water System
TB	Turbine Building
TDS	Total Dissolved Solids
TID	Total Integrated Dose
TIP	NRC's Trial Implementation Project
TMI	Three Mile Island
TNT	Trinitrotoluene
TOC	Total Organic Carbon
TOXDISP	Computer model, estimates vapor concentrations at distance/elevation from toxic spills
TS	Technical Specification(s)
TSC	Technical Support Center
TWA	Time Weighted Average
UBC	Uniform Building Code
UCSS	Updated Charleston Seismic Source
UFL	Upper Flammability Limit
UFSAR	Updated Final Safety Analysis Report
UHRS	Uniform Hazard Response Spectra
UHS	Ultimate Heat Sink
UL	Underwriters Laboratories
UPS	Uninterruptible Power Supply
URD	Utility Requirements Document
USACE	U.S. Army Corps of Engineers
USCS	United Soil Classification System
USDA	U.S. Department of Agriculture

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Table 1.1-1 (Sheet 7 of 7)
AP1000 Acronyms

USGS	U.S. Geological Survey
USI	Unresolved Safety Issue
UST	Underground Storage Tank
UTM	Universal Transverse Mercator Coordinate System
VCIS	U.S. Forrest Service Ventilation Climate Information System
VCSNS	V. C. Summer Nuclear Station
V_s	Shear Wave Velocity
WB	Wet Bulb
WCAP	Westinghouse report designator, originally Westinghouse Commercial Atomic Power
WT	Water Treatment
XOQDOQ	Dispersion Model Outlined in Regulatory Guide 1.111
ZRA	Zone of River Anomalies

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Table 1.1-201
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Table 1.1-202
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Table 1.1-203
Estimated Completion Dates and Commercial Operation Dates for VCSNS Units 2 and 3

Unit 2

Commencement of Construction	2nd Q 2011	
Construction Completion	1st Q 2019	RN-15-086
Commercial Operation	2nd Q 2019	

Unit 3

Commencement of Construction	2nd Q 2011	
Construction Completion	4th Q 2019	RN-15-086
Commercial Operation	2nd Q 2020	

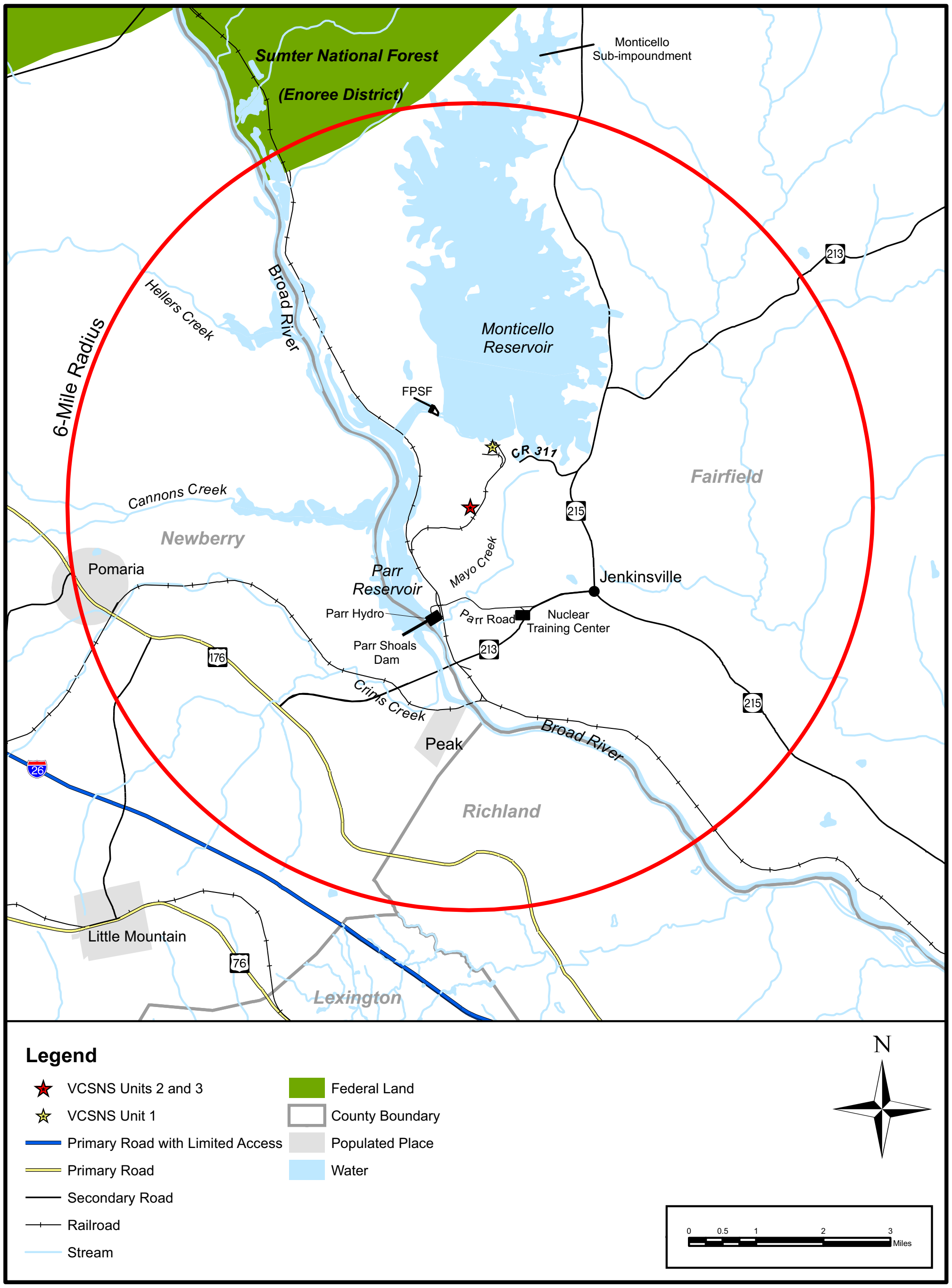

Figure 1.1-201 Site Location Map

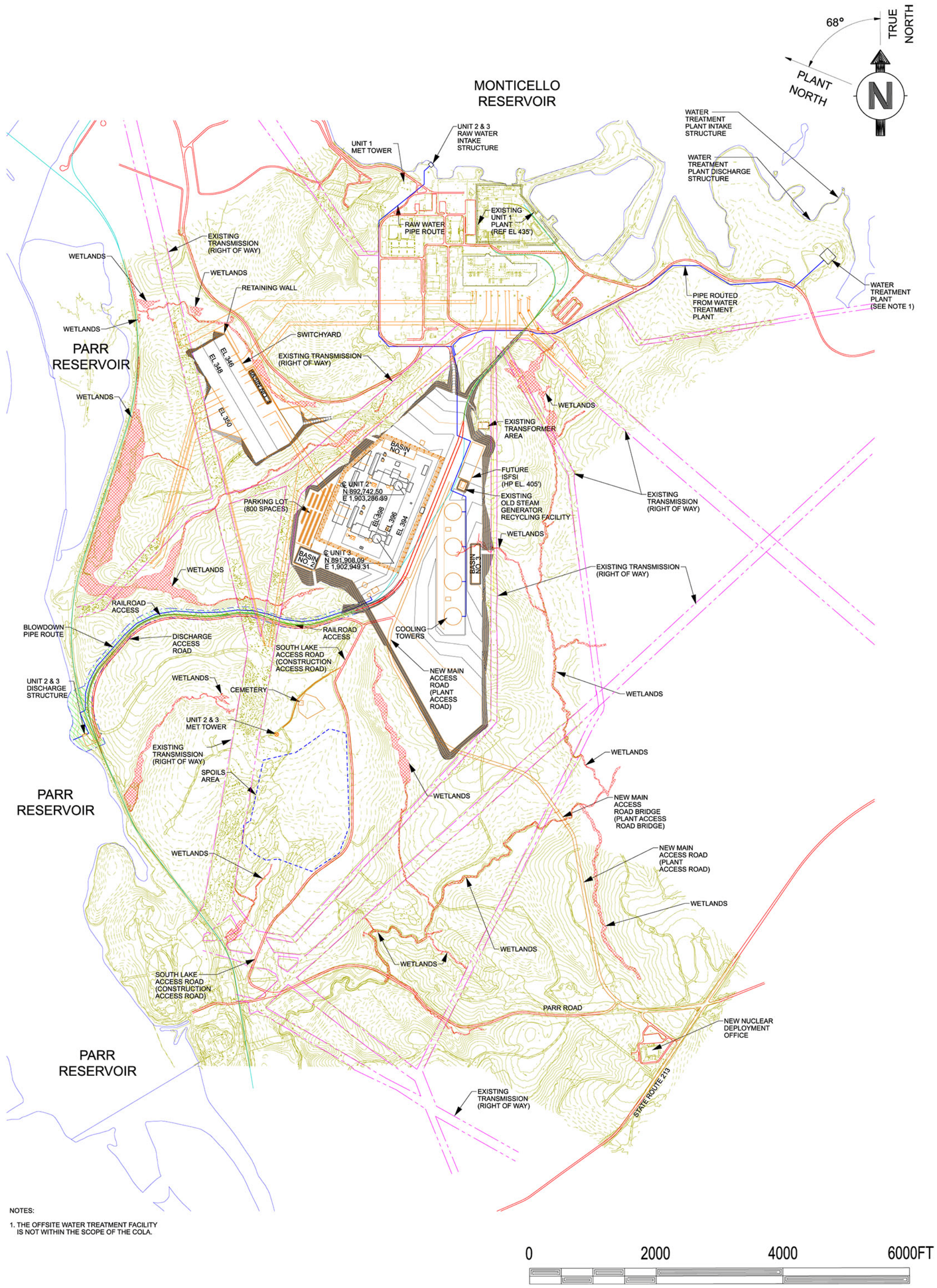

Figure 1.1-202 Site Layout

1.2 General Plant Description

This section includes a general discussion of the objectives, design criteria, operating characteristics and safety considerations for the AP1000 and provides a general description of the plant site, the site criteria, the general plant arrangement, the plant arrangement criteria and key features of each of the individual buildings that are collectively defined as the power generation complex.

Design Certification is sought for the power generation complex, excluding those elements and features considered site-specific. The AP1000 design extends beyond those structures, systems, and equipment which are safety-related. All safety-related structures, systems, and components are located on the nuclear island and are to be included in the design certification. To provide a better understanding of the safety-related features of the AP1000, nonsafety-related features are also described in this DCD. In addition, some plant design features which are outside the boundary of the AP1000, and considered to be site-specific, are described for completeness and to provide a basis for quantification of the required interfaces, as required by 10 CFR 52.47 (a)(1)(ix). The site-specific structures located off the nuclear island are neither safety-related nor seismic Category I. A more complete description of interfaces for the standard design is contained in Section 1.8.

1.2.1 Design Criteria, Operating Characteristics, and Safety Considerations

This section provides an overview of the AP1000 design objectives, design criteria, operating characteristics and safety considerations.

1.2.1.1 Overall Plant

The primary objective of the AP1000 design is to meet applicable safety requirements and goals defined for advanced light water pressurized water reactors with passive safety features. Since the AP600 has already received a Design Certification, it is also a design objective for AP1000 to be as similar as possible to the AP600.

Westinghouse was a principal participant in the development of the EPRI sponsored Utility Requirements Document (URD) and continues to be involved with EPRI on changes to that document. Therefore, an objective of the AP1000 design is to remain as consistent as possible with the EPRI URD. Additional design objectives for the AP1000 are to provide a greatly simplified plant with respect to design, licensing, construction, operation, inspection and maintenance. Specific design objectives follow.

1.2.1.1.1 Power Capability Objectives

- The plant's net electrical power to the grid is at least 1000 MWe with a nuclear steam supply system power rating (core plus reactor coolant pump heat) of about 3415 MWt.
- The plant is designed for rated performance with up to 10 percent of the steam generator tubes plugged and with a maximum hot leg temperature of 610°F.
- The plant is designed to accept a step load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam dump system actuation provided the rated power level is not exceeded.
- The plant is designed to accept a 100 percent load rejection from full power to house loads without reactor trip or operation of the pressurizer or steam generator safety valves. The design provides for a turbine capable of continued stable operation at house loads.

- The plant is designed to accept ramp load changes of 5 percent per minute while operating in the range of 25 to 100 percent of full power without reactor trip or steam dump actuation subject to core power distribution limits and provided the rated power level is not exceeded.

- The plant is designed to permit a design basis daily load follow cycle for at least 90 percent of the fuel cycle length. The design basis daily load follow cycle is defined as the daily (24 hour period) cycle of operation at 100 percent power, followed by a 2-hour linear ramp to 50 percent power, operation at 50 percent power and a 2-hour linear ramp back to 100 percent power. The duration of time at 50 percent power can vary between 2 and 10 hours. This load follow capability is achievable during 90 percent of each fuel cycle.

- During load follow the plant is designed to routinely make load changes of ≤ 10 percent at ± 2 percent per minute between 50 and 100 percent power without exceeding the core power distribution limits for the purpose of responding to grid frequency changes. No change to the reactor coolant boron concentration is required during these load follow maneuvers.

1.2.1.1.2 Reliability and Availability Objectives

- The overall plant availability goal is greater than 90 percent considering all forced and planned outages.

- The rate of unplanned reactor trips goal is less than one per year.

- The plant is designed with significantly fewer components and significantly fewer safety-related components than a current pressurized water reactor of a comparable size.

- The plant design objective is 60 years without the planned replacement of the reactor vessel which itself has a 60 year design objective based on conservative assumptions. The design provides for the replaceability of other major components, including the steam generators.

- The design of the major components required for power generation such as the steam generators, reactor coolant pumps, fuel, internals, turbine and generator is based on equipment that has successfully operated in power plants. Modifications to these proven designs were based on similar equipment that had successful operating experience in similar or more severe conditions.

1.2.1.1.3 Safety Design Criteria

- The plant design conforms to applicable regulations as discussed in Sections 1.9 and 3.1.

- The plant is designed to be fabricated, erected, and operated in such a manner that the release of radioactive materials to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations and for design basis transients and accidents.

- Gaseous and liquid waste disposal facilities are designed so that the discharge of radioactive effluents can be made in accordance with applicable regulations.

- The design provides means by which plant operators are alerted when limits on the release of radioactive effluent are approached.

- The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.

- The reactor is designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate plant systems.
- Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis.
- Essential safety actions are provided by equipment of sufficient redundancy and independence so that no single failure of active components can prevent the required actions.
- Provisions are made for control of active components of nuclear safety systems and engineered safety features from the control room.
- Those portions of the nuclear steam supply system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material containment barrier following design basis operational transients and accidents.
- Nuclear safety systems and engineered safety features functions are designed so that no damage to the reactor coolant pressure boundary results from internal pressures caused by design basis operational transients and accidents.
- Nuclear safety systems and engineered safety features are designed to permit demonstrations of their functional performance requirements.
- The design of nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.
- Standby electrical power sources have sufficient capacity to power the nuclear safety systems and engineered safety features requiring electrical power. Safety-related electrical power requirements needed during a loss of offsite power are supplied via Class 1E dc power.
- Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.
- A containment is provided which completely encloses the reactor system.
- The containment is designed to allow periodic integrity and leak tightness testing.
- The containment, in conjunction with other engineered safety features, limits the release of radioactivity from inside the containment, in the event of a design basis accident. This has the effect of limiting radiological consequences of a design basis accident to within an appropriate fraction of regulatory guidelines.
- Piping that penetrates the containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation is effected in time to limit radiological effects to less than the specified acceptable limits.
- Provisions are made for passively removing energy from the containment to maintain the integrity of the containment system following accidents that release energy to the containment.

- Passive core cooling systems are provided to limit fuel cladding temperature to less than the limits established by 10 CFR 50.46 in the event of a loss-of-coolant accident.
- The passive core cooling system provides for core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary.
- Actuation of the passive core cooling system occurs automatically when required, regardless of the availability of offsite power supplies and the normal generating system.
- The control room is shielded against radiation so that continued occupancy under accident conditions is possible.
- In the event that the control room becomes uninhabitable, it is possible to bring the reactor from power range operation to safe shutdown conditions by utilizing the remote shutdown workstation located outside the control room.
- Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shutdown the reactor from any normal operating conditions and subsequently to maintain the shutdown condition.
- The fuel handling and storage facility is designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.

1.2.1.1.4 Site Objectives

- The plant is designed for location at a site with the parameters set forth in Chapter 2, Site Characteristics.

1.2.1.1.5 Other Objectives

- The radiation exposure goal for plant personnel resulting from normal operation, inspection and maintenance is less than 100 man-Rem/year. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operations.
- The total low level radioactive waste volume goal is less than 1,970 cubic feet per year after de-watering. This waste includes items such as spent resins, spent filter elements, tank sludge, chemical wastes, and clothing. Spent condensate polishing resins are not included. The total wet radioactive waste volume produced from spent resin and filter elements, tank sludge and chemical waste is designed not to exceed 550 cubic feet per year (de-watered).

1.2.1.2 Reactor Coolant System Design

The AP1000 reactor coolant system (Figure 1.2-1) is designed to remove or to enable removal of heat from the reactor during all modes of operation, including shutdown and accident conditions.

The system consists of two heat transfer circuits, each with a steam generator, two reactor coolant pumps, a single hot leg and two cold legs, for circulating reactor coolant. In addition the system includes a pressurizer, interconnecting piping, valves and instrumentation necessary for operational control and safeguards actuation. All system equipment is located in the reactor containment.

During operation, the reactor coolant pumps circulate pressurized water through the reactor vessel and the steam generators. The water, which serves as coolant, moderator and solvent for boric acid (chemical shim control), is heated as it passes through the core to the steam generators where the heat is transferred to the steam system. The water is then is returned to the reactor (core) by the pumps to repeat the process.

The reactor coolant system pressure is controlled by operation of the pressurizer, where water and steam are maintained in equilibrium by the activation of electrical heaters and/or a water spray. Steam is formed by the heaters or condensed by the water spray to control pressure variations due to expansion and contraction of the reactor coolant.

Overpressure protection for the reactor coolant system is provided by the spring loaded safety valves installed on the pressurizer. These valves discharge to the containment atmosphere. The valves for the first three stages of automatic depressurization are also mounted on the pressurizer. These valves discharge steam through spargers to the in-containment refueling water storage tank. The discharged steam is condensed and cooled by mixing with water in the tank.

The reactor coolant system is also served by a number of auxiliary systems, including the chemical and volume control system, the passive core cooling system, the spent fuel pit cooling system, the steam generator system, the primary sampling system, the liquid radwaste system and the component cooling water system.

1.2.1.2.1 Reactor Design

- The core is designed for an 18-month fuel cycle.
- There are no reactor vessel penetrations below the top of the core.
- The core is designed for a moderator temperature coefficient that is non-positive over the entire fuel cycle at any power level with the reactor coolant at the normal operating temperature.
- A core design is maintained for projected fuel cycles.
- The core design provides adequate margin so that departure from nucleate boiling will not occur with a 95 percent probability and 95 percent confidence basis for all Condition I and II events.
- The core is located low in the vessel to minimize core temperature during loss-of-coolant accidents.
- The vessel and internals are designed so coolant at approximately the average of T_{cold} and T_{hot} is maintained in the head and control rod drive mechanism regions.
- The lower internals are designed to prevent flow jetting into the core.
- Bottom mounted incore instrumentation is not used. No vessel penetrations exist below the top of the core.
- An integrated head package which contains the control rod drive mechanisms, instrument columns, insulation, seismic support and package lift rig is employed.
- A permanent welded seal ring is used to provide the seal between the vessel flange and the refueling cavity floor.

1.2.1.2.2 Steam Generator Design

- The Model Delta 125 steam generator of proven design is employed. The steam generator employs thermally treated nickel-chromium-iron Alloy 690 tubes and a steam separator area sludge trap with clean out provisions.
- The channel head is designed for the direct attachment of two reactor coolant pumps.
- The channel head is designed for both manual and robotic accessibility for inspection, plugging, sleeving and nozzle dam placement operations.

1.2.1.2.3 Reactor Coolant Pump Design

- Sealless pumps of proven design are used.
- Two reactor coolant pumps are attached directly to each steam generator channel head with the motor located below the channel head to simplify the loop piping and eliminate fuel uncovery during small loss-of-coolant accidents.
- Each reactor coolant pump includes sufficient internal rotating inertia to provide a flow coastdown to avoid departure from nucleate boiling following a loss of reactor coolant flow accident.
- Each reactor coolant pump impeller and diffuser vanes are ground and polished to minimize radioactive crud deposition and to maximize pump efficiency.
- The reactor coolant pump motors are designed with appropriate lifting and handling attachments (lugs and trunnions) to facilitate maintenance.
- The reactor coolant pumps are designed such that they are not damaged due to a loss of all cooling water until a safety-related pump trip occurs on high bearing water temperature. This automatic protection is provided to protect the reactor coolant pumps from an extended loss of coolant water.

1.2.1.2.4 Pressurizer and Loop Arrangement

- The piping layout is designed for adequate thermal expansion flexibility assuming a fixed vessel and a free floating steam generator/reactor coolant pump support system.
- The reactor coolant loop and surge line piping are designed to leak-before-break criteria.
- The pressurizer is designed such that, with design spray flow rates, the power-operated relief valve function is not required nor provided.

1.2.1.3 Steam and Power Conversion System Design

1.2.1.3.1 Turbine Design

- The turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work which rotates a generator to provide electrical power. It consists of a double flow high pressure cylinder (high pressure turbine) and three double flow low pressure cylinders (low pressure turbines) which exhaust to the condenser. It is a six flow tandem compound, 1800 rpm machine. The turbine system includes stop, control and intercept valves directly attached to the turbine and in the steam flow path, crossover and crossunder piping between the turbine cylinders and the moisture separator reheater.

- The high pressure turbine has connections for two stages of feedwater heating. The high pressure turbine exhaust steam provides steam for one stage of feedwater heating in the deaerator. The low pressure turbines have extraction connections for four stages of feedwater heating.
- The moisture separator reheaters are an integral component of the turbine system which extracts moisture from the steam and reheats the steam to improve the turbine system performance. There are two moisture separator reheaters located between the high pressure turbine exhaust and the low pressure turbine inlet. The reheater has two stages of reheat.
- The turbine orientation minimizes potential interaction between turbine missiles and safety-related structures and components.

1.2.1.3.2 Main Steam System Design

- The main steam system is designed to supply steam from the steam generators to the high pressure turbine over a range of flows and pressures for the entire plant operating range, i.e., from system warmup to valves-wide-open turbine conditions.
- The main steam system is also designed to dissipate heat generated by the nuclear steam supply system to the condenser through steam dump valves or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves when either the turbine-generator or the condenser is not available.
- Six steam generator safety valves are utilized per steam header. There are two steam headers.

1.2.1.3.3 Main Feedwater and Condensate System Design

- The main feedwater system is designed to supply the steam generators with adequate feedwater during all modes of plant operation including transient conditions. The condensate system is designed to condense and collect steam from the low-pressure turbines and turbine steam bypass systems and then, transfer this condensate from the main condenser to the deaerator.
- The main feedwater and condensate systems are designed for increased availability and improved dissolved oxygen control.
- A deaerating heater is employed.

1.2.1.4 Auxiliary Fluid Systems Design

1.2.1.4.1 Engineered Safeguards Systems Design

- The safety systems are designed to mitigate design basis accidents with a single failure, as defined in Chapter 15.

- The safety systems are designed to maximize the use of natural driving forces such as pressurized nitrogen, gravity flow and natural circulation flow. They do not use active components such as pumps, fans or diesel generators. A minimum number of valves are used for the purpose of initially aligning the safety systems.

- The safety systems are designed to function without safety-related support systems such as alternating current, component cooling water, service water, heating, ventilation and air conditioning.

- The number and complexity of operator actions required to control the safety systems are minimized. In meeting this objective, the approach was to eliminate the required action and not to automate them.

- An automatic reactor coolant system depressurization feature is included in the design and meets the following criteria:

  - The reliability (redundancy and diversity) of the automatic depressurization system valves and controls satisfies single failure criteria as well as the failure tolerance required by the low core melt frequency goals.

  - The design provides for both real demands (such as reactor coolant system leaks and failure of the chemical and volume control system makeup pumps) and spurious instrumentation signals. The probability of significant flooding of the containment due to the use of the automatic depressurization system is less than once in 600 years.

- The design is such that, for small break loss-of-coolant accidents up to 8 inches in diameter, the core remains covered.

- The passive safety-related systems can operate for at least 1 hour following anticipated transients without release of contaminants that require significant plant cleanup. The automatic depressurization system is designed not to activate for anticipated transients.

- The passive safety-related systems are designed to cool the reactor coolant system from normal operating temperatures to safe shutdown conditions.

- The passive containment cooling system maintains the containment pressure and temperature within the appropriate design limits for both design basis and severe accident scenarios.

1.2.1.4.2 Nonsafety-Related Systems Designs

- The nonsafety-related systems designs are simplified; the number of systems and components and the complexity of operation and maintenance are reduced from current operating plants.

- The nonsafety-related systems are not relied upon to provide safety functions required to mitigate design basis accidents.

- Nonsafety-related systems that are required for normal plant operation provide high plant availability. These systems have appropriate redundancy, are powered by onsite standby power supplies and have sufficient capacity to prevent automatic passive safety system actuation following anticipated Condition II events.

- The reactor coolant system makeup capability design is sufficient for reactor coolant leaks up to 3/8 inch.
- Steam generator feedwater capability from the startup feedwater system is designed to provide sufficient flow for a loss of main feedwater event.
- The normal containment sump pumps (part of the liquid radwaste system) are designed to assist in recovery from leakage to the containment sump. RN-13-023

- Boric acid solutions are designed to be stored at concentrations that do not require heat tracing or room temperatures above normal values.

1.2.1.5 Electrical and Control Systems Designs

1.2.1.5.1 Control and Protection Systems Designs

- The design provides that during normal operation, a single failure in the protection and safety monitoring system does not result in a reactor trip. This is true even with a channel under maintenance or test.
- The potential for reactor trip and for safeguards actuation due to failures in the plant control system is reduced relative to current operating plants.
- The number of measured plant variables used for reactor trip and for safeguards actuation is minimized relative to current operating plants.
- The margin between the normal operating conditions and the protection system setpoints is increased relative to current operating plants.
- The potential for interaction between the protection and safety monitoring system and the plant control system is reduced relative to current operating plants by incorporating a signal selector function that selects signals for control and for protection.
- A distributed logic system utilizing multiplexing techniques is used to significantly reduce the amount of wiring required in the plant.

1.2.1.5.2 Alternating Current and Direct Current Power Design

- Safety-related direct current (dc) power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary dc power and uninterruptible ac power for items such as the protection and safety monitoring system actuation, the control room functions including habitability, dc-powered valves in the passive safety-related systems and containment isolation.
- All safety-related electrical power is provided from the Class 1E dc power system. No separate safety-related ac power system is required.

1.2.1.5.3 Control Room Design

- A main control room is provided that is able to control the plant during normal and anticipated transients and design basis accidents. The main control room includes indications and controls capable of monitoring and controlling the plant safety systems as well as the nonsafety-related control systems.

- A remote shutdown capability is provided. The remote shutdown workstation contains the indications and controls that allow an operator to achieve and maintain safe shutdown of the plant following an event when the main control room is unavailable. Additional nonsafety-related indications and controls are provided as described in Chapter 7.

- The remote shutdown workstation contains indications and controls consistent with its intended use; i.e., the remote shutdown workstation is to be used in the unlikely event that the main control room is not available.

- Access to the remote shutdown workstation transfer mechanism is under strict administrative control.

- The main control room is serviced by reliable and redundant nonsafety-related power sources and heating, ventilation and air conditioning systems during normal operation.

- In the unlikely event that the normal power source or the heating, ventilating and air conditioning system becomes unavailable, there are passive systems (batteries, compressed air) to support the main control room for up to 3 days.

- The main control room contains the safety-related instrumentation and controls to allow the operator to achieve and maintain safe shutdown following any design basis accident.

- The safety-related power sources and passive cooling system are designed to provide a habitable environment for the operating staff assuming that no ac power is available. Installed equipment provides for at least 3 days of operation, as stated above. After 3 days, it is possible to continue operation with the control room cooled and ventilated with circulation of outside air.

- A mechanism is provided to allow the operating staff to transfer control from the main control room to the remote shutdown workstation.

- The system prevents spurious signals caused by fire damage from being issued to components once transfer to the remote shutdown workstation has been effected.

- The transfer of the control of components to the remote shutdown workstation is alarmed in the main control room.

- Both the main control room and the remote shutdown workstation are designed in accordance with human factors engineering principles and practices.

- Human factors considerations are utilized so that the indications and controls for the remote shutdown workstation are similar to those provided in the main control room.

- The safety-related instrumentation (equipment racks) is maintained at acceptable ambient conditions for 3 days following a loss of all ac power by using a passive cooling system. After 3 days, it is possible to continue operation with the instrumentation and control rooms cooled by circulation of outside air.

- A technical support center is provided.

1.2.1.6 Plant Arrangement and Construction

1.2.1.6.1 Plant Arrangement

- The plant arrangement is comprised of five principal building structures; the nuclear island, the turbine building, the annex building, the diesel generator building, and the radwaste building (see Figure 1.2-3).

- The nuclear island is structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29. The nuclear island consists of a free-standing steel containment building, a concrete shield building, and an auxiliary building. The foundation for the nuclear island is an integral basemat which supports these buildings.

- The nuclear island structures are designed to withstand the effects of natural phenomena such as hurricanes, floods, tornados, tsunamis, and earthquakes without loss of capability to perform safety functions. Design for natural phenomena is based on the industry standards as described in Chapters 2 and 3.

- The nuclear island is designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions.

- The main area of the turbine building structure is designed to International Building Code requirements. The first bay of the turbine building is designed to seismic Category II requirements. The turbine building is supported on a single basemat foundation. RN-13-003

- The annex building area outlined by Column lines E – I.1 and 2 – 13 is designed to seismic Category II requirements. The rest of the annex building area is designed to International Building Code requirements. The annex building includes functions such as the health physics area, the control support area, access control, and personnel facilities (shower and locker rooms). RN-13-003

- The diesel generator building houses two diesel generators and their associated heating, ventilation and air conditioning equipment. The building is a nonseismic structure designed for wind and seismic loads in accordance with the Uniform Building Code.

- The radwaste building contains facilities for the handling and storage of plant wastes. It is a nonseismic structure designed for wind and seismic loads in accordance with the Uniform Building Code. The foundation for the building is a reinforced concrete mat on grade.

- Radioactive equipment and piping in all buildings are arranged and shielded to minimize radiation exposure.

- The overall plant arrangement utilizes building configurations and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, rebar) consistent with safety, operational, maintenance, and structural needs.

- The plant arrangement provides separation between safety-related and nonsafety-related systems to preclude adverse interaction between safety-related and nonsafety-related equipment. Separation between redundant safety-related equipment and systems provides confidence that the safety design functions can be performed. In general this separation is provided by partitioning an area with concrete walls.

- The plant arrangement provides separation for radioactive and non-radioactive equipment and provides separate pathways to these areas for personnel access.

- Pathways through the plant are designed to accommodate equipment maintenance and equipment removal from within the plant. The size of the pathways is dictated by the largest appropriate piece of equipment that may have to be removed or installed after initial installation. Where required, laydown space is provided for disassembling large pieces of equipment to accommodate the removal or installation process.

- Adequate space is provided for equipment maintenance, laydown, removal and inspection. Hatches, monorails, hoists, and removable shield walls are provided to facilitate maintenance.

1.2.2 Site Description

Site Characteristics

The AP1000 is a standard plant that is to be placed on a site with parameters bounded by those used as a basis for design certification as described in Chapter 2, Site Characteristics. The site parameters relate to the seismology, hydrology, meteorology, geology, heat sink and other site-related aspects. The allowable site interface parameters bound a large percentage of potential sites.

The AP1000 is designed on the basis that the equipment, modules, structures, and bulk material can be shipped to the site by commercial rail or truck. This does not preclude the shipment of large equipment or structures by barges should a specific site be accessible by water.

Site Plan

A typical site plan for a single-unit AP1000 reference unit is shown in Figure 1.2-2. The directions north, south, east, and west used in this description are the conventions used in the DCD for the orientation of AP1000 structures and equipment and differ from geographic north, south, east, and west. For Units 2 and 3, the plant orientation is rotated with respect to true north. Units 2 and 3 plant north is 68 degrees counter clockwise from true north. To differentiate between the site-specific and DCD orientation systems, orientations in the site-specific portions of the FSAR will be preceded with the word "plant" (as in "plant north") or, in some subsections, true north will be used. When true north is used, it will be specified in the subsection.

The AP1000 convention throughout the DCD is that design plant grade is given a reference elevation of 100 feet; see, for example, Figure 1.2-13. For Units 2 and 3, this plant elevation of 100 feet correlates to the North American Vertical Datum 1988 (NAVD88) elevation of 400 feet. Site-specific elevations are given in NAVD88. To differentiate between the DCD and site-specific elevations, elevations provided in the site-specific portions of the FSAR include the acronym SEL preceding the elevation unless it is otherwise specified in the FSAR Section or Subsection.

Figure 1.1-202 provides the general arrangement of major plant buildings and structures including the circulating water pump structure, cooling towers, water intake, plant discharge point, settling basins, switchyard, and access roads.

As stated in Subsection 1.2.1.6.1, the power block complex consists of five principal building structures; the nuclear island, the turbine building, the annex building, the diesel generator building and the radwaste building. Each of these building structures is constructed on an individual basemat. The nuclear island consists of the containment, the shield building, and the auxiliary building, all of which are constructed on a common basemat. The power block complex for both units is within a common perimeter.

Figure 1.2-3 provides a functional representation of the principal systems and components that are located in each of the key AP1000 buildings. This figure identifies major systems and components that are contained in these structures.

Units 2 and 3 each use two mechanical draft cooling towers rather than natural draft cooling towers provided in the DCD conceptual design and shown on Figure 1.2-2. The cooling towers are shown on Figure 1.1-202 and the details of the cooling towers are discussed in Section 10.4.

Road access to the site is from SC 213. Rail access is also available from the Norfolk Southern Railroad as shown on Figure 1.1-202. The Norfolk Southern Railroad line parallels the Broad River west of the site, along the east bank of the Broad River. This line provides rail access to the site by having a spur track owned by SCE&G leading off the main line from a switch southwest of the site.

1.2.3 Plant Arrangement Description

Building Definition

A set of the general arrangement drawings for the AP1000 is provided in Figures 1.2-4 through 1.2-30.

Figure 1.2-201 replaces Figure 1.2-18 to reflect the relocation of the Operations Support Center.

The AP1000 consists of the following five principal structures. Each of these buildings is constructed on an individual basemat:

- Nuclear island
- Turbine building
- Annex building
- Diesel generator building
- Radwaste building

The structures that make up the nuclear island are:

- Containment building
- Shield building
- Auxiliary building

These nuclear island buildings are depicted on the site plan. The safety-related equipment designed to perform accident mitigation functions is located in the nuclear island.

1.2.4 Nuclear Island

1.2.4.1 Containment Building

Building Function

The containment building is the containment vessel and the structures contained within the containment vessel. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations.

The containment vessel is an integral part of the passive containment cooling system. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

The containment building is designed to house the reactor coolant system and other related systems and provides a high degree of leak tightness.

Civil/Structural Features

The containment building, a seismic Category I structure, is a freestanding cylindrical steel containment vessel with elliptical upper and lower heads. It is surrounded by a seismic Category I shield building.

Figures 1.2-13 through 1.2-16 provide sectional views through the containment that show the configuration of the containment vessel and the internal structures of the containment.

There are two floor elevations (grade access maintenance floor and operating deck) and four lower equipment compartments within the containment building. Removable hatches are provided for access to equipment at other elevations.

Figures 1.2-7, 1.2-8, 1.2-9, 1.2-15, and 1.2-16 depict the configuration of the refueling water storage tank. This tank is located below the operating deck. The capacity of the refueling water storage tank exceeds the quantity of water required to accomplish safety functions or to fill the refueling cavity during refueling operations. The refueling cavity has two floor elevations. The upper and lower reactor internals storage area is at the lower elevation as is the fuel transfer tube.

Equipment Arrangement

The principal system located within the containment building is the reactor coolant system that consists of two main coolant loops, a reactor vessel, two steam generators, four sealless reactor coolant pumps, and a pressurizer. Figures 1.2-9, 1.2-14 and 1.2-16 depict the reactor coolant system component locations in the containment.

The main steam and feedwater lines are routed from the steam generators to a horizontal run below the operating deck. The steam and feedwater lines penetrate the north side of the containment vessel and are routed through the main steam isolation valve area in the auxiliary building to the turbine island.

The passive core cooling system is also located in the containment building. The primary components of the passive core cooling system are two core makeup tanks, two accumulators, the refueling water storage tank, the passive residual heat removal heat exchanger, and two spargers. The first three stages of the automatic depressurization valves are located above the pressurizer and consist of a two-tier valve module.

The passive residual heat removal heat exchanger and the spargers are located within the refueling water storage tank (Figures 1.2-7 and 1.2-9). The core makeup tanks are located on floor elevation 107′2″ level (Figures 1.2-7 and 1.2-9).

The chemical and volume control system equipment module is located in the containment below the maintenance floor level. This module represents the high pressure purification loop of the chemical and volume control system (Figure 1.2-14).

The reactor coolant drain tank, the reactor coolant drain tank heat exchanger and the containment sump pumps are located in the compartment adjacent to the reactor vessel cavity. Access to the reactor vessel cavity is via a stairwell that descends from the maintenance floor (Figure 1.2-14).

Two containment recirculation cooling units are located adjacent to the steam generator compartments. Each unit consists of two vane axial fans, cooling coils and the associated exit ducts and inlet plenum. The four recirculation fans are connected to the common exit plenum (ring header). Several vertical ducts branch off from the ring header to provide cooling flow to the lower compartments in the containment while other vertical ducts are directed up to provide cooling flow to the upper regions of the containment vessel.

Equipment and Material Handling

A seismic Category I polar crane is provided in the containment and its bridge is sized for lifting the steam generator during a steam generator removal operation. A temporary construction trolley is required for this operation. The polar crane support is attached to the steel cylindrical shell of the containment as shown in Figures 1.2-14 and 1.2-16.

The layout of the containment is designed to permit the removal of either steam generator through a temporary opening cut through the top of containment, then through the center of the passive containment cooling air diffuser. During a steam generator removal operation, the steam generator is lifted from the steam generator compartment by a temporary construction trolley and then through containment by a large mobile crane.

The polar crane trolley is designed for normal refueling operations such as lifting the integrated head package, the lower internals package and the upper internals package.

An auxiliary hook is provided with the polar crane for easier movement of smaller equipment. The polar crane is used for lifting reactor coolant pump motor/impeller assemblies from the steam generator/loop compartments to the operating deck in the event that the reactor coolant pump motor/impeller assemblies have to be removed from the containment for major maintenance.

A reactor coolant pump maintenance cart is provided for use in either of the two steam generator/loop compartments for removing the reactor coolant pump motor/impeller assemblies from the bottom head of the steam generators. This maintenance cart transports the reactor coolant pump motor/impeller assemblies to a designated area in each of the steam generator/loop compartments where the assemblies are lifted from the compartment to the operating deck by the polar crane. Removable sections of grating at all platform levels in the steam generator/loop compartments permit direct access to the pumps. From the operating deck level, the reactor coolant pump motor/impeller assemblies are removed from the containment via the main equipment hatch into the annex building maintenance area.

A refueling machine is provided to move fuel between the fuel transfer system and the reactor core (Figure 1.2-14). The refueling machine consists of a rectilinear bridge and a trolley crane with a vertical mast extending down into the refueling cavity. The bridge spans the refueling cavity and runs on rails set into the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly. In addition, the refueling machine is equipped with an auxiliary hoist which provides additional capability for other refueling operations.

A fuel transfer system is provided to transfer nuclear fuel assemblies between the refueling cavity in the containment building and the fuel transfer canal/spent fuel pit located in the fuel handling area of the auxiliary building. The fuel transfer system also has the capability to transfer control rod clusters.

Building Access and Exit

Access to the containment is provided through a personnel airlock and the main equipment hatch located at the operating deck level and a personnel airlock and a maintenance hatch at the maintenance floor level. Access to the containment can be controlled by the health physics office in the annex building.

In the event that large numbers of temporary personnel require access to the containment during a major outage, temporary personnel facilities can be provided immediately adjacent to the health physics area in the annex building.

1.2.4.2 Shield Building

Building Function

The shield building is the structure that surrounds the containment vessel. During normal operations, a primary function of the shield building is to provide shielding for the containment vessel and the radioactive systems and components located in the containment building. The shield building, in conjunction with the internal structures of the containment building, provides the required shielding for the reactor coolant system and the other radioactive systems and components housed in the containment.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

During accident conditions, the shield building provides the required shielding for radioactive airborne materials that may be dispersed in the containment as well as radioactive particles in the water distributed throughout the containment.

The shield building is an integral part of the passive containment cooling system.

Civil/Structural Features

The shield building is a seismic Category I structure. It shares a common basemat with the containment building and the auxiliary building.

Figures 1.2-13 through 1.2-16 provide sectional views of the shield building which show the basic configuration of the shield building and the annulus area between the containment vessel and the shield building.

The following items represent the significant features of the shield building and the annulus area:

- Shield building cylindrical structure
- Shield building roof structure
- Connections between reinforced concrete and steel concrete composite modules
- Tension ring at the interface between the roof and shield building cylinder
- Knuckle Region at interface of roof and outer wall of passive containment cooling system water storage tank
- Compression ring below the inner wall of the passive containment cooling system water storage tank
- Lower annulus area
- Middle annulus area
- Upper annulus area
- Passive containment cooling system air inlets at the top of the shield building cylinder
- Passive containment cooling system air inlet plenum
- Passive containment cooling system water storage tank
- Passive containment cooling system air diffuser
- Passive containment cooling system air baffle

The cylindrical section of the shield building serves as shielding and a missile barrier and is a key component of the passive containment cooling system. It structurally supports the roof and is a major structural member for the entire nuclear island. Floor slabs and structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

A watertight seal is provided between the upper and middle annulus areas to provide an environmental barrier. The middle annulus area contains the majority of containment penetrations and radioactive piping. This environmental barrier is provided to protect against the following:

- In the event of an accident or spurious actuation, the passive containment cooling system drains the system water storage tank. The water, which runs down the outside of the containment vessel, is prevented from draining into the middle annulus area by the watertight seal. Drains are provided to direct the passive containment cooling system runoff water out of the shield building.

- The passive containment cooling system is designed to perform with the upper annulus permanently open to the environment to permit sufficient air flow through the shield building in the event of an accident. The watertight seal protects the middle annulus area from ambient environmental conditions.

The shield building roof is a reinforced concrete conical shell supporting the passive containment cooling system water storage tank and air diffuser. Air intakes are located at the top of the cylindrical portion of the shield building. The conical roof supports the passive containment cooling system water storage tank which is constructed with a stainless steel liner attached to reinforced concrete walls. The air diffuser in the center of the roof discharges containment cooling air directly upwards.

The passive containment cooling system air baffle is located in the upper annulus area. It is attached to the cylindrical section of the containment vessel. The function of the passive containment cooling system air baffle is to provide a pathway for natural circulation of cooling air in the event that a design basis accident results in a large release of energy into the containment. In this event the outer surface of the containment vessel transfers heat to the air between the baffle and the containment shell. This heated and thus, lower density air flows up through the air baffle to the air diffuser and cooler and higher density air is drawn into the shield building through the air inlets at the top cylindrical portion of the shield building.

Equipment and Material Handling

A monorail is provided in the upper annulus area of the shield building to facilitate the initial installation of the passive containment cooling system air baffle panels and to permit the removal of these air baffle panels when an inspection or repainting of the containment vessel is required.

Two personnel workstation platforms are provided for transporting staff and equipment from the operating deck floor level of the upper annulus area to the top of the shield building. The work station platforms are powered from their respective monorail sections and are able to be positioned at any circumferential position or height beneath the monorail sections. Figures 1.2-14 and 1.2-16 depict the monorail system and the personnel work station platforms.

1.2.4.3 Auxiliary Building

Building Function

The primary function of the auxiliary building is to provide protection and separation for the seismic Category I mechanical and electrical equipment located outside the containment building.

The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event. The auxiliary building also provides shielding for the radioactive equipment and piping that is housed within the building.

The most significant equipment, systems, and functions contained within the auxiliary building are the following:

- Main control room
- Class 1E instrumentation and control systems
- Class 1E electrical system
- Fuel handling area
- Mechanical equipment areas
- Containment penetration areas
- Main steam and feedwater isolation valve compartment

Main control room: The main control room provides the human system interfaces required to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions. The main control room includes the main control area, the operations work area, the operations break area, and an office for the shift manager.

Instrumentation and control systems: The protection and safety monitoring system and the plant control system provide monitoring and control of the plant during startup, ascent to power, powered operation, and shutdown. The instrumentation and control systems include the protection and safety monitoring system, the plant control system, and the data display and processing system.

Class 1E electrical system: The Class 1E system provides 250 volts dc power for safety-related and vital control instrumentation loads including monitoring and control room emergency lighting. It is required for safe shutdown of the plant during a loss of ac power and during a design basis accident with or without concurrent loss of offsite power.

Fuel handling area: The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel. The fuel handling area in conjunction with the annex building provides the means for receiving, inspecting and storing the new fuel assemblies. It also provides for safe storage of spent fuel as described in Section 9.1, Fuel Storage and Handling.

The fuel handling area provides for transferring new fuel assemblies from the auxiliary building rail car bay to and from the new fuel storage area to the containment building and for transferring spent fuel assemblies from the containment building to the spent fuel storage pit within the auxiliary building.

The fuel handling area provides the means for removing the spent fuel assemblies from the spent fuel storage pit and loading the assemblies into a shipping cask for transfer from the facility.

The fuel handling area is protected from external events such as tornadoes and tornado produced missiles. Protection is provided for the spent fuel assemblies, the new fuel assemblies and the associated radioactive systems from external events.

The fuel handling area is constructed so that the release of airborne radiation following any postulated design basis accident that could result in damage to the fuel assemblies or associated radioactive systems does not result in unacceptable site boundary radiation levels.

Mechanical equipment areas: The mechanical equipment located in radiological control areas of the auxiliary building are the normal residual heat removal pumps and heat exchangers, the spent fuel cooling system pumps and heat exchangers, the solid, liquid, and gaseous radwaste pumps, tanks, demineralizers and filters, the chemical and volume control pumps, and the heating, ventilating and air conditioning exhaust fans.

The mechanical equipment located in the clean areas of the auxiliary building are the heating, ventilating and air conditioning air handling units, associated equipment that service the main control room, instrumentation and control cabinet rooms, the battery rooms, the passive containment cooling system recirculation pumps and heating unit and the equipment associated with the air cooled chillers that are an integral part of the chilled water system.

Containment penetration areas: The auxiliary building contains all of the containment penetration areas for mechanical, electrical, and instrumentation and control penetrations. The auxiliary building provides separation of the radioactive piping penetration areas from the non-radioactive penetration areas and separation of the electrical and instrumentation and control penetration areas from the mechanical penetration areas. Also provided is separation of redundant divisions of instrumentation and control and electrical equipment.

Main steam and feedwater isolation valve compartment: The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steam line or feedwater line.

Civil/Structural Features

The auxiliary building is a seismic Category I reinforced concrete structure. It shares a common basemat with the containment building and the shield building.

The auxiliary building wraps around approximately 70 percent of the circumference of the shield building. Floor slabs and the structural walls of the auxiliary building are structurally connected to the cylindrical section of the shield building.

Equipment and Material Handling

A cask handling crane is located in the fuel handling area of the auxiliary building. The cask handling crane is designed to transport the spent fuel cask between the rail car bay, the cask loading pit, and the cask washdown pit. The crane rail length and rail stop limits the crane travel and thus precludes the movement of this crane in the near vicinity of the spent fuel pit. A fuel handling machine is provided to transfer new fuel from the rail car bay to the new fuel rack to the new fuel elevator. A bridge crane is provided in the rail car bay for handling the spent resin waste container fill station cover, the spent resin waste container, and the high activity filter transfer casks.

The major components of the fuel transfer system are located in the fuel transfer canal. The fuel transfer system is designed to transfer fuel assemblies between the fuel transfer canal located in the fuel handling area and the refueling cavity located in the containment building. The fuel transfer system consists of a transfer car/fuel container, a drive car, a traverse drive mechanism, an upending mechanism, the transfer tube, a quick opening hatch on the containment side of the transfer tube and a valve on the fuel handling area side of the transfer tube.

A fuel handling machine is provided to move the spent fuel assemblies between the fuel transfer canal, the spent fuel pool and the cask loading pit. The fuel handling machine is a gantry crane, which has an extension to the south wall to reach the new fuel racks.

The high bay area is designed for a trailer to enter the building through a slide-up door. When used to transport a spent fuel cask to the fuel handling area, the trailer is positioned in the high bay area and the cask lifting rig is attached to the cask handling crane. When the cask is in the vertical position, it is disconnected from the trunnion and lifted to the operating deck through the equipment hatch and placed in the cask loading pit.

1.2.5 Annex Building

Building Function

The annex building (Figures 1.2-17 through 1.2-20) provides the main personnel entrance to the power generation complex. It includes accessways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms. The building also contains the non-1E ac and dc electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the control support area, and various heating, ventilating and air conditioning systems. No safety-related equipment is located in the annex building.

The annex building includes the health physics facilities and provides personnel and equipment accessways to and from the containment building and the rest of the radiological control area via the auxiliary building. Provided are large, direct accessways to the upper and lower equipment hatches of the containment building for personnel access during outages and for large equipment entry and exit. The building includes a hot machine shop for servicing radiological control area equipment. The hot machine shop includes decontamination facilities including a portable decontamination system that may be used for decontamination operations throughout the nuclear island.

Civil/Structural Features

The seismic classification of the annex building can be found in Table 3.2-2. No protection against missile penetration is required. However, certain areas of the building, such as the hot machine shop and the control support area, are provided with shielding for protection against low level radiation from either internal sources or external sources under accident conditions. This is accomplished by either reinforced concrete walls or reinforced masonry walls. The control support area is designed so that it may be used as a technical support center (TSC) if desired.

The annex building is a combination of reinforced concrete structure and steel framed structure with insulated metal siding. Floor slabs are reinforced concrete supported by metal decking. Roof slabs are either reinforced concrete or rigid insulation supported on metal decking. Floors are designed to act as diaphragms to transmit horizontal loads to side wall bracing and to concrete shear walls. The building foundation is a reinforced concrete mat. RN-14-133

1.2.6 Diesel Generator Building

Building Function

The diesel generator building (Figure 1.2-21) houses two identical slide along diesel generators separated by a three hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No safety-related equipment is located in the diesel generator building.

Civil/Structural Features

The diesel generator building houses the two diesel generators and their associated heating, ventilating and air conditioning equipment, none of which are required for the safe shutdown of the plant. The seismic classification of the diesel generator building can be found in Table 3.2-2. The building is designed as a structure subject to wind loads in accordance with the Uniform Building Code.

The building is a single story steel framed structure with insulated metal siding. The roof is composed of a metal deck supporting a concrete slab and serves as a horizontal diaphragm to transmit lateral loads to sidewall bracing and thereby to the foundation.

The foundation consists of a reinforced concrete mat. The diesel generators are skid-mounted and rest on vibration isolators supported directly from the mat.

1.2.7 Radwaste Building

Building Function

The radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building. Dedicated floor areas and trailer parking space for mobile processing systems is provided for the following:

- Contaminated laundry shipping for offsite processing
- Dry waste processing and packaging
- Hazardous/mixed waste shipping for offsite processing
- Chemical waste treatment
- Empty waste container receiving and storage
- Storage and loading packaged wastes for shipment

The radwaste building also provides for temporary storage of other categories of plant wastes.

Three liquid waste monitor tanks are located within the radwaste building. These tanks contain processed effluents which are ready for release to the environment.

Civil/Structural Features

The radwaste building general arrangement is shown on Figure 1.2-22. The seismic design of the radwaste building can be found in Table 3.2-2. The liquid radwaste processing areas are designed to contain any liquid spills. These provisions include a raised perimeter and floor drains that lead to the liquid radwaste system waste holdup tanks. The foundation for the entire building is a reinforced concrete mat on grade.

1.2.8 Turbine Building

Building Function

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

Civil/Structure Features

The turbine building, shown in Figures 1.2-23 through 1.2-30, consists of two sections, the first bay and the main area which houses the turbine. The first bay is immediately adjacent to the auxiliary building, and it consists of reinforced concrete walls and steel framing with reinforced concrete and steel grated floors. The main area is a steel framed building with reinforced concrete and steel grated floors. The first bay and the main area are two independent structures. The turbine building ground floor (structural mat) is a reinforced concrete slab shared by the first bay and main area structure. The seismic design of the turbine building can be found in Table 3.2-2.

The turbine-generator is low-tuned by means of spring supports. The design consists of a reinforced concrete deck mounted on springs. The springs are supported on a structural steel framework that forms an integral part of the turbine building structural system. Lateral bracing serves to provide lateral support for the building as well as the turbine-generator support. The spring-supported concept isolates dynamically the turbine-generator deck from the remainder of the structure for operating frequencies, thus allowing for an integrated structure below the deck. This includes an integrated reinforced concrete foundation mat that supports both the turbine generator and the building. The condenser is attached rigidly to the low pressure turbine exhaust and is supported on springs. The foundation for the entire building is a reinforced concrete mat.

1.2.9 Combined License Information

This section contained no requirement for additional information.

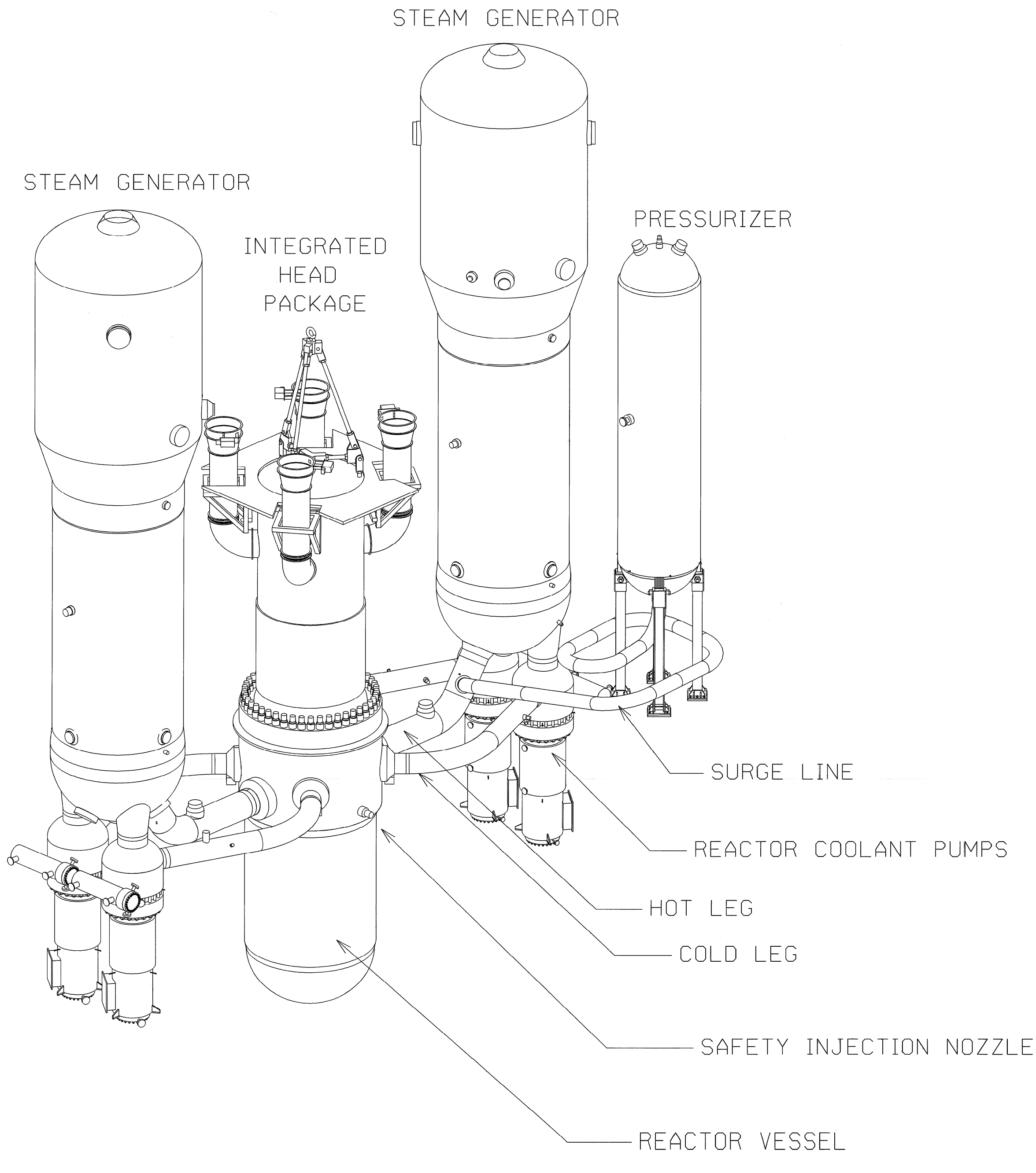

Figure 1.2-1   Reactor Coolant System

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-2 Site Plan RN-12-017

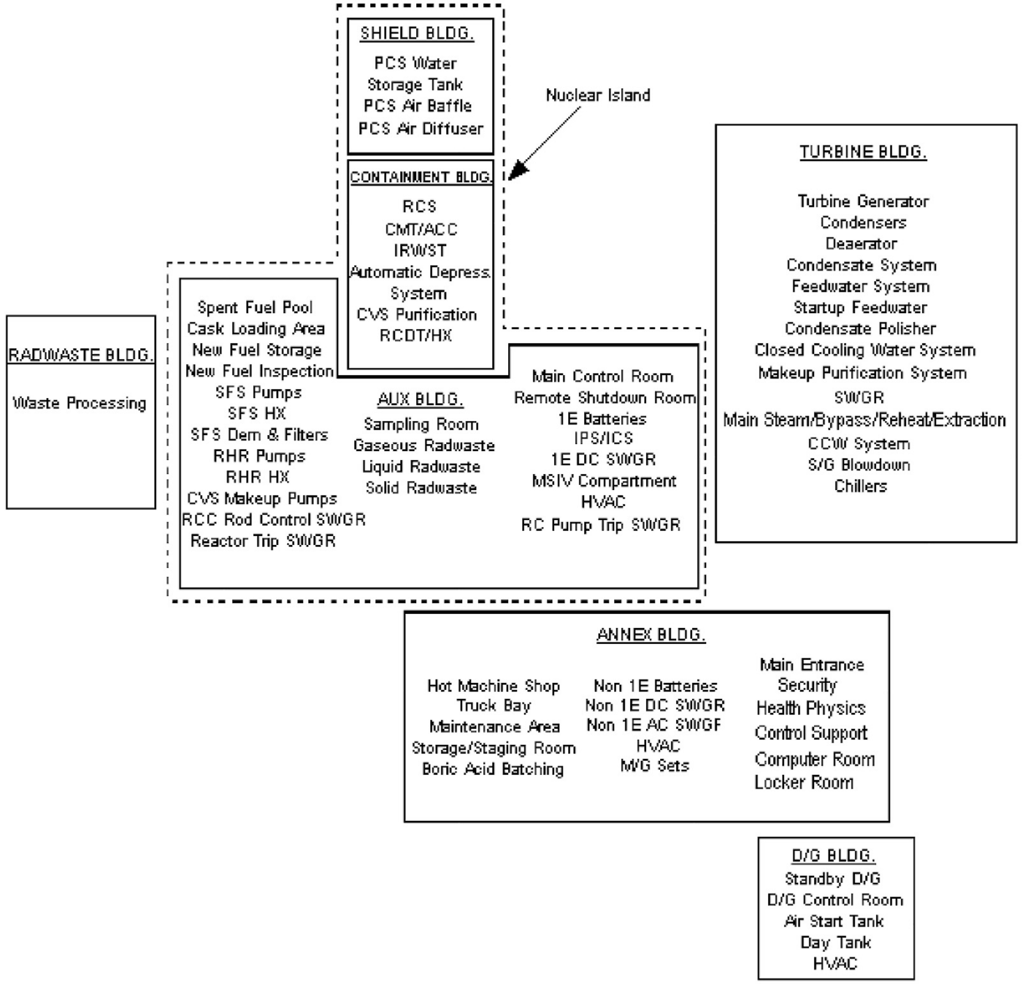

Figure 1.2-3 Functional Allocation of System Components of AP1000 Power Generation Complex

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-4 Nuclear Island General Arrangement Plan at Elevation 66′-6″

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-5 Nuclear Island General Arrangement Plan at Elevation 82′-6″

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-6 Nuclear Island General Arrangement Plan at Elevation 96′-6″ RN-13-088

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-7 Nuclear Island General Arrangement Plan at Elevation 107′-2″ & 111′-0″

RN-14-022
RN-14-131
RN-14-155

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-8 Nuclear Island General Arrangement Plan at Elevation 117′-6″ & 130′-0″ RN-13-046

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-9 Nuclear Island General Arrangement Plan at Elevation 117′-6″ with Equipment RN-14-131

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-10 Nuclear Island General Arrangement Plan at El. 135′-3″ RN-14-031 RN-13-099

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-11 Nuclear Island General Arrangement Plan at Elevation 153′-0″ & 160′-6″ RN-12-072

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-12 Nuclear Island General Arrangement Plan at Elevation 160′-6″ & 180′-0″ RN-12-072

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-13 Nuclear Island General Arrangement Section A-A RN-12-072

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-14 Nuclear Island General Arrangement Section A-A with Equipment RN-14-131 RN-12-072

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-15 Nuclear Island General Arrangement Section B-B

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-16   Nuclear Island General Arrangement Section B-B with Equipment

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-17 Annex Building General Arrangement Section A-A RN-14-133

(Note: This figure replaces Figure 1.2-18.)

Figure 1.2-201 Annex Building General Arrangement Plan at Elevation 100'-0" & 107'-2"

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-19 Annex Building General Arrangement Plan at Elevation 117′-6″ & 126′-3″

RN-12-017
RN-13-067
RN-14-133

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-20 Annex Building General Arrangement Plan at Elevation 135′-3″, 156′-0″ & 158′-0″ RN-13-067 RN-14-133

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-21 Diesel Generator Building General Arrangement Plan at Elevation 100′-0″ & Section A-A

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-22 Radwaste Building General Arrangement Plan at Elevation 100′-0″

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-23 Turbine Building General Arrangement Plan at Elevation 100′-0″

RN-12-017
RN-12-065
RN-14-052
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-24 Turbine Building General Arrangement Plan at Elevation 120′-6″

RN-12-017
RN-12-077
RN-13-057
RN-14-052
RN-14-033
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-25 Turbine Building General Arrangement Plan at Elevation 141′-3″

RN-12-017
RN-12-077
RN-14-033
RN-14-052
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-26 Turbine Building General Arrangement Plan at Elevation 170′-0″

RN-12-107
RN-13-057
RN-14-033
RN-14-052
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-27 Turbine Building General Arrangement Plan at Elevation 170′-0″ with Equipment

RN-12-017
RN-13-057
RN-14-033
RN-14-052
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-28 Turbine Building General Arrangement Plan at Elevation 255′-3" RN-12-017 RN-14-052 RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-29 Turbine Building General Arrangement Section A-A

RN-12-017
RN-14-052
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

Figure 1.2-30 Turbine Building General Arrangement Section B-B

RN-12-077
RN-14-052
RN-14-138

Redacted Information, Withheld Under 10 CFR 2.390d

(Note: This figure replaces Figure 1.2-18.)

Figure 1.2-201 Annex Building General Arrangement Plan at Elevation 100’-0” & 107’-2”

RN-13-067
RN-14-083
RN-14-133

1.3 Comparisons with Similar Facility Designs

A comparison of the major AP1000 design features and nominal parameters with the certified AP600 and a typical two-loop Westinghouse plant is provided in Table 1.3-1. The values provided for AP1000 are nominal and provided for comparison. Design parameter values for design certification are delineated in the sections referenced. The values provided in Table 1.3-1 for the reference AP600 and two-loop plants are typical. The two-loop plant parameters are represented by Waterford Unit 3.

Table 1.3-1 (Sheet 1 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Plant design objective	1.2	60 yrs	60 yrs	40 yrs
NSSS power	4.0	3,415 MWt	1,940 MWt	3,410 MWt
Core power	4.0	3,400 MWt	1,933 MWt	3,390 MWt
Net electrical output	1.2	≥ 1,000 MWe	600 MWe	1,075 MWe
Reactor operating pressure	5.1	2,250 psia	2,250 psia	2,250 psia
Hot leg temp	5.1	610°F	600°F	611°F (Cycle 1) 603°F (current)
Steam generator design pressure	5.4	1200 psia	1200 psia	1100 psia
Main feedwater temp	10.3	440°F	435°F	445°F
Core	4.0			
Number fuel assem.		157	145	217
Active fuel length		168 in	144 in	150 in
Fuel assembly array		17 x 17	17 x 17	16 x 16
Fuel rod OD		0.374 in	0.374 in	0.382 in
Number control assem.		53	45	83
– Absorber material		Ag-In-Cd	Ag-In-Cd	B_4C/Ag-In-Cd
Number gray rod assem.		16	16	8 (part length)
– Absorber material		SS-304/Ag-In-Cd	SS-304/Ag-In-Cd	Inconel 625/ B_4C
Avg linear power		5.707 kW/ft	4.10 kW/ft	5.34 kW/ft
Heat flux hot channel factor, FQ		2.60	2.60	2.35

Table 1.3-1 (Sheet 2 of 6) AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Reactor Vessel	5.3			
Vessel ID		159 in	157 in	172 in
Construction		forged rings	forged rings	welded plate
Number hot leg nozzles		2	2	2
– ID		31.0 in	31.0 in	42 in
Number cold leg nozzles		4	4	4
– ID		22.0 in	22.0 in	30 in
Number safety injection nozzles		2	2	0
Steam Generators	5.4.2			
Type		Vertical U-tube Recirc. design	Vertical U-tube Recirc. design	Vertical U-tube Recirc. design
Model		Delta-125	Delta-75	–
Number		2	2	2
Heat transfer area/SG		123,538 ft^2	75,180 ft^2	103,574 ft^2
Number tubes/SG		10,025	6,307	9,300
Tube material		I 690 TT	I 690 TT	I 600 TT
Separate startup feedwater nozzle		Yes	Yes	No
Reactor Coolant Pumps	5.4.1			
Type		canned	canned	shaft seal
Number		4	4	4
Rated HP		7,300 hp/pump	≤3,500 hp/pump	9,700 hp/pump
Estimated flow/loop		150,000 gpm	102,000 gpm	198,000 gpm

Table 1.3-1 (Sheet 3 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Pressurizer	5.4.5			
Total volume		2,100 ft^3	1,600 ft^3	1,500 ft^3
Volume/MWt		0.618 ft^3/MWt	0.825 ft^3/MWt	0.440 ft^3/MWt
Safety valves #/size		2 – 6"x8"	2 – 6"x6"	3 – 6"
PORV #/size		no	no	no
PRT volume		no	no	2,400 ft^3
Auto depressurization		yes	yes	no
Turbine Island	10.2			
Turbine – # HP cylinder		1	1	1
# LP cylinders		3	2	3
Max blade length		52 in	47 in	40 in
Number reheat stages		2	1	1
Feedwater heating stages				
– # LP stages		4	4	5
– # HP stages		2	2	1
Deaerator		yes	yes	no
Main feedwater pumps		3 motor driven	2 motor driven	2 turbine driven
Condensate pumps		3	3	3
Condenser tube material		Ti	Ti	SS
Condensate polishing		0–33%	33%	0–100%

**Table 1.3-1 (Sheet 4 of 6)
AP1000 Plant Comparison With Similar Facilities**

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Containment	6.2			
Type		Steel	Steel	Steel
Inside dia.		130 ft	130 ft	140 ft
Volume		2.06 E+06 ft^3	1.76 E+06 ft^3	2.677 E+06 ft^3
Volume/MWt		606 ft^3/MWt	910 ft^3/MWt	785 ft^3/MWt
Post accident cooling		Air and water on outside of steel containment vessel	Air and water on outside of steel containment vessel	Component cooling water cooled fan coolers
Safety Injection	6.3			
Accumulator – #/volume		2/2,000 ft^3	2/2,000 ft^3	4/2,250 ft^3
Core makeup tank – #/volume		2/2,500 ft^3	2/2,000 ft^3	no
High head pumps – #		none	none	3
– runout flow		–	–	380 gpm
– shutoff head		–	–	1,365 psi
Low head pumps – #		none	none	See RHR pumps
Refuel water storage tank – #		1	1	1
– location		in containment	in containment	ex-containment
– volume		590,000 gal	530,000 gal	475,000 gal
Boron inject tank #/vol		no	no	1/630 gal (batching) 2/11,800 gal (makeup)
Normal Residual Heat Removal (NRHR)	5.4.7			
Design pressure		900 psig	900 psig	650 psig
Normal RHR pumps – #/design flow		2/1,000 gpm per pump	2/1,000 gpm per pump	2/4,050 gpm per pump

Table 1.3-1 (Sheet 5 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Cooling Water Systems	9.2			
Safety-related		no	no	yes
Component cooling water pumps		2	2	3
Service water pumps		2	2	none
Heat sink		Separate mechanical draft cooling towers	Separate mechanical draft cooling towers	Separate mechanical draft cooling towers
Startup/Auxiliary Feedwater	10.4			
Motor pumps – #/flow per pump/ safety-related		2/520 gpm/no	2/380 gpm/no	2/350 gpm/yes 1/900 gpm/no
Turbine pumps – #/flow		none/–	none/–	1/700 gpm
Passive RHR HX – #/heat removal/ safety-related		1/60 MW/Yes	1/42 MW/Yes	None/–/–
Chemical and Volume Control	9.3.6			
Purification/Letdown flow				
– normal		100 gpm	100 gpm	38 gpm
– max		100 gpm	100 gpm	126 gpm
Purification location		IRC	IRC	ORC
RCP seal injection/pump		none	none	5 – 8 gpm
Charging pumps		2 @ 100 gpm	2 @ 100 gpm	3 @ 44 gpm
– SI use		no	no	no
– safe shutdown use		no	no	yes
– continuous oper.		no	no	yes
Boron thermal regeneration		no	no	no
Boron recycle evaporator		no	no	no

Table 1.3-1 (Sheet 6 of 6)
AP1000 Plant Comparison With Similar Facilities

Systems – Components	DCD	AP1000	AP600	Reference 2 Loop
Instrumentation and Control	7.7			
Type I&C system		digital	digital	analog
Type control room		work station	work station	control boards
Electrical				
Diesels – #	8.31	2	2	2
– safety-related		no	no	yes
– capacity		4,000 kW	4,000 kW	4,400 kW
1E batteries – total capacity	8.32	14,400 amp-hr	28,800 amp-hr	3 x 2,320 amp-hr (@ 8 hour rate)

1.4 Identification of Agents and Contractors

1.4.1 Applicant – Program Manager

SCE&G has been authorized by the South Carolina Public Service Authority (Santee Cooper) to act as their agent in applying for a Combined License (COL) for Units 2 and 3. SCE&G and Santee Cooper will jointly own the facility and share in the costs (including the cost of decommissioning) and output of the facility as follows: SCE&G, 55%; Santee Cooper, 45%. SCE&G will retain sole responsibility for operation of VCSNS Units 2 and 3 after the requirements of 10 CFR 52.103(g) are satisfied.

SCE&G is the principal subsidiary of SCANA Corporation, an energy-based holding company with headquarters in Cayce, South Carolina. SCE&G is a regulated public utility engaged in the generation, transmission, distribution, and sale of electricity in South Carolina. SCE&G is also engaged in the purchase and sale of natural gas in central and southern South Carolina. SCE&G has constructed and currently operates V.C. Summer Unit 1. Unit 1 began construction in April 1973 and has been commercially operated since January 1984.

Santee Cooper is South Carolina's state-owned electric and water utility with headquarters in Moncks Corner, South Carolina. Santee Cooper also generates the power distributed by the state's 20 electric cooperatives.

SCE&G has an Engineering, Procurement, and Construction (EPC) contract with a Consortium comprised of Westinghouse Electric Company, LLC and Stone and Webster, Inc. Stone and Webster, Inc. was acquired by the Shaw Group, Inc. (also referred to as Shaw) in 2000. The Consortium will act as the AP1000 provider, architect-engineer and constructor for VCSNS Units 2 and 3.

Westinghouse Electric Company, LLC (Westinghouse), is responsible for the overall design and design certification of the AP1000 nuclear power plant. A significant portion of the AP1000 design is the same as the design of AP600. Westinghouse Electric Company was also responsible for the overall design and design certification of AP600.

Westinghouse has designed, developed, and manufactured nuclear facilities since the 1950s, beginning with the world's first large central station nuclear plant (Shippingport), which produced power from 1957.

Westinghouse has designed and delivered more than 100 commercial nuclear power plants with a combined electrical generating capacity in excess of 90,000 MW. The company's manufacturing facilities include the commercial nuclear fuel fabrication facility at Columbia, South Carolina; and nuclear component manufacturing facilities at Blairsville, Pennsylvania; and Newington, New Hampshire.

Westinghouse has been involved with advanced light water reactor plant design efforts for over fifteen years. Included is the development of the advanced, passive pressurized water reactors known as the AP600 and AP1000.

Westinghouse has substantial, proven experience, knowledge, and capability to design, manufacture and furnish technical assistance for the installation, startup and service of nuclear power plants.

Shaw is a Fortune 500 company which has been an active participant in the nuclear industry for nearly 60 years (as Stone and Webster, Inc.), from providing engineering and design services for Shippingport, the nation's first commercial nuclear power plant, to the restart of the Tennessee Valley Authority's Browns Ferry Unit 1, which at the time was the largest nuclear construction project in the western hemisphere. Shaw continues to prove its leadership role in the nuclear industry by being part of the AP1000 Consortium.

Westinghouse Electric Company, LLC (Westinghouse) offers a wide range of nuclear plant products and services to utilities throughout the world, including fuel, service and maintenance, instrumentation and control, and advanced nuclear plant designs, including the AP1000 certified reactor design. Westinghouse has operations in twelve states and fourteen countries. After designing the world's first commercial pressurized water reactor nuclear power plant at Shippingport in 1957, Westinghouse and its licensees provided more than 40 percent of the world's 434 operating commercial nuclear plants. By the end of 2003, reactors based on Westinghouse technology had amassed over 2500 reactor-years of power generation.

Westinghouse is responsible for the overall plant design, AP1000 Design Certification revisions, procurement of primary NSSS equipment and power block major components including the Turbine Generator, and plant training simulator. Shaw is responsible for site development, construction, site specific design related work, secondary equipment procurement, module fabrication, and supply of bulk materials and commodities. Westinghouse and Shaw are jointly responsible for testing and startup.

1.4.2 Other Contractors and Participants

Under the direction of Westinghouse, a number of highly qualified organizations provide design and analysis in support of the AP600 and AP1000. Each has a specific responsibility to Westinghouse as defined by various contracts and agreements. Where design features are the same between AP600 and AP1000, the design and analysis performed for AP600 by organizations other than Westinghouse are applied directly to AP1000. The major contributors are identified in this section. They are included here if they have contributed to the base AP600 design or if they have contributed specifically to the AP1000 design.

Throughout the design process, lines of communication have been established among all participants. Design information is generated using common formats, electronic tools and software. Common requirement and compliance documentation has been established and followed. This has allowed design to progress in a complete and consistent manner with interfaces explicitly managed.

1.4.2.1 Bechtel North American Power Corporation

Bechtel North American Power Corporation (Bechtel) is one of the foremost architect-engineering firms in the United States, with the design and construction of 150 nuclear power projects in 25 countries to its credit. In addition to new construction, Bechtel has first-hand experience in operating plant retrofit design and construction, as well as maintenance and management.

1.4.2.2 Southern Electric International

Southern Electric International (SEI) is a wholly owned subsidiary of The Southern Company. The Southern Company is comprised of Southern Company Services, Inc. (an engineering technical services company), six operating utility companies, and Southern Electric International (a commercial engineering consulting services company).

Southern Electric International has benefited from over 99 years of engineering and consulting services experience with the Southern electric utility system. This expertise is derived from experience in designing, constructing, operating, maintaining, and modernizing the 251 generating units of the Southern electric system and those of Southern Electric International's clients. Southern Electric International provides a unique perspective and expertise of an operating electric utility.

1.4.2.3 Burns & Roe Company

Burns & Roe Company is an architect-engineering firm with considerable nuclear expertise. Burns & Roe has provided design, construction management and modernization services to a wide variety of domestic and foreign operating utilities. Burns & Roe contributed to the design and installation of a number of commercial nuclear power plants. Burns & Roe has also been involved with the development of advanced light water reactors since their inception.

1.4.2.4 Washington Group (MK-Ferguson Company)

MK-Ferguson Company is one of the larger construction firms in the world. Their planning and construction management work extends to commercial and industrial projects as well as power generation units. They are a DOE-approved subcontractor on the defense waste processing facility at Savannah River and have worked on such diverse nuclear plant challenges as the replacement of the steam generator at the D.C. Cook Nuclear Plant, decommissioning of the Shippingport Plant and nuclear reactor modifications at Bettis Atomic Power Laboratory.

1.4.2.5 Avondale Industries, Inc.

Avondale Industries, Inc. is the United States pioneer and leader in modular construction. Their modern shipyards prove ideally suited for the modular construction of industrial and commercial facilities. They have the sophisticated infrastructure in engineering, program management, materials and cost control needed to support large, complex projects.

1.4.2.6 Chicago Bridge & Iron Services, Inc.

Chicago Bridge & Iron Services (CBI) is the leading designer and maker of nuclear reactor containment vessels and liners. They have successfully erected 107 containment structures, 70 percent of all containments built in the United States. Chicago Bridge & Iron Services also specializes in operating plant modification and maintenance upgrades; their service expertise includes planning, development, scheduling and implementation of work procedures, ALARA, and decontamination.

1.4.2.7 Other Participants

Westinghouse has also received support from a variety of engineering and testing firms on a subcontract basis. The organizations providing important design or testing services include: SOPREN/ANSALDO of Italy, University of Western Ontario of Canada, ENEL of Italy, BATAN of Indonesia, ENEA of Italy, BPPT of Indonesia, FIAT of Italy, INITEC of Spain, UNESA of Spain, UTE of Spain, PLN/BPPT of Indonesia, Oregon State University, EdF of France, SNERDI of China, MHI of Japan, UAK of Switzerland, DTN of Spain, Fortum of Finland, IBF of Italy, Tioga of the United States, Pennsylvania State University, Ishikawajima-Harima Heavy Industries Co., Ltd. of Japan, SPX/Copes Vulcan of the United States, Doosan of the Republic of Korea, KOPEC of the Republic of Korea, KSB of Germany, EMD of the United States, Toshiba of Japan, Obayashi of Japan, and Shaw Stone & Webster of the United States.

Support was received in the development of the COL Application for Units 2 and 3 from a project team consisting of Bechtel Power Corporation; Westinghouse Electric Company LLC; NuStart Energy, Inc.; MACTEC Engineering & Consultants; William A. Lettis and Associates, Inc., and Tetra Tech NUS, Inc.

Bechtel Power Corporation

Bechtel Power Corporation prepared and published the COL Application and acted as the program manager for the COL Application.

MACTEC Engineering and Consulting, Inc.

MACTEC Engineering and Consulting, Inc. performed geotechnical field investigations and laboratory testing in support of the COL Application. This included performing standard penetration tests, obtaining core samples and rock cores, and installing ground water observation wells as a subcontractor to Bechtel Power Corporation.

NuStart Energy, Inc.

NuStart Energy, Inc. prepared the Reference COLA used as a template for preparation of the non-site-specific portions of the COL Application.

Risk Engineering, Inc.

Risk Engineering, Inc. performed the probabilistic seismic hazard analyses for development of the site-specific ground motion response spectra as a subcontractor to Bechtel Power Corporation.

Tetra Tech NUS, Inc.

Tetra Tech NUS, Inc. provided services for site investigations and preparation of the Environmental Report and portions of the FSAR as a subcontractor to Bechtel Power Corporation.

William A. Lettis and Associates, Inc.

William Lettis & Associates, Inc. performed the investigations and analyses required to prepare the geology, seismology, and geotechnical engineering section of the COL Application as a subcontractor to Bechtel Power Corporation. This included investigating the subsurface materials present at the site, performing a comprehensive geotechnical exploration, and performing geophysical surveys to assess the dynamic response of soil and rock.

1.4.3 Combined License Information

This section contained no requirement for additional information.

1.5 Requirements for Further Technical Information

Introduction

Tests were conducted during the AP600 Conceptual Design Program (1986 through 1989) to provide input for plant design and to demonstrate the feasibility of unique design features. Tests for the AP600 design certification and design program were devised to provide input for the final safety analyses, to verify the safety analysis models (computer codes), and to provide data for final design and verification of plant components. An AP1000 specific Phenomena Identification and Ranking Table (PIRT) and scaling analysis (Reference 25) and a review of safety analysis evaluations of AP1000 (Chapter 15) show that AP600 and AP1000 exhibit a similar range of conditions for the events analyzed. This provides justification that the database of test information generated during the AP600 Conceptual Design Program is sufficient to meet the requirements of 10 CFR Part 52 for AP1000. Table 1.5-1 is a list of the AP600 tests and AP1000 evaluations with references to test and evaluation documentation. Note that Reference 25 reviews each of the AP600 tests described and assesses their applicability to AP1000. The evaluations of Reference 25 show that the AP600 tests are sufficient to support AP1000 safety analysis.

The AP600 tests related to the plant safety functions were selected based on the plant features that are different from current PWRs and where directly applicable experimental data are not available. The tests simulate plant features as required to demonstrate the phenomena being examined. To validate the computer models, these experiments are modeled using the same computer codes used for plant analyses.

Testing of some plant component designs is required to verify their reliability and manufacturability. Other component tests provide data for design optimization. The completed component design tests are described below.

1.5.1 AP600 Safety-Related Tests

The AP600 safety-related experiments are designed to meet several goals:

- Provide input for safety analysis
- Provide data on the passive safeguards systems to validate the safety analysis methods and computer codes
- Assess the design margin in the passive safety system performance

To accomplish these goals, the AP600 test program utilizes the available data from the NRC and industry light water reactor safety research programs as well as specific tests which address the uniqueness of the AP600. The AP600 safety-related test program utilizes component experiments and integral tests to determine the transient behavior of the AP600 safety system components such that computer models can be developed and verified.

The range of plant conditions for design basis accidents and transients, and the new features of the AP600 design were evaluated against current Westinghouse designs and safety-related data available in the literature (NUREG-1230). The results of this assessment were used to determine the data needs, and to define the experiments to support the AP600 safety analysis. Based upon the experiments performed for AP600 and the AP1000 range of plant conditions for design basis accidents and transients, the tests were shown to be sufficient to support AP1000 safety analysis as well.

1.5.1.1 Large-Break LOCA

For large-break LOCA safety analysis, the relevant new features of the AP600 were the core makeup tanks (CMTs), which drain by gravity, and the use of hermetically sealed, high inertia centrifugal canned-motor reactor coolant pumps (RCPs). Two-phase pump flow data exists for Westinghouse designed pumps and others (NUREG-1230) that can be used to characterize the AP600 pumps. The core makeup tank is unique to the AP600 and AP1000 design. A specific AP600 test was conducted for this component. In addition, a test of passive safety injection system check valve flow vs. ΔP with low differential pressure has been completed. The evaluation of Reference 25 shows that these tests are applicable to AP1000.

Core Makeup Tank Performance Test

The purpose of this experiment is to verify the natural circulation and draining behavior of the core makeup tank over a full range of flowrates, pressures and temperatures, and to provide data to support the design and operation of the tank level indication which acts as a control for the automatic depressurization system (ADS). When actuated, the CMT adds water mass to the reactor coolant system (RCS) by natural circulation when the cold leg contains hot water. The water in the core makeup tank drains by gravity head into the RCS when steam is provided from the cold leg to the top of the core makeup tank. This steam replaces the water drained from the core makeup tank. Some of the steam condenses upon entry into the core makeup tank and can affect the tank draining performance. The objective of the test is to verify that the tank will drain as predicted.

A one-eighth diameter and one-half height scale core makeup tank was constructed and instrumented to obtain the condensation rates within the tank to verify the computer model. The core makeup tank water delivery was examined.

Passive Safety Injection System Check Valve Tests

The AP1000 uses check valves to isolate passive systems from the reactor coolant system. Tests have been performed in the AP600 test program on these check valves to demonstrate their operability.

Tests were conducted to measure check valve pressure drop from very low flow to full flow conditions. Detailed data on initial valve opening, valve disk behavior and flow versus differential pressure were obtained for individual check valves as well as for valves installed in series.

Initial check valve low differential opening tests have determined the characteristic valve flow under the expected gravity drain conditions. A review of existing utility information has been conducted to assess check valve performance under conditions similar to those which would be experienced by the gravity drain check valves.

1.5.1.2 Small-Break LOCA

For small-break LOCA safety analysis, the relevant new features of the AP600 and the AP1000 designs are the core makeup tank, and the automatic depressurization system which depressurizes the primary system to near containment pressure.

The core makeup tank provides injection flow to the reactor vessel at any reactor coolant system pressure. The core makeup tank tests described above duplicated small-break conditions as well as the large-break conditions. The automatic depressurization system provides controlled venting of the reactor coolant system to reduce pressure to allow transition to gravity driven injection from the IRWST. Full-scale tests were conducted in the AP600 test program to obtain data on the performance of the automatic depressurization system. As shown by the AP1000 evaluations, these tests also support AP1000 safety analysis.

Automatic Depressurization System Hydraulic Tests

The purpose of these tests is to simulate the automatic depressurization system, to confirm the capacity of the automatic depressurization system valves and spargers, and to determine the dynamic effects on the IRWST structure.

A pressurized, heated water/steam source was used to simulate the water/steam flow rate from the AP600 reactor coolant system during various stages of the automatic depressurization system blowdown. Two test phases were conducted. Phase A consisted of steam only blowdowns at bounding volumetric flowrates. The flow is piped to a full sized sparger submerged in a quench tank simulating the IRWST. The Phase B1 portion of the test included steam/water blowdowns at bounding mass flowrates through a simulation of one of the two ADS stage 1,2,3 flowpaths. Instrumentation to measure water and steam flow rate, and IRWST dynamic loads was installed. Sparger behavior was obtained from ambient to fully saturated IRWST water temperatures.

1.5.1.3 Containment Cooling

Tests to characterize the heat removal capabilities of the AP600 containment design were performed to provide the database for the containment cooling models. These include the following:

- Study of water film behavior and wetting of a steel plate simulating the containment exterior surface
- Heated plate tests to examine the evaporating heat transfer of water from the steel surface of the containment and heat transfer with only air cooling
- Containment external cooling air flow path pressure drop tests to characterize the hydraulic losses
- Steam condensation heat transfer experiments on a flat cool surface at different angles of inclination to simulate the condensation on the inside of the containment in the presence of noncondensible gases

In addition, tests were performed to examine the integrated behavior of the steam condensation on the inside, and the evaporative film cooling and air cooling on the outside of a pressure vessel. The cylindrical vessel used for this integral test was 3 feet wide and 24 feet high. These experiments included transient and steady-state tests which have been used as the basis for the containment analyses. The limits of coolability and the effect of cold weather conditions were also examined.

As shown by the AP1000 evaluations, these tests also support AP1000 safety analysis.

Integral Containment Cooling Tests

This test examines the combined effect of natural convection and condensation on the interior of the containment while the exterior is cooled by film evaporation and air flow. This test demonstrated the operation of the passive containment cooling system over a range of operating conditions, including operation at low environmental temperatures. This test, in conjunction with completed conceptual design phase testing and the large scale containment test described below, characterize the passive containment cooling system design and performance.

Passive Containment Cooling System Heat Transfer Test

A one-eighth scale steel containment structure with external water film and natural circulation air cooling and modeled containment internal compartments was constructed.

This test accurately models both the containment dome and side wall heat transfer areas. It complements the integral containment experiment which simulates the side wall condensation and evaporating film heat transfer. This test was used to verify the containment analysis analytical methods.

Instrumentation measured the condensation heat flux distribution, the resulting heat transfer coefficients, the air/steam mass ratios, and the resulting liquid film evaporation rates. Both the current integral containment cooling test and this larger scale containment test have been modeled to verify the Chapter 15 analysis computer code and to demonstrate the scalability of the results.

Passive Containment Cooling System Water Distribution Test

A passive containment cooling system water distribution experiment was performed to examine and finalize the AP600 containment water distribution. The results provide input into the containment safety analysis computer codes for water coverage of the containment shell.

The test was performed on a full-scale 1/8th sector of the containment dome. The AP600 water supply/distribution arrangement was modeled. Tests were conducted to demonstrate and measure the water spreading from the top center of the dome to the outer edges. Tests have been conducted to verify the performance of the water distribution system design. Tests were conducted with the surface coated with the prototypic AP600 containment coating. Measurements of water film velocities and film thickness variation as a function of flow rate and radial distance on the dome were obtained.

Passive Containment Cooling System Wind Tunnel Tests

Containment cooling relies on natural circulation of air to enhance evaporative cooling of the containment shell during a design basis event. Wind tunnel tests were performed to demonstrate that wind does not adversely affect natural circulation air cooling through the shield building and around the containment shell.

An approximately 1/100-scale model of the AP600 plant, including the adjacent buildings and cooling tower structure, was constructed and instrumented with pressure taps. The model was placed in a boundary layer wind tunnel and tested at different wind directions. The results were used to design the shield building air inlet and exhaust arrangement and to determine the loads on the air baffle. Variations in site layout and topography have been addressed using an approximately 1/800-scale model of the site buildings and local topography.

Tests were also conducted in a larger, higher speed wind tunnel on an approximately 1/30-scale model. These tests were conducted to confirm that the early test results conservatively represented those expected at full scale Reynolds numbers and to obtain better estimates of the baffle loads in the presence of a cooling tower.

1.5.1.4 Non-LOCA Transient Analysis

The non-LOCA accidents are evaluated using the same transient analysis methods used on existing Westinghouse PWR designs. Passive core cooling system computer models have been developed and added to the transient analysis codes. These models consist of a core makeup tank model and a passive residual heat removal (PRHR) heat exchanger model. As shown by the AP1000 evaluations, these tests also support AP1000 safety analysis.

Passive Residual Heat Removal Heat Exchanger Performance Test

The PRHR heat exchanger is located in the IRWST. This heat exchanger, which is connected directly to the reactor coolant system, transfers core decay heat and sensible heat energy to the IRWST water and depends only on natural circulation driving forces.

The passive residual heat removal heat exchanger test determined the heat transfer characteristics of the PRHR heat exchanger and the mixing characteristics in the IRWST. These results confirm the heat exchanger size and configuration.

The test facility consisted of three full-length heat exchanger tubes placed vertically in a cylindrical tank filled with water and baffled to simulate the AP600 IRWST. Water at prototypic natural circulation and forced flow rates was run through the heat exchanger tubes at prototypic system pressure and temperatures. Data was taken with IRWST water cold to saturation temperature to define the PRHR heat transfer correlation. Tests were also conducted using a baffle to simulate the effect of other rows of tubes have on heat exchanger thermal performance and tank mixing.

Departure from Nucleate Boiling Test

Due to the shorter coastdown of the AP600 canned motor reactor coolant pumps, the flow rates at the time of minimum DNBR are somewhat below previously correlated flow rates. DNB testing was performed to extend the DNB correlation to these lower flows.

These critical heat flux tests were conducted using a 5x5, full length heated rod bundle with non-uniform radial and axial heating distributions.

1.5.1.5 Integral Systems Testing

In the AP600, the water injected into the reactor coolant system comes from the CMTs, accumulators, and the IRWST. Two integral systems tests were conducted in the AP600 test program, a low-pressure scaled test and a full-height, full-pressure test. In addition, the NRC conducted tests in the low-pressure scaled test facility (Reference 27). As shown in the AP1000 evaluations (Reference 25), these three test programs are sufficient to support AP1000 safety analysis.

Low-Pressure Integral Systems Test

The primary purpose of this experiment was to examine the operation of the long-term makeup path from the in-containment refueling water storage tank. In addition, analysis of this experiment demonstrates water flow through the core to limit the long-term concentration of boric acid. The facility is capable of simulating high-pressure system responses.

The test models the reactor vessel, steam generators, reactor coolant pumps, in-containment refueling water storage tank, the automatic depressurization system vent paths, the lower containment, and the connecting piping. The hot legs and cold legs are modeled as are the core makeup tanks, PRHR heat exchangers, accumulators, and pressurizer.

Water is the working fluid and the core is simulated with electric heater rods scaled to match the core power levels consistent with the test scaling approach. Tests were performed to simulate various small-break LOCAs with different break locations, break sizes, with and without nonsafety systems operating. The analysis methods in Chapter 15 were compared to the test.

Full-Height, Full-Pressure Integral Systems Test

A test was performed to provide data on system performance at high pressure. This test facility is configured as a full-height, full-pressure integral test with AP600 features including two loops with one hot leg and two cold legs per loop, two core makeup tanks, two accumulators, a PRHR heat exchanger and an automatic depressurization system. The facility includes a scaled reactor vessel, steam generators, pressurizer and reactor coolant pumps. Water is the working fluid and the core is simulated with electric heater rods.

Tests were performed simulating small break LOCAs, steam generator tube ruptures and a steam line break transient. The analysis methods in Chapter 15 were compared to the test results.

1.5.2 AP600 Component Design Tests

The component design tests will provide a larger database for design optimization during the detailed design of the plant. Tests on selected plant components were performed to confirm their reliability or that materials and fabrication methods meet ASME requirements. These tests are also applicable to the AP1000 design and analysis.

Incore Instrumentation System Tests

Systems similar to the AP600 and AP1000 top mounted fixed incore detector (FID) instrumentation have been demonstrated in operating plants. A test was performed to demonstrate that the system will not be susceptible to electro-magnetic interference (EMI) from the nearby control rod drive mechanisms.

The electro-magnetic interference test was performed by mocking up instrument cables, bringing them into close proximity with an operating control rod drive mechanism, and measuring the resulting noise induced on simulated flux signals.

Reactor Coolant Pump/Steam Generator Airflow Test

The airflow test was performed to identify effects on the pump performance due to non-uniform channel head flow distribution, pressure losses of the channel head nozzle dams and pump suction nozzle, and possible vortices in the channel head induced by the pump impeller rotation.

The air test facility was constructed as an approximate one-half scale mockup of the outlet half of the channel head, the two pump suction nozzles, and two pump impellers and diffusers. The channel head tube sheet was constructed from clear plastic to allow smoke flow stream patterns to be seen.

The results of the test showed no flow anomalies or vortices in the channel head were induced by the dual impellers.

Reactor Coolant Pump High Inertia Rotor/Bearing Tests

In the AP600 a rotor, manufactured of depleted uranium clad with stainless steel, was incorporated into the hermetically sealed, high inertia centrifugal canned motor reactor coolant pump to provide the required flow coastdown performance for loss of flow transients.

Tests have been performed to verify manufacturability of the rotor, to determine friction and drag losses, to verify the operating performance of the pivoted-pad bearings, and to develop a detailed quantitative knowledge of the factors influencing bearing design and performance.

Tests were performed to verify the drag losses of the rotor with the journal bearing located on the pump shaft. Approximately 1000 cycles of starts and stops were also performed as a life test to demonstrate that the rotor will maintain its dimensional stability. These tests were performed on the specially-constructed, full-scale rotor/bearing test rig.

The results of these tests were used to check analytical methods used in the design of the AP1000 reactor coolant pump.

1.5.3 Combined License Information

This section contained no requirement for additional information.

1.5.4 References

1. WCAP-14217, “Core Makeup Tank Final Data Report,” (Proprietary), WCAP-14218 (Nonproprietary), November 1994.

2. WCAP-13286, “AP600 Passive Core Cooling System Check Valve Test Final Report,” (Proprietary), WCAP-13287 (Nonproprietary), April 1992.

3. WCAP-13891, “AP600 Automatic Depressurization System Phase A Test Data Report,” (Proprietary), WCAP-14095 (Nonproprietary), May 1994.

4. WCAP-14324, “Final Data Report for ADS Phase B1 Tests,” (Proprietary), WCAP-14325 (Nonproprietary), April 1995.

5. WCAP-14134, “AP600 Passive Containment Cooling System Integral Small-Scale Tests Final Report,” (Proprietary), WCAP-14137 (Nonproprietary), August 1994.

6. WCAP-13566, “AP600 1/8th Large Scale Passive Containment Cooling System Heat Transfer Baseline Data Report,” (Proprietary), Revision 1, WCAP-13567 (Nonproprietary), Revision 0, December 1992.

7. WCAP-14135, “Final Test Report for PCS Large Scale Phase 2 and Phase 3 Tests,” (Proprietary), WCAP-14138 (Nonproprietary), Revision 3, September 1998.

8. WCAP-13353, “Passive Containment Cooling System Water Distribution Phase 1 Test Data Report,” (Proprietary), WCAP-13354 (Nonproprietary), April 1992.

9. WCAP-13296, “PCS Water Distribution Test Phase II Test Data Report,” (Proprietary), WCAP-13297 (Nonproprietary), March 1992.

10. WCAP-13960, “PCS Water Distribution Phase 3 Test Data Report,” (Proprietary), WCAP-13961 (Nonproprietary), December 1993.

11. WCAP-13294, “Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-13295 (Nonproprietary), April 1992.

12. WCAP-13323, “Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-13324 (Nonproprietary), August 1992.

13. WCAP-14068, “Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-14084 (Nonproprietary), May 1994.

14. WCAP-14169, “Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report,” (Proprietary), WCAP-14170 (Nonproprietary), September 1994.

15. WCAP-14091, “Phase IVb Wind Tunnel Testing for the Westinghouse AP600 Reactor,” (Proprietary), WCAP-14092 (Nonproprietary), July 1994.

16. WCAP-12980, “AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report,” (Proprietary), WCAP-13573 (Nonproprietary), Revision 3, April 1997.

17. WCAP-14371, “AP600 Low Flow Critical Heat Flux (CHF) Test Data Analysis,” (Proprietary), WCAP-14372 (Nonproprietary), May 1995.

18. WCAP-14252, “AP600 Low Pressure 1/4 Height Integral Systems Tests - Final Data Report,” (Proprietary), WCAP-14253 (Nonproprietary), Revision 1, November 1998.

19. WCAP-14309, “AP600 Design Certification Program, SPES-2 Tests Final Data Report,” (Proprietary), WCAP-14310 (Nonproprietary), Revision 2, May 1997.

20. WCAP-12648, “AP600 Incore Instrumentation System Electromagnetic Interference Test Report,” (Proprietary), WCAP-13322 (Nonproprietary), Revision 1, April 1992.

21. WCAP-13298, “RCP Air Model Test Report,” (Proprietary), WCAP-13299 (Nonproprietary), August 1991.

22. WCAP-12668, “AP600 High Inertia Rotor Testing - Phase I, Test Report,” (Proprietary), WCAP-13321 (Nonproprietary), March 1990.

23. WCAP-13319, “AP600 High Inertia Rotor Testing - Phase 2 Report,” (Proprietary), WCAP-13320 (Nonproprietary), August 1991.

24. WCAP-13758, “High Inertia Rotor Test Phase 3 Report,” (Proprietary), WCAP-13759 (Nonproprietary), June 1993.

25. WCAP-15613, “AP1000 PIRT and Scaling Assessment,” (Proprietary), WCAP-15706 (Nonproprietary), March 2001.

26. Not used.

27. NUREG/CR-6641, “Final Report of NRC Research Conducted at Oregon State University,” August 1999.

Table 1.5-1
AP600 Design Tests and AP1000 Evaluation

Test	Reference
LOCA Mitigation	
Core Makeup Tank Performance Test	(1)
Passive Safety Injection System Check Valve Test	(2)
Automatic Depressurization System Hydraulic Test	(3), (4)
Containment Cooling	
Integral Containment Cooling Test	(5)
Passive Containment Cooling System Heat Transfer Test	(6), (7)
Passive Containment Cooling System Water Distribution Test	(8), (9), (10)
Passive Containment Cooling System Wind Tunnel Test	(11), (12), (13), (14), (15)
Non-LOCA Transients	
Passive Residual Heat Removal Heat Exchanger Performance Test	(16)
Departure from Nucleate Boiling Test	(17)
Integral Systems Tests	
Low Pressure Integral Systems Test	(18)
Full Height Full Pressure Integral Systems Test	(19)
NRC Low Pressure Integral Systems Test	(27)
Component Design Tests	
Incore Instrumentation System Test	(20)
Reactor Coolant Pump/Steam Generator Airflow Test	(21)
Reactor Coolant Pump High Inertia Rotor/Bearing Test	(22), (23), (24)
AP1000 Evaluation	
AP1000 PIRT and Scaling Assessment	(25)

1.6 Material Referenced

The AP1000 Design Control Document references various Westinghouse technical support documents; these documents are listed by DCD section in Table 1.6-1.

Table 1.6-201 provides a list of the various technical documents incorporated by reference in the FSAR in addition to those technical documents incorporated by reference in the AP1000 DCD.

Table 1.6-1 (Sheet 1 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
1.5	WCAP-14217 (P) WCAP-14218	Core Makeup Tank Final Data Report, November 1994
	WCAP-13286 (P) WCAP-13287	AP600 Passive Core Cooling System Check Valve Test Final Report, April 1992
	WCAP-13891 (P) WCAP-14095	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324 (P) WCAP-14325	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-14134 (P) WCAP-14137	AP600 Passive Containment Cooling System Integral Small-Scale Tests Final Report, August 1994
	WCAP-13566 (P) WCAP-13567	AP600 1/8th Large Scale Passive Containment Cooling System Heat Transfer Baseline Data Report, Revision 1, December 1992
	WCAP-14135 (P) WCAP-14138	Final Test Report for PCS Large Scale Phase 2 and Phase 3 Tests, Revision 3, September 1998
	WCAP-13353 (P) WCAP-13354	Passive Containment Cooling System Water Distribution Phase 1 Test Data Report, April 1992
	WCAP-13296 (P) WCAP-13297	PCS Water Distribution Test Phase II Test Data Report, March 1992
	WCAP-13960 (P) WCAP-13961	PCS Water Distribution Phase 3 Test Data Report, December 1993
	WCAP-13294 (P) WCAP-13295	Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor, April 1992
	WCAP-13323 (P) WCAP-13324	Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor, August 1992
	WCAP-14068 (P) WCAP-14084	Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor, May 1994

(P) Denotes Document is Proprietary

Table 1.6-1 (Sheet 2 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
1.5	WCAP-14169 (P) WCAP-14170	Phase IVa Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report, September 1994
	WCAP-14091 (P) WCAP-14092	Phase IVb Wind Tunnel Testing for the Westinghouse AP600 Reactor, July 1994
	WCAP-12980 (P) WCAP-13573	AP600 Passive Residual Heat Removal Heat Exchanger Test Final Report, Revision 3, April 1997
	WCAP-14371 (P) WCAP-14372	AP600 Low Flow Critical Heat Flux (CHF) Test Data Analysis, May 1995
	WCAP-14252 (P) WCAP-14253	AP600 Low Pressure 1/4 Height Integral Systems Tests - Final Data Report, Revision 1, November 1998
	WCAP-14309 (P) WCAP-14310	AP600 Design Certification Program, SPES-2 Tests Final Data Report, Revision 2, May 1997
	WCAP-12648 (P) WCAP-13322	AP600 Incore Instrumentation System Electromagnetic Interference Test Report, Revision 1, April 1992
	WCAP-13298 (P) WCAP-13299	RCP Air Model Test Report, August 1991
	WCAP-12668 (P) WCAP-13321	AP600 High Inertia Rotor Testing - Phase I, Test Report, March 1990
	WCAP-13319 (P) WCAP-13320	AP600 High Inertia Rotor Testing - Phase 2 Report, August 1991
	WCAP-13758 (P) WCAP-13759	High Inertia Rotor Test - Phase 3 Report, June 1993
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment Report, March 2001
1.9	WCAP-15993	Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria, Revision 1, March 2003

(P) Denotes Document is Proprietary

Table 1.6-1 (Sheet 3 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
1.9	WCAP-15799	AP1000 Compliance with SRP Acceptance Criteria, Revision 1, August 2003
	WCAP-15800	Operational Assessment for AP1000, Revision 3, July 2004
	WCAP-15992	AP1000 Adverse Systems Interactions Evaluation Report, Revision 1, February 2003
	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems
1A	WCAP-8577	The Application of Pre-Heat Temperature After Welding of Pressure Vessel Steels, September 1975
	WCAP-16650-P (P) WCAP-16650-NP	Analysis of the Probability of the Generation of Missiles from Fully Integral Nuclear Low Pressure Turbines, Revision 0, February 2007
3.3	WCAP-13323 (P) WCAP-13324	Phase II Wind Tunnel Testing for the Westinghouse AP600 Reactor, August 1992
	WCAP-14068 (P) WCAP-14084	Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, May 1994
	WCAP-14169 (P) WCAP-14170	Phase IVA Wind Tunnel Testing for the Westinghouse AP600 Reactor, Supplemental Report, September 1994
	WCAP-13294-P (P) WCAP-13295-NP	Phase I Wind Tunnel Testing for the Westinghouse AP600 Reactor, April 1992
3.6	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods, March 1973
	WCAP-8708 (P) WCAP-8709-A	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8252	Documentation of Selected Westinghouse Structural Analysis Computer Codes, Revision 1, May 1977
3.7	WCAP 7921-AR	Damping Values of Nuclear Power Plant Components, May 1974
	WCAP-9903 (P)	Justification of the Westinghouse Equivalent Static Analysis Method for Seismic Qualification of Nuclear Power Plant Auxiliary Mechanical Equipment, August 1980

(P) Denotes Document is Proprietary

Table 1.6-1 (Sheet 4 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
3.8	WCAP-13891 (P) WCAP-14095	AP600 Automatic Depressurization System Phase A Test Data Report, May 1994
	WCAP-14324 (P) WCAP-14325	Final Data Report for ADS Phase B1 Tests, April 1995
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment, March 2001
	[*APP-GW-GLR-602*	*AP1000 Shield Building Design Details for Select Wall and RC/SC Connections, Revision 5, Westinghouse Electric Company LLC*]*
3.9	WCAP-7765-AR	Westinghouse PWR Internals Vibrations Summary Three-Loop Internals Assurance, November 1973
	WCAP-8766 (P) WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant, May 1976
	WCAP-8516-P (P) WCAP-8517	UHI Plant Internals Vibrations Measurement Program and Pre- and Post-Hot Functional Examinations, March 1975
	WCAP-10846 (P)	Doel 4 Reactor Internals Flow-Induced Vibration Measurement Program, March 1985
	WCAP-10865 (P) WCAP-10866	South Texas Plant (TGX) Reactor Internals Flow-Induced Vibration Assessment, February 1985
	WCAP-8708-P-A (P) Volumes 1 and 2 WCAP-8709-A Volumes 1 and 2	MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics, February 1976
	WCAP-8446 (P) WCAP-8449	17x17 Drive Line Components Tests – Phase 1B 11, 111 D-Loop Drop and Deflection, December 1974
	WCAP-9693 (P)	Investigation of Feedwater Line Cracking in Pressurized Water Reactor Plants, June 1980
	WCAP-15949-P (P) WCAP-15949-NP	AP1000 Reactor Internals Flow-Induced Vibration Assessment Program, Revision 1, July 2003
	WCAP-16687-P (P)	AP1000 Reactor Internals Expected and Acceptable Responses During Preoperational Vibration Measurement Program, March 2007

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Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
3H	[*APP-GW-GLR-602*	*AP1000 Shield Building Design Details for Select Wall and RC/SC Connections, Revision 5, Westinghouse Electric Company LLC*]*
4.1	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985, and VANTAGE 5H Fuel Assembly, Addendum 2A, February 1989
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, April 1995
	[*WCAP-12488-A (P)* [*WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
4.2	[*WCAP-12488-A (P)* [*WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-10125-P-A (P) WCAP-10126-NP-A	Extended Burnup Evaluation of Westinghouse Fuel, December 1985
	WCAP-8183	Operational Experience with Westinghouse Cores (Revised Annually)
	WCAP-9179 (P) WCAP-9224	Properties of Fuel and Core Component Materials, July 1978
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, June 1990/ April 1995
	WCAP-8218-P-A (P) WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application, March 1975
	WCAP-10851-P-A (P) WCAP-11873-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, August 1988
	WCAP-13589-A (P) WCAP-14297-A	Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995
	WCAP-8963-P-A (P) WCAP-8964-A	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis, August 1977
	WCAP-10021-P-A (P) WCAP-10377-NP-A	Westinghouse Wet Annular Burnable Absorber Evaluation Report, Revision 1, October 1983
	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985

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Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
4.2	WCAP-8278 (P) WCAP-8279	Hydraulic Flow Test of the 17x17 Fuel Assembly, February 1974
	WCAP-8691 (P) WCAP-8692	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-9500-P-A (P) WCAP-9500-A	Reference Core Report 17x17 Optimized Fuel Assembly, May 1982
	WCAP-8236 (P) WCAP-8288	Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident, December 1973
	WCAP-9401-P-A (P) WCAP-9402-A	Verification, Testing, and Analysis of the 17x17 Optimized Fuel Assembly, August 1981
	WCAP-9283	Integrity of Primary Piping Systems of Westinghouse Nuclear Power Plants During Postulated Seismic Events, March 1978
	WCAP-15063-P-A (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Revision 1, July 2000
	WCAP-8377 (P)	Revised Clad Flattening Model, July 1974
	WCAP-7113	Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors, October 1967
	WCAP-16652-P	AP1000 Core & Fuel Design Technical Report, Revision 0
4.3	WCAP-9272-P-A (P) WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	[*WCAP-12488-P-A (P)* [*WCAP-14204-A*]*	*Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-12472-P-A (P) WCAP-12473-A	BEACON: Core Monitoring and Operations Support System, August 1994; Addendum 1, May 1996; Addendum 2, March 2001
	WCAP-8330	Westinghouse Anticipated Transients Without Reactor Trip Analysis, August 1974
	WCAP-7308-L-P-A (P) WCAP-7308-L-A	Evaluation of Nuclear Hot Channel Factor Uncertainties, June 1988
	WCAP-8218-P-A (P) WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application, March 1975

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Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
4.3	WCAP-8359	Effects of Fuel Densification Power Spikes on Clad Thermal Transients, July 1974
	WCAP-7811	Power Distribution Control of Westinghouse Pressurized Water Reactors, December 1971
	WCAP-8385 (P) WCAP-8403	Power Distribution Control and Load Following Procedures, September 1974
	WCAP-10216-P-A (P) WCAP-10217-A	Relaxation of Constant Axial Offset Control, F_Q Surveillance Technical Specification, Revision 1A, February 1994
	WCAP-7912-P-A (P) WCAP-7912-A	Power Peaking Factors, January 1975
	WCAP-8498	Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors, July 1975
	WCAP-9217 (P) WCAP-9218	Results of Control Rod Worth Program, October 1977
	WCAP-3696-8 (P)	Pressurized Water Reactor pH – Reactivity Effect Final Report, October 1968
	WCAP-3680-20 (P)	Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors, March 1968
	WCAP-3680-21 (P)	Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors, February 1969
	WCAP-3680-22 (P)	Xenon-Induced Spatial Instabilities in Three Dimensions, September 1969
	WCAP-7964	Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor, June 1971
	WCAP-7048-P-A (P) WCAP-7757-A	The PANDA Code, February 1975
	WCAP-7213-A (P) WCAP-7758-A	The TURTLE 24.0 Diffusion Depletion Code, February 1975
	WCAP-8768	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries – Winter 1977 – Summer 1978, Revision 2, October 1978

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Table 1.6-1 (Sheet 8 of 21) Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
4.3	WCAP-6073 (P)	LASER – A Depletion Program for Lattice Calculations Based on MUFT and THERMOS, April 1966
	WCAP-2048 (P)	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements, July 1962
	WCAP-11596-P-A (P) WCAP-11597-A	Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, June 1988
	WCAP-10841 (P) WCAP-10842	Qualification of the PHOENIX/POLCA Nuclear Design and Analysis Program for Boiling Water Reactors, June 1985
	WCAP-7806	Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods, December 1971
	WCAP-3385-56 Part II	Saxton Core II Fuel Performance Evaluation Part II: Evaluation of Mass Spectrometric and Radiochemical Analysis of Irradiated Saxton Plutonium Fuel, July 1973
	WCAP-3385-56 Part I	Saxton Core II - Fuel Performance Evaluation Part I: Materials, September 1971
	WCAP-3385-36	Saxton Plutonium Project - Quarterly Progress Report for the Period Ending June 20, 1973, July 1973
	WCAP-3385-37	Saxton Plutonium Project - Quarterly Progress Report for the Period Ending September 30, 1973, December 1973
	WCAP-3017-6094	Yankee Core Evaluation Program Final Report, January 1971
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-3726-1	PuO_2-UO_2 Fueled Critical Experiments, July 1967
	WCAP-13589-A (P) WCAP-14297-A	Assessment of Clad Flattening and Densification Power Spike Factor Elimination in Westinghouse Nuclear Fuel, March 1995
	WCAP-13524 (P) WCAP-14952-NP-A	APOLLO - A One Dimensional Neutron Theory Program, Revision 1, August 1994
	WCAP-16652-P	AP1000 Core & Fuel Design Technical Report, Revision 0

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Material Referenced**

DCD Section Number	Westinghouse Topical Report Number	Title
4.3	WCAP-3385-54	Saxton Plutonium Program Critical Experiments for the Saxton Partial Plutonium Core, Westinghouse Electric Corp., Atomic Power Division, December 1965
4.4	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-6065 (P)	Melting Point of Irradiated UO_2, February 1965
	WCAP-10444-P-A (P) WCAP-10445-NP-A	Reference Core Report VANTAGE 5 Fuel Assembly, September 1985
	WCAP-9226-P (P) WCAP-9227-NP	Reactor Core Response to Excessive Secondary Steam Releases, January 1989
	WCAP-7695-L (P)	DNB Test Results for R-Grid Thimble Cold Wall Cells, Addendum 1, October 1972
	[*WCAP-12488-A (P)*	*Westinghouse Fuel Criteria Evaluation Process, October 1994*]*
	WCAP-7941-P-A (P) WCAP-7959-A	Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid, January 1975
	WCAP-8298-P-A (P) WCAP-8290-A	The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing, January 1975
	WCAP-8174 (P) WCAP-8202-A	Effect of Local Heat Flux Spikes on DNB in Non Uniform Heated Rod Bundles, August 1973
	WCAP-7667-P-A (P) WCAP-7755-A	Interchannel Thermal Mixing with Mixing Vane Grids, January 1975
	WCAP-8691 (P) WCAP-8692	Fuel Rod Bow Evaluation, Revision 1, July 1979
	WCAP-8054-P-A (P) WCAP-8195-A	Applications of THINC-IV Program to PWR Design, October 1973
	WCAP-7956-P-A (P)	THINC-IV, An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores, February 1989
	WCAP-2923	In-Pile Measurement of UO_2 Thermal Conductivity, March 1966

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Material Referenced**

DCD Section Number	Westinghouse Topical Report Number	Title
4.4	WCAP-10851-P-A (P) WCAP-11873-A	Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations, August 1988
	WCAP-8720 Addendum 2	Revised PAD Code Thermal Safety Model, October 1982
	WCAP-6069	Burnup Physics of Heterogeneous Reactor Lattices, June 1965
	WCAP-3385-56 Part II	Saxton Core II Fuel Performance Evaluation: Evaluation of Mass Spectrometric and Radiochemical Analyses of Irradiated Saxton Plutonium Fuel, July 1970
	WCAP-7912-P-A (P) WCAP-7912-A	Power Peaking Factors, January 1975
	WCAP-8453-A	Analysis of Data from the Zion (Unit 1) THINC Verification Test, May 1976
	WCAP-12610-P-A (P) WCAP-14342-A	VANTAGE+ Fuel Assembly Reference Core Report, April 1995
	WCAP-15025-P-A (P) WCAP-15026-NP-A	Modified WRB-2 Correlation, WRB-2M, for Predicting Critical Heat Flux in 17x17 Rod Bundles with Modified LPD Mixing Vane Grids, April 1999
	WCAP-14565-P-A (P) WCAP-15306-NP-A	VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis, October 1999
	WCAP-15063-P-A (P) WCAP-15064-NP-A	Westinghouse Improved Performance Analysis and Design Model (PAD 4.0), Revision 1, July 2000
	WCAP-16652-NP	AP1000 Core & Fuel Design Technical Report, Revision 0
5.2	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-9292	Dynamic Fracture Toughness of ASME SA-508 Class 2a and ASME SA-533 Grade A Class 2 Base and Heat-Affected Zone Material and Applicable Weld Metals, March 1978
	WCAP-7477-L (P) WCAP-7735	Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems, March 1970 (P), August 1971 (Non-P)

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DCD Section Number	Westinghouse Topical Report Number	Title
5.2	WCAP-8324-A	Control of Delta Ferrite in Austenitic Stainless Steel Weldments, June 1975
	WCAP-8693	Delta Ferrite in Production Austenitic Stainless Steel Weldments, January 1976
5.3	WCAP-15557	Qualification of the Westinghouse Pressure Vessel Neutron Fluence Evaluation Methodology, August 2000
	WCAP-14040-NP-A	Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, Revision 2, January 1996
5.4	WCAP-15994-P (P) WCAP-15994-NP	Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel, March 2003
6.2	WCAP-8077 (P) WCAP-8078	Ice Condenser Containment Pressure Transient Analysis Methods, March 1973
	WCAP-8264-P-A (P) WCAP-8312-A	Westinghouse Mass and Energy Release Data for Containment Design, June 1975 (P), August 1975 (Non-P)
	WCAP-10325-P-A (P) WCAP-10326-A	Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version, May 1983
	WCAP-8822 (P) WCAP-8860 WCAP-8822-P-S1 (P) WCAP-8822-S2-P-A (P)	Mass and Energy Releases Following A Steam Line Rupture, September 1976 Supplement 1 - Calculations of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture, January 1985 Supplement 2 - Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs, September 1986
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-15846 (P) WCAP-15862	WGOTHIC Application to AP600 and AP1000, Revision 1, March 2004
	WCAP-15965-P (P) WCAP-15965-NP	AP1000 Subcompartment Models, November 2002
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997

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DCD Section Number	Westinghouse Topical Report Number	Title
6.2	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
6.3	WCAP-8966 (P)	Evaluation of Mispositioned ECCS Valves, September 1977
	WCAP-13594 (P) WCAP-13662 (NP)	FMEA of Advanced Passive Plant Protection System, Revision 1, June 1998
6A	WCAP-15846 (P) WCAP-15862	WGOTHIC Application to AP600 and AP1000, Revision 1, March 2004
	WCAP-14135 (P) WCAP-14138	Final Data Report for Passive Containment Cooling System Large Scale Test, Phase 2 and Phase 3, Revision 3, November 1998
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment Report, March 2001
7.1	WCAP-14605 (P) WCAP-14606 (NP)	Westinghouse Setpoint Methodology for Protection Systems – AP600, April 1996
	WCAP-16361-P WCAP-16361-NP	Westinghouse Setpoint Methodology for Protection Systems - AP1000, February 2011
	WCAP-15775	AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report
	[*WCAP-16096-NP-A*	*Software Program Manual for Common Q Systems, Revision 01A, December 2004*]*
	[*WCAP-16097-P-A* *WCAP-16097-NP-A*	*Common Qualified Platform, Revision 01, May 2003*]*
	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, April 2002
	WCAP-16674-P WCAP-16674-NP	AP1000 I&C Data Communication and Manual Control of Safety Systems and Components, Revision 4
	WCAP-16675-P WCAP-16675-NP	AP1000 Protection and Safety Monitoring System Architecture Technical Report, Revision 5
	APP-GW-GLR-017	AP1000 Standard Combined License Technical Report, Resolution of Common Q NRC Items

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DCD Section Number	Westinghouse Topical Report Number	Title
7.1	[*WCAP-17179-P* *WCAP-17179-NP*	*AP1000 Component Interface Module Technical Report*]*
	[*WCAP-15927 (NP)*	*Design Process for AP1000 Common Q Safety Systems, Revision 2, November 2008*]*
		Westinghouse Electric Company Quality Management System (QMS), (Non-Proprietary), Revision 5, October 2002
	APP-GW-J0R-012	AP1000 Protection and Safety Monitoring System Computer Security Plan, Revision 1
	[*WCAP-17201-P*	*AC160 High Speed Link Communication Compliance to DI&C-ISG-04 Staff Positions 9, 12, 13 and 15, Revision 0, February 2010*]*
	WCAP-17184-P (P)	AP1000™ Diverse Actuation System Planning and Functional Design Summary Technical Report
7.2	WCAP-16438-P WCAP-16438-NP	FMEA of AP1000 Protection and Safety Monitoring System, Revision 3
	WCAP-16592-P WCAP-16592-NP	Software Hazards Analysis of AP1000 Protection and Safety Monitoring System, Revision 2
	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, April 2002
	WCAP-16097-P-A WCAP-16097-NP-A	Common Qualified Platform, Digital Plant Protection System, Appendix 3, May 2003
7.3	WCAP-15776	Safety Criteria for the AP1000 Instrumentation and Control Systems, April 2002
7.7	WCAP-17184-P	AP1000™ Diverse Actuation System Planning and Functional Design Summary Technical Report
9.5	WCAP-15871	AP1000 Assessment Against NFPA 804, Revision 1, December 2002
10.2	WCAP-16650-P (P) WCAP-16650-NP	Analysis of the Probability of the Generation of Missiles for AP1000 Fully Integral Low Pressure Turbines, Revision 0, February 2007
	WCAP-16651-P (P) WCAP-16651-NP	Probabilistic Evaluation of Turbine Valve Test Frequency, Revision 1, May 2009
13	WCAP-14690	Designer's Input to Procedure Development for the AP600, Revision 1, June 1997

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Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
13	WCAP-13864	Rod Control System Evaluation Program, Revision 1-A, November 1994
15.0	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10054-P-A (P) WCAP-10081	Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate LOCA Analysis, Revision 1, March 1998
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE – A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
15.1	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-9226 (P) WCAP-9227	Reactor Core Response to Excessive Secondary Steam Releases, January 1978
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004

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DCD Section Number	Westinghouse Topical Report Number	Title
15.2	WCAP-7769	Overpressure Protection for Westinghouse Pressurized Water Reactors, Revision 1, June 1972
	WCAP-16779-NP	Overpressure Protection Report for AP1000 Nuclear Power Plant
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-9230 (P) WCAP-9231	Report on the Consequences of a Postulated Main Feedline Rupture, January 1978
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
15.3	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-8424	An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs, Revision 1, May 1975
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
15.4	WCAP-7979-P-A (P) WCAP-8028-A	TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, January 1975
	WCAP-7908-A	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984

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Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
15.4	WCAP-7588	An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, Revision 1A, January 1975
	WCAP-10965-P-A (P) WCAP-10966-A	ANC: A Westinghouse Advanced Nodal Computer Code, September 1986
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
15.5	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
15.6		Letter from R. C. Jones, Jr. (USNRC), to N. J. Liparulo (W), Subject: Acceptance for Referencing of the Topical Report, WCAP-12945 (P), Westinghouse CQD for Best Estimate LOCA Analysis, June 28, 1996
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate Analysis, Revision 2, March 1998
	WCAP-10079-P-A (P) WCAP-10080-A	NOTRUMP – A Nodal Transient Small Break and General Network Code, August 1985
	WCAP-10054-P-A (P) WCAP-10081-A	Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985
	WCAP-7907-P-A (P) WCAP-7907-A	LOFTRAN Code Description, April 1984
	WCAP-7908-A	FACTRAN – A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod, December 1989
	WCAP-11397-P-A (P) WCAP-11397-A	Revised Thermal Design Procedure, April 1989
	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14206 (P) WCAP-14207	Applicability of the NOTRUMP Computer Code to AP600 SSAR Small-Break LOCA Analyses, November 1994

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DCD Section Number	Westinghouse Topical Report Number	Title
15.6	WCAP-14601 (P) WCAP-15062	AP600 Accident Analyses – Evaluation Models, Revision 2, May 1998
	WCAP-14234 (P) WCAP-14235	LOFTRAN and LOFTTR2 AP600 Code Applicability Document, Revision 1, August 1997
	WCAP-14171 (P) WCAP-14172	WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident, Revision 2, March 1998
	WCAP-14807 (P) WCAP-14808	NOTRUMP Final Validation Report for AP600, Revision 5, August 1998
	WCAP-14776 (P) WCAP-14777	WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report, Revision 4, April 1998
	WCAP-15644-P (P) WCAP-15644-NP	AP1000 Code Applicability Report, Revision 2, March 2004
	WCAP-15613 (P) WCAP-15706	AP1000 PIRT and Scaling Assessment, March 2001
	WCAP-16009-P-A (P) WCAP-16009-NP-A	Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)
	WCAP-14449-P-A (P) WCAP-14450	Application of Best Estimate Large Break LOCA Methodology to Westinghouse PWRs with Upper Plenum Injection, Revision 1
16.1	WCAP-9272-P-A (P) WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	WCAP-11618	Methodically Engineered, Restructured and Improved, Technical Specifications, Merits Program – Phase II Task 5 Criteria Application, November 1987, including Addendum 1, April 1989
	WCAP-8385 (P) WCAP-8403	Power Distribution Control and Load Following Procedures, September 1974
	WCAP-10216-P-A (P) WCAP-10217-A	Relaxation of Constant Axial Offset Control F_Q Surveillance Technical Specifications, Revision 1A, February 1994
	WCAP-12945-P-A (P) WCAP-14747	Code Qualification Document for Best Estimate Loss of Coolant Accident Analysis, Revision 1, March 1998

(P) Denotes Document is Proprietary

Table 1.6-1 (Sheet 18 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
16.1	WCAP-12472-P-A (P) WCAP-12473-A	BEACON Core Monitoring and Operations Support System, August 1994, and Addendum 1, May 1996
	WCAP-7308-L-P-A (P) WCAP-7308-L-A	Evaluation of Nuclear Hot Channel Factor Uncertainties, June 1988
	WCAP-9273-NP-A	Westinghouse Reload Safety Evaluation Methodology, July 1985
	WCAP-14606	Westinghouse Setpoint Methodology for Protection Systems, April 1996
	WCAP-10271-P-A (P) WCAP-10272-A	Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System, June 1996
	WCAP-7924-A	Basis for Heatup and Cooldown Limit Curves, April 1975
	WCAP-16361-P (P)	Westinghouse Setpoint Methodology for Protection Systems – AP1000, February 2011
	WCAP-13632-P-A (P) WCAP-13787-A	Elimination of Pressure Sensor Response Time Testing Requirements, Revision 2, January 1996
	WCAP-7769	Topical Report on Overpressure Protection, October 1971
	WCAP-15985	AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process, Revision 2, August 2003
	WCAP-16779	AP1000 Overpressure Protection Report, April 2007
17.6	WCAP-8370	Energy Systems Business Unit – Power Generation Business Unit Quality Assurance Plan, Revision 12a
	WCAP-8370/7800	Energy Systems Business Unit – Nuclear Fuel Business Unit Quality Assurance Plan, Revision 11A/7A
	WCAP-12600	AP600 Advanced Light Water Reactor Design Quality Assurance Program Plan, Revision 4, January 1998
18.1	WCAP-14645	Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant, Revision 3
	WCAP-14644	AP600/AP1000 Functional Requirements Analysis and Function Allocation, Revision 1

(P) Denotes Document is Proprietary

Table 1.6-1 (Sheet 19 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
18.1	WCAP-14694	Designer's Input to Determination of the AP600 Main Control Room Staffing Level, July 1996
	[*WCAP-14651*	*Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
	WCAP-14690	Designer's Input to Procedure Development for the AP600, Revision 1, June 1997
	WCAP-14655	Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel, Revision 1, August 1996
	[*WCAP-15860*	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 2, October 2003*]*
18.2	WCAP-14645	Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant, Revision 3
	WCAP-14694	Designer's Input to Determination of the AP600 Main Control Room Staffing Level, July 1996
	[*WCAP-15847*	*AP1000 Quality Assurance Procedures Supporting NRC Review of AP1000 DCD Sections 18.2 and 18.8, Rev. 1, December 2002*]*
	WCAP-14644	AP600/AP1000 Functional Requirements Analysis and Function Allocation, Revision 1
18.3	WCAP-14645	Human Factors Engineering Operating Experience Review Report for the AP1000 Nuclear Power Plant, Revision 3
18.4	WCAP-14644	AP600/AP1000 Functional Requirements Analysis and Function Allocation, Revision 1
18.5	WCAP-10170	Emergency Response Facilities Design and V&V Process, April 1982
	[*WCAP-14695*	*Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology, July 1996*]*
	[*WCAP-14651*	*Integration of Human Reliability Analysis and Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
18.6	WCAP-14694	Designer's Input to Determination of the AP600 Main Control Room Staffing Level, July 1996
18.7	[*WCAP-14651*	*Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*

Table 1.6-1 (Sheet 20 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
18.7	WCAP-16555	AP1000 Identification of Critical Human Actions and Risk Important Tasks, Revision 1
18.8	[*WCAP-14651*	*Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
	[*WCAP-15860*	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 2, October 2003*]*
18.8	[*WCAP-14695*	*Description of the Westinghouse Operator Decision Making Model and Function Based Task Analysis Methodology, July 1996*]*
	WCAP-14655	Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel, Revision 1, August 1996
	WCAP-14690	Designer's Input to Procedure Development for the AP600, Revision 1, June 1997
	WCAP-10170	Emergency Response Facilities Design and V&V Process, April 1982
	WCAP-14694	Designer's Input to Determination of the AP600 Main Control Room Staffing Level, July 1996
	[*WCAP-14396*	*Man-in-the-Loop Test Plan Description, Revision 3, November 2002*]*
18.9	WCAP-14690	Designer's Input to Procedure Development for the AP600, Revision 1, June 1997
18.10	WCAP-14655	Designer's Input to the Training of the Human Factors Engineering Verification and Validation Personnel, Revision 1, August 1996
18.11	[*WCAP-15860*	*Programmatic Level Description of the AP1000 Human Factors Verification and Validation Plan, Revision 2, October 2003*]*
	[*APP-OCS-GEH-120*	*AP1000 Human Factors Engineering Design Verification Plan,]* Revision 2, Westinghouse Electric Company LLC*
	[*APP-OCS-GEH-220*	*AP1000 Human Factors Engineering Task Support Verification Plan,]* Revision 2, Westinghouse Electric Company LLC*
	[*APP-OCS-GEH-320*	*AP1000 Human Factors Engineering Integrated System Validation Plan,]* Revision 4, Westinghouse Electric Company LLC*
	[*APP-OCS-GEH-420*	*AP1000 Human Factors Engineering Discrepancy Resolution Process,]*,Revision 2, Westinghouse Electric Company LLC*]

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RN-13-031
RN-13-052
RN-14-154

Table 1.6-1 (Sheet 21 of 21)
Material Referenced

DCD Section Number	Westinghouse Topical Report Number	Title
18.11	[*APP-OCS-GEH-520*	*AP1000 Plant Startup Human Factors Engineering Design Verification Plan,]* Revision 4, Westinghouse Electric Company LLC*
18.12	[*WCAP-14651*	*Integration of Human Reliability Analysis with Human Factors Engineering Design Implementation Plan, Revision 2, May 1997*]*
	WCAP-13793	The AP600 System/Event Matrix, June 1994
19.41.13	WCAP-13388 (P) WCAP-13389	AP600 Phenomenological Evaluation Summaries, (Prop – Revision 0, June 1992, Non-Prop - Revision 1, 1994)
19.59	APP-GW-GJR-400 (NT)	Framework for AP1000 Severe Accident Management Guidance, Revision 0, January 2007
19B	WCAP-13388 (P) WCAP-13389	AP600 Phenomenological Evaluation Summaries (Prop - Revision 0, June 1992, Non-Prop - Revision 1, 1994)
19D	APP-GW-GJR-400 (NT)	Framework for AP1000 Severe Accident Management Guidance, Revision 0, January 2007
19E	WCAP-10698-P-A (P) WCAP-10750-A	SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill, August 1987
	WCAP-14171 (P) WCAP-14172	WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident, Revision 2, March 1998

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(P) Denotes Document is Proprietary
(NT)Denotes Document is Non-Topical Westinghouse Report

Table 1.6-201
Additional Material Referenced

Author / Report Number[a]	Title	Revision	Section	Document Transmittal	ADAMS Accession Number
Westinghouse / APP-GW-GL-700	AP1000 Design Control Document	19	All	June 2011	ML11171A500
NEI 07-08A	Generic FSAR Template Guidance for Ensuring That Occupational Radiation Exposures Are As Low As Is Reasonably Achievable (ALARA)	0	12.1	October 2009	ML093220164
NEI 07-03A	Generic FSAR Template Guidance for Radiation Protection Program Description	0	Appendix 12AA	May 2009	ML091490684
NEI 06-13A	Template for an Industry Training Program Description	2	13.2	March 2009	ML090910554
NEI 07-02A	Generic FSAR Template Guidance for Maintenance Rule Program Description for Plants Licensed Under 10 CFR Part 52	0	17.6	March 2008	ML080910149
10 CFR Part 52 Appendix D	Design Certification Rule for the AP1000 Design	–	1.1	–	–
QAPD	SCE&G Quality Assurance Program Description	4	17.5	October 2014	ML13290A631
Emergency Plan	VCSNS 2 and 3 Emergency Plan	6	13.3	December 2014	ML14356A277
Security Plan	VCSNS 2 and 3 Physical Security Plan	2	13.6	August 2010	Not Applicable (Safeguards)
Cyber Security Plan	VCSNS 2 and 3 Cyber Security Plan	1	13.6	June 2011	Not Applicable (SUNSI)

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(a) The NRC-accepted NEI documents identified by the A in the document number include the accepted template, the NRC safety evaluation, and corresponding responses to the NRC Requests for Additional Information. Only the accepted template is incorporated by reference. The remainder of the document is referenced but not incorporated into the FSAR.

(A) Denotes NRC approved document.

1.7 Drawings and Other Detailed Information

The figures referenced in Subsections 1.7.1 and 1.7.2 may represent a functional diagram, general structural representation, or another general illustration. For instrumentation and control (I&C) systems, figures may also represent aspects of the relevant logic of the system or part of the system. Unless specified explicitly, the figures are not indicative of the scale, location, dimensions, shape, or spatial relationships of as-built structures, systems, and components. In particular, the as-built attributes of structures, systems, and components may vary from the attributes depicted on the figures, provided that those safety functions discussed in the design description pertaining to the figure are not adversely affected.

1.7.1 Electrical and Instrumentation and Control Drawings

Instrument and control functional diagrams, electrical one-line diagrams, and onsite standby diesel generator loading sequence and initiating circuit logic diagrams are listed in Table 1.7-1.

The legend for electrical power, control, lighting, and communication drawings is provided in Figure 1.7-1, sheets 1, 2, and 3. The index, notes, and symbols for instrument and control functional diagrams are provided in Figure 7.2-1.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 contains a list of piping and instrumentation diagrams (P&IDs) and the corresponding figure numbers. The three letter system names are provided in Table 1.7-2. Figures appear at the end of the respective text section. The P&ID legend, Figure 1.7-2, sheets 1, 2, and 3, provides an explanation of AP1000 symbols and characters used in these figures.

1.7.3 Combined License Information

This section contained no requirement for additional information.

Table 1.7-1
I&C Functional and Electrical One-Line Diagrams

DCD Figure Number	Title
7.2-1 (Sheet 1)	Index and Symbols
7.2-1 (Sheet 2)	Reactor Trip Function
7.2-1 (Sheet 3)	Nuclear Startup Protection
7.2-1 (Sheet 4)	Nuclear Overpower Protection
7.2-1 (Sheet 5)	Core Heat Removal Protection and Reactor Coolant Pump Trip
7.2-1 (Sheet 6)	Primary Overpressure & Loss of Heat Sink Protection
7.2-1 (Sheet 7)	Loss of Heat Sink Protection
7.2-1 (Sheet 8)	Loss of Heat Sink Protection
7.2-1 (Sheet 9)	Steam Line Isolation
7.2-1 (Sheet 10)	Feedwater Isolation
7.2-1 (Sheet 11)	Safeguards Actuation
7.2-1 (Sheet 12)	Core Makeup Tank Actuation
7.2-1 (Sheet 13)	Containment and Other Protection
7.2-1 (Sheet 14)	Turbine Trip
7.2-1 (Sheet 15)	Automatic RCS Overpressurization Valve Sequencing
7.2-1 (Sheet 16)	Incontainment Refueling Water Storage Tank Actuations
7.2-1 (Sheet 17)	Passive Residual Heat Removal and Core Makeup Tank Isolation Valve Interlocks
7.2-1 (Sheet 18)	Normal Residual Heat Removal System Isolation Valve Interlocks
7.2-1 (Sheet 19)	Containment Vacuum Relief Protection
7.2-1 (Sheet 20)	Diverse Actuation System Logic, Automatic Actuations
7.2-1 (Sheet 21)	Diverse Actuation System Logic, Manual Actuations
8.3.1-1	AC Power System - Station One-Line Diagram (Sheets 1 & 2)
8.3.1-2	On-site Standby Diesel Generator Initiation Circuit Logic Diagram
8.3.1-3	Post 72 Hours Temporary Electric Power One Line Diagram
8.3.2-1	Class 1E DC System One-Line Diagrams (Sheets 1 & 2)
8.3.2-2	Class 1E 208Y/120V UPS One-Line Diagram
8.3.2-3	Non-Class 1E DC & UPS System One-Line Diagrams (Sheets 1, 2 & 3)

Table 1.7-2 (Sheet 1 of 3)
AP1000 System Designators and System Diagrams

Designator	System (Note 1)	Section	Figure (Note 2)
ASS	Auxiliary Steam Supply System	10.4.10	None
BDS	Steam Generator Blowdown System	10.4.8	10.4.8-1
CAS	Compressed and Instrument Air Systems	9.3.1	9.3.1-1
CCS	Component Cooling Water System	9.2.2	9.2.2-2
CDS	Condensate System	10.4.7	10.4.7-1
CES	Condenser Tube Cleaning System	10.4.1.2.1, 10.4.5.2.3	None
CFS	Turbine Island Chemical Feed System	10.4.11	None
CMS	Condenser Air Removal System	10.4.2	None
CNS	Containment System	6.2.3	None
CPS	Condensate Polishing System	10.4.6	10.4.6-1
CVS	Chemical and Volume Control System	9.3.6	9.3.6-1
CWS	Circulating Water System (Partially out of scope)	10.4.5	10.4-201
DAS	Diverse Actuation System	7.7	7.2-1 (Sh. 19 & 20)
DDS	Data Display and Processing System	7.1 & 7.7	7.1-1
DOS	Standby Diesel Fuel Oil System	9.5.4	9.5.4-1
DRS	Storm Drain System (Wholly out of scope)	None	None
DTS	Demineralized Water Treatment System	9.2.3	None
DWS	Demineralized Water Transfer and Storage System	9.2.4	9.2.4-1
ECS	Main ac Power System	8.3.1	8.3.1-1
EDS	Non Class 1E dc and UPS System	8.3.2	8.3.2-3
EFS	Communication Systems	9.5.2	None
EGS	Grounding and Lightning Protection System	8.3.1.1	None
EHS	Special Process Heat Tracing System	8.3.1.1	None
ELS	Plant Lighting System	9.5.3	None
EQS	Cathodic Protection System (Partially out of scope)	None	None
FHS	Fuel Handling and Refueling System	9.1.1, 9.1.2, 9.1.4	9.1 - various
FPS	Fire Protection System	9.5.1, 6.5.2	9.5.1-1
FWS	Main and Startup Feedwater System	10.4.7, 10.4.9	10.4.7-1
GSS	Gland Seal System	10.4.3	10.4.3-1
HCS	Generator Hydrogen and CO_2 Systems	10.2	None
HDS	Heater Drain System	10.4.7	None
HSS	Hydrogen Seal Oil System	10.2	None
IDS	Class 1E dc and UPS System	8.3.2	8.3.2-1
IIS	In-core Instrumentation System	4.4.6	None

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Table 1.7-2 (Sheet 2 of 3)
AP1000 System Designators and System Diagrams

Designator	System (Note 1)	Section	Figure (Note 2)
LOS	Main Turbine and Generator Lube Oil System	10.2	None
MES	Meteorological and Environmental Monitoring System (Wholly out of scope)	2.3.3	None
MHS	Mechanical Handling System	9.1	None
MSS	Main Steam System	10.3	10.3.2-2
MTS	Main Turbine System	10.2	10.2-1
OCS	Operation and Control Centers System	7.1, Ch. 18	7.1-1
PCS	Passive Containment Cooling System	6.2.2	6.2.2-1
PGS	Plant Gas Systems	9.3.2	None
PLS	Plant Control System	7.1 & 7.7	7.1-1
PMS	Protection and Safety Monitoring System	Ch. 7	7.2-1
PSS	Primary Sampling System	9.3.3	9.3.3-1
PWS	Potable Water System (Partially out of scope)	9.2.5	None
PXS	Passive Core Cooling System	6.3	6.3-1
RCS	Reactor Coolant System	5.1	5.1-5
RDS	Gravity and Roof Drain Collection System (Partially out of scope)	None	None
RMS	Radiation Monitoring System	11.5	None
RNS	Normal Residual Heat Removal System	5.4.7	5.4-7
RWS	Raw Water System (Wholly out of scope)	9.2.1.2.2, 9.2.1.2.3.1, 9.2.3, 9.2.5, 9.2.11	9.2-201
RXS	Reactor System	3.9.4, 3.9.5, 4.2.2.2, 4.2.2.3.1, 5.3	5.3-1
SDS	Sanitary Drainage System (Partially out of scope)	9.2.6	None
SES	Plant Security System (Partially out of scope)	13.6	None
SFS	Spent Fuel Pit Cooling System	9.1.3	9.1-6
SGS	Steam Generator System	10.3, 10.4.7, 10.4.9	10.3.2-1
SJS	Seismic Monitoring System	3.7.4	None
SMS	Special Monitoring System	4.4.6.4	None
SSS	Secondary Sampling System	9.3.4	None
SWS	Service Water System	9.2.1	9.2.1-1
TCS	Turbine Building Closed Cooling Water System	9.2.8	None

Table 1.7-2 (Sheet 3 of 3)
AP1000 System Designators and System Diagrams

Designator	System (Note 1)	Section	Figure (Note 2)
TDS	Turbine Island Vents, Drains and Relief System	9.2.9.2.2, 10.4.2.2.1, 10.4.3.1.2, 10.4.3.2.2, 10.4.6.3	None
TOS	Main Turbine Control and Diagnostics System	10.2.2.4	None
TVS	Closed Circuit TV System (Wholly out of scope)	None	None
VAS	Radiologically Controlled Area Ventilation System	9.4.3	9.4.3-1
VBS	Nuclear Island Nonradioactive Ventilation System	9.4.1	9.4.1-1
VCS	Containment Recirculation Cooling System	9.4.6	9.4.6-1
VES	Main Control Room Emergency Habitability System	6.4	6.4-2
VFS	Containment Air Filtration System	9.4.7	9.4.7-1
VHS	Health Physics and Hot Machine Shop HVAC System	9.4.11	9.4.11-1
VLS	Containment Hydrogen Control System	6.2.4	6.2.4 - various
VRS	Radwaste Building HVAC System	9.4.8	9.4.8-1
VTS	Turbine Building Ventilation System	9.4.9	9.4.9-1
VUS	Containment Leak Rate Test System	6.2.5	6.2.5-1
VWS	Central Chilled Water System	9.2.7	9.2.7-1
VXS	Annex/Auxiliary Non-Radioactive Ventilation System	9.4.2	9.4.2-1
VYS	Hot Water Heating System	9.2.10	None
VZS	Diesel Generator Building Ventilation System	9.4.10	9.4.10-1
WGS	Gaseous Radwaste System	11.3	11.3-2
WLS	Liquid Radwaste System	11.2	11.2-2
WRS	Radioactive Waste Drain System	9.3.5, 11.2	9.3.5-1
WSS	Solid Radwaste System	11.4	11.4-1
WWS	Waste Water System (Partially out of scope)	9.2.9	None
YFS	Yard Fire Water System (Wholly out of scope)	None	None
ZAS	Main Generation System (Note 3)	8.1	None
ZBS	Switchyard Single Line Diagram	8.2	8.2-202
ZBS	Transmission Switchyard and Offsite Power System (Wholly out of scope)	8.2	8.2-201, 8.2-202
ZOS	Onsite Standby Power System	8.2.1, 8.3.1	8.3.1-4, 8.3.1-5
ZRS	Offsite Retail Power System (Wholly out of scope)	None	None
ZVS	Excitation and Voltage Regulation System	10.2.2.3	None

Notes:

1. For the System names:
   a) An entry with the system name only means the system is wholly in the scope of the AP1000 design certification.
   b) An entry with the system name followed by (Partially out of scope) means the system is partially in the scope of the AP1000 design certification.
   c) An entry with the system name followed by (Wholly out of scope) means the system is not in the scope of the AP1000 design certification.
2. For the DCD Figures:
   In the AP1000 design documentation system, Piping and Instrumentation Diagrams are numbered xxx-M6-yyy, where xxx is the system designator and yyy is the sheet number. Electrical One-Line Diagrams are numbered xxx-E3-yyy, where xxx is the system designator and yyy is the sheet number. I&C Functional Logic Diagrams are numbered xxx-J1-yyy, where xxx is the I&C system designator and yyy is the sheet number.
3. For the Main Generation System:
   The high side voltage of the main step-up transformer and the reserve auxiliary transformer is site specific.

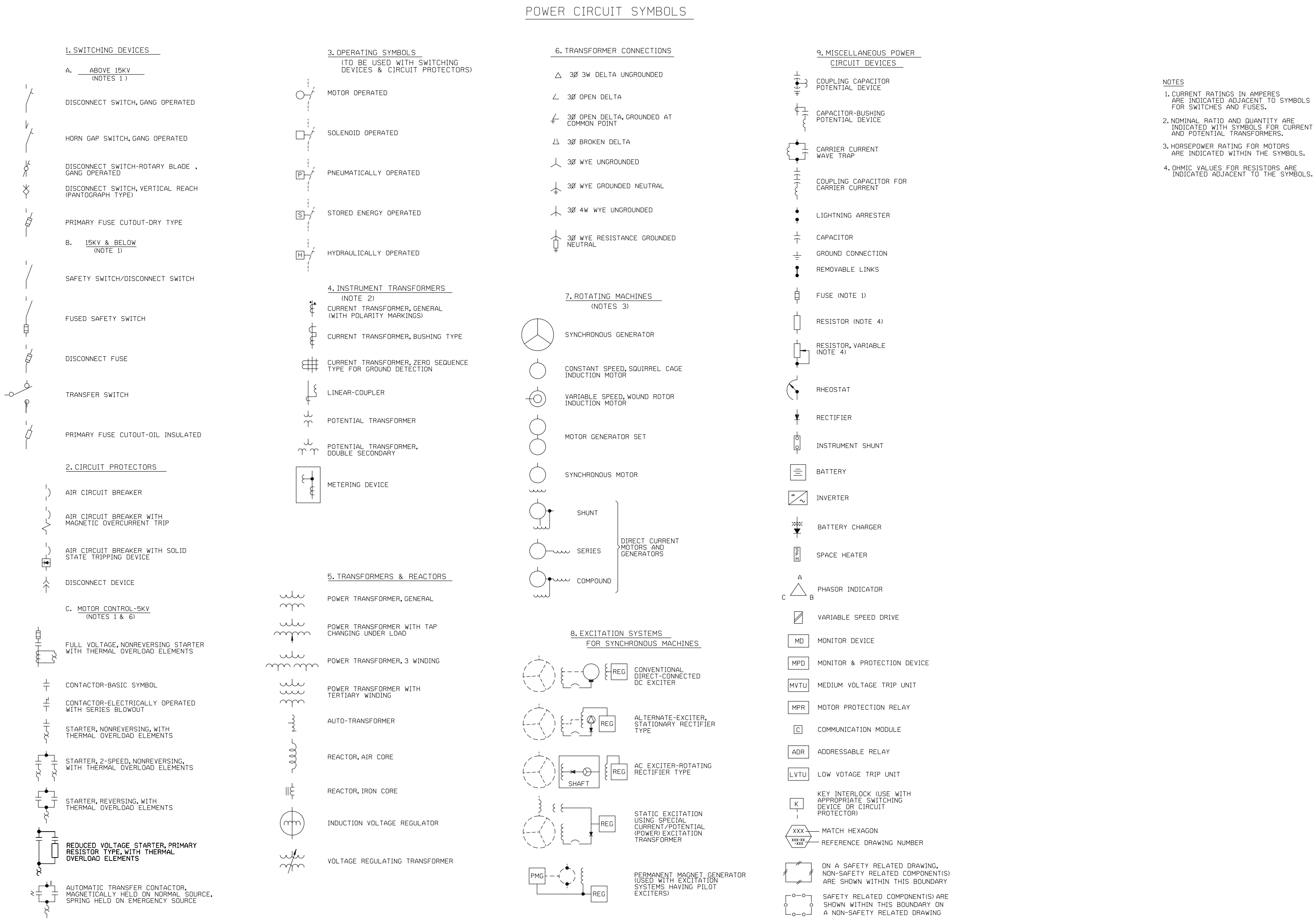


Figure 1.7-1 (Sheet 1 of 3) Legend for Electrical Power, Lighting, and Communication Drawings

POWER CIRCUIT SYMBOLS (CONTINUED)

10. METERS AND INSTRUMENTS

PREFIX R INDICATES RECORDING TYPE

- A — AMMETER
- AH — AMPERE-HOUR METER
- D — DEMAND METER
- F — FREQUENCY METER
- GD — GROUND DETECTOR
- OHM — OHMMETER
- OSC — OSCILLOGRAPH
- PD — PRINTING DEMAND METER
- PF — POWER FACTOR METER
- RT — RUNNING TIME METER
- SY — SYNCHROSCOPE
- T — TEMPERATURE METER
- THC — THERMAL CONVERTER
- TI — TEMPERATURE INDICATOR
- TLM — TELEMETERING
- TM — TRANSMITTER
- TR — TEMPERATURE RECORDER
- */T — TRANSDUCER
  - A-INDICATES AMPS
  - V-INDICATES VOLTS
  - VAR-INDICATES VARS
  - W-INDICATES WATTS
- V — VOLTMETER
- VAR — VARMETER
- VARH — VAR-HOUR METER
- VV — DOUBLE VOLTMETER
- W — WATTMETER
- WH — WATTHOUR METER
- WHD — WATTHOUR DEMAND METER

11. METER & INSTRUMENT SWITCHES

- AS — AMMETER SWITCH
- VS — VOLTMETER SWITCH
- SS — SYNCHRONIZING SWITCH
- MSS — METERING SELECTOR SWITCH

12. CONTROL SWITCHES & PUSHBUTTON STATIONS

- CONTROL SWITCH WITH RED & GREEN INDICATING LIGHTS. P-INDICATES PERMISSIVE
- CONTROL SWITCH WITH RED, GREEN & AMBER INDICATING LIGHTS
- START-STOP MAINTAINED CONTACT PUSH BUTTON
- START-STOP MAINTAINED CONTACT PUSH BUTTON WITH RED & GREEN INDICATING LIGHTS
- START-STOP MOMENTARY CONTACT PUSH BUTTON
- START-STOP MOMENTARY CONTACT PUSH BUTTON WITH RED & GREEN INDICATING LIGHTS
- RED & GREEN INDICATING LIGHTS
- SELECTOR SWITCH WITH RED & GREEN INDICATING LIGHTS
- HAND-OFF-AUTOMATIC SELECTOR SWITCH
- TEST SWITCH WITH RED & GREEN INDICATING LIGHTS
- COMBINATION CONTROL & TEST SWITCH WITH RED & GREEN INDICATING LIGHTS
- LO — LOCK-OUT PUSH BUTTON

13. MISCELLANEOUS DEVICES

- 20 — ELECTRICALLY OPERATED VALVE
- 23 — TEMPERATURE CONTROL DEVICE
- 24 — BUS TIE CIRCUIT BREAKER
- 25 — SYNCHRONIZING DEVICE
- 26 — APPARATUS THERMAL DEVICE
- 28 — FLAME DETECTOR
- 33 — POSITION SWITCH (POSITION OR LIMIT SWITCH)
- 41 — FIELD CIRCUIT BREAKER
- 42 — MOTOR STARTER
- 43 — TRANSFER DEVICE
- 52 — AC CIRCUIT BREAKER
- 63P — SUDDEN PRESSURE SWITCH
- 65 — GOVERNOR DEVICE
- 69 — PERMISSIVE CONTROL DEVICE
- 70 — RHEOSTAT DEVICE
- 71 — LIQUID OR GAS LEVEL SWITCH
- 72 — DC CIRCUIT BREAKER
- 75 — POSITION CHANGING MECHANISM
- 77 — IMPULSE TRANSMITTER DEVICE
- 89 — FUSE DISCONNECT
- 90 — REGULATING DEVICE
- 95 — METERING TRANSFER DEVICE
- 96 — BATTERY MONITOR
- 97 — BATTERY CHARGER FAILURE

14. RELAYS

- 21 — DISTANCE (IMPEDANCE)
- 27 — UNDERVOLTAGE
- 30 — ANNUNCIATOR
- 32 — REVERSE POWER
- 37 — UNDERCURRENT OR UNDERPOWER
- 40 — FIELD
- 46 — REVERSE PHASE (NEGATIVE PHASE SEQUENCE)
- 47 — PHASE SEQUENCE
- 48 — INCOMPLETE SEQUENCE
- 49 — THERMAL
- 50 — INSTANTANEOUS OVERCURRENT
- 50D — HIGHDROPOUT OVERCURRENT USED AS MOTOR DIFFERENTIAL
- 50G — INSTANTANEOUS GROUND CURRENT
- 50/51 — INSTANTANEOUS OVERCURRENT & TIME DELAY
- 51 — AC TIME OVERCURRENT
- 51G — GROUND CURRENT TIME DELAY
- 51N — RESIDUAL GROUND OVERCURRENT
- 51V — OVERCURRENT-TIME DELAY-VOLTAGE RESTRAINT
- 55 — POWER FACTOR
- 56 — FIELD APPLICATION
- 58 — RECTIFICATION FAILURE
- 59 — OVERVOLTAGE
- 60 — VOLTAGE BALANCE
- 62 — TIME DELAY
- 64 — GROUND PROTECTIVE
- 67 — AC DIRECTIONAL OVERCURRENT
- 68 — BLOCKING
- 74 — ALARM
- 76 — DC OVERCURRENT
- 78 — PHASE ANGLE
- 79 — AC RECLOSING
- 81 — FREQUENCY
- 83 — AUTOMATIC SELECTIVE CONTROL
- 85 — CARRIER OR PILOT-WIRE RECEIVER
- 86 — LOCKING-OUT
- 87 — DIFFERENTIAL
- 94 — TRIPPING OR TRIP-FREE

CONTROL SYMBOLS

A. CONTROL DEVICES

- BLOCKING DIODE (SEMICONDUCTOR RECTIFIER)
- PUSH BUTTON-MOMENTARY CONTACT NORMALLY CLOSED
- PUSH BUTTON-MOMENTARY CONTACT NORMALLY OPEN
- PUSH BUTTON LOCK-OUT
- PUSH BUTTON-MAINTAINED CONTACT
- 49 — CONTACTS OF OVERLOAD DEVICES
- OFF — SELECTOR SWITCH (TWO OR THREE POSITIONS)
- SOLENOID
- NORMALLY OPEN CONTACT (NO)
- NORMALLY CLOSED CONTACT (NC)
- MOTOR OPERATED VALVE POSITION LIMIT SWITCH
- MOTOR OPERATED VALVE TORQUE SWITCH

B. OPERATING COIL & COMPONENTS

OR

- CC — CONTACTOR COIL
- CR — CONTROL RELAY
- LR — LATCHING RELAY
- M — MOTOR STARTER OPERATING COIL
- MC — VALVE MOTOR STARTER CLOSE COIL
- MO — VALVE MOTOR STARTER OPEN COIL
- TDDO — TIME DELAY DROPOUT
- TDPU — TIME DELAY PICKUP
- 42-C — REVERSING MAGNETIC CONTACTOR CLOSING COIL
- 42-MX — MOTOR STARTER AUXILIARY RELAY
- 42-O — REVERSING MAGNETIC CONTACTOR OPENING COIL
- 52-CC — CIRCUIT BREAKER CLOSE COIL
- 52-TC — CIRCUIT BREAKER TRIP COIL

C. MISCELLANEOUS DEVICES

OR

- CS — CONTROL SWITCH
- dPS — DIFFERENTIAL PRESSURE SWITCH
- dPSH — DIFFERENTIAL PRESSURE SWITCH HIGH
- E/P — ELECTRO-PNEUMATIC CONVERTER
- EP — ELECTRO-PNEUMATIC SWITCH
- FM — FLOW METER
- FR — FIRE RELAY
- FS — FLOW SWITCH
- FT — FLOW TRANSMITTER
- H — HUMIDISTAT
- LMS — LIMIT SWITCH
- LS — LEVEL SWITCH
- LT — LEVEL TRANSMITTER
- PE — PNEUMATIC-ELECTRIC RELAY
- PIL — POTENTIAL INDICATING LIGHT
- PMS — PERMISSIVE SWITCH
- PS — PRESSURE SWITCH
- PT — PRESSURE TRANSMITTER
- RTD — RESISTANCE TEMPERATURE DETECTOR
- SD — SMOKE DETECTOR
- SS — SELECTOR SWITCH
- SV — SOLENOID VALVE CONTACT
- T — THERMOSTAT
- T/C — THERMOCOUPLE
- TE — TEMPERATURE ELEMENT
- THD — THERMODETECTOR IN MOTOR
- TS — TEMPERATURE SWITCH

- INDICATING LIGHT R=RED G=GREEN W=WHITE A=AMBER B=BLUE C=CLEAR
- INDICATING LIGHT WITH RESISTOR (LIGHT COLOR CODE AS ABOVE)
- BA — BELL ALARM CONTACT
- ANNUNCIATOR POINT
- COMPUTER INPUT OR OUTPUT SIGNAL
- THERMOCOUPLE
- RTD — RESISTANCE TEMPERATURE DETECTOR
- ---E--- ELECTRICAL INTERLOCK
- ---M--- MECHANICAL INTERLOCK
- ---K--- KEY INTERLOCK
- SINGLE CONDUCTOR WITH SHIELD-UNGROUNDED
- SINGLE CONDUCTOR WITH SHIELD-GROUNDED

ABBREVIATIONS
(USED WITH POWER TRANSFORMER SYMBOLS)

- AA — OPEN DRY TYPE
- GA — GAS FILLED DRY TYPE
- OA — OIL FILLED, SELF COOLED TYPE
- FA — OIL FILLED, FORCED AIR COOLED TYPE
- FOA — OIL FILLED, FORCED OIL, FORCED AIR COOLED TYPE

ABBREVIATIONS
(COMMON TO ALL ELECTRICAL DRAWINGS)

- AFF — ABOVE FINISHED FLOOR
- BD — BUS DUCT
- BTG — BOILER TURBINE GENERATOR
- CB — CONTROL BOARD
- CAC — CLOSED AFTER CLOSED
- CAT — CLOSE AFTER TRIP
- CND — CONDUIT
- CL — CURRENT LIMITING
- CP — CONTROL PANEL
- CS — CONTROL SWITCH
- CT — CURRENT TRANSFORMER
- DB — DIRECT BURIAL CABLE
- DT — DUST TIGHT
- DPST — DOUBLE POLE SINGLE THROW
- DPDT — DOUBLE POLE DOUBLE THROW
- ECB — EMERGENCY CONTROL BOARD
- EP — EXPLOSION PROOF
- EXC — EXCITER
- GEN — GENERATOR
- GND — GROUND
- ILC — INTEGRATED LOGIC CABINET
- INTLK — INTERLOCK
- I & C — INSTRUMENT AND CONTROL
- IPB — ISOLATED PHASE BUS DUCT
- IPC — INTEGRATED PROTECTION CABINET
- LB — LOAD BREAK
- LC — LOAD CENTER
- LCD — LOAD CENTER POWERED FROM DIESEL
- LDP — LIGHTING DISTRIBUTION PANEL
- LP — LIGHTING PANEL
- MBB — MECHANICAL BENCH BOARD
- MCB — MAIN CONTROL BOARD
- MCC — MOTOR CONTROL CENTER
- MOV — MOTOR OPERATED VALVE
- MR — MULTI-RATIO
- MTS — MANUAL TRANSFER SWITCH
- MU — MAKE UP
- MUX — MULTIPLEXER
- NEG — NEGATIVE
- NC — NORMALLY CLOSED
- NO — NORMALLY OPEN
- NSR — NUCLEAR SAFETY RELATED
- PB — PUSHBUTTON
- PDP — POWER DISTRIBUTION PANEL
- PELEC — PHOTOELECTRIC
- POS — POSITIVE
- PP — POWER PANEL
- PT — POTENTIAL TRANSFORMER
- REG — REGULATOR
- RL — REMOTE LOCATION
- RMS — RADIATION MONITORING SYSTEM
- RP — RELAY PANEL
- SHLD — SHIELD
- SLS — SELECTOR SWITCH
- SOLV — SOLENOID OPERATED VALVE
- SWGR — SWITCHGEAR
- SWYD — SWITCHYARD
- TDC — TIME DELAY CLOSE
- TDO — TIME DELAY OPEN
- UPS — UNINTERRUPTIBLE POWER SUPPLY
- VCB — VERTICAL CONTROL BOARD
- VP — VAPOR PROOF
- VT — VOLTAGE TRANSFORMER
- WP — WEATHERPROOF OR WEATHER PROTECTED
- WT — WATERTIGHT (SUBMERSIBLE)
- XFMR — TRANSFORMER

Figure 1.7-1 (Sheet 2 of 3) Legend for Electrical Power, Lighting, and Communication Drawings

Figure 1.7-1 (Sheet 3 of 3) Legend for Electrical Power, Lighting, and Communication Drawings

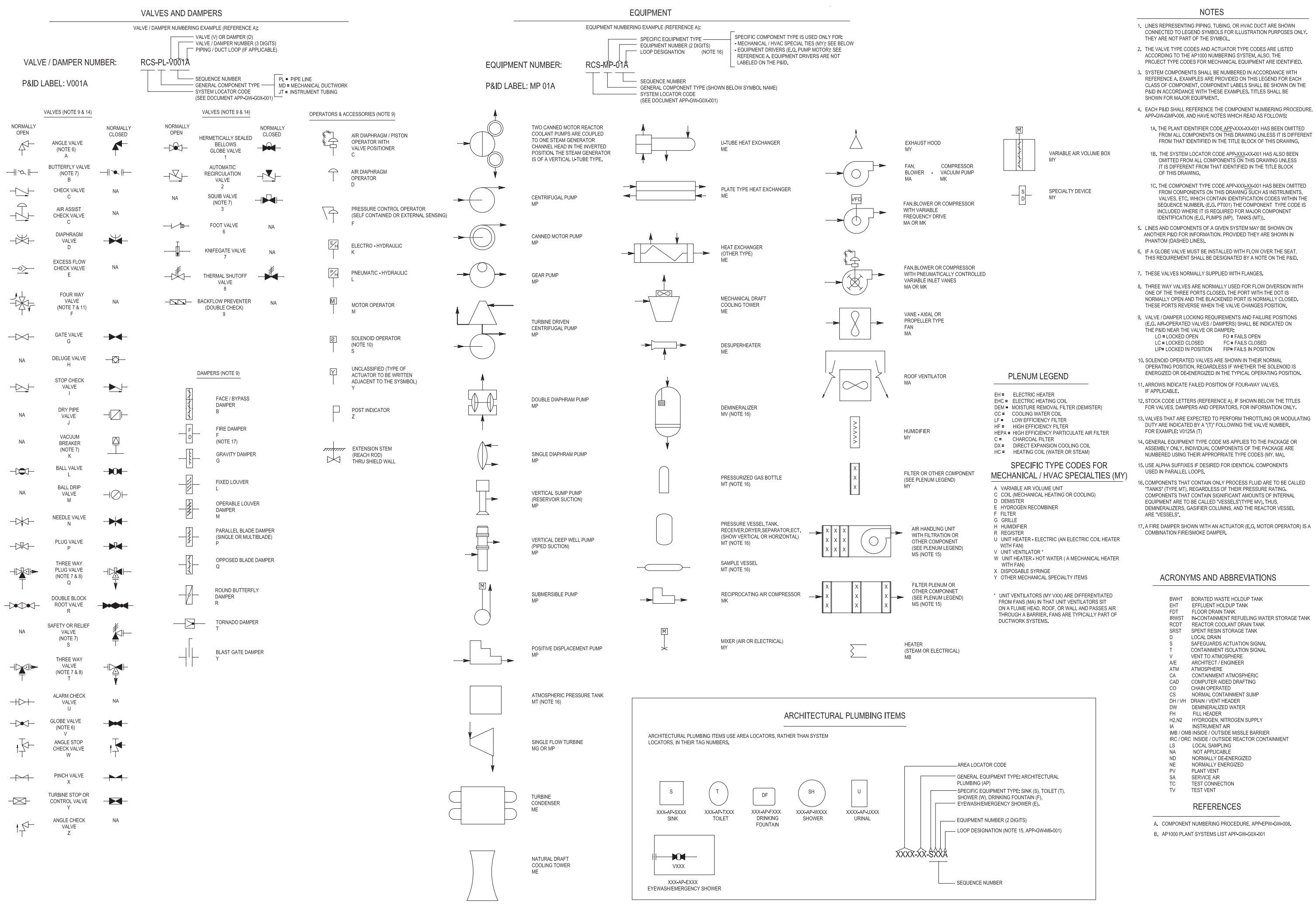
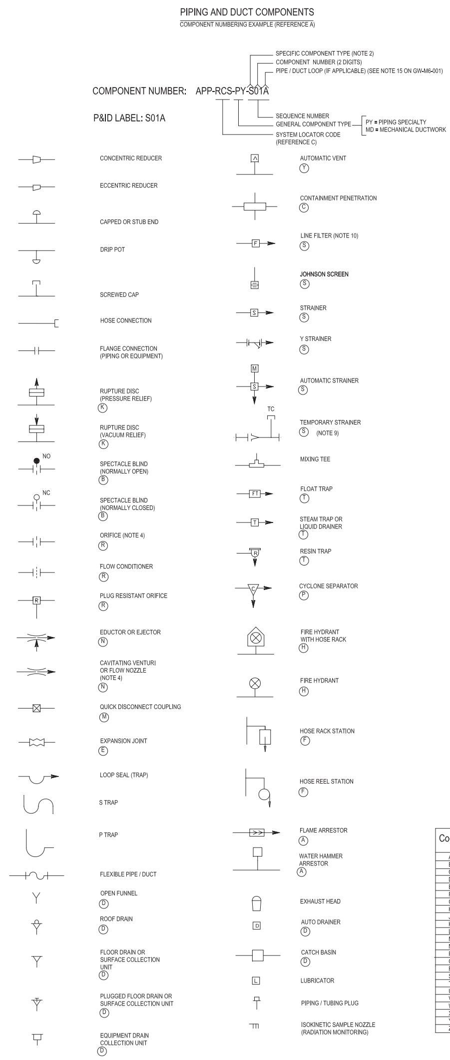
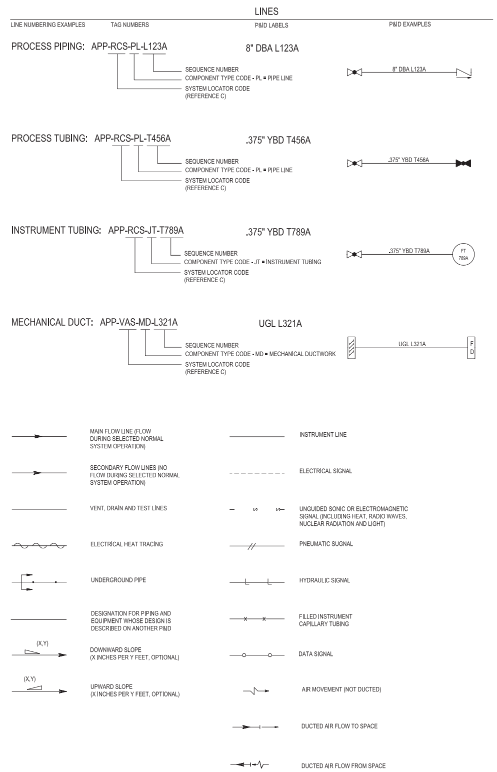


Figure 1.7-2 (Sheet 1 of 3) Piping and Instrumentation Diagram Legend

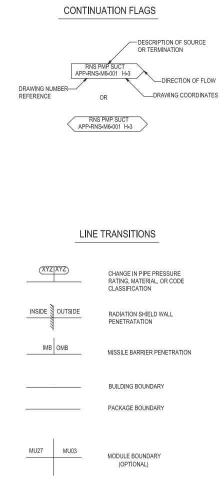
RN-13-039





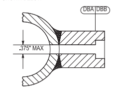
LINE AND VALVE / DAMPER SPECIFICATIONS

Code	Nominal Pressure Rating (First Letter)	Basic Pipe Line Material (Second Letter)	Design Code / Standard (Third Letter)
A	CL 2500 ASME B16.5	Alloy Steel, Standard	ASME Sec.III, Class 1, NRC Quality Group A
B	Special Rating - See Class Sheets	Austenitic Stainless Steel, Standard	ASME Sec.III, Class 2, NRC Quality Group B
C	CL 1500 ASME B16.5	Carbon Steel, Standard	ASME Sec.III, Class 3, NRC Quality Group C
D	CL 900/600 ASME B16.5	Ductile Iron - Cement Lined	ASME B31.1, (pipe)/SMACNA (duct) NRC Quality Group D
E	CL 600/900 ASME B16.5	Ductile Iron	ASME B31.1, Non Q
F	CL 400 ASME B16.5	Ferrous, Cast Iron	NFPA Code, Fire Protection QA
G	CL 300 ASME B16.5	Carbon Steel, Galvanized	NFPA Code, Non Q
H	CL 250 ASME B16.1	Copper - Nickel Alloy	
J	CL 150 ASME B16.5	PE3408/PE3608 HDPE	
K	CL 125 ASME B16.1	Copper, Brass or Bronze	
L	150 PSI, CWP	Lined Pipe	SMACNA Duct Const. Std. Non Q
M	125 PSI, CWP	Aluminum	
N	100 PSI, CWP	Alloy 20 (UNS N08020)	
P	80 PSI, CWP	Plastic & Non-Metallic	National Plumbing Code Non Q
Q	Not Used	Carbon Steel, Rubber Lined	
R	250 PSI, CWP	Concrete	ASME/ANSI AG-1 (ASME N509) Duct Standard
S	200 PSI, CWP	Alloy Steel, High Strength	
T	300 PSI Service (Victaulic Pressfit)	Austenitic Stainless Steel, High Strength	
U	0" - 2" Wg	Carbon Steel, Impact Tested	
V	2" - 6" Wg	PE4710 HDPE	
W	6" - 10" Wg		American Water Works Association (AWWA) Standards-Non Q
X	Over 10" Wg		
Y	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets
Z	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets	Special use as shown on Piping Class Sheets



NOTES:

1. FOR GENERAL NOTES SEE LEGEND DRAWING APP-GW-M6-001
2. IF APPLICABLE, CODE LETTERS FOR SPECIFIC COMPONENT TYPE (REFERENCE A) ARE SHOWN BELOW THE NAME OF THE SYMBOL.
3. IN GENERAL, PIPING AND DUCT COMPONENTS WILL NOT BE LABELED ON THE P&ID, UNLESS SPECIFIC IDENTIFICATION IS REQUIRED FOR MAINTENANCE, OPERATION OR TEST PURPOSES. COMPONENTS TO BE LABELLED ON THE P&ID ARE IDENTIFIED BY A CIRCLE AROUND THE CODE LETTER FOR SPECIFIC COMPONENT TYPE.
4. ORIFICES AND FLOW NOZZLES WHICH ARE USED FOR FLOW MEASUREMENT SHALL BE NUMBERED AND LABELED AS INSTRUMENTS (FLOW ELEMENTS) IN ACCORDANCE WITH DRAWING APP-GW-M6-003.
5. CONTINUATION FLAGS SHALL BE USED TO SHOW INTERFACES BETWEEN P&IDS. SYSTEM INTERFACES SHALL ALWAYS BE REPRESENTED BY CONTINUATION FLAGS.
6. ANNOTATIONS ARE GRAPHICS ONLY. NO ELECTRONIC DATA IS ASSOCIATED WITH ANNOTATIONS.
7. IN SAFETY CLASS I PIPING AND EQUIPMENT, A FLOW RESTRICTION IS REQUIRED IN 1 (ONE) INCH PIPING TO ALLOW TRANSITION FROM SAFETY CLASS 1 TO SAFETY CLASS 2. TYPICAL FOR RESTRICTION SHOWN BELOW:



8. INSTRUMENT TUBING (COMPONENT TYPE JT) IS BETWEEN THE INSTRUMENT ROOT VALVE AND THE TRANSMITTER. THE INSTRUMENT ROOT VALVE AND PIPING TO THE PROCESS LINE ARE COMPONENT TYPE CODE "PL".
9. TEMPORARY STRAINER TEST CONNECTION IS LOCATED ON PUMP SIDE OF THE STRAINER.
10. FOR FILTER VESSEL USE PRESSURE VESSEL SYMBOL.

REFERENCES:

A. AP1000 COMPONENT NUMBERING PROCEDURE. APP-GW-GMP-006.
B. P& ID LEGEND DRAWING, APP-GW-M6-001.
C. AP1000 PLANT SYSTEM LIST, APP-GW-G0X-001.

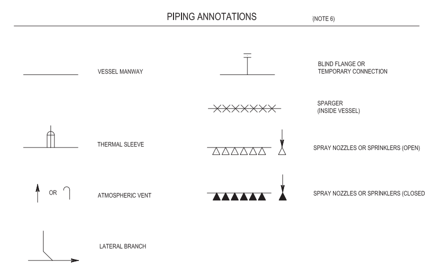
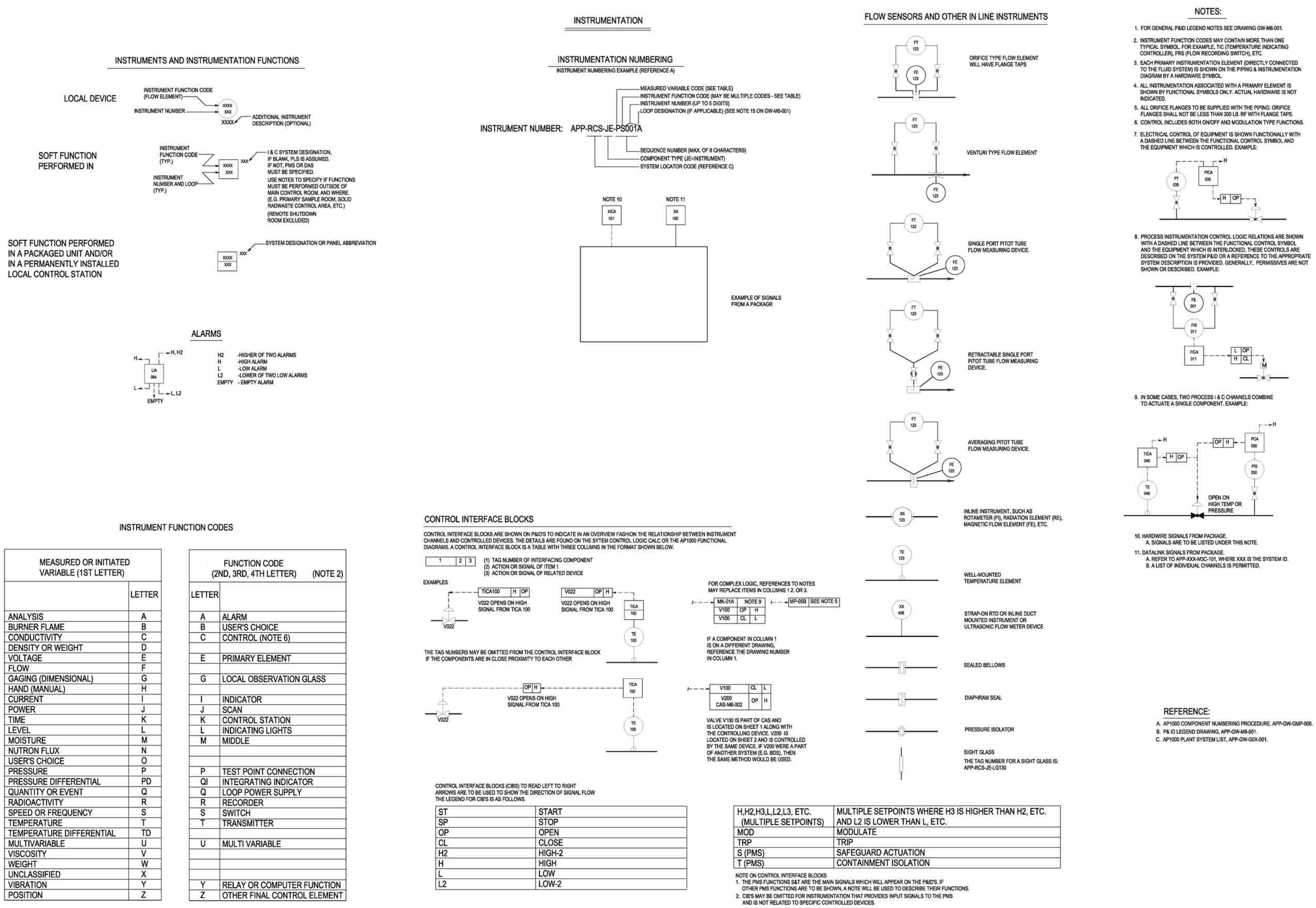


Figure 1.7-2 (Sheet 2 of 3) Piping and Instrumentation Diagram Legend

RN-13-081



INSTRUMENT FUNCTION CODES

MEASURED OR INITIATED VARIABLE (1ST LETTER)	LETTER	LETTER	FUNCTION CODE (2ND, 3RD, 4TH LETTER) (NOTE 2)
ANALYSIS	A	A	ALARM
BURNER FLAME	B	B	USER'S CHOICE
CONDUCTIVITY	C	C	CONTROL (NOTE 6)
DENSITY OR WEIGHT	D		
VOLTAGE	E	E	PRIMARY ELEMENT
FLOW	F		
GAGING (DIMENSIONAL)	G	G	LOCAL OBSERVATION GLASS
HAND (MANUAL)	H		
CURRENT	I	I	INDICATOR
POWER	J	J	SCAN
TIME	K	K	CONTROL STATION
LEVEL	L	L	INDICATING LIGHTS
MOISTURE	M	M	MIDDLE
NUTRON FLUX	N		
USER'S CHOICE	O		
PRESSURE	P	P	TEST POINT CONNECTION
PRESSURE DIFFERENTIAL	PD	QI	INTEGRATING INDICATOR
QUANTITY OR EVENT	Q	Q	LOOP POWER SUPPLY
RADIOACTIVITY	R	R	RECORDER
SPEED OR FREQUENCY	S	S	SWITCH
TEMPERATURE	T	T	TRANSMITTER
TEMPERATURE DIFFERENTIAL	TD		
MULTIVARIABLE	U	U	MULTI VARIABLE
VISCOSITY	V		
WEIGHT	W		
UNCLASSIFIED	X		
VIBRATION	Y	Y	RELAY OR COMPUTER FUNCTION
POSITION	Z	Z	OTHER FINAL CONTROL ELEMENT

ST	START
SP	STOP
OP	OPEN
CL	CLOSE
H2	HIGH-2
H	HIGH
L	LOW
L2	LOW-2

H,H2,H3,L,L2,L3, ETC. (MULTIPLE SETPOINTS)	MULTIPLE SETPOINTS WHERE H3 IS HIGHER THAN H2, ETC. AND L2 IS LOWER THAN L, ETC.
MOD	MODULATE
TRP	TRIP
S (PMS)	SAFEGUARD ACTUATION
T (PMS)	CONTAINMENT ISOLATION

Figure 1.7-2 (Sheet 3 of 3) Piping and Instrumentation Diagram Legend

RN-12-049

1.8 Interfaces for Standard Design

This section identifies the AP1000 standard plant scope, interfaces related to design certification between the AP1000 plant design and the Combined License applicant, and the site-specific items to be included in an application for a Combined License. It is submitted to satisfy the requirements of 10 CFR 52.47(a)(1)(vii).

The AP1000 is a plant incorporating six buildings, the equipment in them, and the associated yard structures and tankage. This includes the nuclear island (consisting of the containment/shield building and the auxiliary building), the annex building and associated equipment, the diesel/generator building and associated equipment, the turbine generator building, the turbine/generator equipment, and the radwaste facilities. The physical boundary of the portion of the AP1000 included in this application is shown on the site plan, Figure 1.2-2. It includes arrangement and placement of structures within the indicated boundary. Additionally, the red zone delay barrier necessary for security is included, but the boundary fence and vehicle barrier are not included since they are site-specific. As a result, no interfaces need to be identified between or among the portions of the plant within the boundary. They are addressed in their appropriate section of this DCD. There are no safety-related interfaces to site-specific elements of the plant outside the scope of this certification application. The following site-specific elements are outside the scope of the AP1000 standard plant:

(1) The portions of the circulating water system and its heat sink outside the AP1000 buildings, the portions of the service water blowdown lines outside the AP1000 buildings, which transfer fluid to the circulating water cooling tower basins as well as the specific design details of the main condenser. A conceptual design is presented, delineated by Double Brackets ([[ ]]), in Subsection 10.4.5, based upon a cooling tower approach. RN-13-006

(2) The offsite power transmission system outside the low voltage terminals of the main and reserve transformers. Location and design of the main switchyard area and the equipment located therein, as well as design details such as voltage level for the main step-up transformers. A conceptual design of this system is included, delineated by Double Brackets ([[ ]]), in Section 8.2 for reference.

(3) Raw water source and treatment outside the turbine building. An interface specification of amount and water chemistry limits is provided.

(4) Sanitary and other drain systems outside the buildings identified above. This DCD is based upon the COL applicant providing adequate overall site drain collection and processing systems

(5) Communications systems and equipment outside the buildings identified above. This DCD is based upon the COL applicant providing adequate external communications.

(6) Location and design of administrative and training structures.

(7) Landscaping features.

(8) Size and location of the waste water retention basins and the associated plant outfall piping.

A more detailed listing of the systems included in the standard AP1000 plant is included in Section 3.2.

There are a number of information interfaces between the AP1000 design and other portions of a completely licensed facility which must be addressed by parties that reference the AP1000 design. These interfaces are identified in Table 1.8-1 in the order they are presented in this DCD.

The safety-related interface requirements in Table 1.8-1 have been selected based on a review of interfaces between the AP1000 plant design and other Combined License applicant or site-specific items. Satisfying the referenced information for each of the interfaces listed will provide confidence that systems, structures and components within the AP1000 can perform their safety functions. The specific details of the interface parameters are identified in the DCD sections identified in Table 1.8-1. The interface specifications have been selected to suit a wide range of potential sites. Values identified by a Combined License applicant to be outside the range of acceptable parameters may be demonstrated to be acceptable. Such cases will be documented in the appropriate sections of the specific Combined License application.

The classification of interface types is based on the sources of interfaces listed in Appendix A of Regulatory Guide 1.70. The first four types below are directly related to the four sources of interfaces. They have been redefined slightly to reflect the fact that AP1000 is an essentially complete plant design. The classification of interface types is as follows:

- **Requirement of AP1000** – Requirements for operation of the AP1000 design that must be satisfied by the matching portion of the site, utility or Combined License applicant administration.

- **AP1000 Interface** – Interface condition used for AP1000 design which must be more precisely defined during the coordination effort between the AP1000 design team and the Combined License applicant.

- **Site Interface** – Site-related interface data upon which the AP1000 design is based.

- **Pertinent Criteria** – Criteria pertinent to the AP1000 design that may be useful for the design and staff review of the matching systems, components and structures.

- **Not an Interface** – Interface items identified in Appendix A of Regulatory Guide 1.70 which are wholly within the boundaries of the AP1000 plant. As a result, the "Matching Interface Item" in Table 1.8-1 is identified as N/A (not applicable).

- **Non-Nuclear Safety (NNS)** – Interface items identified in Appendix A of Regulatory Guide 1.70 which are non-nuclear safety-related because of the design features of AP1000.

Note that all plant interfaces listed in Appendix A of Regulatory Guide 1.70 have been listed in Table 1.8-1. As noted above and in Table 1.8-1, a number of these interfaces do not apply to the AP1000 plant as described in this DCD. In some cases, the interface listed in Appendix A of Regulatory Guide 1.70 is totally within the AP1000 plant and therefore not an interface. Other interfaces from Appendix A of Regulatory Guide 1.70 are identified as non-nuclear safety. The classification of systems, structures and components is described in Section 3.2. Only safety-related interfaces are detailed in Table 1.8-1. An example of an "NNS" (non-nuclear safety) type of interface is any of those associated with site service water. AP1000 does not rely on site service water as a safety grade ultimate heat sink. Neither the cooling tower nor the diesel-generator building is safety-related in AP1000. As such, there are no safety-related interfaces for these features.

Interfaces are listed in the order discussed in the DCD. General interfaces are listed as they relate to a particular section of this DCD. No specific system-by-system interface listings are required due to the complete nature of the AP1000 plant design. All safety-related systems are contained within the AP1000 plant design. The listing includes identification of the interface classification and the matching interface item to be specified by the Combined License applicant. In addition, the section of this DCD which addresses the listed interface is identified. To satisfy the requirements of 10 CFR 52.47(a)(1)(ix), representative conceptual designs are included in this DCD for those portions of the plant for which Westinghouse does not seek certification to aid the NRC staff in its review of the DCD and the probabilistic risk assessment to be submitted in support of the application, and to permit assessment of the adequacy of interface requirements.

Combined License Information

Combined License applicants referencing the AP1000 certified design will be required to provide site-specific information, verification that interface criteria are satisfied, information related to operating procedures, and other information required to support the AP1000 Design Certification. The description of information to be provided by the Combined License applicant is found in the DCD sections applicable to the specific information. Table 1.8-2 is a listing of the Combined License Information Items and the DCD location of the description of the information. In some cases, the activity required by a COL Information Item requires as-built information or other conditions that are not available when the COL application is submitted. These items are noted in the applicable DCD sections and Table 1.8-2. These activities are completed prior to fuel load.

{HISTORICAL - Departures from the referenced DCD are summarized in Table 1.8-201. Table 1.8-201 lists each departure and the section or subsection impacted.- HISTORICAL} RN-13-005

Table 1.8-2 presents Combined License Information for the AP1000. Items requiring COL Applicant or COL Holder action are presented in Table 1.8-202. Section(s) addressing these COL items are tabulated in this table. COL Holder items listed in Table 1.8-202 are regulatory commitments of the COL Holder and these actions will be completed as specified in the appropriate section. Completion of these COL Holder items is the subject of a Combined License Condition as presented in a separate document submitted as part of this COL Application.

Table 1.8-1 presents interface items for the AP1000.

Table 1.8-1 (Sheet 1 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection
1.1	Post accident Radio-Iodine sampling capability per NUREG 0737	Requirement of AP1000	Combined License applicant program	(b)
2.1	Envelope of AP1000 plant site related parameters	Site Interface	Site specific parameters	2.0, Table 2.0-201
2.2	External missiles from man-made hazards and accidents	Site Interface	Site specific parameters	2.2, 2.2.3.1, 3.5
2.3	Maximum loads from man-made hazards and accidents	Site Interface	Site specific parameters	2.2, 2.2.3.1
2.4	Limiting meteorological parameters (χ/Q) for design basis accidents and for routine releases and other extreme meteorological conditions for the design of systems and components exposed to the environment.	Site Interface	Site specific parameters	2.3, Table 2.0-201
2.5	Tornado and operating basis wind loadings	Site Interface	Site specific parameters	2.3, Table 2.0-201
2.6	External missiles generated by natural phenomena	Site Interface	Site specific parameters	2.3, Table 2.0-201
2.7	Snow, ice and rain loads	Site Interface	Site specific parameters	2.3, 2.3.1.3.4, 2.3.1.3.5
2.8	Ambient air temperatures	Site Interface	Site specific parameters	2.3, Table 2.0-201
2.9	Onsite meteorological measurement program	Requirement of AP1000	Combined License applicant program	2.3.3
2.10	Flood and ground water elevations	Site Interface	Site specific parameters	2.4, Table 2.0-201
2.11	Hydrostatic loads on systems, components and structures	Site Interface	Site specific parameters	2.4, Table 2.0-201
2.12	Seismic parameters peak ground acceleration response spectra shear wave velocity	Site Interface	Site specific parameters	 2.5, 2.5, 2.5, Table 2.0-201
2.13	Required bearing capacity of foundation materials	Site Interface	Site specific parameters	2.5, Table 2.0-201
3.1	Deleted			N/A
3.2	Operating procedures to minimize water hammer	Requirement of AP1000	Combined License applicant procedure	3.6, 10, 10.3.2.2.1, 10.4.7.2.1

Table 1.8-1 (Sheet 2 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection
3.3	Site seismic sensor location and "trigger" value	Requirement of AP1000	Onsite implementation	3.7.4, 3.7.4.2.1, 3.7.4.2
3.4	Depth of overburden	Requirement of AP1000	Onsite implementation	3.8, 3.8.5.1, 2.5.4
3.5	Depth of embedment	Requirement of AP1000	Onsite implementation	3.8, 3.8.5.1, 2.5.4
3.6	Specific depth of waterproofing	Requirement of AP1000	Onsite implementation	3.8.5, 2.5.4
3.7	Foundation Settlement Monitoring	Requirement of AP1000	Combined License applicant coordination	3.8.5, 2.5.4.13
3.8	Lateral earth pressure loads	Not an Interface	N/A	3, N/A
3.9	Preoperational piping vibration test parameters	Not an Interface	N/A	3, N/A
3.10	Inservice Inspection requirements and locations	Requirement of AP1000	Combined License applicant program	3.9.6, 5.2.4, 6.6
3.11	Maintenance of preservice and reference test data for inservice testing of pumps and valves	Requirement of AP1000	Combined License applicant program	3.9.6, 5.2.4, 6.6
3.12	Earthquake response procedures	Requirement of AP1000	Combined License applicant program	3.7.4, 3.7.4.4
5.1	Steam Generator Tube Surveillance Requirements	Requirement of AP1000	Combined License applicant program	5.4.2, 5.4.2.5
6.1	Inservice Inspection requirements for the containment	Requirement of AP1000	Combined License applicant program	6.2.1, 6.2, 6.6
6.2	Offsite environmental conditions assumed for Main Control Room and control support area habitability design	AP1000 Interface	Site specific parameter	6.4, 2.2.3
7.1	Listing of all design criteria applied to the design of the I&C systems	Not an Interface	N/A	7, N/A
7.2	Power required for site service water instrumentation	NNS and Not an Interface	N/A	7, N/A

Table 1.8-1 (Sheet 3 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection
7.3	Other provisions for site service water instrumentation	NNS and Not an Interface	N/A	7, N/A
7.4	Post-accident monitoring system	NNS	Combined License applicant coordination	7.5.5
8.1	Listing of design criteria applied to the design of the offsite power system	NNS	Combined License applicant coordination	8, 8.1.4.3
8.2	Offsite ac requirements Steady-state load Inrush kVA for motors Nominal voltage Allowable voltage regulation Nominal frequency Allowable frequency fluctuation Maximum frequency decay rate Limiting under frequency value for RCP	NNS	Combined License applicant coordination	8, 8.2.2
8.3	Offsite transmission system analysis: Loss of AP1000 or largest unit Voltage operating range Transient stability must be maintained and the RCP bus voltage must remain above the voltage required to maintain the flow assumed in Chapter 15 analyses for a minimum of three (3) seconds following a turbine trip. The protective devices controlling the switchyard breakers are set with consideration given to preserving the plant grid connection following a turbine trip.	NNS	Combined License applicant analysis	8.2, 8.2.2, 8.2.1.2.2
8.4	Listing of design criteria applied to the design of onsite ac power systems	NNS and Not an Interface	N/A	8, N/A
8.5	Onsite ac requirements	NNS and Not an Interface	N/A	8, N/A
8.6	Diesel generator room coordination	NNS and Not an Interface	N/A	8, N/A
8.7	Listing of design criteria applied to the design of onsite dc power systems	Not an Interface	N/A	8, N/A
8.8	Provisions of dc power systems to accommodate the site service water system	NNS and Not an Interface	N/A	8, N/A
9.1	Listing of design criteria applied to the design of portions of the site service water within AP1000	NNS and Not an Interface	N/A	9, N/A

Table 1.8-1 (Sheet 4 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection
9.2	Integrated heat load to site service water system	NNS and Not an Interface	N/A	9, N/A
9.3	Plant cooling water systems parameters	NNS and Not an Interface	N/A	9, N/A
9.4	Plant makeup water quality limits	NNS	Site specific parameter	9, 9.2.11
9.5	Requirements for location and arrangement of raw and sanitary water systems	NNS	Site implementation	9, 9.2.5 9.2.6 9.2.11
9.6	Ventilation requirements for diesel-generator room	NNS and Not an Interface	N/A	9, N/A
9.7	Requirements to satisfy fire protection program	AP1000 Interface	Combined License applicant program	9.5.1
9.8	Requirements for location and size of waste water retention basins and associated plant outfall	NNS	Site Implementation	9
11.1	Expected release rates of radioactive material from the Liquid Waste System including: Location of release points Effluent temperature Effluent flow rate Size and shape of flow orifices	Site Interface	Site specific parameters	11.2, 9.2.11.4
11.2	Expected release rates of radioactive materials from the Gaseous Waste System including: Location of release points Height above grade Height relative to adjacent buildings Effluent temperature Effluent flow rate Effluent velocity Size and shape of flow orifices	Site Interface	Site specific parameters	11.3
11.3	Expected release rates of radioactive material from the Solid Waste System including: Location of release points Material types Material qualities Size and shape of material containers	Site Interface	Site specific parameters	11.4, 11.4.6
11.4	Requirements for offsite sampling and monitoring of effluent concentrations	AP1000 Interface	Combined License applicant program	11.5, 11.5.3 11.5.7

**Table 1.8-1 (Sheet 5 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant**

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection
12.1	Identification of miscellaneous radioactive sources	AP1000 Interface	Combined License applicant program	12.2, 12.2.1.1.10
13.1	Features that may affect plans for coping with emergencies as specified in 10 CFR 50, Appendix O	AP1000 Interface	Combined License applicant program	13.3
13.2	Physical Security Plan consistent with AP1000 plant	AP1000 Interface	Combined License applicant program	13.6
14.1	Identification of special features to be considered in development of the initial test program	Requirement of AP1000	Combined License applicant program	14
14.2	Maintenance of preoperational test data and inservice inspection baseline data	AP1000 Interface	Combined License applicant program	14
16.1	Administrative requirements associated with reliability information maintenance	AP1000 Interface	Combined License applicant program	16
16.2	Administrative requirements associated with the Technical Specifications	Requirement of AP1000	Combined License applicant implementation	16
16.3	Site and operator related information associated with the Reliability Assurance Program (D-RAP)	Requirement of AP1000	Combined License applicant program	16.2
18.1	Operating staff consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.6
18.2	Operator training consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.8, 18.10
18.3	Operating Procedures consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	18.8, 18.9, 18.10
18.4	Final coordination and integration of human system interface areas within a specific AP1000 consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	(b)

Table 1.8-1 (Sheet 6 of 6)
Summary of AP1000 Plant Interfaces With Remainder of Plant

Item No.	Interface	Interface Type	Matching Interface Item	Section or Subsection
18.5	Final coordination and integration of Combined License applicant facilities with those of a specific AP1000 consistent with Human Factors evaluations	AP1000 Interface	Combined License applicant program	(b)

b) Westinghouse has determined that this item has been fully addressed by the DCD. Thus, the item is not addressed by the COLA.

Table 1.8-2 (Sheet 1 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
1.1-1	Construction and Startup Schedule	1.1.7	N/A	Yes	–
1.9-1	Regulatory Guide Conformance	1.9.1.5	APP-GW-GLN-129	Yes	–
2.1-1	Geography and Demography	2.1.4	N/A	Yes	–
2.2-1	Identification of Site-specific Potential Hazards	2.2.4	N/A	Yes	–
2.3-1	Regional Climatology	2.3.6.1	N/A	Yes	–
2.3-2	Local Meteorology	2.3.6.2	N/A	Yes	–
2.3-3	Onsite Meteorological Measurements Program	2.3.6.3	N/A	Yes	–
2.3-4	Short-Term Diffusion Estimates	2.3.6.4	N/A	Yes	–
2.3-5	Long-Term Diffusion Estimates	2.3.6.5	N/A	Yes	–
2.4-1	Hydrological Description	2.4.1.1	N/A	Yes	–
2.4-2	Floods	2.4.1.2	N/A	Yes	–
2.4-3	Cooling Water Supply	2.4.1.3	N/A	Yes	–
2.4-4	Groundwater	2.4.1.4	N/A	Yes	–
2.4-5	Accidental Release of Liquid Effluents into Ground and Surface Water	2.4.1.5	N/A	Yes	–
2.4-6	Flood Protection Emergency Operation Procedures	2.4.1.6	N/A	Yes	–

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Table 1.8-2 (Sheet 2 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
2.5-1	Basic Geologic and Seismic Information	2.5.1	N/A	Yes	–
2.5-2	Site Seismic and Tectonic Characteristics Information	2.5.2	N/A	Yes	–
2.5-3	Geoscience Parameters	2.5.2.3	N/A	Yes	–
2.5-4	Surface Faulting	2.5.3	N/A	Yes	–
2.5-5	Site and Structures	2.5.4.6.1	N/A	Yes	–
2.5-6	Properties of Underlying Materials	2.5.4.6.2	N/A	Yes	–
2.5-7	Excavation and Backfill	2.5.4.6.3	N/A	Yes	–
2.5-8	Ground Water Conditions	2.5.4.6.4	N/A	Yes	–
2.5-9	Liquefaction Potential	2.5.4.6.5	N/A	Yes	–
2.5-10	Bearing Capacity	2.5.4.6.6	N/A	Yes	–
2.5-11	Earth Pressures	2.5.4.6.7	N/A	Yes	–
2.5-12	Static and Dynamic Stability of Facilities	2.5.4.6.9	N/A	Yes	–
2.5-13	Subsurface Instrumentation	2.5.4.6.10	N/A	Yes	–
2.5-14	Stability of Slopes	2.5.5	N/A	Yes	–
2.5-15	Embankments and Dams	2.5.6	N/A	Yes	–
2.5-16	Settlement of Nuclear Island	2.5.4.6.11	N/A	Yes	–
2.5-17	Waterproofing System	2.5.4.6.12	N/A	Yes	–

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Table 1.8-2 (Sheet 3 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
3.3-1	Wind and Tornado Site Interface Criteria	3.3.3	APP-GW-GLR-020	Yes	–
3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3	N/A	Yes	–
3.5-1	External Missile Protection Requirements	3.5.4	APP-GW-GLR-020	Yes	–
3.6-1	Pipe Break Hazards Analysis	3.6.4.1	APP-GW-GLR-021 APP-GW-GLR-074	No	Yes
3.6-2	Leak-Before-Break Evaluation of as-Designed Piping	3.6.4.2	APP-GW-GLR-022	No	No
3.6-3	Deleted Leak-Before-Break Evaluation of as-Built Piping	Deleted	APP-GW-GLR-021	N/A	N/A
3.6-4	Primary System Inspection Program for Leak-Before-Break Piping	3.6.4.4	N/A	Yes	–
3.7-1	Seismic Analysis of Dams	3.7.5.1	N/A	Yes	–
3.7-2	Post-Earthquake Procedures	3.7.5.2	N/A	Yes	–
3.7-3	Seismic Interaction Review	3.7.5.3	APP-GW-GLR-021	No	Yes
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4	APP-GW-GLR-021 APP-GW-S2R-010	No	Yes
3.7-5	Location of Free-Field Acceleration Sensor	3.7.5.5	N/A	Yes	–
3.8-1	Containment Vessel Design Adjacent to Large Penetrations	3.8.6.1	APP-GW-GLR-005	No	No

Table 1.8-2 (Sheet 4 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
3.8-2	Deleted Passive Containment Cooling System Water Storage Tank Examination	Deleted	APP-GW-GLR-021	N/A	N/A
3.8-3	Deleted As-Built Summary Report	Deleted	APP-GW-GLR-021	N/A	N/A
3.8-4	Deleted In-Service Inspection of Containment Vessel	Deleted	APP-GW-GLR-021	N/A	N/A
3.8-5	Structures Inspection Program	3.8.6.5	N/A	Yes	–
3.8-6	Construction Procedures Program	3.8.6.6	N/A	No	Yes
3.9-1	Reactor Internal Vibration Response	3.9.8.1	WCAP-16687-P	No	No
3.9-2	Design Specification and Reports	3.9.8.2	APP-GW-GLR-021	No	Yes
3.9-3	Snubber Operability Testing	3.9.8.3	N/A	Yes	–
3.9-4	Valve Inservice Testing	3.9.8.4	APP-GW-GLN-020	Yes	–
3.9-5	Surge Line Thermal Monitoring	3.9.8.5	N/A	Yes	–
3.9-6	Piping Benchmark Program	3.9.8.6	APP-GW-GLR-006	No	No
3.9-7	As-Designed Piping Analysis	3.9.8.7	Various	Yes	Yes
3.10-1	Deleted Experience-Based Qualification	Deleted		N/A	N/A
3.11-1	Equipment Qualification File	3.11.5	APP-GW-GLN-110	No	Yes
4.2-1	Changes to Reference Reactor Design	4.2.5	APP-GW-GLR-059	No	No
4.3-1	Changes to Reference Reactor Design	4.3.4	APP-GW-GLR-059 APP-GW-GLR-119	No	No

Table 1.8-2 (Sheet 5 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
4.4-1	Changes to Reference Reactor Design	4.4.7.1	APP-GW-GLR-059	No	No
4.4-2	Confirm Assumptions for Safety Analyses DNBR Limits	4.4.7.2	N/A	–	Yes
5.2-1	ASME Code and Addenda	5.2.6.1	N/A	Yes	–
5.2-2	Plant Specific Inspection Program	5.2.6.2	N/A	Yes	–
5.2-3	Response to Unidentified Reactor Coolant System Leakage Inside Containment	5.2.6.3	N/A	Yes	–
5.3-1	Reactor Vessel Pressure – Temperature Limit Curves	5.3.6.1	APP-GW-GLR-021	No	Yes
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.6.2	N/A	Yes	–
5.3-3	Surveillance Capsule Lead Factor and Azimuthal Location Confirmation	5.3.6.3	APP-GW-GLR-023	No	No
5.3-4	Reactor Vessel Materials Properties Verification	5.3.6.4.1	N/A	No	Yes
5.3-5	Reactor Vessel Insulation	5.3.6.5	APP-GW-GLR-060	No	No
5.3-6	Analysis of Reactor Vessel Insulation and Support Structure	5.3.6.4.2	APP-GW-GLR-060	No	No
5.4-1	Steam Generator Tube Integrity	5.4.15	N/A	Yes	–
5.3-7	Quickloc Weld Buildup ISI	5.3.6.6	N/A	Yes	–
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.3.1	N/A	Yes	–

Table 1.8-2 (Sheet 6 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
6.1-2	Coating Program	6.1.3.2	N/A	Yes	–
6.2-1	Containment Leak Rate Testing	6.2.6	N/A	Yes	–
6.3-1	Containment Cleanliness Program	6.3.8.1	N/A	Yes	–
6.3-2	Verification of Containment Resident Particulate Debris Characteristics	6.3.8.2	APP-GW-GLR-079	No	No
6.4-1	Local Hazardous Gas Services and Monitoring	6.4.7	N/A	Yes	–
6.4-2	Procedures for Training for Control Room Habitability	6.4.7	N/A	Yes	–
6.4-3	Main Control Room Inleakage Test Frequency	6.4.7	APP-GW-GLR-007	No	No
6.6-1	Inspection Programs	6.6.9.1	N/A	Yes	–
6.6-2	Construction Activities	6.6.9.2	N/A	Yes	–
7.1-1	Setpoint Calculations for Protective Functions	7.1.6.1	WCAP-16361-P	Yes	Yes
7.1-2	Resolution of Generic Open Items and Plant-Specific Action Items	7.1.6.2	APP-GW-GLR-017	No	No
7.2-1	FMEA for Protection System	7.2.3	WCAP-16438-P WCAP-16592-P	No	No
7.5-1	Post-Accident Monitoring	7.5.5	N/A	Yes	No
8.2-1	Offsite Electrical Power	8.2.5	N/A	Yes	–
8.2-2	Technical Interfaces	8.2.5	N/A	Yes	–

**Table 1.8-2 (Sheet 7 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items**

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
8.3-1	Grounding and Lightning Protection	8.3.3	N/A	Yes	–
8.3-2	Onsite Electrical Power Plant Procedures	8.3.3	N/A	Yes	–
9.1-1	New Fuel Rack	9.1.6.1	APP-GW-GLR-026	No	No
9.1-2	Criticality Analysis for New Fuel Rack	9.1.6.2	APP-GW-GLR-030	No	No
9.1-3	Spent Fuel Racks	9.1.6.3	APP-GW-GLR-033	No	No
9.1-4	Criticality Analysis for Spent Fuel Racks	9.1.6.4	APP-GW-GLR-029P	No	No
9.1-5	Inservice Inspection Program of Cranes	9.1.6.5	N/A	Yes	–
9.1-6	Radiation Monitor	9.1.6.6	N/A	Yes	–
9.1-7	Metamic Monitoring Program	9.1.6.7	N/A	No	Yes
9.2-1	Potable Water	9.2.11.1	N/A	Yes	–
9.2-2	Waste Water Retention Basins	9.2.11.2	N/A	Yes	–
9.3-1	Air Systems (NUREG-0933 Issue 43)	9.3.7	N/A	Yes	–
9.4-1	Ventilation Systems Operations	9.4.12	N/A	Yes	–
9.5-1	Qualification Requirements for Fire Protection Program	9.5.1.8.1	N/A	Yes	–
9.5-2	Fire Protection Analysis Information	9.5.1.8.2	N/A	Yes	–
9.5-3	Regulatory Conformance	9.5.1.8.3	N/A	Yes	–

Table 1.8-2 (Sheet 8 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
9.5-4	NFPA Exceptions	9.5.1.8.4	N/A	Yes	–
9.5-5	Operator Actions Minimizing Spurious ADS Actuation	9.5.1.8.5	APP-GW-GLR-027	No	No
9.5-6	Verification of Field Installed Fire Barriers	9.5.1.8.6	N/A	No	Yes
9.5-7	Fire Resistance Test Data	9.5.1.8.8	APP-GW-GLR-019	No	No
9.5-8	Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance	9.5.1.8.7	N/A	Yes	–
9.5-9	Offsite Interfaces	9.5.2.5.1	N/A	Yes	–
9.5-10	Emergency Offsite Communications	9.5.2.5.2	N/A	Yes	–
9.5-11	Security Communications	9.5.2.5.3	N/A	Yes	–
9.5-12	Cathodic Protection	9.5.4.7.1	APP-GW-GLR-120	No	No
9.5-13	Fuel Degradation Protection	9.5.4.7.2	APP-GW-GLR-120	Yes	–
10.1-1	Erosion-Corrosion Monitoring	10.1.3	N/A	No	Yes
10.2-1	Turbine Maintenance and Inspection	10.2.6	APP-GW-GLN-018	No	Yes
10.4-1	Circulating Water Supply	10.4.12.1	N/A	Yes	–
10.4-2	Condensate, Feedwater and Auxiliary Steam System Chemistry Control	10.4.12.2	N/A	Yes	–
10.4-3	Potable Water	10.4.12.3	N/A	Yes	–

Table 1.8-2 (Sheet 9 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
11.2-1	Liquid Radwaste Processing by Mobile Equipment	11.2.5.1	N/A	Yes	–
11.2-2	Cost Benefit Analysis of Population Doses	11.2.5.2	N/A	Yes	–
11.2-3	Identification of Ion Exchange and Adsorbent Media	11.2.5.3	APP-GW-GLR-008	No	No
11.2-4	Dilution and Control of Boric Acid Discharge	11.2.5.4	APP-GW-GLR-014	No	No
11.3-1	Cost Benefit Analysis of Population Doses	11.3.5.1	N/A	Yes	–
11.3-2	Identification of Adsorbent Media	11.3.5.2	APP-GW-GLR-008	No	No
11.4-1	Solid Waste Management System Process Control Program	11.4.6	N/A	Yes	–
11.5-1	Plant Offsite Dose Calculation Manual (ODCM)	11.5.8	N/A	Yes	–
11.5-2	Effluent Monitoring and Sampling	11.5.8	N/A	Yes	–
11.5-3	10 CFR 50, Appendix I	11.5.8	N/A	Yes	–
12.1-1	ALARA and Operational Policies	12.1.3	N/A	Yes	–
12.2-1	Additional Contained Radiation Sources	12.2.3	N/A	Yes	–
12.3-1	Administrative Controls for Radiological Protection	12.3.5.1	N/A	Yes	–
12.3-2	Criteria and Methods for Radiological Protection	12.3.5.2	N/A	Yes	–
12.3-3	Groundwater Monitoring Program	12.3.5.3	N/A	Yes	–

Table 1.8-2 (Sheet 10 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
12.3-4	Record of Operational Events of Interest for Decommissioning	12.3.5.4	N/A	Yes	–
12.5-1	Radiological Protection Organization and Procedures	12.5.5	N/A	Yes	–
13.1-1	Organizational Structure of Combined License Applicant	13.1.1	N/A	Yes	–
13.2-1	Training Program for Plant Personnel	13.2.1	N/A	Yes	–
13.3-1	Emergency Planning and Communications	13.3.1	N/A	Yes	–
13.3-2	Activation of Emergency Operations Facility	13.3.1	N/A	Yes	–
13.4-1	Operational Review	13.4.1	N/A	Yes	–
13.5-1	Plant Procedures	13.5.1	APP-GW-GLR-040	Yes	–
13.6-1	Security	13.6	APP-GW-GLR-062 APP-GW-GLR-066 APP-GW-GLR-068	Yes	–
13.6-2	Deleted Vital Equipment Verification	Deleted	APP-GW-GLR-062 APP-GW-GLR-066 APP-GW-GLR-068	N/A	N/A
13.6-3	Deleted Site-Specific Security System	Deleted	APP-GW-GLR-062 APP-GW-GLR-066 APP-GW-GLR-068	N/A	N/A

**Table 1.8-2 (Sheet 11 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items**

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
13.6-4	Deleted Nuclear Material Control Requirements	Deleted	APP-GW-GLR-062 APP-GW-GLR-066 APP-GW-GLR-068	N/A	N/A
13.6-5	Cyber Security Program	13.6	APP-GW-GLR-104	No	Yes
14.4-1	Organization and Staffing	14.4.1	N/A	Yes	–
14.4-2	Test Specifics and Procedures	14.4.2	APP-GW-GLR-037	No	Yes
14.4-3	Conduct of Test Program	14.4.3	APP-GW-GLR-038	No	Yes
14.4-4	Review and Evaluation of Test Results	14.4.4	N/A	No	Yes
14.4-5	Testing Interface Requirements	14.4.5	N/A	Yes	–
14.4-6	First-Plant-Only and Three-Plant-Only Tests	14.4.6	APP-GW-GLR-021	Yes	Yes
15.0-1	Documentation of Plant Calorimetric Uncertainty Methodology	15.0.15.1	NA	–	Yes
15.7-1	Consequences of Tank Failure	15.7.6	N/A	Yes	–
16.1-1	Technical Specification Preliminary Information	16.1	APP-GW-GLR-064 APP-GW-GLN-075	Yes	–
16.3-1	Procedure to Control Operability of Investment Protection Systems, Structures and Components	16.3.2	N/A	Yes	–
17.5-1	Quality Assurance Design Phase	17.5.1	N/A	Yes	–

**Table 1.8-2 (Sheet 12 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items**

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
17.5-2	Quality Assurance for Procurement, Fabrication, Installation, Construction and Testing	17.5.2	N/A	Yes	–
17.5-3	Design Reliability Assurance Program/Site Specific List of Systems, Structures and Components	17.5.3	APP-GW-GLR-117	No	No
17.5-4	Quality Assurance Program for Operations	17.5.4	N/A	Yes	–
17.5-5	Maintaining Reliability of Risk-Significant SSCs	17.5.5	APP-GW-GLR-117	No	No
17.5-6	Maintenance Activities Relevant to Maintenance Rule	17.5.6	APP-GW-GLR-117	No	No
17.5-7	Operational Reliability Assurance Activities	17.5.7	APP-GW-GLR-117	No	No
17.5-8	Operational Reliability Assurance Program Integration with Quality Assurance Program	17.5.8	N/A	Yes	–
18.2-1	Execution of the NRC Approved Human Factors Engineering Program	18.2.6.1	APP-GW-GLR-012	No	No
18.2-2	Design of the Emergency Operations Facility	18.2.6.2	APP-GW-GLR-136	Yes	No
18.5-1	Task Analysis	18.5.4.1	APP-GW-GLR-081 APP-GW-GLR-090	No	No
18.5-2	Main Control Room	18.5.4.2	APP-GW-GLR-090	No	No
18.6-1	Plant Staffing	18.6.1	APP-GW-GLR-090	Yes	–

Table 1.8-2 (Sheet 13 of 14) Summary of AP1000 Standard Plant Combined License Information Items

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
18.7-1	Execution and Documentation of the Human Reliability Analysis/Human Factors Engineering Integration	18.7.1	APP-GW-GLR-011 APP-GW-GLR-090	No	No
18.8-1	Execution and Documentation of the Human System Interface Design Implementation Plan	18.8.5	APP-GW-GLR-082 APP-GW-GLR-090	No	No
18.9-1	Procedure Development	18.9.1	APP-GW-GLR-040 APP-GW-GLR-090	No	No
18.10-1	Training Program Development	18.10.1	APP-GW-GLR-090	Yes	–
18.11-1	Verification and Validation of AP1000 Human Factors Engineering Program	18.11.1	APP-GW-GLR-084 APP-GW-GLR-090	No	No
18.14-1	Human Performance Monitoring	18.14	APP-GW-GLR-090	Yes	–
19.59.10-1	As-Built SSC HCLPF Comparison to Seismic Margin Evaluation	19.59.10.5	APP-GW-GLR-021	No	Yes
19.59.10-2	Evaluation of As-Built Plant Versus Design in AP1000 PRA and Site-Specific PRA External Events	19.59.10.5	APP-GW-GLR-101	Yes	Yes
19.59.10-3	Internal Fire and Internal Flood Analyses	19.59.10.5	APP-GW-GLR-021	No	Yes
19.59.10-4	Develop and Implement Severe Accident Management Guidance	19.59.10.5	APP-GW-GLR-070	Yes	–
19.59.10-5	Equipment Survivability	19.59.10.5	APP-GW-GLR-021	No	Yes

**Table 1.8-2 (Sheet 14 of 14)
Summary of AP1000 Standard Plant
Combined License Information Items**

Item No.	Subject	Subsection	Addressed by Westinghouse Document	Action Required by COL Applicant	Action Required by COL Holder
19.59.10-6	Confirm that the Seismic Margin Assessment analysis is applicable to the COL site.	19.59.10.5	n/a	Yes	No
	Bulletins and Generic Letters (WCAP-15800, Revision 3, July 2004)	1.9.5.5	APP-GW-GLR-129	Yes	No
	Unresolved Safety Issues and Generic Safety Issues	Table 1.9-2	APP-GW-GLR-129	Yes	No

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{HISTORICAL - Table 1.8-201
Summary of FSAR Departures from the DCD - HISTORICAL}

Departure Number	Departure Description Summary	FSAR Section or Subsection
STD DEP 1.1-1	An administrative departure is established to identify instances where the renumbering of FSAR sections is necessary to effectively include content consistent with Regulatory Guide 1.206, as well as NUREG-0800. See note (a).	9.2.11 9.2.12 9.2.13 9.5.1.8 9.5.1.9 13.1 13.1.4 13.5 13.5.3 13.7 17.5 17.6 17.7 17.8
VCS DEP 2.0-1	An administrative departure is established to identify instances where the renumbering of FSAR sections is necessary to effectively include content consistent with Regulatory Guide 1.206, as well as NUREG-0800 which differs from STD DEP 1.1-1.	2.0 2.1 2.2 2.4 2.5
VCS DEP 2.0-2	The site parameter value provided in the DCD Tier 1, Table 5.0-1 for the air temperature maximum wet bulb (noncoincident) is 86.1°F. This site parameter value is listed as the maximum safety wet bulb (noncoincident) air temperature in DCD Tier 2, Table 2-1. The corresponding site characteristic value is 87.3°F as reported in Subsection 2.3.1.5. This site characteristic exceeds the DCD site parameter by 1.2°F.	2.0 2.3.1.5 5.4.7.1 6.2.1.1.3 6.2.2.3 6.4 6.4.1.1 9.1.3.1.3.1 9.2.2.1 9.2.7.2.4
STD DEP 8.3-1	The Class 1E voltage regulating transformers do not have active components to limit current.	8.3.2.2
VCS DEP 18.8-1	At VCSNS, the Technical Support Center (TSC) is not located in the control support area (CSA) as identified in DCD Subsection 18.8.3.5; the TSC location is as described in the Emergency Plan. Additionally, the Operations Support Center (OSC) is also being moved from the location identified in DCD Subsections 18.8.3.6 and 12.5.2.2 and as identified on DCD figures in Sections 1.2 and 12.3, and Appendix 9A; the OSC location is as described in the Emergency Plan.	1.2.2 12.3 12.3.1.2 12.5.2.2 12.5.3.2 18.8.3.5 18.8.3.6 Appendix 9A

(a) The Departure is standard for AP1000 COLAs but the applicable FSAR Sections or Subsections may vary in the AP1000 Subsequent COLAs.

Table 1.8-202 (Sheet 1 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
1.1-1	Construction and Startup Schedule	1.1.7	1.1.5 1.1.7	A
1.9-1	Regulatory Guide Conformance	1.9.1.5	1.9.1 1.9.1.1 1.9.1.2 1.9.1.3 1.9.1.4 1.9.1.5 Appendix 1A Appendix 1AA	A
1.9-2[a]	Bulletins and Generic Letters	1.9.5.5	1.9.5.5	A
1.9-3[a]	Unresolved Safety Issues and Generic Safety Issues	Table 1.9-2 1.9.4.1	1.9.4.1 1.9.4.2.3	A
2.1-1	Geography and Demography	2.1.1	1.1.1 1.2.2 2.1.1	A
2.2-1	Identification of Site-specific Potential Hazards	2.2.1	2.2	A
2.3-1	Regional Climatology	2.3.6.1	2.3.1	A
2.3-2	Local Meteorology	2.3.6.2	2.3.2	A
2.3-3	Onsite Meteorological Measurements Program	2.3.6.3	2.3.3	A
2.3-4	Short-Term Diffusion Estimates	2.3.6.4	2.3.4.1 15.6.5.3.7.3 15A.3.3	A
2.3-5	Long-Term Diffusion Estimates	2.3.6.5	2.3.5.1	A

Table 1.8-202 (Sheet 2 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
2.4-1	Hydrological Description	2.4.1.1	2.4	A
2.4-2	Floods	2.4.1.2	2.4.2	A
2.4-3	Cooling Water Supply	2.4.1.3	2.4.7 2.4.11	A
2.4-4	Groundwater	2.4.1.4	2.4.12	A
2.4-5	Accidental Release of Liquid Effluents into Ground and Surface Water	2.4.1.5	2.4.13	A
2.4-6	Flood Protection Emergency Operation Procedures	2.4.1.6	2.4.10 2.4.14	A
2.5-1	Basic Geologic and Seismic Information	2.5.1	2.5.1	A
2.5-2	Site Seismic and Tectonic Characteristics Information	2.5.2.1	2.5.2.1	A
2.5-3	Geoscience Parameters	2.5.2.3	2.5.2.6	A
2.5-4	Surface Faulting	2.5.3	2.5.3	A
2.5-5	Site and Structures	2.5.4.6.1	2.5.4.1	A
2.5-6	Properties of Underlying Materials	2.5.4.6.2	2.5.4.2	A
2.5-7	Excavation and Backfill	2.5.4.6.3	2.5.4.5	A
2.5-8	Ground Water Conditions	2.5.4.6.4	2.5.4.6	A
2.5-9	Liquefaction Potential	2.5.4.6.5	2.5.4.8	A
2.5-10	Bearing Capacity	2.5.4.6.6	2.5.4.10.1	A
2.5-11	Earth Pressures	2.5.4.6.7	2.5.4.10.3	A
2.5-12	Static and Dynamic Stability of Facilities	2.5.4.6.9	2.5.4.10.2	A
2.5-13	Subsurface Instrumentation	2.5.4.6.10	2.5.4.13	A
2.5-14	Stability of Slopes	2.5.5	2.5.5	A
2.5-15	Embankments and Dams	2.5.6	2.5.6	A

**Table 1.8-202 (Sheet 3 of 14)
COL Item Tabulation**

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
2.5-16	Settlement of Nuclear Island	2.5.4.6.11	2.5.4.10.2	A
2.5-17	Waterproofing System	2.5.4.6.12	2.5.4.14 3.8.5.1	A
3.3-1	Wind and Tornado Site Interface Criteria	3.3.3	1.2.2 2.2 3.3.1.1 3.3.2.1 3.3.2.3 3.3.3 3.5.1.5 3.5.1.6	A
3.4-1	Site-Specific Flooding Hazards Protective Measures	3.4.3	3.4.1.3 3.4.3	A
3.5-1	External Missile Protection Requirements	3.5.4	1.2.2 2.2 3.3.1.1 3.3.2.1 3.3.2.3 3.5.1.5 3.5.1.6 3.5.4	A
3.6-1	Pipe Break Hazards Analysis	3.6.4.1	3.6.4.1 14.3.3.1	H
3.6-4	Primary System Inspection Program for Leak-Before-Break Piping	3.6.4.4	3.6.4.4	A
3.7-1	Seismic Analysis of Dams	3.7.5.1	3.7.2.12 3.7.5.1	A

Table 1.8-202 (Sheet 4 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
3.7-2	Post-Earthquake Procedures	3.7.5.2	3.7.4.4 3.7.5.2	A
3.7-3	Seismic Interaction Review	3.7.5.3	3.7.5.3	H
3.7-4	Reconciliation of Seismic Analyses of Nuclear Island Structures	3.7.5.4	3.7.5.4	H
3.7-5	Location of Free-Field Acceleration Sensor	3.7.5.5	3.7.4.2.1 3.7.5.5	A
3.8-5	Structures Inspection Program	3.8.6.5	3.8.3.7 3.8.4.7 3.8.5.7 3.8.6.5 17.6	A
3.8-6	Construction Procedures Program	3.8.6.6	3.8.6.6	H
3.9-2	Design Specification and Reports	3.9.8.2	3.9.8.2	H
3.9-3	Snubber Operability Testing	3.9.8.3	3.9.3.4.4 3.9.8.3	A
3.9-4	Valve Inservice Testing	3.9.8.4	3.9.6 3.9.6.2.2 3.9.6.2.3 3.9.6.2.4 3.9.6.2.5 3.9.6.3 3.9.8.4	A
3.9-5	Surge Line Thermal Monitoring	3.9.8.5	3.9.3.1.2 3.9.8.5 14.2.9.2.22	A

Table 1.8-202 (Sheet 5 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
3.9-7	As-Designed Piping Analysis	3.9.8.7	3.9.8.7 14.3.3.2	H
3.11-1	Equipment Qualification File	3.11.5	3.11.5	H
4.4-2	Confirm Assumptions for Safety Analyses DNBR Limits	4.4.7.2	4.4.7	H
5.2-1	ASME Code and Addenda	5.2.6.1	5.2.1.1 5.2.6.1	A
5.2-2	Plant Specific Inspection Program	5.2.6.2	5.2.4 5.2.4.1 5.2.4.3.1 5.2.4.4 5.2.4.5 5.2.4.6 5.2.4.8 5.2.4.9 5.2.4.10 5.2.6.2	A
5.2-3	Response to Unidentified Reactor Coolant System Leakage Inside Containment	5.2.6.3	5.2.6.3 5.2.5.3.5	A
5.3-1	Reactor Vessel Pressure – Temperature Limit Curves	5.3.6.1	5.3.6.1	H
5.3-2	Reactor Vessel Materials Surveillance Program	5.3.6.2	5.3.2.6 5.3.2.6.3 5.3.6.2	A
5.3-4	Reactor Vessel Materials Properties Verification	5.3.6.4.1	5.3.6.4.1	H
5.3-7	Quickloc Weld Build-up ISI	5.3.6.6	5.2.4.1 5.3.6.6	A

Table 1.8-202 (Sheet 6 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
5.4-1	Steam Generator Tube Integrity	5.4.15	5.4.2.5 5.4.15	A
6.1-1	Procedure Review for Austenitic Stainless Steels	6.1.3.1	6.1.1.2 6.1.3.1	A
6.1-2	Coating Program	6.1.3.2	6.1.2.1.6 6.1.3.2	A
6.2-1	Containment Leak Rate Testing	6.2.6	6.2.5.1 6.2.5.2.2 6.2.6	A
6.3-1	Containment Cleanliness Program	6.3.8.1	6.3.8.1	A
6.4-1	Local Hazardous Gas Services and Monitoring	6.4.7	2.2.2.2.1.1 6.4.4 6.4.4.2 6.4.7	A
6.4-2	Procedures for Training for Control Room Habitability	6.4.7	6.4.3 6.4.7	A
6.6-1	Inspection Programs	6.6.9.1	6.6 6.6.1 6.6.3.1 6.6.4 6.6.6 6.6.9.1	A
6.6-2	Construction Activities	6.6.9.2	6.6.2 6.6.9.2	A
7.1-1	Setpoint Calculations for Protective Functions	7.1.6.1	7.1.6.1	B

Table 1.8-202 (Sheet 7 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
7.5-1	Post Accident Monitoring System	7.5.5	7.5.2 7.5.3.5 7.5.5	A
8.2-1	Offsite Electrical Power	8.2.5	8.2.1 8.2.1.1 8.2.1.2 8.2.1.3 8.2.5	A
8.2-2	Technical Interfaces	8.2.5	8.2.1.2.2 8.2.2 8.2.5	A
8.3-1	Grounding and Lightning Protection	8.3.3	8.3.1.1.7 8.3.1.1.8 8.3.3	A
8.3-2	Onsite Electrical Power Plant Procedures	8.3.3	8.3.1.1.2.4 8.3.1.1.6 8.3.2.1.4 8.3.3	A
9.1-5	Inservice Inspection Program of Cranes	9.1.6.5	9.1.4.4 9.1.5.4 9.1.6	A
9.1-6	Radiation Monitor	9.1.6.6	9.1.4.3.8 9.1.5.3 9.1.6	A
9.1-7	Coupon Monitoring Program	9.1.6.7	9.1.6	H

Table 1.8-202 (Sheet 8 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
9.2-1	Potable Water	9.2.11.1	9.2.5.2.1 9.2.5.3 9.2.12.1	A
9.2-2	Waste Water Retention Basins	9.2.11.2	9.2.9.2.2 9.2.12.2	A
9.3-1	Air Systems (NUREG-0933 Issue 43)	9.3.7	9.3.7	A
9.4-1	Ventilation Systems Operations	9.4.12	9.4.1.1.1 9.4.1.2.3.1 9.4.1.4 9.4.7.4 9.4.12	A
9.5-1	Qualification Requirements for Fire Protection Program	9.5.1.8.1	9.5.1.6 9.5.1.8 9.5.1.8.1.2 9.5.1.8.2 9.5.1.8.2.1 9.5.1.8.2.2 9.5.1.8.2.2.1 9.5.1.8.3 9.5.1.8.6 9.5.1.8.7 9.5.1.9.1 13.1.1.2.10 13.1.1.3.2.1.4	A
9.5-2	Fire Protection Analysis Information	9.5.1.8.2	9.5.1.9.2 9A.3.3	A

Table 1.8-202 (Sheet 9 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
9.5-3	Regulatory Conformance	9.5.1.8.3	9.5.1.8.1.1 9.5.1.8.8 9.5.1.9.3 9A.3.3.7 9A.4	A
9.5-4	NFPA Exceptions	9.5.1.8.4	9.5.1.8.1.1 9.5.1.9.4	A
9.5-6	Verification of Field Installed Fire Barriers	9.5.1.8.6	9.5.1.8.6 9.5.1.9.6	H
9.5-8	Establishment of Procedures to Minimize Risk for Fire Areas Breached During Maintenance	9.5.1.8.7	9.5.1.8.1.2.a.3.vi 9.5.1.9.7	A
9.5-9	Offsite Interfaces	9.5.2.5.1	9.5.2.5.1	A
9.5-10	Emergency Offsite Communications	9.5.2.5.2	9.5.2.5.2	A
9.5-11	Security Communications	9.5.2.5.3	9.5.2.5.3	A
9.5-13	Fuel Degradation Protection	9.5.4.7.2	9.5.4.5.2 9.5.4.7.2	A
10.1-1	Erosion-Corrosion Monitoring	10.1.3	10.1.3.1	H
10.2-1	Turbine Maintenance and Inspection	10.2.6	10.2.6	H
10.4-1	Circulating Water Supply	10.4.12.1	10.4.5.2.1 10.4.5.2.2 10.4.5.5 10.4.12.1	A
10.4-2	Condensate, Feedwater and Auxiliary Steam System Chemistry Control	10.4.12.2	10.4.7.2.1 10.4.12.2	A
10.4-3	Potable Water	10.4.12.3	10.4.12.3	A

Table 1.8-202 (Sheet 10 of 14) COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
11.2-1	Liquid Radwaste Processing by Mobile Equipment	11.2.5.1	11.2.1.2.5.2 11.2.5.1	A
11.2-2	Cost Benefit Analysis of Population Doses	11.2.5.2	11.2.3.3 11.2.3.5 11.2.3.5.3 11.2.3.5.4 11.2.5.2	A
11.3-1	Cost Benefit Analysis of Population Doses	11.3.5.1	11.3.3.4 11.3.3.4.3 11.3.3.4.4 11.3.5.1	A
11.4-1	Solid Waste Management System Process Control Program	11.4.6	11.4.6	A
11.5-1	Plant Offsite Dose Calculation Manual (ODCM)	11.5.8	11.5.8	A
11.5-2	Effluent Monitoring and Sampling	11.5.8	11.5.1.2 11.5.2.4 11.5.3 11.5.4 11.5.4.1 11.5.6.5 11.5.8	A
11.5-3	10 CFR 50, Appendix I	11.5.8	11.2.3.3 11.2.3.5 11.3.3.4 11.3.5.1 11.5.8	A

Table 1.8-202 (Sheet 11 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
12.1-1	ALARA and Operational Policies	12.1.3	12.1 12.1.3 Appendix 12AA	A
12.2-1	Additional Contained Radiation Sources	12.2.3	12.2.1.1.10 12.2.3	A
12.3-1	Administrative Controls for Radiological Protection	12.3.5.1	12.3.5.1 12.5.4 Appendix 12AA	A
12.3-2	Criteria and Methods for Radiological Protection	12.3.5.2	12.3.4 12.3.5.2	A
12.3-3	Groundwater Monitoring Program	12.3.5.3	12.3.5.3 Appendix 12AA	A
12.3-4	Record of Operational Events of Interest for Decommissioning	12.3.5.4	12.3.5.4 Appendix 12AA	A
12.5-1	Radiological Protection Organization and Procedures	12.5.5	12.5.5 Appendix 12AA	A
13.1-1	Organizational Structure of Combined License Applicant	13.1.1	13.1 13.1.1.2.10 13.1.1.2.11 13.1.1.3.2.1.4 13.1.1.3.2.2 13.1.1.4 13.1.2.1 13.1.3.1 13.1.4 Appendix 13AA	A

Table 1.8-202 (Sheet 12 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
13.2-1	Training Program for Plant Personnel	13.2.1	13.2 13.2.1	A
13.3-1	Emergency Planning and Communications	13.3.1	13.3 13.3.1	A
13.3-2	Activation of Emergency Operations Facility	13.3.1	13.3 13.3.1	A
13.4-1	Operational Review	13.4.1	13.4 13.4.1	A
13.5-1	Plant Procedures	13.5.1	13.5 13.5.1 13.5.3	A
13.6-1	Security	13.6	13.6 13.6.1 14.3.2.3.2	A
13.6-5	Cyber Security Program	13.6.1	13.6.1	H
14.4-1	Organization and Staffing	14.4.1	14.2.2 14.4.1	A
14.4-2	Test Specifics and Procedures	14.4.2	14.4.2	H
14.4-3	Conduct of Test Program	14.4.3	14.2.1 14.2.1.4 14.2.1.5 14.2.3 14.2.3.1 14.2.6 14.4.3	H

Table 1.8-202 (Sheet 13 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
14.4-4	Review and Evaluation of Test Results	14.4.4	14.2.3.2.1 14.2.3.3.1 14.4.4	H
14.4-5	Testing Interface Requirements	14.4.5	14.2.9.4.15 14.2.9.4.22 14.2.10.4.29 14.4.5	A
14.4-6	First-Plant-Only and Three-Plant-Only Tests	14.4.6	14.4.6	B
15.0-1	Documentation of Plant Calorimetric Uncertainty Methodology	15.0.15.1	15.0.3.2 15.0.15	H
15.7-1	Consequences of Tank Failure	15.7.6	15.7.6	A
16.1-1	Technical Specification Preliminary Information	16.1	16.1.1	A
16.3-1	Procedure to Control Operability of Investment Protection Systems, Structures and Components	16.3.2	16.3.1 16.3.2	A
17.5-1	Quality Assurance Design Phase	17.5.1	17.5 17.7	A
17.5-2	Quality Assurance for Procurement, Fabrication, Installation, Construction and Testing	17.5.2	17.5 17.7	A
17.5-4	Quality Assurance Program for Operations	17.5.4	17.5 17.7	A
17.5-8	Operational Reliability Assurance Program Integration with Quality Assurance Program	17.5.8	17.5 17.7	A
18.2-2	Design of the Emergency Operations Facility	18.2.6.2	18.2.1.3 18.2.6.2	A

Table 1.8-202 (Sheet 14 of 14)
COL Item Tabulation

COL ITEM	SUBJECT	DCD SUBSECTION	FSAR SECTION(S)	COL APPLICANT (A), HOLDER (H) OR BOTH (B)
18.6-1	Plant Staffing	18.6.1	13.1.1.4 13.1.3.1 18.6 18.6.1	A
18.10-1	Training Program Development	18.10.1	13.1.1.3.2.2.1 13.2 18.10 18.10.1	A
18.14-1	Human Performance Monitoring	18.14	18.14	A
19.59.10-1	As-Built SSC HCLPF Comparison to Seismic Margin Evaluation	19.59.10.5	19.59.10.5	H
19.59.10-2	Evaluation of As-Built Plant Versus Design in AP1000 PRA and Site-Specific PRA External Events	19.59.10.5	19.59.10.5	B
19.59.10-3	Internal Fire and Internal Flood Analyses	19.59.10.5	19.59.10.5	H
19.59.10-4	Implement Severe Accident Management Guidance	19.59.10.5	19.59.10.5	H
19.59.10-5	Equipment Survivability	19.59.10.5	19.59.10.5	H
19.59.10-6	Confirm that the Seismic Margin Assessment analysis is applicable to the COL site	19.59.10.5	19.55.6.3 19.59.10.5	A

a) COL Items 1.9-2 and 1.9-3 are un-numbered in the DCD.

1.9 Compliance with Regulatory Criteria

1.9.1 Regulatory Guides

Regulatory guides are issued by the NRC in the following 10 broad divisions:

- Division 1 - Power Reactors
- Division 2 - Research and Test Reactors
- Division 3 - Fuels and Materials Facilities
- Division 4 - Environmental and Siting
- Division 5 - Materials and Plant Protection
- Division 6 - Products
- Division 7 - Transportation
- Division 8 - Occupational Health
- Division 9 - Antitrust and Financial Review
- Division 10 - General

Divisions 2, 3, 6, 7, 9, and 10 of the regulatory guides do not apply to the design and design certification phase of AP1000. The following sections provide a summary discussion of NRC Divisions 1, 4, 5, and 8 of the regulatory guides applicable to the design and design certification phase of AP1000.

Appendix 1A provides a discussion of AP1000 regulatory guide conformance.

Divisions 2, 3, 6, 7, 9, and 10 of the regulatory guides do not apply to the construction or operational safety considerations and are not addressed.

Division 4 of the regulatory guides applies to the Environmental Report and the topics are addressed in the Environmental Report. Two Division 4 Regulatory Guides are addressed in Appendix 1A.

Division 5 of the regulatory guides applies to materials and plant protection. As appropriate, the Division 5 regulatory guide topics are addressed in the DCD and plant-specific security plans (i.e., Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan).

Applicable Division 8 Regulatory Guides are addressed in Appendix 1A.

Appendix 1A provides a discussion of plant specific regulatory guide conformance, addressing new Regulatory Guides and new revisions not addressed by the referenced DCD. Regulatory Guides that are completely addressed by the DCD are not listed.

The following subsections provide a summary discussion of Divisions 1, 4, 5 and 8 of the regulatory guides as applicable to the content of this document, or to the construction and/or operations phases.

1.9.1.1 Division 1 Regulatory Guides - Power Reactors

Currently there are approximately 190 Division 1 regulatory guides that have been issued by the NRC for implementation or for comment.

Appendix 1A provides an evaluation of the degree of AP1000 compliance with NRC Division 1 regulatory guides. The revisions of the regulatory guides against which AP1000 is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. For those regulatory guides applicable to the AP1000 Table 1.9-1 identifies the appropriate cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

The superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

Appendix 1A provides an evaluation of the degree of compliance with Division 1 regulatory guides as applicable to the content of this document, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the degree of compliance is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1A). Table 1.9-1 identifies the appropriate regulatory guide to cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

Superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

1.9.1.2 Division 4 Regulatory Guides - Environmental and Siting

One Division 4 regulatory guide, Regulatory Guide 4.7, merits discussion.

Regulatory Guide 4.7, "General Site Suitability Criteria for Nuclear Power Stations," provides guidelines for identifying suitable candidate sites for nuclear power stations. The guidance of this regulatory guide is considered as appropriate in the establishment of the AP1000 site interface criteria, and is described in Sections 2.1 and 2.5.

Division 4 of the regulatory guides applies to the Environmental Report and the topics are addressed in the Environmental Report. Appendix 1A provides an evaluation of the degree of compliance with Division 4 regulatory guides as applicable to the content of this document, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the plant is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1A). For those regulatory guides applicable, Table 1.9-1 identifies the appropriate cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

1.9.1.3 Division 5 Regulatory Guides - Materials and Plant Protection

Division 5 of the regulatory guides applies to materials and plant protection. Appendix 1A provides an evaluation of the degree of conformance with Division 5 regulatory guides as applicable to the content of the AP1000 DCD and the plant-specific Cyber Security Plan. The plant-specific physical security plans (i.e., Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan) were developed using the template in NEI 03-12, Revision 6, “Template for the Security Plan, Training and Qualification Plan, Safeguards Contingency Plan [and Independent Spent Fuel Storage Installation Security Program],” which was endorsed for use by NRC letter dated April 9, 2009. The plant-specific physical security plans include no substantive deviations from the NRC-endorsed template in NEI 03-12, Revision 6. Therefore, the degree of conformance with Division 5 regulatory guides for the plant-specific physical security plans is consistent with the degree of conformance of NEI 03-12, Revision 6.

Three Division 5 regulatory guides, Regulatory Guides 5.9, 5.12, and 5.65, merit discussion.

Regulatory Guide 5.9, "Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material," provides guidelines for data acquisition systems associated with the use of a lithium-drifted germanium gamma ray spectroscopy system. This regulatory guide is not applicable to AP1000 design certification.

Regulatory Guide 5.12, "General Use of Locks in the Protection and Control of Facilities and Special Nuclear Materials," provides guidelines for the selection and use of commercially available locks in the protection of facilities and special nuclear material. The guidance of this regulatory guide is considered as appropriate in the AP1000 design.

Regulatory Guide 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," is not applicable to design certification.

1.9.1.4 Division 8 Regulatory Guides - Occupational Health

Two Division 8 regulatory guides, Regulatory Guides 8.8 and 8.19 merit discussion.

Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposure at Nuclear Power Stations will be As Low As is Reasonably Achievable (ALARA)," provides NRC guidance for meeting the requirements of 10 CFR Part 20. This regulatory guide includes guidance in the following areas for maintaining radiation exposures ALARA:

- Overall program (e.g., policy, organization, and training)
- Facility and equipment design features
- Radiation protection program
- Radiation protection facilities, instrumentation, and equipment

Regulatory Guide 8.8 is written primarily for utility applicants and licensees. However, Westinghouse has established policy, design, and operational considerations that will be applied in the AP1000 design in accordance with this regulatory guide. These considerations are discussed in Section 12.1.

Regulatory Guide 8.19, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants" describes a method acceptable to the NRC staff for performing an assessment of collective occupational radiation dose as part of the ongoing design review process involved in designing a light-water-cooled power reactor so that occupational radiation exposures will be ALARA. This regulatory guide includes guidance for estimating occupational radiation exposures (principally during the design stage) as a result of:

- Reactor operations and surveillance
- Routine maintenance
- Waste processing
- Refueling
- Inservice inspection
- Special maintenance

Occupational radiation exposure estimates that are in accordance with Regulatory Guide 8.19 are described in Section 12.4.

Appendix 1A provides an evaluation of the degree of compliance with Division 8 regulatory guides as applicable to the content of this document, or to the site-specific design, construction and/or operational aspects. The revisions of the regulatory guides against which the plant is evaluated are indicated. Any exceptions or alternatives to the provisions of the regulatory guides are identified and justification is provided. One such general alternative is the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD in order to preserve the finality of the certified design (see Notes at the end of Appendix 1A). For those regulatory guides applicable, Table 1.9-1 identifies the appropriate cross-references. The cross-referenced sections contain descriptive information applicable to the regulatory guide positions found in Appendix 1A.

Superseded or canceled regulatory guides are not considered in Appendix 1A or Table 1.9-1.

1.9.1.5 Combined License Information

Division 1, 4, 5 and 8 Regulatory Guides applicable to the content of this document, or to the site-specific design, construction and/or operational aspects are listed in Table 1.9-1 and Appendix 1A.

1.9.2 Compliance with Standard Review Plan (NUREG-0800)

WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," provides the results of a review of the AP1000 compliance with the acceptance criteria for each section of the Standard Review Plan, NUREG-0800.

Table 1.9-1 provides the required assessment of conformance with the applicable acceptance criteria and the associated cross-references.

1.9.3 Three Mile Island Issues

This section identifies the Three Mile Island issues of 10 CFR 50.34(f) that are addressed by AP1000 design features or program plans. The additional issues of NUREG-0660 and NUREG-0737 that apply to the AP1000 are resolved in accordance with the guidance of NUREG-0933, with specific details provided in the applicable sections of the DCD.

Some of the 10 CFR 50.34(f) issues initially identified as applicable only to Boiling Water Reactors (BWRs) or Babcock and Wilcox plants have also been addressed for the AP1000 design. For example, the AP1000 design incorporates an automatic depressurization system with some similarity to that utilized for BWRs.

10 CFR 50.34(f):

(1)(i) Plant/Site Specific TMI-Related Risk Assessment (NUREG-0660 Item II.B.8)

"Perform a plant/site specific probabilistic risk assessment, the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant."

AP1000 Response:

A plant-specific Probabilistic Risk Assessment (PRA) performed on the AP1000 design evaluates the plant in terms of core damage frequency and containment integrity. The PRA supports the design effort and establishes the capability of the design to meet established safety goals. Level 1 (Plant), 2 (Containment), and 3 (Site) PRA evaluations, including internal and external events:

- Demonstrate that the plant design meets the NRC safety goals
- Identify design vulnerabilities, evaluate alternate design features and operational strategies, and modify the design to reduce risk

The PRA process has been integrated into the design process to verify that the design effort meets the targeted goals and resolves the identified vulnerabilities. As a result, specific design changes were incorporated into the plant systems to improve the reliability of the core and containment heat removal systems.

Close interaction between the plant designers and PRA analysts is maintained to consider severe accident vulnerabilities as part of the design process. The AP1000 PRA is provided to the NRC as a separate document.

(1)(ii) Auxiliary Feedwater System Evaluation (NUREG-0737 Item II.E.1)

"Perform an evaluation of the proposed auxiliary feedwater system, to include (applicable to pressurized water reactors only): (A) a simplified Auxiliary Feedwater System reliability analysis using event-tree and fault-tree logic techniques, (B) a design review of Auxiliary Feedwater System, and (C) an evaluation of Auxiliary Feedwater System flow design bases and criteria."

AP1000 Response:

The AP1000 design does not utilize an auxiliary feedwater system. A nonsafety-related startup feedwater system is provided to remove the core decay heat after the reactor trip during postulated non-LOCA event. Decay heat removal maintains core subcooling and prevents water relief from the pressurizer safety valves by preventing heatup of the reactor coolant system. The startup feedwater pumps automatically start following anticipated transients resulting in low steam generator level. However, operation of the nonsafety-related startup feedwater system is not credited to mitigate licensing design basis accidents described in Chapter 15.

The safety-related passive core cooling system provides emergency core decay heat removal during transients, accidents, or whenever the normal nonsafety-related heat removal paths are unavailable.

The safety-related passive core cooling system design basis and criteria are described in Section 6.3.

(1)(iii) Reactor Coolant Pump Seals (NUREG-0737 Items II.K.2.16 and II.K.3.25)

"Perform an evaluation of the potential for and impact of reactor coolant pump seal damage following small-break loss of coolant accident with loss of offsite power. If damage cannot be precluded, provide an analysis of the limiting small-break loss of coolant accident with subsequent reactor coolant pump seal damage."

AP1000 Response:

The AP1000 design uses sealless motor pumps for circulating primary reactor coolant through the reactor core, piping, and steam generators. In the sealless design, all rotating components are inside a pressure vessel; therefore, no seal can fail and initiate reactor coolant system leakage.

(1)(iv) Automatic Power-Operated Relief Valve Isolation System (NUREG-0737 Item II.K.3.2)

"Perform an analysis of the probability of a small-break loss of coolant accident caused by a stuck-open power-operated relief valve. If this probability is a significant contributor to the probability of small-break loss of coolant accidents from all causes, provide a description and evaluation of the effect on small-break loss of coolant accident probability of an automatic power-operated relief valve isolation system that would operate when the reactor coolant system pressure falls after the power-operated relief valve has opened."

AP1000 Response:

The AP1000 design does not include power-operated relief valves. The pressurizer volume is about 40 percent larger than the pressurizer volume in current plants with a comparable power rating. The larger pressurizer increases transient operation margins and prevents safety valve actuation in most accident situations. The pressurizer surge line is also larger to permit a more rapid transfer of coolant between the reactor coolant system and the pressurizer, and also to accommodate the automatic depressurization system first- to third-stage flow rates. The surge line limits the pressure drop during maximum anticipated surge (Condition II loss of load transient) to prevent exceeding the maximum reactor coolant system pressure limit.

Overpressure protection is provided by two totally enclosed pop-type safety valves. These valves are spring-loaded and self-actuated and they are designed to meet the requirements of the ASME Code, Section III. If the pressurizer pressure exceeds the set pressure, the safety valves start lifting. A temperature indicator in the discharge piping for each safety valve alarms on high temperature to alert the operator to the presence of high temperature fluid from leakage or when the valves open.

The AP1000 design also includes an automatic depressurization system. The system consists of four stages of valves. Three stages are connected to the pressurizer. The fourth stage is connected to the hot legs. These valves are not actuated on a high pressure signal. Design features are included to reduce the chance of spurious automatic depressurization system actuation including appropriate interlocks, 2-out-of-4 instrument actuation, fail as is valves, redundant, closed first, second, and third stage valves in each line, and redundant series controllers for forth stage valves. Probabilistic risk assessment is used to determine the probability of a loss of coolant accident caused by failure of the automatic depressurization system. Results of this evaluation are factored into the design process. See Chapter 5 and Section 6.3 for additional information.

(1)(v) Separation of HPCI and RCIC System Initiation Levels (NUREG-0737 Item II.K.3.13)

"Perform an evaluation of the safety effectiveness of providing for separation of high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) system initiation levels so that the RCIC system initiates at a higher water level than the HPCI system, and of providing that both systems restart on low water level. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI')."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(vi) Relief Valve Challenges (NUREG-0737 Item II.K.3.16)

"Perform a study to identify practicable system modifications that would reduce challenges and failures of relief valves, without compromising the performance of the valves or other systems."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(vii) Automatic Depressurization System Activation (NUREG-0737 Item II.K.3.18)

"Perform a feasibility and risk assessment study to determine the optimum automatic depressurization system design modifications that would eliminate the need for manual activation to ensure adequate core cooling."

AP1000 Response:

Although this issue is identified as applicable to BWRs only, the AP1000 design uses an automatic depressurization system with some similarity to that used on BWRs.

The automatic depressurization system actuates on Low-1 core makeup tank level, coincident with a core makeup tank actuation signal. Therefore manual actuation of the automatic depressurization system is not required to maintain core cooling. As discussed in Section (1)(i), PRA analysis confirms the reliability of the automatic actuation. Additional information is provided in Section 6.3.

(1)(viii) Core Spray and Low Pressure Coolant Injection Systems (NUREG-0737 Item II.K.3.21)

"Perform a study of the effect on all core-cooling modes under accident conditions of designing the core spray and low pressure coolant injection systems to ensure that the systems will automatically restart on loss of water level, after having been manually stopped, if an initiation signal is still present."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(ix) RCIC and HPCI Additional Space Cooling (NUREG-0737 Item II.K.3.24)

"Perform a study to determine the need for additional space cooling to ensure reliable long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) systems, following a complete loss of offsite power to the plant for at least two (2) hours. (For plants with high pressure core spray systems in lieu of high pressure coolant injection systems, substitute the words 'high pressure core spray' for 'high pressure coolant injection' and 'HPCS' for 'HPCI')."

AP1000 Response:

This issue is applicable to BWRs only and is not applicable to AP1000.

(1)(x) Automatic Depressurization System Functionality During Accidents (NUREG-0737 Item II.K.3.28)

"Perform a study to ensure that the Automatic Depressurization System, valves, accumulators, and associated equipment and instrumentation will be capable of performing their intended functions during and following an accident situation, taking no credit for non-safety related equipment or instrumentation, and accounting for normal expected air (or nitrogen) leakage through valves."

AP1000 Response:

Although this issue is identified as applicable to BWRs only, the AP1000 uses a safety-related automatic depressurization system that is different from that presently used on BWRs. The AP1000 automatic depressurization system uses safety-related dc motor-operated valves and squib valves to initiate depressurization. The motive power for these valves is safety-related dc power. There is no nonsafety-related equipment or instrumentation, including instrument air or nitrogen supply, relied on in the operation of these valves.

These valves are designed and qualified to function in the conditions of an accident. They will also be subject of pre-operational and in-service testing. They will be included in the reliability assurance program. Additional information is provided in Section 6.3 for the passive core cooling system, Subsection 3.9.3 for valve operability requirements, Chapter 14 for the initial test program, Subsection 3.9.6 for in-service testing, and Section 16.2 for the reliability assurance program.

(1)(xi) Depressurization Methods/Rapid Cooldown (NUREG-0737 Item II.K.3.45)

"Provide an evaluation of depressurization methods, other than by full actuation of the automatic depressurization system, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown."

AP1000 Response:

This issue is applicable to BWRs only.

(1)(xii) Hydrogen Control System Evaluation (NUREG-0660 Item II.B.8)

"Perform an evaluation of alternative hydrogen control systems that would satisfy the requirements of paragraph (f)(2)(ix) of this section (50.34). As a minimum include consideration of a hydrogen ignition and post-accident inerting system. The evaluation shall include: (A) a comparison of costs and benefits of the alternative systems considered, (B) for the selected system, analyses and test data to verify compliance with the requirements of (f)(2)(ix) of this section (50.34), and (C) for the selected system, preliminary design descriptions of equipment, function, and layout."

AP1000 Response:

Continuous indication of hydrogen concentration in the containment atmosphere is provided. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the active zircaloy cladding.

Hydrogen igniters control rapid releases of hydrogen during and after postulated degraded core and core melt accidents to maintain concentration below 10 percent.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

See Subsection 6.2.4 for additional information.

(2)(i) Simulator Capability (NUREG-0933 Item I.A.4.2)

"Provide simulator capability that correctly models the control room and includes the capability to simulate small-break loss of coolant accidents."

AP1000 Response:

Simulator capability is not included within the scope of the AP1000 design certification. Functional requirements for simulator capability are derived from Human Factors Engineering Program described in Chapter 18.

(2)(ii) Plant Procedures (NUREG-0933 Item I.C.9)

"Establish a program to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include emergency procedures, reliability analyses, human factors engineering, crisis management, operator training, and coordination with INPO and other industry efforts."

AP1000 Response:

See Chapter 13 for a discussion of plant procedures, training of operations personnel and emergency planning.

(2)(iii) Control Room Design (NUREG-0737 Item I.D.1)

"Provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to fabrication or revision of fabricated control room panels and layouts."

AP1000 Response:

The human factors engineering design process of the AP1000 has been developed to conform with NUREG-0711, "Human Factors Engineering Program Review Model." The elements of the design process provide a structured top-down system analysis using accepted human factors engineering principles. The design of the main control room and the other operation and control centers reflect state-of-the-art human factors principles. See Appendix 1A for information on conformance with applicable regulatory guides. See Chapter 18 for additional information on the AP1000 human factors engineering design process.

(2)(iv) Safety Parameter Display System (NUREG-0737 Item I.D.2)

"Provide a plant safety parameter display console that will display to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached or exceeded."

AP1000 Response:

The purpose of the plant safety parameter display console (or safety parameter display system) is to display important plant variables in the main control room in order to assist in rapidly and reliably determining the safety status of the plant.

The requirements for the safety parameter display system are specified during the main control room design process, and are met by the main control room design, specifically as part of the alarms, displays, and controls. The requirements for a safety parameter display system (NUREG-0696, Reference 1) are met by grouping the alarms by plant process or purpose, as directly related to the critical safety functions.

The process data presented on the graphic displays is similarly grouped, facilitating an easy transition for the operators. The safety parameter display system requirement for presentation of plant data in an analog fashion prior to reactor trip is met by the design of the graphic CRT displays.

Displays are available at the operator workstations, the remote shutdown workstation, and at the technical support center. See Chapter 18 for additional information pertaining to the safety parameter display system design.

(2)(v) Safety System Status Indication (NUREG-0933 Item I.D.3)

"Provide for automatic indication of the bypassed and [in]operable status of safety systems."

AP1000 Response:

The AP1000 main control room meets the NRC Regulatory Guide 1.47 recommendations, including automatic indication of bypassed and inoperable status of plant safety systems, as described in Appendix 1A.

Plant safety parameters, protection system status, and plant component status signals are processed by the protection and safety monitoring system and made available to the entire instrumentation and control system via the redundant monitor bus.

Class 1E signals are provided to the qualified data processor, which is part of the protection and safety monitoring system, for accident monitoring displays. The display of this data is incorporated in the process data displays on the graphic CRTs in the AP1000 main control room.

See Chapters 7 and 18 for additional information pertaining to bypass inoperable status indication. Appendix 1A describes conformance with Regulatory Guide 1.47.

(2)(vi) Reactor Coolant System High Point Vents (NUREG-0737 Item II.B.1)

"Provide the capability of high point venting of noncondensible gases from the reactor coolant system, and other systems that may be required to maintain adequate core cooling. Systems to achieve this capability shall be capable of being operated from the control room and their operation shall not lead to an unacceptable increase in the probability of loss-of-coolant accident or an unacceptable challenge to containment integrity."

AP1000 Response:

In the AP1000 design, the capability for remotely operated high point venting of the reactor coolant system is provided by the safety-related automatic depressurization system valves and the safety-related reactor vessel head vent system. Both of these vent paths discharge to the in-containment refueling water storage tank.

During loss of cooling accident events, the automatic depressurization system automatically depressurizes the reactor coolant system so that the passive core cooling system may effectively deliver core cooling flow. Depressurization via the automatic depressurization system results in creation of a gas-steam volume in the upper region of the vessel. This vapor volume expands down to the inside of the hot leg before it begins venting through the hot leg either via the automatic depressurization system paths connected to the pressurizer or directly from the hot legs via the fourth stage automatic depressurization system paths. This process provides an open injection and steam venting flow path through the reactor vessel, maintaining required core cooling flow.

The reactor vessel head vent system can also be operated from the main control room to directly vent from the top of the reactor vessel head. Subsection 5.4.12 provides additional information pertaining to the reactor coolant system venting capabilities.

(2)(vii) Plant Radiation Shielding (NUREG-0737 Item II.B.2)

"Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain TID-14844 source term radioactive materials, and design as necessary to permit adequate access to important areas and to protect safety equipment from the radiation environment."

AP1000 Response:

Post-accident radiation sources, used in the shield design and assessment of post-accident access to vital areas, are addressed in Subsection 12.2.1.3. The post-LOCA instantaneous and integrated source strengths as a function of time are also included as Tables 12.2-20 and 12.2-21, respectively. The sources are based on the core activity release model from Regulatory Guide 1.183, which supersedes the TID-14844 source term assumptions as reflected in Regulatory Guide 1.4.

Vital areas for post-accident personnel access are addressed in Section 12.3, including radiation zone maps that show projected dose rates in these areas and access routes for the various post-accident actions in vital areas. Time estimates have been made for ingress, egress, and performance of actions at the vital area locations and have been used in demonstrating that total individual radiation doses are limited to less than 5 rem and that Item II.B.2 of NUREG-0737 and GDC-19 requirements are met.

Environmental qualification of safety-related equipment is addressed in Section 3.11. The determination of the radiation environments during postulated accident situations considers the activity release model based on NUREG-1465, which supersedes the source term definition of Parts 1 and 4 of Item II.B.2 of NUREG-0737.

Subsection 12.2.3 defines the responsibility to address any additional contained radiation sources not identified in 12.2.1. Thus, appropriate source terms have been identified and used in establishing that the requirements of Item II.B.2 of NUREG-0737 and GDC 19 are met and the issues are resolved.

(2)(viii) Post-Accident Sampling (NUREG-0737 Item II.B.3)

"Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain TID-14844 source term radioactive materials without radiation exposures to any individual exceeding 5 rem to the whole-body or 50 rem to the extremities. Materials to be analyzed and quantified include certain radionuclides that are indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and non-volatile isotopes), hydrogen in the containment atmosphere, dissolved gases, chloride, and boron concentrations."

AP1000 Response:

Recently the NRC published a model Safety Evaluation Report on eliminating post-accident sampling system requirements from technical specifications for operating plants (Federal Register Volume 65, Number 211, October 31, 2000). The AP1000 sampling design is consistent with the approach in the Model safety evaluation report and not the guidance outlined in NUREG-0737 and Regulatory Guide 1.97. The primary sampling system design is consistent with contingency plans to obtain and analyze highly radioactive post-accident samples from the reactor coolant system, the containment sump, and the containment atmosphere.

(2)(ix) Hydrogen Control (NUREG-0660 Item II.B.8)

"Provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction. Preliminary design information on the tentatively preferred system option of those being evaluated in paragraph (1)(xii) of this section (50.34) is sufficient at the construction permit stage. The hydrogen control system and associated systems shall provide, with reasonable assurance, that:

(A) Uniformly distributed hydrogen concentrations in the containment do not exceed 10 percent during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100 percent fuel-clad metal-water reaction, or that the post-accident atmosphere will not support hydrogen combustion.

(B) Combustible concentrations of hydrogen will not collect in areas where unintended combustion or detonation could cause loss of containment integrity or loss of appropriate mitigating features.

(C) Equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

(D) If the method chosen for hydrogen control is a post-accident inerting system, inadvertent actuation of the system can be safely accommodated during plant operation."

AP1000 Response:

See the response provided for issue (1)(xii).

(2)(x) Reactor Coolant System Valve Testing (NUREG-0737 Item II.D.1)

"Provide a test program and associated model development and conduct tests to qualify reactor coolant system relief and safety valves and, for pressurized water reactors, power-operated relief valves, block valves, for all fluid conditions expected under operating conditions, transients and accidents. Consideration of anticipated transients without scram (ATWS) conditions shall be included in the test program. Actual testing under ATWS conditions need not be carried out until subsequent phases of the test program are developed."

AP1000 Response:

The AP1000 reactor coolant system design does not include power-operated relief valves and their associated block valves. However, the safety valve and discharge piping used in the AP1000 design will be either of design similar to those items tested by EPRI and documented in EPRI Report EPRI NP-2770-LD (Reference 2) or will be tested in accordance with the guidelines of Item [II.D.1] of NUREG-0737.

The AP1000 design includes automatic depressurization system valves which are used to depressurize the plant and establish conditions for injection from the accumulators and the in-containment refueling water storage tank. The operability of the automatic depressurization system valves and spargers is confirmed by a test program. See Section 1.5 for information pertaining to the testing program.

Accident analyses for the AP1000 determine fluid conditions expected under operating conditions, transients, and accidents, and the postulated system responses to these conditions, including the operation of reactor coolant system safety valves. Anticipated transients without scram events are analyzed. Appropriate valve qualification documentation is maintained.

(2)(xi) Valve Position Indication (NUREG-0737 Item II.D.3)

"Provide direct indication of relief and safety valve position (open or closed) in the control room."

AP1000 Response:

The AP1000 design does not include power-operated relief valves and their associated block valves from the reactor coolant system.

Direct indication of relief and safety valve position (open or closed) is provided in the main control room.

(2)(xii) Auxiliary Feedwater System Initiation and Indication (NUREG-0737 Item II.E.1.2)

"Provide automatic and manual auxiliary feedwater system initiation, and provide auxiliary feedwater system flow indication in the control room."

AP1000 Response:

As previously noted in the AP1000 response to Issue (1)(ii), the AP1000 design includes a nonsafety-related startup feedwater system, but not an auxiliary feedwater system. Flow indication of the startup feedwater system is provided in the main control room.

The startup feedwater pumps automatically start following anticipated transients resulting in low steam generator level. The startup feedwater control valves automatically control feedwater flow to the steam generators during operation. They can also be operated manually from the main control room.

The safety-related passive core cooling system provides for emergency core decay heat removal during transients, accidents, or whenever the normal heat removal paths are unavailable. Automatic and manual actuation and flow rate indication are available in the main control room.

(2)(xiii) Pressurizer Heater Power Supplies (NUREG-0737 Item II.E.3.1)

"Provide pressurizer heater power supply and associated motive and control power interfaces sufficient to establish and maintain natural circulation in hot standby conditions with only onsite power available."

AP1000 Response:

The AP1000 pressurizer heaters are powered from the nonsafety-related ac power system. During loss of offsite power events, a portion of the pressurizer heaters is capable of being powered from the nonsafety-related onsite standby power system. The pressurizer heaters are capable of establishing and maintaining natural circulation in hot standby condition, with only the diesel generators supplying electrical power.

With only safety-related dc (Class 1E dc) power available, the safety-related passive core cooling system can establish and maintain natural circulation cooling using the passive residual heat removal heat exchangers, transferring the decay heat to the in-containment refueling water storage tank water and to the passive containment cooling system.

Therefore, the nonsafety-related pressurizer heaters are not required for core decay heat removal following a loss of offsite power. See Section 8.3 for additional information.

(2)(xiv) Containment Isolation System (NUREG-0737 Item II.E.4.2)

"Provide containment isolation systems that: (A) ensure all nonessential systems are isolated automatically by the containment isolation system, (B) for each non-essential penetration (except instrument lines) have two isolation barriers in series, (C) do not result in reopening of the containment isolation valves on resetting of the isolation signal, (D) utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation, and (E) include automatic closing on a high radiation signal for all systems that provide a path to the environs."

AP1000 Response:

The AP1000 containment isolation design satisfies NRC requirements, including post-TMI requirements. In general, this means that two barriers are provided -- one inside containment and the other outside containment. Usually these barriers are valves, but in some cases they are closed, seismic Category I piping systems not connected to the reactor coolant system or to the containment atmosphere. Table 6.2.3-1 identifies containment isolation design provisions for mechanical penetrations. The isolation signal and maximum closure times are defined for each remotely operated valve. Containment penetrations, other than equipment hatches and flanges, incorporate two isolation barriers in series.

The AP1000 design incorporates a reduction in the number of required penetrations compared to the number in previous plant designs. The majority of these penetrations are normally closed. Those few that are normally open, use automatically closed isolation valves.

Containment isolation is automatically actuated by a safeguards actuation signal, using two-out-of-four coincident logic. The containment isolation actuation is set as low as reasonable without creating potential for spurious trips during normal operations. Containment isolation can also be initiated manually from the main control room. Containment penetrations do not automatically reopen on the resetting of the isolation signal. See Subsection 6.2.3 for additional information.

(2)(xv) Containment Purging/Venting (NUREG-0933 Item II.E.4.4)

"Provide a capability for containment purging/venting designed to minimize the purging time consistent with ALARA principles for occupational exposure. Provide and demonstrate high assurance that the purge system will reliably isolate under accident conditions."

AP1000 Response:

Containment purging for the AP1000 is provided by the nonsafety-related containment air filtration system. The function of the system is to clean up the containment atmosphere to acceptable radiation levels during plant operation and prior to personnel entry. It can also be used for containment pressure equalization.

The containment air filtration system is designed to reliably isolate under accident conditions. There are two penetrations and two containment filtration subsystems for AP1000.

See Subsection 9.4.7 for additional information.

(2)(xvi) ECCS Actuation Cycles (NUREG-0933 Item II.E.5.1)

"Establish a design criterion for the allowable number of actuation cycles of the emergency core cooling system and reactor protection system consistent with the expected occurrence rates of severe overcooling events (considering both the expected transients and accidents)."

AP1000 Response:

This issue is applicable to Babcock & Wilcox designs only.

The AP1000 design uses the passive core cooling system to provide emergency reactor coolant inventory control and emergency decay heat removal. Component design criteria have been established for the number of actuation cycles for the passive core cooling system. The identified actuation cycles include inadvertent actuation, as well as the system response to expected plant trip occurrences, including overcooling events.

Automatic depressurization system operation is not expected for either design basis or best estimate overcooling events. See Subsection 3.9.1 for additional information.

(2)(xvii) Specific Accident Monitoring Instrumentation (NUREG-0737 Item II.F.1)

"Provide instrumentation to measure, record and readout in the control room: (A) containment pressure, (B) containment water level, (C) containment hydrogen concentration, (D) containment radiation intensity (high level), and (E) noble gas effluents at all potential accident release points. Provide for continuous sampling of radioactive iodines and particulates in gaseous effluents from all potential accident release points, and for onsite capability to analyze and measure these samples."

AP1000 Response:

AP1000 post-accident monitoring is described in Chapter 7.

AP1000 post-accident monitoring provides for indication of the specified parameters as follows:

- Containment pressure
- Containment water level

- Containment radiation intensity (high level)
- Noble gas effluents - to ascertain reactor coolant system integrity

Other noble gas effluents are designated Type E variables and include information to permit the operators to:

- Monitor the habitability of the main control room
- Monitor plant areas where access may be required to service equipment necessary to monitor or mitigate the consequences of an accident
- Estimate the magnitude of release of radioactive materials through identified pathways
- Monitor radiation levels and radioactivity in the environment surrounding the plant

Subsection 11.5.5 has additional information on measurement of radioactive effluents and conformance with Regulatory Guide 1.97.

The AP1000 primary sampling system is designed to provide post accident sampling functions. See Subsection 9.3.3.1 for additional information on the post accident sampling system.

The human factors aspects of the AP1000 are discussed in Chapter 18.

(2)(xviii) Inadequate Core Cooling Instrumentation (NUREG-0737 Item II.F.2)

"Provide instruments that provide in the control room an unambiguous indication of inadequate core cooling, such as primary coolant saturation meters in PWRs, and a suitable combination of signals from indicators of coolant level in the reactor vessel and in-core thermocouples in PWRs and BWRs."

AP1000 Response:

The AP1000 reactor system includes instrumentation for detecting voids in the reactor vessel head and other reactor vessel inventory deficits that could lead to inadequate core cooling.

The available instrumentation includes core subcooling margin monitors, core exit thermocouples, pressurizer level indicators, and reactor coolant system hot leg level. RN-14-132

The AP100 design uses hot leg level instrumentation to provide those functions necessary to support monitoring of the passive plant design features and to provide indication that the reactor vessel water level remains above the elevation of the fuel. RN-14-132

The AP1000 features that provide margin to or indication of inadequate core cooling include the following:

- A larger pressurizer than most current PWRs, with a pressurizer that is located above the reactor pressure vessel head
- No automatic power-operated relief valves
- An improved reactor vessel head venting capability
- A passive core cooling system
- A passive containment cooling system

- No dependence on ac power to maintain adequate core and containment cooling
- Reactor coolant system hot leg level instrumentation
- Improved reactor system instrumentation
- Core subcooling monitoring

See Sections 6.3 and 7.5 for additional information.

(2)(xix) Post-Accident Monitoring Instrumentation (NUREG-0933 Item II.F.3)

"Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage."

AP1000 Response:

The AP1000 post-accident monitoring system was developed by using Regulatory Guide 1.97 as a guidance document.

Data used for post-accident monitoring is displayed either by the normal control room display system or by the qualified data processing system.

The normal control room display system is used for display of nonsafety-related signals which are not required to be displayed by a qualified system. The qualified data processing system provides for the display of signals which must be displayed by a qualified system.

The qualified data processing system is a microprocessor-based, safety-related system that provides instrumentation to monitor the plant variables and systems during and following an accident. The system consists of two independent, electrically isolated, physically separated divisions.

Additional details pertaining to this system are provided in the AP1000 response to issue (2)(xvii) and in Chapter 7.

(2)(xx) Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators (NUREG-0737 Item II.G.1)

"Provide power supplies for pressurizer relief valves, block valves, and level indicators such that: (A) level indicators are powered from vital buses, (B) motive and control power connections to the emergency power sources are through devices qualified in accordance with requirements applicable to systems important to safety, and (C) electric power is provided from emergency power sources."

AP1000 Response:

The AP1000 design does not include power-operated relief valves and their associated block valves from the reactor coolant system.

Pressurizer level indication is provided by instrumentation powered from the Class 1E dc and UPS system. The system provides safety-related, uninterruptible power for the Class 1E plant instrumentation, control, monitoring, and other vital functions, including safety-related components that are essential for safe shutdown of the plant.

The Class 1E direct current system is designed such that these critical plant loads are powered during emergency plant conditions when both onsite and offsite ac power sources are unavailable.

See Chapter 7 and Section 8.3 for additional information.

(2)(xxi) Auxiliary Heat Removal Systems (NUREG-0933 Item II.K.1.22)

"Design auxiliary heat removal systems such that necessary automatic and manual actions can be taken to ensure proper functioning when the main feedwater system is not operable."

AP1000 Response:

Although this issue is applicable to BWRs only, there are some considerations for AP1000.

Following a loss of main feedwater for the AP1000, there are a number of plant systems that automatically actuate to provide decay heat removal. The startup feedwater system is a nonsafety-related system, that can be powered by the nonsafety-related diesel generators, and is automatically actuated and controlled by steam generator level.

For design basis events, the safety-related passive core cooling system includes a passive residual heat removal heat exchanger which automatically actuates to provide emergency core decay heat removal if the nonsafety-related systems are not available.

The AP1000 main control room meets the NRC guidelines for manual actuation of protective functions including those that are used in the event of a loss of normal feedwater.

See Sections 6.3 and 10.4 for additional information.

(2)(xxii) Failure Mode and Effects Analysis for Control Systems (NUREG-0933 Item II.K.2.9)

"Provide a failure modes and effects analysis of the integrated control system to include consideration of failures and effects of input and output signals to the integrated control system."

AP1000 Response:

This issue is applicable to Babcock & Wilcox plants only.

(2)(xxiii) Safety-Grade Anticipatory Reactor Trip (NUREG-0737 Item II.K.2.10)

"Provide, as part of the reactor protection system, an anticipatory reactor trip that would be actuated on loss of main feedwater and on turbine trip."

AP1000 Response:

This issue is applicable to Babcock & Wilcox plants only.

The AP1000 trip logic includes an anticipatory reactor trip for loss of main feedwater using low steam generator water level. See Section 7.2 for additional information.

Since the AP1000 design does not include power-operated relief valves and their associated block valves in the reactor coolant system, the anticipatory reactor trip on turbine trip is not required for AP1000.

(2)(xxiv) Central Water Level Recording (NUREG-0933 Item II.K.3.23)

"Provide the capability to record reactor vessel water level in one location on recorders that meet normal post-accident recording requirements."

AP1000 Response:

This issue is applicable to BWRs only.

(2)(xxv) Emergency Response Facilities (NUREG-0737 Item III.A.1.2)

"Provide an onsite technical support center, an onsite operational support center, and, for construction permit applications only, a nearsite emergency operations facility."

AP1000 Response:

The AP1000 provides for an onsite technical support center and an operational support center. See the figures in Section 1.2 for additional information on the location. The detailed design of the workstations and the associated man-machine interface for the technical support center and the operational support center is guided by the human factors engineering design process described in Chapter 18. The offsite emergency response facility is discussed in subsection 18.2.6.

(2)(xxvi) Leakage Control Outside Containment (NUREG-0737 Item III.D.1.1)

"Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) TID-14844 source term radioactive materials following an accident. Applicants shall submit a leakage control program, including an initial test program, a schedule for retesting these systems, and the actions to be taken for minimizing leakage from such systems. The goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency."

AP1000 Response:

As described in issue (2)(vii), the safety-related AP1000 passive systems do not recirculate radioactive fluids outside of containment following an accident. A nonsafety-related system can be used to recirculate coolant outside of containment following an accident, but this system is not operated when high containment radiation levels exist.

(2)(xxvii) In-Plant Monitoring (NUREG-0737 Item III.D.3.3)

"Provide for monitoring of inplant radiation and airborne radioactivity as appropriate for a broad range of routine and accident conditions."

AP1000 Response:

Area radiation monitors (ARMs) are provided to supplement the personnel and area radiation survey provisions of the AP1000 health physics program described in Section 12.5 and to comply with the personnel radiation protection guidelines of 10 CFR 20, 10 CFR 50, 10 CFR 70, and Regulatory Guides 1.97, 8.2, and 8.8. In addition to the installed detectors, periodic plant environmental surveillance is established.

(2)(xxviii) Control Room Habitability (NUREG-0737 Item III.D.3.4)

"Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in a TID-14844 source term release, and make necessary design provisions to preclude such problems."

AP1000 Response:

Normally, a nonsafety-related HVAC system keeps the AP1000 main control room slightly pressurized to prevent infiltration of air from other plant areas. During accident conditions, a safety-related isolation of the main control room is automatically actuated.

Upon the loss of nonsafety-related ac power, the main control room environment is sufficient to protect the operators and support the man-machine interfaces necessary to establish and maintain safe shutdown conditions for the plant following postulated design basis accident conditions. The sources are based on the core activity release model from Regulatory Guide 1.183, which supersedes the TID-14844 source term assumptions as reflected in Regulatory Guide 1.4.

The main control room is sealed with safety-related connections to a safety-related compressed air breathing source. This compressed air system provides continued pressurization and a source of fresh air for operator habitability. The air supply is sized to last for 72 hours following an accident. It is expected that the onsite nonsafety-related normal HVAC system will be operational before the installed compressed air supply is exhausted.

The nonsafety-related HVAC system, equipped with a refrigeration-type air conditioning unit, normally provides main control room cooling. This equipment is powered from the onsite diesel generators. If the normal HVAC system is not available, outside air is not allowed into the main control room, and the safety-related compressed air storage system is actuated.

(3)(i) Industry Experience (NUREG-0737 Item I.C.5)

"Provide administrative procedures for evaluating operating, design, and construction experience and for ensuring that applicable important industry experiences will be provided in a timely manner to those designing and constructing the plant."

AP1000 Response:

AP1000 design engineers are continually involved in reviewing industry experiences from sources such as NRC Bulletins, Licensee Event Reports, NRC request for information letters to holders of operating licenses for nuclear power reactors, Federal Register information, and generic letters. Lessons learned experience was incorporated in the AP600 through the Westinghouse participation in developing Volume III of the ALWR Utility Requirements Document and participation in the ALWR Utility Steering Committee activities. The AP1000 design is closely based on the AP600. See Subsection 1.9.5.5 for additional information.

(3)(ii) Quality Assurance List (NUREG-0933 Item I.F.1)

"Ensure that the quality assurance list required by Criterion II, Appendix B, 10 CFR Part 50 includes all structures, systems and components important to safety."

AP1000 Response:

The AP1000 Quality Assurance Plan is described in Chapter 17. Structures, systems, and components are classified as described in Section 3.2.

(3)(iii) Quality Assurance Program (NUREG-0737 Item I.F.2)

"Establish a quality assurance program based on consideration of: (A) ensuring independence of the organization performing checking functions from the organization responsible for performing the functions; (B) performing quality assurance/quality control functions at construction sites to the maximum feasible extent; (C) including Quality Assurance personnel in the documented review of and concurrence in quality related procedures associated with design, construction and installation; (D) establishing criteria for determining Quality Assurance programmatic requirements; (E) establishing qualification requirements for Quality Assurance and Quality Control personnel; (F) sizing the Quality Assurance staff commensurate with its duties and responsibilities; (G) establishing procedures for maintenance of "as-built" documentation; and (H) providing a Quality Assurance role in design and analysis activities."

AP1000 Response:

The AP1000 Quality Assurance Plan described in Chapter 17 meets the requirements of issue 1.F.2.

(3)(iv) Dedicated Containment Penetrations (NUREG-0660 Item II.B.8)

"Provide one or more dedicated containment penetrations, equivalent in size to a single 3-foot diameter opening, in order not to preclude future installation of systems to prevent containment failure, such as a filtered vented containment system."

AP1000 Response:

The containment analysis for the AP1000, including PRA and severe accident assessments, demonstrate that the containment, with its passive heat rejection capability, does not need a filtered vent to prevent overpressurization.

The 36-inch diameter containment air filtration system penetration provided for AP1000 meets the requirement of 10 CFR 50.34(f)(3)(iv). See Figure 9.4.7-1, note 6, for additional information.

(3)(v) Containment Design (NUREG-0660 Item II.B.8)

"Provide preliminary design information at a level of detail consistent with that normally required at the construction permit stage of review sufficient to demonstrate that:

(A)(1) Containment integrity will be maintained (i.e., for steel containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level C Limits, except that evaluation of instability is not required, considering pressure and dead load alone. For concrete containments by meeting the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2 Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone) during an accident that releases hydrogen generated from 100 percent fuel clad metal-water reaction accompanied by either hydrogen burning or the added pressure from post-accident inerting assuming carbon dioxide is the inerting agent. As a minimum, the specific code requirements set forth above, appropriate for each type of containment, will be met for a combination of dead load and an internal pressure of 45 psig. Modest deviations from these criteria will be considered by the staff, if good cause is shown by an applicant. Systems necessary to ensure containment integrity shall also be demonstrated to perform their function under these conditions.

(2) Subarticle NE-3220, Division 1, and subarticle CC-3720, Division 2, of Section III of the July 1, 1980 ASME Boiler and Pressure Vessel Code, which are referenced in paragraph (f)(3)(v)(A)(1) and (f)(3) (v)(B)(1) of this section, were approved for incorporation by reference by the Director of the Office of the Federal Register. A notice of any changes made to the material incorporated by reference will be published in the Federal Register. . . .

(B)(1) Containment structure loadings produced by an inadvertent full actuation of a post-accident inerting hydrogen control system (assuming carbon dioxide), but not including seismic or design basis accident loadings will not produce stresses in steel containments in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subarticle NE-3220, Service Level A Limits, except that evaluation of instability is not required (for concrete containments the loadings specified above will not produce strains in the containment liner in excess of the limits set forth in the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Service Load Category), (2) The containment has the capability to safely withstand pressure tests at 1.10 and 1.15 times (for steel and concrete containments, respectively) the pressure calculated to result from carbon dioxide inerting."

AP1000 Response:

The AP1000 containment vessel is designed to meet the requirements of the ASME Code, Section III, Division I, Subsection NE. A severe accident containment analysis is conducted to support the design effort. The results of the analysis are fission product source terms and plant thermal-hydraulic response for each of the accident sequences chosen to be representative of the plant damage states determined in level 1 PRA analysis.

Results of the analysis indicate that containment failure is not predicted for cases in which the passive containment cooling system cooling water is available. The hydrogen igniter system controls hydrogen and mitigates threats to the containment due to hydrogen.

See Section 6.2 for additional information.

(3)(vi) Hydrogen Recombiners (NUREG-0737 Item II.E.4.1)

"For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere."

AP1000 Response:

Since external hydrogen recombiners are not provided for the AP1000, this requirement is not applicable. See Section 6.2 for additional information.

(3)(vii) Management Plan (NUREG-0933 Item II.J.3.1)

"Provide a description of the management plan for design and construction activities, to include: (A) the organizational and management structure singularly responsible for direction of design and construction of the proposed plant; (B) technical resources director by the applicant; (C) details of the interaction of design and construction within the applicant's organization and the manner by which the applicant will ensure close integration of the architect engineer and the nuclear steam supply vendor; (D) proposed procedures for handling the transition to operation; (E) the degree of top level management oversight and technical control to be exercised by the applicant during design and construction, including the preparation and implementation of procedures necessary to guide the effort."

AP1000 Response:

The AP1000 design team has developed a management plan for the AP1000 project which consists of a properly structured organization with open lines of communication, clearly defined responsibilities, well-coordinated technical efforts, and appropriate control channels. The procedures to be used in the construction, startup, and operation phases of the plant are provided in accordance with the Master Plan and Procedure Development Process identified in APP-GW-GLR-040 (Reference 72).

1.9.4 Unresolved Safety Issues and Generic Safety Issues

Proposed technical resolutions of Unresolved Safety Issues and medium- and high-priority Generic Safety Issues, as identified in NUREG-0933, Reference 3 are required for new plants as part of the NRC policy on severe accidents and are required for design certification in accordance with 10 CFR 52.47(a)(1)(iv).

The current program for identifying and establishing the priority of open safety issues is summarized in NUREG-0933. This program provides for the prioritization and tracking of previously categorized

Unresolved Safety Issues and Generic Safety Issues, New Generic Issues, TMI Action Plan Items Under Development, and Human Factors Program Plan Issues.

The following subsection reviews each of the NUREG-0933 safety issues and identifies the safety issues that are applicable to the AP1000. For each of these issues guidance is provided on how the issue is addressed for the AP1000.

1.9.4.1 Review of NRC List of Unresolved Safety Issues and Generic Safety Issues

Applicants for design certification are required by 10 CFR 52.47(a)(1)(iv) to identify:

"Proposed technical resolutions of those Unresolved Safety Issues and medium- and high-priority Generic Safety Issues which are identified in the version of NUREG-0933 current on the date six months prior to application and which are technically relevant to the design."

NUREG-0933, "A Prioritization of Generic Safety Issues," through Supplement 25 identifies hundreds of issues. The issues tabulated in Supplement 25 were reviewed to determine which issues are technically relevant to the AP1000 design. In this review process, the following screening criteria were applied:

a. Issue has been prioritized as **Low**, **Drop**, or has not been prioritized.

b. Issue is not an AP1000 design issue. Issue is applicable to GE, B&W, or CE designs only.

c. Issue resolved with no new requirements.

d. Issue is not a design issue (Environmental Issue, Licensing Issue, Regulatory Impact Issue, or covered in an existing NRC program).

e. Issue superseded by one or more issues.

f. Issue is not an AP1000 design certification issue. Issue is applicable to NTOL plants only, responsibility of combined license applicant, or issue is limited to current generation operating plants.

Issues meeting one or more of the preceding screening criteria were screened out of the review process as issues that are not applicable to the AP1000 design. The remaining issues fall into one of the following two categories:

g. Issue is resolved by establishment of new regulatory requirements and/or guidance.

h. Issue is unresolved pending generic resolution (e.g., prioritized as **High**, **Medium**, or possible resolution identified).

Table 1.9-2 identifies the results of the screening review. For those issues identified as relevant to the AP1000 design (i.e., issues screened as **g** or **h**), Table 1.9-2 identifies the subsection that addresses the issue.

Table 1.9-2 addresses the second un-numbered COL Information Item identified at the end of Table 1.8-2 and listed in Table 1.8-202 as COL Information Item 1.9-3, "Unresolved Safety Issues and Generic Safety Issues." As such, Table 1.9-2 lists those issues identified by Note "d," which apply to other than design issues, Note "f," which apply either to resolution of Combined License (COL) Information Items or to nuclear power plant operations issues, Note "h," which apply to issues unresolved pending generic resolution at the time of submittal of the AP1000 DCD, and any new

Unresolved Safety Issues and Generic Safety Issues that have been included in NUREG-0933 (through supplement 30) since the DCD was developed. Many of these have since been resolved and incorporated into the applicable licensing regulations or guidance (e.g., the standard review plans). These resolved items (as indicated by NUREG-0933) are identified only as "Resolved per NUREG-0933." Many others are not in the list of items in NUREG-0933 Appendix B identified as applicable to new plants. These items are identified only as "Not applicable to new plants." For the remaining items, the table provides the sections that address the topic.

1.9.4.2 AP1000 Resolution of Unresolved Safety Issues and Generic Safety Issues

1.9.4.2.1 TMI Action Plan Issues

TMI Action Plan issues that were not incorporated in 10CFR50.34(f) are addressed in the following. Those issues incorporated into 10CFR50.34(f) are addressed in Subsection 1.9.3.

I.D.5(2) Plant Status and Post-Accident Monitoring Discussion:

TMI action plant item I.D.5(2) addresses the need to improve the operators' ability to prevent, diagnose and properly respond to accidents. The emphasis is on the information needs (i.e., indication of plant status) of the operator. This issue was resolved with the issuance of Revision 2 to Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Environs Conditions During and Following an Accident."

AP1000 Response:

The AP1000 conforms to and meets the intent of Regulatory Guide 1.97. Regulatory Guide 1.97 provides the requirements for post-accident monitoring of nuclear reactor safety parameters, including plant process parameters important to safety and the monitoring of effluent paths and plant environs for radioactivity. These guidelines include definition and categorization of plant variables that are available to the main control room operators for monitoring the plant safety status following a design basis event.

For the AP1000, an analysis is conducted to identify the appropriate variables and to establish the appropriate design basis and qualification criteria for instrumentation used by the operator for monitoring conditions in the reactor coolant system, the secondary heat removal system, the containment, and the systems used for attaining a safe shutdown condition, as discussed in Section 7.5.

The instrumentation is used by the operator to monitor and maintain the safety of the plant during operating conditions, including anticipated operational occurrences and accident and post-accident conditions. A set of plant parameters identified according to the Regulatory Guide 1.97 guidelines are processed and displayed by the qualified data processing system (QDPS), which is discussed in Section 18.8. The verification and validation (V&V) of the QDPS complies with the V&V process described in Section 18.11.

I.D.5(3) On-Line Reactor Surveillance System

Discussion:

TMI action plan item I.D.5(3) addresses the benefit to plant safety and operations of continuous on-line automated surveillance systems. Continuous on-line surveillance systems that automatically monitor reactors can assist plant operations by providing diagnostic information which can predict anomalous behavior.

Various methods of on-line reactor surveillance have been used, including neutron noise monitoring in boiling water reactors (BWRs) to detect internals vibration, and pressure noise surveillance at TMI-2 to monitor primary loop degasification.

AP1000 Response:

The AP1000 reactor coolant pressure boundary is monitored for leaks from the reactor coolant and associated systems by a variety of components located in multiple systems. The leak detection system provides information permitting the plant operators to take corrective action if any detected leakage exceeds technical specifications. The leak detection system is designed according to the requirements of 10 CFR 50, Appendix A, General Design Criterion 30. The system provides a means to detect and, to the extent practical, to identify the source of the reactor coolant pressure boundary leakage. Subsection 5.2.5 provides further discussion of leak detection.

A digital metal impact monitoring system (DMIMS) monitors the reactor coolant system for the presence of loose metallic parts. This system conforms with the guidance provided in Regulatory Guide 1.133, Rev. 1, May 1981. An advanced microprocessor-based system, employing digital technology, automatically actuates audible and visual alarms if a signal exceeds the preset alarm level.

I.F.1 Expand Quality Assurance List

Discussion:

Item I.F.1 addressed the issue of systems that are "important to safety" that are not on the Quality Assurance List. The suggestion was made that equipment important to safety be ranked and that ranking used to determine systems that should be added to the Quality Assurance List. This approach has not been implemented by the NRC on either a generic or cases-by case basis. In NUREG-0933 this item was classified as resolved with no additional requirements established.

AP1000 Response:

The requirements of 10 CFR Appendix B apply to safety-related systems and components. See Subsection 3.2.2 for a discussion of the AP1000 equipment classification system and the associated quality assurance requirements, including requirements for nonsafety-related systems.

I.G.1 Training Requirements

Discussion:

Item I.G.1 included the issue of natural circulation testing for use as input into operator training.

AP1000 Response:

For the AP1000, natural circulation heat removal using the steam generators is not safety-related, as in current plants. This safety-related function is performed by the passive residual heat removal system. Natural circulation heat removal via the passive residual heat removal heat exchanger is tested for every plant during hot functional testing. This testing of passive residual heat removal system meets the intent of the requirement to perform natural circulation testing and the results of this testing is factored into the operator training.

For the AP1000, the tests outlined below are contained in the AP1000 initial test plan and demonstrate the effectiveness of natural circulation cooling.

1. During hot functional testing, prior to fuel load, with the reactor coolant pumps not running and no onsite power available, the heat removal capability of the passive residual heat removal heat exchanger with natural circulation flow is verified (See Subsection 14.2.9.1.3, item e).

2. After fuel loading, but prior to criticality, with the reactor system at no-load operating temperature and pressure and all reactor coolant pumps operating, the depressurization rate is determined by de-energizing the heaters and pressure is further reduced through use of sprays (See Subsection 14.2.10.1.19).

3. After criticality is achieved and the plant is at ~ 3% power, the plant is placed in a natural circulation mode by tripping all reactor coolant pumps and observing the plant response using the steam generators (See Subsection 14.2.10.3.6) and then using the PRHR (see Subsection 14.2.10.3.7) as the primary heat sinks. These tests are performed for the first plant only.

4. A loss-of-offsite power test is performed with the plant at minimum power level supplying normal house loads. The turbine is tripped and the plant is placed in a stable condition using batteries and the diesel generator (See Subsection 14.2.10.4.26).

5. Data obtained from the first plant only natural circulation tests using the steam generators and PRHR is provided for operator training on a plant simulator at the earliest opportunity. Operating training for subsequent plants is also obtained while performing the hot functional PRHR natural circulation test described in item 1 above.

This response as modified for the AP1000 design is consistent with the response to NUREG-0737, action item I.G.1 which provided a proposal for low power testing of existing and future Westinghouse pressurized water reactors in Attachment 4 to letter NS-EPR-2465 from Westinghouse (E. P. Rahe) to the NRC (H. R. Denton) dated July 8, 1981.

I.G.2 Scope of Test Program

Discussion:

TMI Action Plan Items I.G.2 recommended additional testing during preoperational and startup programs to search for anomalies in a plants response to transients. The Standard Review Plan, Section 14 was revised to provide additional guidance for preoperational and startup test programs.

AP1000 Response:

The program plan for preoperational and startup testing of the AP1000 is in Section 14.2. This section addresses the Standard Review Plan, Section 14. The conformance with Standard Review Plan, Section 14 is outlined in AP1000 Compliance with SRP Acceptance Criteria, WCAP-15799.

II.E.1.3 Update Standard Review Plan and Develop Regulatory Guide

Discussion:

This item was a requirement to update Section 10.4.9 of the Standard Review Plan to address the requirements of Items II.E.1.1 and II.F.1.2 for auxiliary feedwater systems. Standard Review Plan 10.4.9 was revised and this issue is classified as resolved.

AP1000 Response:

The AP1000 does not have a safety-related auxiliary feedwater system. For conformance of the AP1000 with Items II.E.1.1 and II.E.1.2 see the write-up for (1)(ii) and (2)(xii) in Subsection 1.9.3. For conformance with Standard Review Plan Section 10.4.9 see WCAP-15799.

II.E.6.1 Test Adequacy Study

Discussion:

This item was intended to establish the adequacy of requirements for safety-related valve testing. Subsequent to this item, expanded requirements were written into the ASME OM Code for valve testing.

AP1000 Response:

The AP1000 is designed for an in-service test program in accordance with the ASME OM Code. See Subsection 3.9.6 for additional information on the in-service testing program plan.

II.K.1(10) Review and Modify Procedures for Removing Safety-related Systems from Service

Discussion:

This item required operating plants to review and modify (as required) their procedures for removing safety-related systems from service to assure operability status is known.

AP1000 Response:

Section 13.5 describes the AP1000 procedure development, preparation, and responsibility.

II.K.1(13) Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items.

Discussion:

This item required that operating plants propose technical specification changes to address Bulletin items.

AP1000 Response:

The AP1000 Technical Specifications (Section 16.1) are based on and were reviewed against the Westinghouse Standard Technical Specifications, which incorporated the requirements of the bulletins for the TMI Action Plan.

II.K.1(17) Trip PZR Level Bistable So That Low Pressure Will Initiate Safety Injection

Discussion:

This item required operating licensees and operating license applicants with Westinghouse designed nuclear steam supply systems to trip the pressurizer level bistable so that the pressurizer low pressure (rather than the pressurizer low pressure and pressurizer low level coincidence) would initiate safety injection.

AP1000 Response:

This issue does not apply to AP1000. The AP1000 does not rely on coincident low pressurizer pressure and low pressurizer level for actuation. See Section 6.3 for a discussion of actuation of the passive core cooling system.

II.K.1(24) Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and Reactor Coolant Pump Trip

Discussion:

This item requires analyses to provide the basis for the comparison of analytical methods.

AP1000 Response:

The analyses documented in Chapter 15 cover a range of small break sizes. The AP1000 automatically trips the reactor coolant pump on an SI signal. The need to look at time lapses between reactor trip and pump trip is not required.

II.K.3(5) Automatic Trip of Reactor Coolant Pumps

Discussion:

This item requires that operating plants and operating plant applicants study the need for automatic trip of reactor coolant pumps and to modify procedures of designs as appropriate.

AP1000 Response:

The AP1000 design provides for an automatic trip of the reactor coolant pumps on actuation of the passive core cooling system. This trip is provided to prevent reactor coolant pump interaction with the operation of the core makeup tank. See Section 6.3 for additional information.

II.K.3(9) Proportional Integral Derivative Controller Modification

Discussion:

TMI action plan item II.K.3(9) required all Westinghouse plants to raise the interlock bistable trip setting to preclude derivative action from opening the PORVs.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not include power-operated relief valves. See Subsections 5.1.2 and 5.2.2 for additional information.

1.9.4.2.2 Task Action Plan Items

A-1 Water Hammer

Discussion:

Generic Safety Issue A-1 was raised after the occurrence of various incidents of water hammer that involved steam generator feedrings and piping, emergency core cooling systems, residual heat removal systems, containment spray, service water, feedwater, and steam lines. The incidents have been attributed to such causes as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Most of the damage has been relatively minor and involved pipe hangers and restraints. However, several incidents have resulted in piping and valve damage. This item was originally identified in NUREG-0371, (Reference 4) and was later determined to be an Unresolved Safety Issue.

AP1000 Response:

Specific sections of the Standard Review Plan (NUREG-0800) address criteria for mitigation of water hammer concerns. The applicable Standard Review Plan sections as well as information provided in NUREG-0927 (Reference 5) were reviewed. The AP1000 meets the water hammer provisions as specified. The discussion that follows provides a brief description of selected systems identified as being subject to water hammer occurrences and special design features that mitigate or prevent water hammer damage.

Design features are incorporated as appropriate to prevent water hammer damage in applicable systems including steam generator feedrings and piping, passive core cooling system, passive residual heat removal system, service water system, feedwater system, and steam lines.

Water hammer issues are considered in the design of the AP1000 passive core cooling system. The passive core cooling system design includes a number of design features specifically to prevent or mitigate water hammer.

The automatic depressurization system operation uses multiple, sequenced valve stages to provide a relatively slow, controlled depressurization of the reactor coolant system, which helps to reduce the potential for water hammer.

Once the depressurization is complete, gravity injection from the in-containment refueling water storage tank is initiated by opening squib valves and then check valves, which reposition slowly. Gravity injection flow actuates slowly, without water hammer, as the pressure differential across the gravity injection check valves equalizes, and the valves open and initiate flow.

The passive residual heat removal heat exchanger is normally aligned with an open inlet valve and closed discharge valves. This alignment keeps the system piping at reactor coolant system pressure, preventing water hammer upon initiation of flow through the heat exchanger. Instrumentation is provided at the system high point to detect a void in the system.

The core makeup tanks are normally aligned with an open inlet line from the reactor coolant cold leg to keep the tanks at reactor coolant system pressure. This alignment keeps the system piping at reactor coolant pressure, preventing water hammer upon initiation of flow through the tank. In addition, instrumentation is provided at each high point to detect voids within the system. Section 6.3 provides additional information on the passive core cooling system.

The potential for water hammer in the feedwater line is minimized by the improved design and operation of the feedwater delivery system. The steam generator features include introducing feedwater into the steam generator at an elevation above the top of the tube bundles and below the normal water level by a top discharge spray tube feedring. The feedring is welded to the feedwater nozzle to limit the potential for inadvertent draining. The layout of the feedwater line is consistent with industry standard recommendations to reduce the potential of a steam generator water hammer.

The startup feedwater system is a nonsafety-related system that provides feedwater during normal plant startup, shutdown, and hot standby. The startup feedwater line is separate from the main feedwater line and therefore does not contribute to the potential of water hammer in the feedwater piping or steam generator feedring.

The main steam line drains are designed to remove accumulated condensate from the main steam lines and to maintain the turbine bypass header at operating temperature during plant operation. The system is designed to accommodate drain flows during startup, shutdown, transient, and normal operation to protect the turbine and the turbine bypass valves from water slug damage.

A-2 Asymmetric Blowdown Loads on Reactor Primary Coolant Systems

Discussion:

Generic Safety Issue A-2 pertains to asymmetric loadings that could act on a pressurized water reactor's primary system as the result of a postulated double-ended rupture of the piping in the primary coolant system. The magnitude of these loads is potentially large enough to damage the supports of the reactor vessel, the reactor internals, and other primary components of the system. Therefore, the NRC initiated a generic study to develop criteria for an evaluation of the response of the primary systems in pressurized water reactors to these loads.

AP1000 Response:

The use of mechanistic pipe break criteria permits elimination of the evaluation of dynamic effects of sudden circumferential and longitudinal pipe breaks in the structural analysis of structures, systems, and components. General Design Criterion 4 allows the use of analyses to eliminate from the design basis the dynamic effects of pipe ruptures postulated at locations defined in Subsection 3.6.2. Dynamic effects include jet impingement, pipe whip, jet reaction forces on other portions of the piping and components, subcompartment pressurization including reactor cavity asymmetric pressurization transients, and traveling pressure waves from the depressurization of the system.

The AP1000 reactor coolant loop and pressurizer surge line are designed in accordance with mechanistic pipe break criteria. In addition, other high energy ASME Code, Section III, Class 1 and 2 piping of 6 inches and greater nominal diameter is evaluated against leak-before-break criteria. The evaluation methodology is described in Subsection 3.6.3 and Appendix 3B.

A-3 Steam Generator Tube Integrity

Discussion:

Pressurized water reactor steam generator tube integrity is subject to various degradation mechanisms, including corrosion-induced wastage, cracking, reduction in tube diameter, denting, (which leads to primary side stress corrosion cracking), vibration-induced fatigue cracks, and wear or fretting due to loose parts in the secondary system. The primary concern is the capability of degraded tubes to maintain their integrity during normal operation and under accident conditions (LOCA or a main steam line break) with adequate safety margins.

Steam generator tube integrity concerns for the three steam generator suppliers, Westinghouse, Combustion Engineering, and Babcock and Wilcox, are addressed by an integrated NRC program for Generic Safety Issues A3, A4, and A5. This program addresses the areas of steam generator integrity, plant systems response, human factors, radiological consequences, and the response of various organizations to a steam generator tube rupture.

AP1000 Response:

The AP1000 steam generators are designed in accordance with the recommendations of Generic Letter 85-02 and NUREG-0844 (References 6 and 7). The AP1000 steam generator is equipped with a number of features to enhance steam generator tube performance and reliability. These features are described in Subsection 5.4.2.

A-9 Anticipated Transients Without Scram

Discussion:

Generic Safety Issue A-9 was resolved with the publication of 10 CFR 50.62. This regulation sets forth the requirements for reduction of risks from anticipated transients without scram.

AP1000 Response:

The AP1000 complies with the requirements of 10 CFR 50.62 except that the AP1000 does not have a safety-related auxiliary feedwater system. In lieu of the automatic initiation of the auxiliary feedwater system under conditions indicative of an ATWS as required by 10 CFR 50.62 (c)(1), the AP1000 automatically initiates the passive residual heat removal system as discussed in Section 6.3.

A discussion of the AP1000 design features used to address the probability of an ATWS is presented in Subsection 1.9.5 and Section 7.7.

A-11 Reactor Vessel Materials Toughness

Discussion:

Generic Issue A-11 addresses a concern with the reduction of reactor vessel fracture toughness as plants accumulate more and more service time. 10 CFR 50, Appendix G provides requirements for reactor vessel material toughness.

AP1000 Response:

The AP1000 reactor vessel design complies with the requirements of 10 CFR 50, Appendix G and includes numerous features to reduce neutron fluence, enhance material toughness at low temperature and eliminate weld seams in critical areas. Material requirements are provided in Subsection 5.3.2. Pressure and temperature limits are provided in Subsection 5.3.3.

A-12 Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports

Discussion:

Generic Safety Issue A-12 addresses a concern with the potential for lamellar tearing of steam generator and RCP support material. NUREG-0577 (Reference 8) categorizes operating plants relative to the adequacy of the plant's steam generator and reactor coolant pump supports with respect to fracture toughness.

AP1000 Response:

The steam generator and reactor coolant pump supports are described in Subsection 5.4.10. The supports are designed in accordance with subsection NF of Section III of the ASME Code. Design and fabrication of these supports in accordance with Subsection NF requirements provide acceptable fracture toughness of materials, and conform with NUREG-0577.

A-13 Snubber Operability Assurance

Discussion:

Generic Issue A-13 addresses snubber operability concerns. Snubbers are utilized primarily as seismic and pipe whip restraints at nuclear power plants. Their safety function is to operate as rigid supports for restraining the motion of attached systems or components under rapidly applied load conditions such as earthquakes, pipe breaks, and severe hydraulic transients.

Operating experience reports show that a substantial number of snubbers have leaked hydraulic fluid and that the rejection rate from functional testing and inspection is high. This has led to an NRC and ACRS concern regarding the effect of snubber malfunctions on plant safety.

AP1000 Response:

The use of snubbers is minimized in the AP1000. Gapped support devices, leak-before-break considerations, and state-of-the-art piping analysis methods are used to minimize the use of snubbers. Snubbers applied in safety-related applications are constructed to ASME Code, Section III, Subsection NF as discussed in Subsection 3.9.3.4.3.

A-17 Systems Interactions in Nuclear Power Plants

Discussion:

This item addresses the potential systems interactions among systems including safety-related and nonsafety-related structures, systems, and components. There can be unintended and unrecognized dependencies among structures, systems, and components. A number of specific types of interactions have been addressed in other generic safety issues and NRC staff activities. These include guidance for inclusion of internal flooding in the IPE program, requirements that address seismically-induced systems interactions, and evaluation of electric power supplies for electric power reliability. NUREG-0933 classifies this item as resolved with no new requirements.

AP1000 Response:

In addition to addressing the specific system interaction guidance mentioned above, the AP1000 was the subject of a systematic evaluation of potential adverse systems interactions documented in WCAP-15992, "AP1000 Adverse Systems Interactions Evaluation Report" (Reference 69).

A-24 Qualification of Class 1E Safety-Related Equipment

Discussion:

Generic Issue A-24 was resolved with the publication of 10 CFR 50.49, prescribing aging and testing for synergistic effects. The NRC has also issued Revision 1 to Regulatory Guide 1.89 for comment. The proposed revision describes a method acceptable to the NRC staff to demonstrate compliance with the requirements of 10 CFR 50.49.

AP1000 Response:

The AP1000 environmental qualification methodology described in Appendix 3D is based on the generic Westinghouse qualification program approved by the NRC. The Westinghouse methodology addresses the requirements of General Design Criteria 4 and 10 CFR 50.49, as well as the guidance of Regulatory Guide 1.89 and IEEE Standard 323-1974. See Appendix 1A and Reference 9.

A-25 Non-Safety Loads on Class 1E Power Sources

Discussion:

Generic Issue A-25 addresses whether nonsafety-related loads should be allowed to share Class 1E power sources with safety-related plant systems. Past regulatory practice has allowed the connection of nonsafety-related loads in addition to the required safety loads to Class 1E power sources by imposing some restrictions. The purpose of this issue is for the NRC to determine whether the reliability of the Class 1E power sources is significantly affected by the sharing of safety and nonsafety-related loads.

The NRC considers this issue as technically resolved with the issuance of Revision 2 to Regulatory Guide 1.75. This regulatory guide includes special requirements for connection of nonsafety-related loads to a Class 1E source.

AP1000 Response:

The AP1000 conforms with the criteria of Regulatory Guide 1.75 with minor exceptions (see Appendix 1A and IEEE 384-1974). The AP1000 safety-related power source is the Class 1E dc and UPS system, which supplies power to the ac inverters for the plant instrumentation and control systems. The system also provides power to dc loads associated with the four protection channels and the accident monitoring system. Non-Class 1E loads powered from Class 1E sources are limited to loads that need connection to a reliable power source. No Credible failure of non-Class 1E equipment or systems will degrade the Class 1E system below an acceptable level. Subsection 8.3.2.1.1 provides a discussion on the Class 1E power source.

A-26 Reactor Vessel Pressure Transient Protection

Discussion:

Generic Issue A-26 addresses the need to provide reactor vessel overpressure protection whenever plants are in a cold shutdown condition. Branch Technical Position RSB 5-2 establishes the current NRC criteria for a low-temperature overpressurization protection system.

AP1000 Response:

The AP1000 conforms with the criteria established in Branch Technical Position RSB 5-2. The AP1000 pressurizer is sized to accommodate most pressure transients. Overpressure protection for the reactor coolant system is provided by either the pressurizer safety valves or the normal residual heat removal relief valves, as described in Subsection 5.2.2.

A-28 Increase in Spent Fuel Pool Storage Capacity

Discussion:

Generic Issue A-28 addresses the safety significance of damage to spent fuel, primarily from a lack of adequate cooling, that could result in the release of radioactivity.

AP1000 Response:

The AP1000 incorporates the NRC criteria. The heat load is evaluated for the spent fuel storage capacity.

A-29 Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage

Description

This item addresses potential methods to reduce vulnerability to sabotage. The NRC staff concluded that existing requirements dealing with plant physical security, controlled access to vital areas, screening for reliable personnel appear to be effective. This item was resolved with no new requirements.

AP1000 Response:

The passive systems in the AP1000 provided to mitigate the effects of potential accidents may have an inherent advantage when considering potential acts of sabotage compared to the active systems in operating plants. The AP1000 includes provisions for access control to the vital area. The provisions for security are discussed in the AP1000 Security Design Report and outlined in Section 13.6.

A-31 Residual Heat Removal Requirements

Discussion:

Generic Issue A-31 addresses the desire for plants to be able to go from hot-standby to cold-shutdown conditions (when this is determined to be the safest course of action) under an accident condition. The safe shutdown of a nuclear power plant following an accident not related to a loss-of-coolant accident has been typically interpreted as achieving a hot standby condition (the reactor is shut down, but system temperature and pressure are at or near normal operating values).

There are events that require eventual cooldown and long-term cooling to perform inspection and repairs.

AP1000 Response:

The AP1000 employs safety-related core decay heat removal systems that establish and maintain the plant in a safe shutdown condition following design basis events. It is not necessary that these passive systems achieve cold shutdown as defined by Regulatory Guide 1.139.

The AP1000 complies with General Design Criteria 34 by using a more reliable and simplified system design. The passive core cooling system is employed for both hot-standby and long-term cooling modes. Hot-standby conditions are achieved immediately and a temperature of 420°F is reached within 36 hours. Reactor pressure is controlled and can be reduced to about 250 psig. The passive residual heat removal system provides a closed cooling system to maintain long-term core cooling. Passive feed and bleed cooling, using the passive injection features for the feed and the automatic depressurization system for bleed, provides another closed-loop safety-related cooling capability. This capability eliminates dependency on open-loop cooling systems, which have limited ability to remain in hot standby for long-term core cooling. See Section 7.4 for a discussion of safe shutdown and Section 6.3 for a description of the passive core cooling system.

Since the passive core cooling system maintains safe conditions indefinitely, cold shutdown is necessary only to gain access to the reactor coolant system for inspection or repair. On the AP1000, cold shutdown is accomplished by using non-safety-related systems. These systems are highly reliable. They have similar redundancy as current generation safety-related systems and are supplied with ac power from either onsite or offsite sources. See Subsection 5.4.7 for a description of the normal residual heat removal system and Subsection 7.4.1.3 for a discussion of cold shutdown achieved by use of non-safety-related systems.

A-35 Adequacy of Offsite Power Systems

Discussion:

Generic Issue A-35 addresses the susceptibility of safety-related electric equipment to offsite power source degradation. The NRC considers this issue as technically resolved with the issuance of the Standard Review Plan, Section 8.3.1 criteria specified in Appendix A, Branch Technical Position BTP PSB 1, "Adequacy of Station Electric Distribution System Voltages."

AP1000 Response:

The AP1000 ac power system is discussed in Sections 8.1 through 8.3. The AP1000 does not require any ac power source to achieve and maintain safe shutdown.

A-36 Control of Heavy Loads Near Spent Fuel

Discussion:

Generic Issue A-36 addresses the need to review requirements, facility designs, and Technical Specifications regarding the movement of heavy loads near spent fuel. The NRC has documented its technical position on this issue in NUREG-0612 (Reference 10) and that issued Standard Review Plan, Section 9.1.5, which includes NUREG-0612 as a part of the review plan.

AP1000 Response:

The AP1000 design conforms to NUREG-0612 and Standard Review Plan, Section 9.1.5. Light load handling systems are described in Subsection 9.1.4, and overhead heavy-load handling systems are described in Subsection 9.1.5.

A-39 Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits for BWR Containments

Discussion:

Generic Issue A-39 addresses operation of BWR primary system pressure relief valves whose operation can result in hydrodynamic loads on the suppression pool retaining structures or those structures located within the pool. These loads result from initial vent clearing of relief valve piping and steam quenching due to high local pool temperatures. This USI was resolved with the issuance of SRP Section 6.2.1.1.C and a series of NUREG reports.

Generic Issue A-39 is not directly applicable to the AP1000. However, the AP1000 in-containment refueling water storage tank (IRWST) has some functional similarity to a suppression pool when the automatic depressurization system (ADS) is actuated.

AP1000 Response:

The AP1000 in-containment refueling water storage tank design includes consideration of loads due to automatic depressurization system operation. The effect of hydrodynamic loads is addressed in Subsection 3.8.3.4.2.

A-40 Seismic Design Criteria - Short Term Program

Discussion:

Generic Issue A-40 addresses a desire to identify and quantify conservatism in the seismic design process. The Standard Review Plan, Section 3.7 provides clarification of development of site-specific spectra, justification for use of single synthetic time-history by power spectral density function, location and reductions of input ground motion for soil-structure interaction, and design of above-ground vertical tanks. The revised provisions are used for margin studies and re-evaluations or individual plant examination for external events.

AP1000 Response:

The AP1000 conforms to the criteria outlined in the Standard Review Plan, Section 3.7. The seismic design criteria and seismic evaluation methodology are described in Section 3.7.

The AP1000 employs generic, enveloping seismic design criteria and applies established seismic evaluation methodology that complies with current regulations and regulatory guidance. For sites having specific characteristics outside the range of the selected parameters, the AP1000 is evaluated to demonstrate acceptability to the site-specific characteristics.

A-43 Containment Emergency Sump Performance

Discussion:

Generic Issue A-43 addresses technical concerns as follows:

- Pressurized water reactor sump (or boiling water reactor residual heat removal system suction intake) hydraulic performance under post-loss-of-coolant accident adverse conditions resulting from potential vortex formation, air ingestion, and subsequent pump failure

- The possible transport of large quantities of insulation debris generated by a loss-of-coolant accident resulting from a pipe break to the sump debris screen(s), and the potential for sump screen (or suction strainer) blockage to reduce net positive suction head (NPSH) margin below that required for the recirculation pumps to maintain long-term cooling

- The capability of residual heat removal and containment spray system pumps to continue pumping when subjected to possible air, debris, or other effects, such as particulate ingestion on pump seal and bearing systems

AP1000 Response:

Air ingestion, vortexing, and debris blockage are not significant concerns for the AP1000. Containment recirculation includes sump screens that conform to the criteria specified in Regulatory Guide 1.82. The recirculation screens have a large cross-sectional area to reduce the fluid flow velocity through the screen and to provide a large screening area to accommodate accumulated debris. Horizontal plates located above the recirculation screens preclude debris being deposited in the water directly adjacent to the screens. Pipe subject of loss of coolant pipe breaks and in the vicinity of these breaks use reflective metallic insulation to preclude the generation of fibrous insulation debris. See Subsection 6.3.2.2.7 for additional information on the design of the screens and limits on use of fibrous insulation.

Since the AP1000 design does not use pumps to provide safety injection flow, the passive core cooling system injection flow rates are substantially lower than those for plants with pumped injection flow. This results in lower fluid flow velocities through the screens, reducing the potential to draw debris into the sump screens.

The containment recirculation sump piping inlet is located slightly above the compartment floor, which is substantially below the expected flood-up water level. This precludes air ingestion in the piping since recirculation does not initiate until the flood-up water level is well above the piping inlet.

The elimination of pumps also eliminates concerns about the effects on safety injection capability for vortexing, air ingestion, and blockage effects on pump net positive suction head.

The AP1000 includes the capability to use nonsafety-related normal residual heat removal pumps to take a suction from the containment recirculation sump to provide reactor coolant system injection. The sump screen design addresses concerns with screen debris, vortexing, and air ingestion.

Section 6.3 provides additional information on the operation of the passive core cooling system. Appendix 1A describes conformance with Regulatory Guide 1.82. Section 6.2 provides additional information on the containment recirculation sump.

A-44 Station Blackout

Discussion:

Generic Issue A-44 was resolved with the publication of 10 CFR 50.63, which provides requirements that light-water-cooled nuclear power plants be able to withstand for a specified duration and recover from a station blackout. It specifies that an alternate ac power source constitutes acceptable capability to withstand station blackout provided an analysis is performed that demonstrates that the plant has this capability from the onset of the station blackout until the alternate ac source(s) and required shutdown equipment are started and lined up to operate.

10 CFR 50.2 for the alternate ac source notes that the alternate ac power source must have sufficient capability and reliability for operation of all systems required for coping with station blackout for the time required to place and maintain the plant in safe shutdown.

AP1000 Response:

AC electrical power is not needed to establish or maintain a plant safe shutdown condition for the AP1000. The ac power system is discussed in Chapter 8. In addition, two nonsafety-related standby diesel generators are provided as alternate sources of electrical power to nonsafety-related active systems that provide a defense-in-depth function.

A-46 Seismic Qualification of Equipment in Operating Plants

Discussion:

Generic Issue A-46 addresses the variability among operating plants in the margins of safety provided in equipment to resist seismically induced loads and perform the intended safety functions. The NRC believes that the seismic qualification of equipment in operating plants must, therefore, be reassessed to confirm the ability to bring the plant to a safe shutdown condition when it is subject to a seismic event.

AP1000 Response:

This issue applies to operating plants and, as such, does not specifically apply to the AP1000, which is designed in accordance with current seismic requirements. The seismic Category I mechanical and electrical equipment utilized for the AP1000 is qualified in accordance with the AP1000 qualification methodology discussed in Section 3.10. The methodology is based on the generic Westinghouse qualification program previously approved by the NRC. This methodology addresses IEEE Standard 344-1987 (Reference 13) and Regulatory Guide 1.100. See Subsection 1.9.1 (Appendix 1A).

A-47 Safety Implications of Control Systems

Discussion:

Generic Issue A-47 addresses the safety impact of non-safety-related control systems on plant dynamics. Instrumentation and control systems used by nuclear plants comprise safety-related protection systems and nonsafety-related control systems. Safety-related systems are used to trip the reactor when specified parameters exceed allowable limits and to protect the core from overheating by initiating emergency core cooling systems. Nonsafety-related control systems are used to maintain the plant within prescribed parameters during shutdown, startup, normal load, and varying power operation. Nonsafety-related systems are not relied on to perform any safety functions during or following postulated accidents, but are used to control plant processes.

AP1000 Response:

For the AP1000, control system failures are considered as potential initiating events. The analyses of these transients demonstrate that the consequences of such failures are bounded by ANS Condition II criteria. No design basis failure of a control system violates Condition II criteria.

The integrated control system for the AP1000 obtains certain control input signals from signals used in the integrated protection system. With the integrated control and protection system, functional independence of the control and protection systems is maintained by providing a signal selection device in the control system for those signals used in the protection system. The purpose of the signal selection device is to prevent a failed signal, caused by the failure of a protection channel, from resulting in a control action that could lead to a plant condition requiring that protective action. The signal selection device provides this capability by comparing the redundant signals and automatically eliminating an aberrant signal from use in the control system. This capability exists for bypassed sensors or for sensors whose signals diverge from the expected error tolerance.

The plant control system incorporates design features such as redundancy, automatic testing, and self-diagnostics to prevent challenges to the protection and safety monitoring system. Chapter 7 provides a discussion of the AP1000 instrumentation and controls. The surveillance requirements for the main and startup feedwater control are found in Technical Specifications 3.7.3 and 3.7.7.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Discussion:

Generic Issue A-48 addresses postulated light water reactor accidents resulting in a degraded or melted core that could result in the generation and release to the containment of large quantities of hydrogen. One source of hydrogen is from the reaction of the zirconium fuel cladding with the steam at high temperatures. The NRC requires design provisions for handling hydrogen releases associated with rapid reaction of a large portion of fuel cladding (10 CFR 50.44 and 10 CFR 50.34).

AP1000 Response:

The AP1000 design complies with the provisions of draft changes to 10 CFR 50.44 and 10 CFR 50.34 (f). The mechanisms used to monitor and control hydrogen inside containment are discussed in Subsection 6.2.4.

A-49 Pressurized Thermal Shock

Discussion:

Generic Issue A-49 addresses transients and accidents postulated to occur in pressurized water reactors that can result in severe overcooling (thermal shock) of the reactor vessel, concurrent with high pressure. In these pressurized thermal shock events, rapid cooling of the reactor vessel internal surface causes a temperature distribution across the reactor vessel wall that produces a thermal stress with maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress varies with the rate of change of temperature and is compounded by coincident pressure stresses.

As long as the fracture resistance of the reactor vessel material is relatively high, these events are not expected to cause vessel failure. The fracture resistance of the reactor vessel material decreases with the integrated exposure to fast neutrons. The rate of decrease is dependent on the chemical composition of the vessel wall and weld materials.

AP1000 Response:

The AP1000 complies with the requirements of 10 CFR 50.61. Material requirements and pressure-temperature limits are discussed in Subsections 5.3.2 and 5.3.3.

B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments

Discussion:

Part I - Ductility of Two-Way Slabs and Shells

Generic Issue B-5 involved a concern over the lack of information on the behavior of two-way reinforced concrete slabs loaded dynamically in biaxial membrane tension, flexure, and shear. The NRC Staff concluded that there is sufficient information pertaining to the design of two-way slabs subjected to dynamic loads and biaxial tension to enable a reasonably accurate analysis.

Part II - Buckling Behavior of Steel Containments

Generic Issue B-5 involves a concern over the lack of a uniform, well defined approach for design evaluation of steel containments. Of particular interest was potential instability of the shell during dynamic loadings. Based on the conclusion of the NRC Staff that existing steel containments had adequate margins against buckling and that the issue of steel containment buckling had very little safety impact, this item was classified as resolved with no new requirements.

AP1000 Response:

The design requirements and analysis methods used for two-way reinforced concrete slabs and for the steel containment are outlined in Section 3.8.

B-17 Criteria for Safety-Related Operator Actions

Discussion:

Generic Issue B-17 addresses the development of a time criterion for safety-related operator actions including a determination of whether or not automatic actuation is required. The evaluation of this issue includes Issue 27, Manual versus Automated Actions.

AP1000 Response:

The AP1000 automatically initiates the safety-related actions required to protect the plant during design basis events. The plant systems are designed to provide the required information to the operator to monitor plant conditions and to evaluate the performance of the safety-related passive systems, as well as the nonsafety-related active systems. The active systems are designed to automatically actuate and provide defense-in-depth for various plant events, to preclude unnecessary actuation of the safety-related passive systems. The plant design also provides the capability for a backup manual initiation of both the safety-related systems and the nonsafety-related defense-in-depth systems.

As described in Chapter 15, the AP1000 safety systems maintain the plant in a safe condition following design basis events. For the design basis events described in Chapter 15, this is accomplished without the need for operator action for up to 72 hours. Operator action is planned and expected during plant events to achieve the most effective plant response consistent with event conditions and equipment availability. For events where operator action is taken, the plant design maximizes the time available to complete actions for events. For example, during a steam generator tube rupture, no operator action is required to establish safe shutdown conditions or prevent steam generator overfill. It is expected that the main control room operators take actions similar to those taken in current plants to identify and isolate the faulted steam generator and to stabilize plant conditions.

For events where operator actions are taken, the AP1000 design is based on previous experience and the guidance of ANSI 58.8-1984 (Reference 21). At least 30 minutes is available following design basis events for the operator to initiate planned actions.

B-22 LWR Fuel

Discussion:

Generic Issue B-22 addresses the reliability of fuel behavior predictions during normal operation and postulated accidents. Standard Review Plan, Section 4.2 provides detailed NRC criteria for the design of fuel and core components.

AP1000 Response:

The AP1000 reactor core design complies with the Standard Review Plan, Section 4.2. See Section 4.2 for a discussion of the fuel system design.

B-29 Effectiveness of Ultimate Heat Sinks

Discussion:

Generic Issue B-29 addresses NRC confirmation of currently used mathematical models for prediction of ultimate heat sink performance by comparing model performance with field data and development of better guidance regarding the criteria for weather record selection to define ultimate heat sink design basis meteorology.

The NRC considers this issue to be technically resolved with the publication of three reports: NUREG-0693, NUREG-0733, and NUREG-0858 (References 23, 24 and 25).

AP1000 Response:

The AP1000 passive containment cooling system complies with Standard Review Plan, Section 9.2.5 by providing passive decay heat removal that transfers heat to the atmosphere, which is the ultimate heat sink for accident conditions. The passive containment cooling system is described in Subsection 6.2.2.

B-32 Ice Effects on Safety-Related Water Supplies

Discussion:

Generic Issue B-32 addresses the potential effects of extreme cold weather and ice buildup on the reliability of various plant water supplies. Current NRC criteria are provided in Standard Review Plan, Section 2.4.7, "Ice Effects."

AP1000 Response:

Subsection 6.2.2 describes the ultimate heat sink design and discusses the features that prevent freezing in the passive containment cooling system.

B-36 Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Features Systems and for Normal Ventilation Systems

Discussion:

Generic Issue B-36 addresses the development of revisions to current guidance and technical positions regarding engineered safety features and normal ventilation system air filtration and adsorption units. The NRC considers this issue technically resolved with the issuance of Revision 2 to Regulatory Guide 1.52 and Revision 1 to Regulatory Guide 1.140.

AP1000 Response:

The AP1000 main control room emergency habitability system (VES) includes a passive filtration system that is contained entirely within the main control room envelope. Regulatory Guide 1.52 was written for active safety-related filtration systems. To the extent applicable, system design criteria are established in accordance with Regulatory Guide 1.52 Revision 3. The passive filtration portion of the AP1000 VES contains no active equipment.

B-53 Load Break Switch

Discussion:

Generic Issue B-53 addresses the use of the generator load break switch for isolating the generator from the step-up transformer following turbine trip. Plant designs that utilize generator load circuit breakers to satisfy the requirement for an immediate access circuit stated in General Design Criterion

17, "Electric Power Systems," must prototype-test the generator load circuit breaker to demonstrate functional capability.

AP1000 Response:

The AP1000 design incorporates a generator load circuit breaker to provide a reliable source of ac power to the electrical systems. Exceptions to General Design Criteria 17, as discussed in Section 3.1, are due to the AP1000 design not requiring ac power sources for a design basis accident. Subsection 8.2.2.5 provides further discussion.

B-56 Diesel Reliability

Discussion:

Generic Safety Issue B-56 addresses the reliability of emergency onsite diesel-generators. Diesel reliability is a factor in the criteria associated with the resolution of Unresolved Safety Issue A-44. The resolution of issue B-56 is the development of guidelines for an acceptable emergency diesel-generator reliability program to ensure conformance with the emergency diesel-generator target reliability (0.95 to 0.975) identified in the proposed resolution of Unresolved Safety Issue A-44.

AP1000 Response:

The AP1000 diesel-generators are not safety related. The AP1000 diesel-generator reliability is based on diesel-generator industry standards and practices. The diesel generator is discussed in Subsection 8.3.1. The diesel generator reliability is modeled in the PRA. The reliability assurance program is discussed in Section 16.2.

B-61 Allowable ECCS Equipment Outage Periods

Discussion:

Generic Safety Issue B-61 addresses surveillance test intervals and allowable equipment outage periods in the technical specifications for safety-related systems. This task involves the NRC development of analytically based criteria for use in confirming or modifying these surveillance intervals and allowable equipment outage periods.

AP1000 Response:

The AP1000 surveillance test intervals and allowable outage times help to meet plant safety goals while maximizing plant availability and operability. In determining these limits for the AP1000 technical specifications, a combination of NUREG-1431 precedent, system design, and safety-related function is considered.

B-63 Isolation of Low-Pressure Systems Connected to the Reactor Coolant Pressure Boundary

Discussion:

Generic Issue B-63 addresses the adequacy of the isolation of low-pressure systems that are connected to the reactor coolant pressure boundary. The NRC staff requires that valves forming the interface between high- and low-pressure systems associated with the reactor coolant boundary have sufficient redundancy to prevent the low-pressure systems from being subjected to pressures that exceed their design limits.

AP1000 Response:

The AP1000 includes interconnections between high- and low-pressure systems. Each of these systems interfaces contains appropriate isolation provisions. Valves at the interface between high-and low-pressure systems have redundancy to prevent low-pressure systems from being subjected to pressures that exceed their design limits. The AP1000 design meets the provisions of the Standard Review Plan, Section 3.9.6.

The normal residual heat removal system interface is addressed in Subsection 5.4.7. WCAP-15993 (Reference 56) provides an evaluation of the AP1000 conformance to intersystem loss-of-coolant accident regulatory criteria.

B-66 Control Room Infiltration Measurements

Discussion:

Generic Safety Issue B-66 addresses the adequacy of control room area ventilation systems and control building layout to ensure that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases. The NRC considers this issue as being technically resolved, and criteria have been incorporated in Standard Review Plan, Section 6.4.

AP1000 Response:

The AP1000 main control room is essentially leak-tight. A description of the control room habitability systems is contained in Section 6.4.

Verification of design infiltration rates is as specified in Standard Review Plan, Section 6.4. The AP1000 minimizes unfiltered in-leakage by maintaining the main control room at a slightly positive pressure.

C-1 Assurance of Continuous Long-Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment

Discussion:

Generic Issue C-1 addresses the long-term capability of hermetically sealed instruments and equipment that must function in post-accident environments. The NRC considers this issue as being technically resolved with the issuance of current criteria for qualification of safety-related electrical equipment.

AP1000 Response:

The AP1000 environmental qualification program described in response to Unresolved Safety Issue A-24 addresses qualification of safety-related instrumentation and electrical equipment that must function under accident conditions. This program confirms the integrity of seals employed in the design of Class 1E equipment. See item A-24 of this subsection and Section 3.11 for AP1000 qualification methodology.

C-4 Statistical Methods for ECCS Analysis

Discussion:

Generic Issue C-4 addresses NRC development of a statistical assessment of the certainty level of the peak clad temperature limit. Appendix K, "ECCS Evaluation Models," to 10 CFR 50 specifies the requirements for ECCS analysis. These requirements call for conservatisms to be applied to certain models and assumptions used in the analysis to account for data uncertainties at the time Appendix K was written. The resulting conservatism in the calculated peak clad temperature (PCT) has not been thoroughly compared against the uncertainty in peak clad temperature obtained from a realistically calculated (best-estimate) LOCA. The staff allows voluntary use of statistical uncertainty analysis to justify relaxation of all but the required conservatisms contained in current ECCS evaluation models.

AP1000 Response:

Chapter 15 discusses the LOCA analysis for the AP1000.

C-5 Decay Heat Update

Discussion:

Generic Issue C-5 involves following the work of research groups in determining best-estimate decay heat data and associated uncertainties for use in LOCA calculations.

The staff has determined that the 1979 ANSI 5.1 is technically acceptable and has allowed the use of this data to justify relaxation of non-required conservatisms in current ECCS evaluation models. The ECCS rule change allows the use of this new data. This issue was determined to be resolved.

AP1000 Response:

The large-break LOCA analyses for the AP1000, which employ the best-estimate W COBRA/TRAC analysis methodology (Subsection 15.6.5), use the decay heat model identified in the 1979 ANSI 5.1 (Reference 26).

C-6 LOCA Heat Sources

Discussion:

Generic Issue C-6 addresses the impact on LOCA calculations of LOCA heat sources, their associated uncertainties, and the manner in which they are combined. An evaluation was made of the combined effect of power density, decay heat, stored energy, fission power decay, and their associated uncertainties with regard to calculations of LOCA heat sources.

AP1000 Response:

See Subsection 15.6.5 for a discussion of LOCA heat sources.

C-10 Effective Operation of Containment Sprays in a LOCA

Discussion:

Generic Issue C-10 addresses the effectiveness of containment sprays to remove airborne radioactive materials that could be present within the containment following a LOCA. The NRC considers this issue as being technically resolved with the issuance of ANSI 56.5-1979 (Reference 28), which is referenced in Standard Review Plan, Section 6.5.2.

AP1000 Response:

The AP1000 design does not employ a safety-related containment spray system for removal of airborne radioactive materials in containment. Subsection 15.6.5.3 provides details of source term and mitigation techniques.

C-17 Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Discussion:

Generic Issue C-17 addresses the development of criteria for acceptability of radwaste solidification agents. The NRC considers this issue as technically resolved with the issuance of 10 CFR 61.56.

AP1000 Response:

The AP1000 solid radwaste system transfers, stores, and prepares spent ion exchange resins for disposal. It also provides for disposal of filter elements; sorting, shredding, and compaction of compressible dry active wastes. The solid radwaste system does not provide for liquid waste concentration or solidification. These functions, if used, are provided using mobile systems. Solidification of wastes is not performed by permanently installed systems.

1.9.4.2.3 New Generic Issues

These items were identified in NUREG-0933 as New Generic Issues and surfaced after the publication of the NUREGs that included the Task Action Plan items other unresolved safety issues.

Issue 14 PWR Pipe Cracks

Discussion:

This issue addresses the occurrences of main feedwater line cracking found in operating plants. This issue was classified as resolved with no new requirements.

AP1000 Response:

The design and inspection requirements for the feedwater lines are discussed in Subsection 10.4.7.

Issue 15 Radiation Effects on Reactor Vessel Supports

Discussion:

Generic Safety Issue 15 addresses the potential problem of radiation embrittlement of reactor vessel support structures. There is a potential for radiation embrittlement of the reactor vessel support structure from long-term exposure to neutrons with an energy of 1 MeV or greater. Embrittlement due to neutron damage may increase the potential for propagation of existing flaws.

AP1000 Response:

The supports for the AP1000 reactor pressure vessel are designed for loading conditions and environmental factors including consideration of neutron fluence levels. The material requirements include fracture toughness requirements and impact testing requirements in compliance with ASME Code, Section III, Subsection NF requirements. The reactor pressure vessel supports are not in the region of high neutron fluence where neutron embrittlement of the supports would be a significant concern.

Issue 22 Inadvertent Boron Dilution Event

Discussion:

Some operating plants do not have provisions to detect boron dilution during cold shutdown. This could result in inadvertent criticality. The NRC staff concluded that existing review criteria are adequate. This issue was classified as resolved with no new requirements.

AP1000 Response:

The provisions in the design to preclude inadvertent boron dilution events are outlined in Subsection 9.3.6.

Issue 23 Reactor Coolant Pump Seal Failures

Discussion:

Generic Safety Issue 23 addresses reactor coolant pump seal failures that challenge the makeup capacity in PWRs. Such seal failures represent small-break loss-of-coolant accidents.

AP1000 Response:

The AP1000 reactor coolant pumps are sealless pumps. A sealless pump contains the motor and all rotating components inside a pressure vessel designed for full reactor coolant system pressure. The shaft for the impeller and rotor is contained within the pressure boundary; therefore, seals are not required in order to restrict leakage out of the pump into containment. Subsection 5.4.1 provides additional information on the sealless pump design for the AP1000 reactor coolant pumps. Since the reactor coolant pumps do not rely on seals as a reactor coolant pressure boundary, this issue is not applicable to the AP1000.

Issue 24 Automatic ECCS Switchover to Recirculation

Discussion:

This issue addresses the issue of switchover from safety injection to recirculation using manual valve alignment or automatic valve alignment.

AP1000 Response:

The AP1000 does not switch from injection to recirculation in the sense that injection is not isolated when recirculation is opened. The AP1000 does provide for automatic opening of the recirculation line on a low level signal from the in-containment refueling water storage tank. See Section 6.3 for additional details.

Issue 29 Bolting Degradation or Failure in Nuclear Power Plants

Discussion:

Generic Safety Issue 29 addresses a concern about pressure boundary integrity and component support reliability associated with bolt failures.

As documented in Generic Letter 91-17, the NRC has provided resolution of this issue. The resolution is documented in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," and NUREG-1445, "Regulatory Analysis for the Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants." The resolution was based on a number of industry initiatives and NRC staff actions. NRC staff actions include issuing a number of bolting-related bulletins, generic letters and information notices. Industry initiatives include the publishing of EPRI Reports NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants," and NP-5067, "Good Bolting Practices, A Reference Manual for Nuclear Power Plant Maintenance Personnel."

EPRI Report NP-5769 establishes the characteristic that bolted connections exhibit leakage prior to failure resulting from bolt degradation. The NRC has endorsed the recommendation in NP-5769 that plant-specific bolting integrity programs be established that encompass safety-related bolting. NUREG-1339 includes recommendations and guidelines for the content of a comprehensive bolting integrity program.

AP1000 Response:

The elements of resolution pertain to the design, material selection, fabrication, and in-service inspection of the bolted connections found in the AP1000. To address this, resolutions found in NUREG-1339 are incorporated into the design, material selection, fabrication, and maintenance of the bolted connections. The maintenance practices are addressed by the maintenance program of the combined license holder. Conformance to ASME Code, Section III requirements for pressure boundary components and related supports provides safe operation in the event of bolting degradation. Because of the emphasis in the AP1000 design on access for maintenance and inspection, the recommended maintenance practices can be implemented.

Issue 43 Reliability of Air Systems

Discussion:

This issue addresses the concern that compressed air system degradation or malfunction may cause malfunction of safety-related systems and components. Of particular interest are air operated valves because of problems with the quality of the air supply or the manner in which the compressed air system fails. Generic Letter 88-14 and NUREG-1275 were issued in response to this issue.

AP1000 Response:

The compressed air systems are described in Subsection 9.3.1. Provisions are included to maintain the quality of the air supply. The AP1000 safety-related, air-operated valves do not rely on the air supply to perform their safety-related function.

Issue 45 Inoperability of Instrumentation Due to Extreme Cold Weather

Discussion:

Generic Safety Issue 45 addresses the inoperability of instrumentation due to extreme cold weather. This issue was resolved with the issuance of changes to Standard Review Plan, Section 7.1, Appendix A to Section 7.1, Section 7.5, and Section 7.7.

AP1000 Response:

The AP1000 complies with Standard Review Plan Section 7.1, Appendix A to Section 7.1, Section 7.5, and Section 7.7.

Issue 51 Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems

Discussion:

Generic Safety Issue 51 addresses the susceptibility of open cycle service water systems to fouling including the buildup of aquatic bivalves and corrosion products that can significantly degrade the performance of the system. In operating plants, the service water system is typically used to cool safety-related equipment and to transfer decay heat to the ultimate heat sink.

AP1000 Response:

The service water system in the AP1000 provides cooling water to the component cooling water system and has no safety-related functions. None of the safety-related equipment requires cooling water to effect a safe shutdown or mitigate the effects of design basis events. Heat transfer to the ultimate heat sink is accomplished by heat transfer through the containment shell to air and water flowing on the outside of the shell.

The design of the service water system and the provisions for minimizing long-tern corrosion and organic fouling are described in Subsection 9.2.1.

Issue 57 Effects of Fire Protection System Actuation

Discussion:

Generic Safety Issue 57 addresses the potential for adverse interactions from actuation of the fire protection system with safety-related equipment. Operating experience has shown that safety-related equipment subject to fire protection system water spray and other suppressant chemicals can be rendered inoperable.

AP1000 Response:

The fire protection system and fire protection program in the AP1000 minimize the potential for adverse interactions of safety-related equipment with the fire protection system. The means used to achieve this result include: isolating combustible material and limiting the spread of fire by subdividing the plant into fire areas separated by fire barriers, providing separate and redundant safe shut down components and associated electrical divisions to preserve the ability to safely shutdown the plant following a fire, and providing floor drains sized to remove expected firefighting water without flooding safety-related equipment. The design of the fire protection system is described in Subsection 9.5.1.

Issue 67.3.3 Improved Accident Monitoring

Discussion:

This issue addresses weaknesses in accident monitoring. The recommended solution is to implement Regulatory Guide 1.97.

AP1000 Response:

The guidance of Regulatory Guide 1.97 is followed to determine the appropriate parameters to monitor in the AP1000.

Issue 73 Detached Thermal Sleeves

Discussion:

This issue addresses problems with "generation 3" thermal sleeves.

AP1000 Response:

The AP1000 does not use generation 3 thermal sleeves and includes design provisions to preclude failures of thermal sleeves.

Issue 75 Generic Implications of ATWS Events at the Salem Nuclear Plant

Discussion:

This issue considers the failure of reactor trip breakers to open and issues related to design and testing of the reactor protection system. Issues to be considered include the capability to record and display reactor trip system parameters, equipment classification information, post-maintenance testing, and reliability improvements in operating plants. Generic letter 83-28 and IE Bulletins 83-01 and 83-04 were issued by the staff with specific requirements.

AP1000 Response

The design of the reactor trip breakers and the reactor protection system is outlined in Section 7.1. Information on the functional requirements for reactor trip and conformance with industry and regulatory guidance is outlined in Section 7.2.

The provisions provided to display and record parameters used by the reactor trip system are outlined in Subsections 7.1.2.6 and 7.1.2.13. Section 7.5 also provides information on requirements for safety-related display information.

Subsection 7.1.1 identifies the safety-related functions provided by the protection and safety monitoring system and the items that are included in the system including the reactor trip switchgear. Conformance of safety-related systems and components to industry and regulatory criteria is identified in Subsection 7.1.4.

The reliability and fault tolerance of the protection and safety monitoring system for test maintenance and bypass conditions are outlined in Subsection 7.1.2.10.

The changes in the design of the reactor trip breakers and associated logic to enhance reliability in operating nuclear power plants have been incorporated in the AP1000 design as appropriate. The reactor trip system includes built-in test capability.

WCAP-15800 addresses conformance with generic letters and bulletins.

Issue 79 Unanalyzed Reactor Vessel Thermal Stress During Natural Convection Cooldown

Discussion:

Generic Safety Issue 79 addresses the thermal stresses that occur in the reactor vessel head flange during a natural circulation cooldown. High stresses in the flange or studs during a natural circulation cooldown in PWRs could violate ASME code allowables. Cycling of the stresses could reduce the fatigue margin. Generic Letter 92-02 repeated the reporting requirements of 10CFR 50.73 (a)(2)(ii)(B), "Licensee event report system."

AP1000 Response:

The natural circulation cooldown transient is evaluated as part of ASME Code vessel evaluations and is discussed in Subsection 3.9.1.1.2.11. The reporting requirements to address the requirements of 10CFR 50.73 (a)(2)(ii)(B) referenced in Generic Letter 92-02 are the responsibility of the Combined License holder.

Issue 82 Beyond Design Basis Accidents in Spent Fuel Pools

Discussion:

This issue addresses the concern of a beyond design basis accident in which the spent fuel pool is drained and spent fuel stored there subsequently catches on fire releasing very large amounts of radioactive contamination. This issue is classified as resolved with no new requirements.

AP1000 Response:

The AP1000 includes design provisions that preclude draining of the spent fuel pool. Also, provisions are available to supply water to the pool in the event the water covering the spent fuel begins to boil off.

Issue 83 Control Room Habitability

Discussion:

Loss of control room habitability following an accidental release of external toxic or radioactive material or smoke can impair or cause loss of the control room operators' capability to safely control the reactor. Use of the remote shutdown workstation outside the control room following such events is unreliable since this station has no emergency habitability or radiation protection provisions.

AP1000 Response:

Habitability of the main control room is provided by the main control room/control support area HVAC subsystem of the nonsafety-related nuclear island nonradioactive ventilation system (VBS). If ac power is unavailable for more than 10 minutes or if "high-high" particulate or iodine radioactivity is detected in the main control room supply air duct, which would lead to exceeding General Design Criteria 19 operator dose limits, the protection and safety monitoring system automatically isolates the main control room and operator habitability requirements are then met by the main control room emergency habitability system (VES). The safety-related main control room emergency habitability system supplies breathable quality air for the main control room operators while the main control room is isolated.

In the event of external smoke or radiation release, the nonsafety-related nuclear island nonradioactive ventilation system provides for a supplemental filtration mode of operation, as discussed in Section 9.4. In the unlikely event of a toxic chemical release, the safety-related main control room emergency habitability system has the capability to be manually actuated by the operators. Further, a 6-hour supply of self-contained portable breathing equipment is stored inside the main control room pressure boundary.

Issue 87 Failure of HPCI Steam Line Without Isolation

Discussion:

Generic Safety Issue 87 addresses the uncertainty regarding the operability of the motor-operated isolation valves for the steam supply lines of the high-pressure coolant injection (HPCI) system in boiling water reactors following a postulated break in the supply line. A break in the line could lead to high flow or high differential pressure that may inhibit closure of the isolation valve. These valves typically cannot be tested in-situ for the design flow rates and pressures. Although the AP1000 does not have a high-pressure coolant injection system, it does have isolation valves designed to close against high flow or high pressure differential in the event of a postulated pipe break.

The issue of the operability of motor-operated valves has received considerable attention since Generic Safety Issue 87 was initiated. The NRC provided guidance for inservice testing of motor-operated, safety-related valves in Generic Letter 89-10. SECY-93-087 identifies the proposed position on inservice testing of safety-related valves for advance light water reactors. The guidance in these documents recommends that safety-related valves be tested under full flow under actual plant conditions where practical. EPRI has a program to demonstrate operation of motor-operated valves.

AP1000 Response:

Safety-related valves must meet the requirements of ASME Code, Section III to provide pressure boundary integrity. Valves and valve operators are sized to provide operation under a full range of design basis flow and pressure drop conditions. For the AP1000, safety-related motor-operated valve designs are subject to qualification testing to demonstrate the capability of the valve to open, close, and seat against maximum pressure differential and flow. The requirements for this testing are based on ASME QME-1-2007, “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants.” See Subsection 5.4.8 for an outline of AP1000 valve requirements.

The in-service testing program for safety-related valves is discussed in Subsection 3.9.6. Motor-operated valves are to be operability tested as outlined in Subsection 3.9.6.2.2. Subsection 3.9.6.2.2 includes a discussion of the factors to be considered to determine which valves and test conditions are to be used for operability testing of power-operated valves. Sufficient flow is provided to fully open check valves during testing unless the maximum accident flows are not sufficient to fully open the check valve. The valves built to ASME Code, Section III are tested in compliance with the requirements found in the ASME code, "Code for Operation and Maintenance of Nuclear Power Plants." For additional information on inservice testing of safety-related valves, see Subsection 3.9.6.

Issue 93 Steam Binding of Auxiliary Feedwater Pumps

Discussion:

Generic Safety Issue 93 addresses the potential for a common mode failure of the pumps in an auxiliary or emergency feedwater system. Hot water leaking through one or more isolation valves can flash to steam at the auxiliary feedwater pump potentially resulting in the failure of the pump to operate if required because of steam binding. The NRC addressed this issue in Bulletin 85-01, and reinforced it in Generic Letter 88-03, by requesting that the fluid conditions in the auxiliary feedwater system be monitored and procedures be developed to recognize steam binding and restore the auxiliary feedwater system to operable status if steam binding should occur.

AP1000 Response:

The AP1000 does not have a safety-related auxiliary feedwater system. The passive core cooling system provides the safety-related function of cooling the reactor coolant system in the event of loss of feedwater. The startup feedwater system provides the steam generators with feedwater during plant conditions of startup, hot standby, and cooldown and when the main feedwater pumps are unavailable. The startup feedwater system has no safety-related function.

The startup feedwater system includes temperature instrumentation in the pump discharge for monitoring of the temperature of the startup feedwater system. The system also includes a normally closed isolation valve and a normally closed check valve for each pump limiting potential back leakage.

Issue 94 Additional Low-Temperature Overpressure Protection for Light Water Reactors

Discussion:

Generic Safety Issue 94 addresses the establishment of additional guidance for reactor coolant system low-temperature overpressure protection to ensure reactor vessel and reactor coolant system integrity beyond that identified in the resolution to Generic Safety Issue (GSI) A-26. Low-pressure overpressurization events that occurred subsequent to the implementation of the guidelines for resolution of GSI A-26 indicated a need for additional low-temperature overpressure protection. To resolve this issue, the NRC issued Generic Letter 90-06 which required a revision to plant technical specifications for operability of the low-temperature overpressure protection system. Other possible solutions identified in GL 90-06 included hardware modifications including use of residual heat removal system relief valves and requiring the low temperature overpressure protection system to be fully safety related.

AP1000 Response:

The reactor vessel for the AP1000 is designed to be less susceptible to brittle fracture during low temperature overpressure events. The material requirements and welding processes are developed to enhance resistance to embrittlement. See Subsection 5.3.2 for additional information on the requirements to address fracture toughness of the reactor vessel.

The normal residual heat removal system is designed to provide the safety-related function of low temperature overpressure protection for the reactor coolant system during refueling, startup, and shutdown operations. The system is designed to limit the reactor coolant system pressure within the limits specified in 10 CFR 50, Appendix G. The relief valve in the normal residual heat removal system is used to provide the overpressure protection. See Subsection 5.4.7 for additional information on the design of the normal residual heat removal system and the overpressure protection function.

Issue 103 Design for Probable Maximum Precipitation

Discussion:

Generic Safety Issue 103 addresses the methodology used for determining the design flood level for a particular reactor site. This issue was resolved by incorporating the methodology into the Standard Review Plan.

AP1000 Response:

This is a site-related parameter. The AP1000 is designed for air temperatures, humidity, precipitation, snow, wind, and tornado conditions as specified in Table 2.0-201. The site is acceptable if the site characteristics fall within the AP1000 plant site design parameters in Table 2.0-201. For cases where a site characteristic exceeds the envelope parameter, see Chapter 2.

Issue 105 Interfacing System LOCA at BWRs

Discussion:

Generic Safety Issue 105 addresses concerns over the adequacy of isolation valves between the reactor coolant system and low-pressure interfacing systems in BWRs. This issue, which is limited to pressure isolation valves in BWRs, is related to Generic Safety Issue 96, which considers the failure of the pressure isolation valves between the reactor coolant system and the RHR system in PWRs. Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release. Designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue.

AP1000 Response:

For information on this issue, see Subsection 1.9.5.1 SECY-90-016 Issues. See Subsection 5.4.7 for additional information on the normal residual heat removal system design.

Issue 106 Piping and Use of Highly Combustible Gases in Vital Areas

Discussion:

Generic Safety Issue 106 addresses the normal process system use of relatively small amounts of combustible gases on site and also addresses leaks or breaks in the hydrogen piping and supply system that could result in the accumulation of a combustible or an explosive mixture of air and hydrogen within the auxiliary systems building. The accumulation of combustible or explosive mixtures of gas in the auxiliary systems building could represent a threat to safety-related equipment if the combustible gases are inadvertently ignited.

AP1000 Response:

The AP1000 uses small amounts of combustible gases for normal plant operation. Most of these gases are used in limited quantities and are associated with plant functions or activities that do not jeopardize any safety-related equipment. These gases are found in areas of the plant that are removed from the Nuclear Island (see Subsection 9.3.2 for a description of the plant gas system). The exception to this is the hydrogen supply line to the chemical and volume control system (CVS).

The chemical and volume control system is the only system on the nuclear island that uses hydrogen gas. Hydrogen is supplied to the AP1000 CVS inside containment from a single hydrogen bottle. The release of the contents of an entire bottle of hydrogen in the most limiting building volumes (both inside containment and in the auxiliary building) would not result in a volume percent of hydrogen large enough to reach a detonable level.

The chemical and volume control system hydrogen supply piping is routed through the turbine building and into the auxiliary building and then into containment. The H_2 supply line is routed through the piping/valve room on elevation 100′-0″ of the auxiliary building. The piping/valve penetration room in the auxiliary building on elevation 100′-0″ is designed as a 3-hour fire zone. A fire in this area would not inhibit the safe shutdown of the plant. More information is contained in Appendix 9A.

The turbine building does not house any safety-related systems or equipment. The release of hydrogen into an area of the turbine building does not represent a threat to the safety of the plant.

The AP1000 containment has hydrogen sensors that would detect hydrogen leaks. The containment hydrogen concentration monitoring subsystem is described in Subsection 6.2.4.1.

Issue 113 Dynamic Qualification Testing of Large-Bore Hydraulic Snubbers

Discussion:

Generic Safety Issue 113 addresses the requirements for qualification and periodic operability testing of large bore hydraulic snubber for operating plants. Large-bore hydraulic snubbers are used to a limited extent on the AP1000 to provide support, particularly for seismic events, of piping systems and components while allowing for movement due to thermal expansion. The NRC, in a draft regulatory guide (SC-708-4, "Qualification and Acceptance Test for Snubbers Used in Systems Important to Safety"), has established recommendations for testing of hydraulic snubbers on a forward-fit basis; that is, units without a license at the time the recommendations were established.

AP1000 Response:

The AP1000 plant uses significantly fewer hydraulic snubbers than do currently operating plants. In addition to the recommendations in the draft regulatory guide, testing requirements have been established in ASME OM Code – 1995 Edition up to and including the 1996 Addenda, "Code for Operation and Maintenance of Nuclear Power Plants." Subsection 3.9.3.4.3 discusses requirements for production and qualification testing. The design of the hydraulic snubbers permits required preoperational and inservice testing.

Subsection 3.9.8.3 defines the responsibility to provide information on snubber operability testing.

Issue 120 On-Line Testability of Protection System

Discussion:

This issue is related to the protection system of some older plants that do not provide for as complete a degree of on-line protection system testing surveillance capability as is now required. Testing requirements and guidance are found in GDC 21, Regulatory Guides 1.22 and 1.118 and IEEE Standard 338. This item is classified as resolved with no additional requirements.

AP1000 Response:

This item does not apply to the AP1000. The provision for testing of the protection system in conformance with the regulatory guidance is found in Section 7.1.

Issue 121 Hydrogen Control for Large, Dry PWR Containments

Discussion:

Generic Safety Issue 121 concerns ongoing NRC experimental and analytical programs addressing the likelihood of safe shutdown equipment surviving a hydrogen burn. The staff also intends to explore the possibility and probable consequences of the formation of local detonable concentrations in large, dry PWRs. The concerns are prediction of conditions in realistic configurations, and containment and equipment survivability.

AP1000 Response:

The AP1000 includes provisions for hydrogen control for the unlikely severe accident cases in which large amounts of hydrogen could be generated because of degraded core events. Analyses were performed to examine the consequences of hydrogen burn and to evaluate the likelihood of deflagration to detonable transitions.

For severe accident cases, the containment hydrogen control system prevents hydrogen burn initiation at high hydrogen concentration levels. Hydrogen igniters promote burning when the lower flammability limit is reached and limits the containment hydrogen concentration to less than 10 volume percent during and following a degraded core or core melt.

Thus, for severe accident cases, the AP1000 is designed to prevent the occurrence of hydrogen detonation, thereby preventing the possibility of the resultant large pressure spikes in containment, which is the source of concern for containment integrity and equipment survival. Details of the hydrogen ignition subsystem are provided in Subsection 6.2.4.2.3. Placement of the hydrogen igniters is discussed in Subsection 6.2.4.

A hydrogen burn analysis shows that the AP1000 hydrogen igniter system is effective in maintaining the hydrogen concentration throughout the containment close to the lower flammability limit, and that the peak pressure in the containment during and following hydrogen burn remains well below ASME service level C stress intensity limits. The hydrogen concentration is similar in all compartments analyzed, indicating that the hydrogen released mixes well in the AP1000 containment. The analyses predict conditions in realistic configurations. Peak gas temperatures and pressures in each compartment for each case analyzed are provided, thus providing the hydrogen burn thermal environment that containment equipment will experience. Details are provided in Chapter 14 of the PRA report.

The challenge to the AP1000 containment integrity from hydrogen deflagrations and detonations during core damage events is examined in the hydrogen deflagration and detonation analyses. This bounding evaluation assumes that an amount of hydrogen equivalent to 100-percent active cladding oxidation burns all at once in the AP1000 containment, with no credit taken for the hydrogen igniters. The evaluation concludes that a hydrogen deflagration is unlikely to cause containment failure. Other analyses show that a deflagration to detonation transition in any part of the AP1000 containment is unlikely. Containment failure from a detonation is not considered a credible event for the AP1000 because of the lack of conditions supporting a deflagration to detonation transition, the provision and placement of hydrogen igniters, and the containment design features resulting in a well-mixed atmosphere. Details are provided in subsection 10.2.5 of the PRA evaluation report.

The hydrogen igniters and the containment electrical and mechanical penetrations are designed to operate in the most limiting severe accident environment, including a hydrogen burn. (See subsection 10.2.5 of the PRA evaluation report.) The approach of using controlled burning to prevent accidental hydrogen burn initiation provides confidence that safety-related equipment will continue to operate during and after hydrogen burns. (See Subsection 6.2.4.)

Issue 124 Auxiliary Feedwater System Reliability

Discussion:

Generic Safety Issue 124 addresses the use of probabilistic risk assessment to evaluate the reliability of the auxiliary feedwater system. The issue was resolved by the NRC's issuing plant-specific requirements for a few plants that did not initially have a reliability higher than a minimum criteria.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not have a safety-related auxiliary feedwater system. The passive core cooling system provides the safety-related function of cooling of the reactor coolant system in the event of loss of feedwater. The startup feedwater system provides the steam generators with feedwater during plant conditions of startup, hot standby, and cooldown and when the main feedwater pumps are unavailable. The startup feedwater system has no safety-related function beyond containment isolation.

Issue 128 Electrical Power Reliability

Discussion:

Generic Safety Issue 128 addresses the reliability of onsite electrical systems and encompasses GSI 48, GSI 49, and GSI A-30.

AP1000 Response:

The design basis and design criteria for the Class 1E dc and UPS system is provided in Subsections 8.1.4.2.1 and 8.1.4.3. The class 1E dc and UPS system design is described in Subsection 8.3.2.1.1. Specifically, this design addresses IEEE Standards 603 and 308. This includes the following generic issues:

- Generic Safety Issue 48, LCO for Class 1E vital instrument buses in operating reactors. Chapter 16 provides the AP1000 technical specifications. Subsections 16.1.3.8.3 and 16.1.3.8.4 provide the limiting conditions for operation in the event of a loss of one or more Class 1E 120-vac vital instrument buses and the associated inverters. The AP1000 Class 1E buses have no tie breakers

- Generic Safety Issue 49, interlocks and LCOs for Class 1E tie breakers. Based on the historical background, this issue is not applicable to the AP1000 design. There are no tie breakers between the four class 1E divisions.

- Generic Safety Issue A-30, adequacy of safety-related dc power supplies. The AP1000 incorporates the following recommended enhancements:

  – The Class 1E dc distribution system design is in accordance with the guidelines of IEEE Standard 384 and Regulatory Guide 1.75.

  – Four separate divisions of Class 1E dc power are provided.

The AP1000 design provides additional testing capability through the installed spare battery bank with one installed battery charger. The spare battery bank permits frequent full-component testing without compromising plant availability. Battery equalization can be performed off-line. The battery and battery charger can be tested and maintained separately.

Issue 130 Essential Service Water Pump Failure at Multiple Plant Sites

Discussion:

Generic Safety Issue 130 addresses the use of shared or cross-connected essential service water systems at sites with two or more reactor plants. During some situations the crosstied pumps may not be available for accident mitigation operations.

AP1000 Response:

The AP1000 is a single, independent plant that does not share or cross-tie systems or components with another plant. See Section 1.2 for a general description of the plant. This issue is not applicable to the AP1000.

Issue 135 Integrated Steam Generator Issues

Discussion:

Generic Safety Issue 135 was initiated to provide an integrated work plan for the resolution of steam generator issues including steam generator overfill consequences, water hammer, and eddy current testing. The issue was divided into the following four tasks:

1. Assessing current capabilities of eddy current testing and developing recommendations.

2. Reviewing SGTR results and conclusions to develop regulatory analysis supporting Standard Review Plan changes.

3. Reassessing SGTR associated issues including radiological, design basis, tube integrity, procedures, and RCS pressure control.

4. Reviewing the effects of water hammer, overfill and water carryover.

The results of the tasks will provide the staff with a basis to develop a position on offsite dose, operator action, tube integrity, water hammer, and valve operability.

AP1000 Response:

The AP1000 design features are discussed below.

TASK 1: Appendix 1A identifies the level of conformance with Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes." As detailed in Appendix 1A, the AP1000 conforms with the regulatory guidance except where state-of-the-art advances have enhanced inservice inspection techniques. Further, as specified in Subsection 5.4.2.5, the steam generators permit access to tubes for inspection and/or repair or plugging, if necessary, per the guidelines described in Regulatory Guide 1.83. The AP1000 steam generator includes features to enhance robotics inspection of steam generator tubes without manned entry of the channel head.

TASK 2: Subsection 15.6.3.1.4 discusses anticipated operator recovery actions and the effects of those actions in the mitigation of a steam generator tube rupture (SGTR). As discussed in Subsection 15.6.3.2, the AP1000 incorporates automatic steam generator overfill protection. The details of the design are provided in Subsection 15.6.3.2, with the control logic provided in Section 7.2.

TASK 3: The following sections provide pertinent details on SGTR issues.

- Reassessment of radiological consequences: Subsection 15.6.3 provides details of the scenario, analysis assumptions, and results.
- Re-evaluation of design basis SGTR: The design basis SGTR evaluated on the AP1000 is discussed in Subsection 15.6.3, providing details of the scenario, analysis assumptions and results.
- Supplemental Tube Inspections: See Subsection 5.4.2.5, Appendix 1A, Regulatory Guide 1.83.
- Denting criteria: Subsection 5.4.2.4.3 provides a discussion of steam generator design and tubing compatibility with secondary coolants.
- Improved accident monitoring and reactor vessel inventory measurement: Section 7.5 discusses the safety related display information.
- Reactor coolant pump trip: Subsection 7.3.1.2.5 discusses reactor coolant pump trip.
- Control room design: Sections 7.5 and 18.8 discuss the control room design and design process.
- Emergency operating procedures: Section 18.9 addresses the development of emergency operating procedures.
- Organizational responses: Chapter 13 identifies the requirements for organizational responses.
- Reactor coolant pressure control: Subsection 7.7.1.6 addresses primary system pressure control.

TASK 4: Steam generator overfill, water carryover and water hammer are addressed as discussed in Subsection 15.6.3.2, with the control logic provided in Section 7.2.

Issue 142 Leakage Through Electrical Isolators in Instrumentation Circuits

Discussion:

Generic Issue 142 addresses the susceptibility to leakage of isolation devices between safety- and nonsafety-related electrical systems. The NRC requires that licensees identify isolation devices in instrumentation circuits that are potentially susceptible to electrical leakage, define and perform an inspection and test program, replace failed or unacceptable isolators, and implement an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems.

AP1000 Response:

The use of isolation devices in the AP1000 Instrumentation and Control Architecture is described in Subsections 7.1.2.10, "Isolation Devices," 7.7.1.11, "Diverse Actuation System," and WCAP-15776 (Reference 70), Section 3.9, “Conformance to the Requirements to Maintain Independence Between Safety Systems and Other Interconnected Equipment (Paragraph 5.6.3.1 of IEEE 603-1991).” As stated in WCAP-15776, Section 3.9, the isolation devices are tested to conform to requirements. This testing meets the requirement for an inspection and test program and identifies those devices that are potentially susceptible to electrical leakage. Implementation of an annual program to inspect and test all electronic isolators between Class 1E and non-Class 1E systems is the responsibility of the

Combined License holder. The use of fiber-optic data links eliminates electrically conductive paths between receiving and transmitting terminals, and eliminates the potential for electrically generated noise caused by leakage through an isolator. These communication links also use extensive testing and error checking to minimize erroneous transmissions. These data links are described in Subsection 7.1.2.8, "Communication Functions." In addition, electromagnetic design, testing, and qualification is performed as described in WCAP-15776, Section 2.6, "Design Basis: Range of Conditions for Safety System Performance (Paragraph 4.7 of IEEE 603-1991.)"

Issue 143 Availability of Chilled Water System and Room Cooling

Discussion:

This issue relates to the need to maintain air cooling systems in some rooms containing safety-related system components.

AP1000 Response:

This issue does not apply to the AP1000. The AP1000 does not rely on active safety systems to provide safe shutdown of the plant. A total loss of HVAC systems will not prevent a safe shutdown.

Issue 153 Loss of Essential Service Water in LWRs

Discussion:

This issue is related to the reliability of essential service water and the failure of such systems due to fouling mechanisms, ice effects, design deficiencies, flooding, multiple equipment failures, and personnel errors. This issue has been the subject of a number of generic communications from the NRC staff.

AP1000 Response:

This issue is not applicable to the AP1000. The AP1000 does not rely on the service water and component cooling water systems to provide safety-related safe shutdown.

Issue 163 Multiple Steam Generator Tube Leakage

Discussion:

This issue identifies a safety concern associated with potential multiple steam generator tube leaks triggered by a main steam line break outside containment that cannot be isolated. This sequence of events could lead to core damage due to the loss of all primary system coolant and safety injection fluid in the refueling water storage tank.

AP1000 Response:

The AP1000 plant response to a main steam line break (MSLB) scrams the reactor automatically and removes decay heat via the intact generator or the PRHR heat exchanger. If the MSLB is not isolated, the RCS will continue to lose coolant after shutdown through leaking steam generator tubes; the plant responds to the scenario as a small LOCA. The core makeup tanks drain and produce a low level signal. The plant protection and monitoring system depressurizes the RCS via the automatic depressurization system (ADS). The core remains covered throughout the scenario. Once the RCS is depressurized, the much lower reactor coolant system pressure stops the water loss through the leaking steam generator tubes. Therefore, no long-term core uncovery is expected.

Issue 168 Environmental Qualification of Electrical Equipment

Discussion:

This issue is related to the effects of cable aging and whether the licensing basis for older plants should be reassessed or enhanced in connection with license renewal, or whether they should be reassessed for the current license term.

AP1000 Position:

This issue applies to operating plants and does not apply to the AP1000.

Issue 185 Control of Recriticality Following Small-Break LOCAs in PWRs

Discussion:

This issue is related to the potential for large reactivity transients, including prompt criticality, and significant heat generation resulting from natural circulation flow of unborated water formed in steam generators following small-break LOCAs.

AP1000 Position:

This scenario is not a safety concern for the AP1000 because of the passive safety systems designed to mitigate the consequences of a LOCA. Specifically, the automatic depressurization system operates to reduce primary system pressure and, thus, prevents significant heat transfer in the steam generators. Consequently, the steam generators should not generate any significant amount of boron-free condensate via reflux condensation over an extended period during a LOCA event. In the AP1000 design, the steam generator functions as a "heat source" as the RCS depressurizes, rather than a "heat sink" as it does in conventional PWR designs. Therefore, the differential temperature across the primary and secondary side of the generators is such that steam from the reactor will not condense on the tubes.

Another important design feature of the AP1000 that reduces the significance of this event is the elimination of the loop seal in the inlet to the reactor coolant pump. By elimination of the crossover leg piping, a large volume of boron-free condensate cannot collect in the loop piping. Thus, restart of the reactor coolant pumps following a LOCA will not result in a large slug of unborated water entering the core.

Post-LOCA, the PRHR heat exchanger can act as a heat sink and potentially could be a source of unborated water post-LOCA. However, condensate from the PRHR heat exchanger outlet mixes with the borated injection from the core makeup tanks and accumulators, and adequately mixes in the reactor vessel downcomer to prevent post-LOCA boron dilution. Long-term boration of the core is provided by the injection from the borated IRWST.

Issue 186 Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants

Discussion:

This issue concerns licensees operating within the regulatory guidelines of Generic Letter 85-11 that may not have taken adequate measures to assess and mitigate the consequences of dropped heavy loads.

FSAR Position:

There are no planned heavy load lifts outside those already described in the DCD. However, over the plant life there may be occasions when heavy loads not presently addressed need to be lifted (i.e. in support of special maintenance/ repairs). For these occasions, special procedures are generated that address the activity. Further discussion is provided in Subsection 9.1.5.3.

Issue 189 Susceptibility of Ice Condenser and Mark III Containments to Early Failure From Hydrogen Combustion During a Severe Accident Description

Discussion:

This issue concerns the early containment failure probability for ice condenser and BWR MARK III containments given the relatively low containment free volume and low containment strength in these designs.

FSAR Position:

The AP1000 design does not have an ice condenser containment or a Mark III containment. Therefore, this issue is not addressed here.

Issue 191 Assessment of Debris Accumulation on PWR Sump Performance

Discussion:

This issue addresses new contributors to debris and possible blockage of PWR sumps. Generic Letter (GL) 2004-02 (Reference 2), issued in September 2004, identified actions that utilities must take to address the sump blockage issue. The NRC position is that plants must be able to demonstrate that debris transported to the sump screen after a LOCA will not lead to unacceptable head loss for the recirculating flow. For the AP1000, this requirement is interpreted as demonstrating that debris transported to recirculating screens will not significantly impede flow through the PXS and will not adversely affect the long-term operation of the PXS.

AP1000 Position:

The AP1000 Nuclear Power Plant uses natural recirculation for cooling the core following a loss of coolant accident (LOCA).

Screens are provided in strategic areas of the plant to remove debris that might migrate with the water in containment and adversely affect core cooling. Accordingly, it must be assured that the screens themselves are not susceptible to plugging.

Technical report APP-GW-GLR-079 (Reference 71) evaluates the potential for debris to plug the AP1000 screens consistent with Regulatory Guide 1.82 Revision 3 and subsequently issued Nuclear Regulatory Commission guidance. The evaluation considers the various potential contributors to screen plugging. It considers debris that could be produced by a LOCA as well as resident fibers and particles that could be present in containment prior to the LOCA. It considers the AP1000 containment design, equipment locations, and containment cleanliness program. The evaluation uses debris characteristics based on sample measurements from operating plants and evaluates the generation of chemical precipitants considering materials used inside the AP1000 containment, the post-accident water chemistry, and applicable research and testing. The AP1000 screen designs are acceptable.

Issue 191 Assessment of Debris Accumulation on PWR Sump Performance (REV. 1)

Discussion:

Results of research on BWR ECCS suction strainer blockage identified new phenomena and failure modes that were not considered in the resolution of Issue A-43. In addition, operating experience identified new contributors to debris and possible blockage of PWR sumps, such as degraded or failed containment paint coatings.

FSAR Position:

The design aspects of this issue are addressed by the DCD. The protective coatings program controls the procurement, application, inspection, and monitoring of Service Level I and Service Level III coatings with the quality assurance features discussed above. The protective coatings program complies with Regulatory Guide 1.54, and is controlled and implemented by administrative procedures. The program is discussed in Subsection 6.1.2.1.6.

Administrative procedures implement the containment cleanliness program. Implementation of the program minimizes the amount of debris that might be left in containment following refueling and maintenance outages. The program is consistent with the containment cleanliness program used in the evaluation discussed in Subsection 6.3.8.2. The program is discussed in Subsection 6.3.8.1.

Issue 196 Boral Degradation

Discussion:

The issue specifically addresses the use of Boral in long-term dry storage casks for spent reactor fuel.

FSAR Position:

Long-term dry storage casks for spent reactor fuel are not used and therefore this issue is not addressed here.

1.9.4.2.4 Human Factors Issues

These issues were outlined in the Human Factors Program Plan and are documented in NUREG-0985, Revision 1. The Human Factors Program Plan includes the human factors tasks required to address NUREG-0660.

HF4.4 Guidelines for Upgrading Other Procedures

Discussion:

The need was evaluated to develop technical guidance for use in upgrading normal operating procedures and abnormal operating procedures, similar to what the NRC staff completed for emergency operating procedures. NUREG-0933 classified this item as resolved with no new requirements.

AP1000 Response:

The process to manage the development, review and approval of AP1000 Normal Operating, Abnormal Operating, Emergency Operating, Refueling and outage planning, Alarm response, Administrative, Maintenance, Inspection, Test and Surveillance Procedures as well as the procedures which address the operation of post-72 hour equipment is delineated in APP-GW-GLR-040 (Reference 72).

Writer's Guidelines have been developed which control the preparation of Normal Operating Procedures and Two-Column Format Procedures. The Writer's Guidelines establish programmatic guidelines. The criteria and methodology for procedure development is described in this technical report and in Westinghouse Writer's Guidelines, and Human Factors-related procedures have been developed in accordance with these criteria/guidelines.

HF5.1 Local Control Stations

Discussion:

Human Factors Issue 5.1 addresses the need to develop additional guidance for the design of local control stations.

AP1000 Response:

The AP1000 local control stations are designed using the same human factors engineering (HFE) design process as is used for the main control room (MCR). The human factors engineering design process is described in Chapter 18. Section 18.8 provides a description of the human system interface (HSI) design element of the overall design process. As part of the human system interface design process, design guidelines for each interface, such as workstation displays, are generated. These guidelines are used when designing the respective interface and control stations. This provides consistency of human system interface design, including local control stations, with the main control room.

HF5.2 Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation

Discussion:

Human Factors Issue 5.2 addresses review criteria for human factors aspects of advanced controls and instrumentation.

AP1000 Response:

Chapter 18 describes the human factors engineering (HFE) program for the AP1000. Section 18.4 includes a description of the Functional Requirements Analysis and Allocation (element 3) for the AP1000. The objective of this allocation process is to define the AP1000 safety function requirements and allocate functions between the human and the machine appropriately. Section 18.8 also presents the implementation plan for the human system interface (HSI) design. This description of the human system interface design process includes the development of design guidelines, the execution of man-in-the-loop concept testing, review of human system interface design, and the use of a full-scale mockup.

The AP1000 human system interface (HSI)/man-machine interface (MMI) includes the following resources:

- Alarm system
- Computerized Procedure System
- Plant Information System
- Qualified Data Process System (QDPS)
- Controls (dedicated and soft)
- Wall Panel Information System (WPIS)

The implementation plan for the design of each of these human system interfaces (HSI design) is described in Section 18.8. The mission statements and high-level information for each of these resources is also provided in Section 18.8. The plant information system provides display at the operators workstation. The qualified data process system provides qualified (Class 1E) displays to operator, located at the dedicated safety panel. The alarm system provides alarm overviews which are integrated into the wall panel information system and it provides alarm support displays at the operator’s workstation. Alarms are integrated into the workstation displays. There will be a navigational link from an alarm support display for a specific alarm to its associated alarm response procedure as presented to the operator by the computerized procedure system. Design guidelines for each human system interface is developed as part of the human system interface design (as described in Section 18.8). These design guidelines are developed from existing industry guidelines and considerations specific to the technology planned for the human system interface. Human factors engineering specialists are part of the human factors engineering/ man-machine interphase design team (Section 18.2) and will be involved in the development of the design guidelines.

1.9.5 Advanced Light Water Reactor Certification Issues

This subsection addresses the advanced light water reactor issues identified by the NRC in SECY-90-016 (Reference 29), in the February 27, 1992 NRC letter from D. M. Crutchfield to E. E. Kintner (Reference 30).

1.9.5.1 SECY-90-016 Issues

The following issues were outlined in SECY-90-016 (Reference 29).

1.9.5.1.1 Advanced Light Water Reactor Public Safety Goal

NRC Position:

Based on current regulatory guidance, including the NRC Severe Accident Policy Statement, Standardization Policy Statement, and Safety Goal Policy Statement, it is expected that any new standard plant design will result in a higher level of severe accident safety than current plant designs. This is achieved by improving safety and by striking a balance between accident prevention and mitigation.

The overall objective of the public safety goal is to significantly reduce or eliminate the likelihood of known major safety issues.

The safety goals approved by the NRC in the Staff Requirements Memorandum to SECY-90-016 (Reference 31) are as follows:

- The mean core damage frequency target for each design should be less than 1.0×10^{-4} per reactor year.
- The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation, where a large release is defined as one that has a potential for causing an early offsite fatality.

AP1000 Response:

The AP1000 level 1, 2, and 3 PRA evaluations for both internal and external events (excluding seismic events) demonstrate conformance with the NRC safety goals. The AP1000 PRA evaluates shutdown events and provides additional information and specific results.

1.9.5.1.2 Use of Physically-Based Source Term

NRC Position:

As noted in SECY-95-172 (Reference 57), the NRC plans to use the accident source term model from NUREG-1465 (Reference 58). This source term model provides a physically based approach to modelling of activity releases from the reactor core to the containment in the event of a core degradation accident. As discussed in SECY-94-302 (Reference 59), for the design basis accident, release of activity from the core will not be assumed to extend beyond the in-vessel release phase.

In calculating the radiological consequences of accidents, as stated in Reference 57, the NRC intends to use the model presented in SECY-94-194 (Reference 60) which identifies the proposed changes to 10 CFR Parts 50 and 100. The pertinent features that will be applied to the determination of accident radiological consequences are:

- In place of thyroid and whole body dose limits, dose limits are specified as total effective dose equivalent (TEDE). The offsite dose limits of 25 rem whole body and 300 rem thyroid are replaced by a limit of 25 rem TEDE. The dose limit for the control room operators (currently identified in SRP Section 6.4 as 5 rem whole body, 30 rem thyroid, and 30 rem beta skin) is replaced by 5 rem TEDE which is consistent with GDC 19.

- Instead of calculating the site boundary dose over the first two hours of the accident, the dose is to be calculated for the two hour interval over which the highest dose would be calculated.

AP1000 Response:

The AP1000 radiological consequence analysis utilizes the accident source term provided in Regulatory Guide 1.183.

1.9.5.1.3 Anticipated Transients Without Scram (ATWS)

NRC Position:

This former unresolved safety issue was resolved with the issuance of Rule 10 CFR 50.62. Requirements for currently operating pressurized water reactors include diverse reactor trip (except for Westinghouse plants) and diverse actuation of auxiliary feedwater and turbine trips.

The Staff Requirements Memorandum to SECY-90-016 (Reference 31) approved the requirement for diverse reactor trip systems for evolutionary advanced light water reactors. However, it added that if the applicant can demonstrate that the consequences of an anticipated transient without reactor trip are acceptable, the NRC should accept the demonstration as an alternative to the diverse reactor trip system.

AP1000 Response:

The AP1000 complies with the current rules on an anticipated transient without reactor trip as specified in 10 CFR 50.62.

The AP1000 design includes the following design features aimed at minimizing the probability of occurrence of an anticipated transient without reactor trip and at mitigating the consequences if it occurs.

- The design of the protection and safety monitoring system is highly reliable, using a two out of four coincidence logic and featuring continuous diagnostic testing. The system incorporates fail-safe features to the extent practical. It is designed to generate a reactor trip signal and to generate an actuation signal for most engineered safety features components when protection system failures occur.

- For a reactor trip, the switchgear consists of eight circuit breakers arranged in a two out of four matrix located in two separate cabinets. The trip is implemented by undervoltage trip attachments and diverse shunt trip devices on the circuit breakers. To initiate a reactor trip, power is interrupted to the undervoltage trip attachment, while the shunt trip attachment is energized. Either device trips the breaker. The eight-breaker configuration permits testing of the reactor trip breakers without the use of auxiliary bypass breakers.

- The reactor trip switchgear can be actuated manually from the main control room by reactor trip switches hard-wired to the shunt trip attachment and undervoltage coils for each reactor trip breaker. In addition, it is possible to manually initiate a reactor trip from the main control room by turning off the motor-generators that provide power for control rod operation.

- A nonsafety-related diverse actuation system is included in the AP1000 design. The diverse actuation system inserts control rods by de-energizing the field windings of the control rod motor-generators.

- The diverse actuation system trips the turbine and diversely actuates selected other engineered safeguards functions. Additional details of the diverse actuation system are included in Section 7.7.

Section 15.8 describes the evaluation of an anticipated transient without reactor trip.

1.9.5.1.4 Midloop Operation

NRC Position:

Loss of decay heat removal function has occurred on a number of occasions in operating plants. In response to these events, the NRC issued Generic Letter 87-12 requesting that operating plants provide information regarding mid-loop operation. Generic Letter 88-17 requested additional information and provided guidance to operating utilities. Subsequent NRC evaluations have indicated that loss of decay heat removal during midloop operation may contribute significantly to public risk.

It is the NRC position that for future plants, conformance with Generic Letter 88-17 is insufficient, and additional hardware features should be incorporated into the design.

The Staff Requirements Memorandum to SECY-90-016 (Reference 31) approved the proposed NRC position, with the following four additional recommendations made by the ACRS:

- Design provisions to help ensure continuity of flow through the core and residual heat removal system with low liquid levels at the junction of the decay heat removal system suction lines and the reactor coolant system

- Provisions to ensure availability of reliable systems for decay heat removal

- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the decay heat removal system suction lines and the reactor coolant system

- Provisions for maintaining containment closure or for rapid closure of containment openings.

AP1000 Response:

The following features are incorporated into the design of the reactor coolant system and the normal residual heat removal system for continued performance of the residual heat removal function during midloop operation:

- The layout of the reactor coolant system hot leg piping and the steam generator channel head is such that installation of the nozzle dams can be performed with an 80 percent level in the hot leg piping. This is about 9 inches above the actual hot leg piping midplane elevation. (The hot leg piping has a 31-inch inside diameter.)

- A specially designed vortex breaker is used for the normal residual heat removal system suction nozzle. This vortex breaker connects vertically to the bottom of the hot leg piping. The normal residual heat removal system suction piping is connected to the bottom of this vortex breaker. With the vortex breaker, the amount of air entrainment remains below 10 percent unless the hot leg is essentially drained. Therefore, the potential for a loss of normal residual heat removal system flow and damage to the normal residual heat removal pump is substantially reduced.

- The normal residual heat removal pump suction piping is designed to be self-venting by sloping the lines continuously upward from the pump to the hot leg connection at the vortex breaker. If the pump should stop during midloop operation, any air bubbles present in the pump or suction piping are vented back up through the suction line to the water surface in the hot leg. This feature allows the operator to rapidly restart the pump with an air-free suction line.

- The normal residual heat removal pumps are designed to minimize cavitation and other adverse conditions when operating with minimal subcooling of the reactor coolant. Specifically, the plant piping layout configuration (such as piping elevations and routing) and the available and required pump net positive suction head characteristics allow the normal residual heat removal pumps to be started and operated at their full design flow rates, with saturation conditions in the reactor coolant system (associated with boiling in the reactor vessel). Therefore, the normal residual heat removal system is readily restored after a temporary loss of decay heat removal.

- The core makeup tanks, accumulators, and the in-containment refueling water storage tank are isolated, but can be manually actuated during midloop operations. In addition, the in-containment refueling water storage tank is automatically actuated on a sustained loss of shutdown decay heat removal. This arrangement provides a reliable water source for maintaining the reactor coolant system inventory that is either automatically or manually actuated.

- Redundant narrow-range level instrumentation indicates the reactor coolant system water level between the bottom of the hot leg and the top of the steam generator inlet elbow. Indication and low level alarms are provided in the main control room. In addition, this instrumentation actuates the in-containment refueling water storage tank makeup.

- Wide-range pressurizer level instrumentation used during cold plant operations is expanded to the bottom of the hot legs. This provides a continuous level indication in the main control room, from the normal level in the pressurizer to the range of the two narrow-range hot leg level instrumentation.

- Normal residual heat removal system heat exchanger discharge flow instrumentation provides main control room indication of return flow to the reactor vessel. A low-flow alarm alerts the operator to a decrease in normal residual heat removal system return flow from either heat exchanger.

- The drain-down of the reactor coolant system to the midloop operating level and the subsequent reactor coolant system inventory control during midloop operation are performed by the operator from the main control room.

The plant design precludes the need to locally coordinate actions in the containment with the main control room operators to control the reactor coolant system drain-down rate and level.

- Reactor coolant system hot leg wide range temperature instruments are provided in each hot leg. The orientation of the wide range thermowell-mounted resistance temperature detectors enable measurement of the reactor coolant fluid in the hot leg when in reduced inventory conditions. In addition, at least two incore thermocouple channels are available to directly measure the core exit temperature during midloop residual heat removal operation. These two thermocouple channels are associated with separate electrical divisions.

- The automatic depressurization system first-, second-, and third-stage valves, connected to the top of the pressurizer, are open whenever the core makeup tanks are blocked during shutdown conditions while the reactor vessel upper internals are in place. This provides a vent flow path to preclude pressurization of the reactor coolant system during shutdown conditions when decay heat removal is lost. This also allows the in-containment refueling water storage tank to automatically provide injection flow if it is actuated on a sustained loss of decay heat removal.

Administrative controls require containment closure capability in modes 5 and 6, during reduced inventory operations, and when the upper internals are in place. Containment closure capability is defined as the capability to close the containment prior to core uncovery following a loss of the normal decay heat removal system (that is, normal residual heat removal system). The containment design also includes penetrations for temporary cables and hoses needed for shutdown operations. These penetrations are isolated in an emergency.

In addition to these design features, appropriate procedures are defined to guide and direct the operator in the proper conduct of midloop operation and to aid in identifying and correcting abnormal conditions that might occur during shutdown operations.

1.9.5.1.5 Station Blackout

NRC Position:

The NRC has issued NUREG-0649 (Reference 34), NUREG-1032 (Reference 35), and NUREG-1109 (Reference 36) to address the unresolved safety issue of station blackout (USI-44). See Subsection 1.9.4 for a discussion of USI-44.

To resolve this issue, the NRC published 10 CFR 50.63 and Regulatory Guide 1.155, which establish new requirements so that an operating plant can safely shut down following a loss of all ac power. SECY-94-084 (Reference 67), discusses station blackout for passive plants.

AP1000 Response:

The AP1000 is in conformance with the NRC guidelines for station blackout.

The AP1000 design minimizes the potential risk contribution of station blackout by not requiring ac power sources for design basis events. Safety-related systems do not need nonsafety-related ac power sources to perform safety-related functions.

The AP1000 safety-related passive systems automatically establish and maintain safe shutdown conditions for the plant following design basis events, including an extended loss of ac power sources. The passive systems can maintain these safe shutdown conditions after design basis events, without operator action, following a loss of both onsite and offsite ac power sources. Subsection 1.9.5.4 provides additional information on long-term actions following an extended station blackout beyond 72 hours.

The AP1000 also includes redundant nonsafety-related onsite ac power sources (diesel-generators) to provide electrical power for the nonsafety-related active systems which provide defense in depth.

AP1000 design features that mitigate the consequences of a station blackout are as follows:

- A full-load rejection capability to reduce the probability of loss of onsite power
- Safety-related passive residual heat removal heat exchanger
- Safety-related passive containment cooling
- Bleed and feed capability, using the safety-related automatic depressurization system in conjunction with the water available from the core makeup tanks, the accumulators, and the in-containment refueling water storage tank
- Class 1E batteries sized for 72 hours of operation under station blackout conditions
- Nonsafety-related reserve auxiliary transformers to provide power to selected ac power systems
- A nonsafety-related ac power system that includes two diesel-generators that automatically start on loss of offsite power
- An automatic nonsafety-related load-sequencing circuit that starts the following redundant nonsafety-related equipment after a loss of offsite power, once the associated diesel-generator is started:
  - Startup feedwater pump
  - Component cooling water pump
  - Service water pump
- Reactor coolant pumps without shaft seals
- Passive cooling for the rooms containing equipment assumed to operate during station blackout conditions (the protection and safety monitoring system cabinet rooms and the main control room) so that this equipment continues to operate. (Section 6.4 provides additional information.)

Training and procedures to mitigate a 10 CFR 50.63 “loss of all alternating current power” (or station blackout (SBO)) event are implemented in accordance with Sections 13.2 and 13.5, respectively. As recommended by NUMARC 87-00 (Reference 201), the SBO event mitigation procedures address response (e.g., restoration of onsite power sources), ac power restoration (e.g., coordination with transmission system load dispatcher), and severe weather guidance (e.g., identification of actions to prepare for the onset of severe weather such as an impending tornado), as applicable. The AP1000 is a passive design and does not rely on offsite or onsite ac sources of power for at least 72 hours after an SBO event, as described above. In addition, there are no nearby large power sources, such as gas turbine or black start fossil fuel plant, that can directly connect to the station to mitigate the event.

Restoration from an SBO event will be contingent upon ac power being made available from any one of the transmission lines described in Section 8.2 or any one of the standby diesel generators.

1.9.5.1.6 Fire Protection

NRC Position:

Current fire protection criteria are contained in GDC 3 and 10 CFR 50.48, guidelines for compliance with these criteria are provided in the Standard Review Plan, Section 9.5.1, including Branch Technical Position CMEB 9.5-1. Reference 9 identifies the following enhancements:

- Alternative, dedicated shutdown capability for main control room fires.
- Safe shutdown capability required for a fire in any other fire area, without reliance on any equipment in that area or re-entry into that area for repairs or for performance of operator actions.
- Fire protection for redundant shutdown systems in the reactor containment building must be provided to ensure, to the extent practicable, that on shutdown the division will be free of fire damage.
- Migration of smoke, hot gases, or fire-suppressant chemicals into other applicable fire areas must be minimized by design to prevent any adverse impact on safe shutdown capability, including operator actions.

SECY-98-161 (Reference 66) presents the results of the NRC review of the AP1000 Fire Protection System.

AP1000 Response:

Enhanced fire protection has been one of the goals of the AP1000 design. The following physical separation philosophy is used:

Outside Containment:

- Within the nuclear island, redundant divisions of safety-related equipment outside containment are located in safety-related areas separated from each other and from other areas in the plant by fire barriers with a minimum fire resistance rating of 3-hours to provide that safe shutdown can be achieved. Since most safety-related mechanical equipment is located inside containment, this applies primarily to the protection and safety monitoring system and the Class 1E dc and UPS system.
- Each safety-related area is provided with ventilation isolation provisions at the fire barrier boundaries to minimize the migration of smoke, hot gasses, or fire suppressant chemicals into other safety-related areas. Fiber-optic cables are used to provide communication between redundant protection and safety monitoring divisions.
- Exceptions to the use of three-hour fire barriers outside containment are made only in cases where physical separation conflicts with other requirements or where the equipment is not clearly division oriented, such as the main control room, the remote shutdown room, the main steam tunnel, and the passive containment cooling system valve room.

Inside Containment:

- The containment is a single fire area. Separation by three-hour fire barriers inside containment is not practical due to issues of hydrogen venting, compartment pressure equalization, and during high-energy line breaks and for system functionality. To the extent practical, separation is provided between redundant safety-related equipment.

- Separation between redundant safety-related equipment is accomplished by using existing structural walls. Where this is not possible, other methods are used, such as physical separation with no intervening combustibles.

- To the extent practical, the containment is split into two different fire zones for the purpose of routing of protection and safety monitoring system cabling and electrical power cabling. Divisions A and C cabling is routed below the operating deck, while Divisions B and D cabling is routed above the operating deck. Additional separation is provided by existing floors and walls and by the physical separation of cabling runs. Protection for the primary input sensors and the final actuation devices is accomplished by the physical separation of the various sensors and components using existing containment walls as barriers.

- The in-containment fire area contains reduced combustible material due to the use of sealless reactor coolant pump motors that do not use oil lubrication and due to strict combustible material limitations.

Main Control Room:

- Functionality requirements dictate that the main control room be a single fire zone. Features are included in the main control room to:

  – Reduce the probability of fire initiation
  – Reduce the likelihood of fire spreading
  – Increase the probability of fire detection
  – Effectively mitigate the effects of a fire

- In the event of main control room evacuation, safe shutdown conditions are established and maintained using the remote shutdown workstation.

See Appendix 9A.3 for information on the main steam tunnel and the passive containment cooling system valve room. See Subsection 9.5.1 and Appendix 9A for additional information.

1.9.5.1.7 Intersystem LOCA

NRC Position:

Overpressurization of low-pressure piping systems due to reactor coolant system boundary isolation failure could result in rupture of the low-pressure piping outside containment. This may result in a core melt accident with an energetic release outside the containment building that could cause a significant offsite radiation release.

It is the NRC position that designing interfacing systems to withstand full reactor pressure is an acceptable means of resolving this issue. The Staff Requirements Memorandum to SECY-90-016 (Reference 31) added that consideration should be given to all elements of the low-pressure system (such as instrument lines, pump seals, heat exchanger tubes, and valve bonnets). For interfacing systems not designed to withstand full reactor coolant system pressure, it is necessary to provide leak testing capability for the pressure isolation valves, main control room position indication for de-energized reactor coolant system isolation valves, and high pressure alarms to alert control room operators when increasing reactor coolant system pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

AP1000 Response:

The AP1000 has incorporated various design features to address intersystem loss-of-coolant accident challenges. These design features result in very low AP1000 core damage frequency for intersystem loss-of-coolant accidents compared with operating nuclear power plants. The design features are primarily associated with the normal residual heat removal system and are discussed in Section 3 of WCAP-15993 (Reference 56) as well as Subsection 5.4.7. WCAP-15993 was prepared to document the evaluation of the AP1000 for conformance to the intersystem loss-of-coolant accident regulatory criteria identified in various NRC documents. See that document for additional information on conformance to intersystem loss-of-coolant accident regulatory criteria.

1.9.5.1.8 Hydrogen Generation and Control

NRC Position:

It is the NRC position that the likelihood of early containment failure from hydrogen combustion should be reduced. Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, advanced light water reactors should be designed to:

- Accommodate hydrogen equivalent to 100 percent metal-water reaction of the fuel cladding
- Limit containment hydrogen concentration to no greater than 10 percent

Further, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits, and given uncertainties in present analytical capabilities, advanced light water reactors should provide containment-wide hydrogen control (such as igniters or inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment.

AP1000 Response:

The AP1000 design includes mechanisms for monitoring and controlling hydrogen inside the containment. The containment hydrogen control system maintains hydrogen concentrations below 10 percent following the reaction of 100 percent of the zircaloy cladding.

Passive autocatalytic hydrogen recombiners control hydrogen concentration following design basis events. Nonsafety-related hydrogen igniters control rapid releases of hydrogen during and after postulated events with degraded core conditions or with core melt.

Sufficient vent area is provided for each subcompartment in the containment to prevent high local concentrations of hydrogen.

The containment air filtration system provides a capability to purge the containment atmosphere.

See Subsection 6.2.4 for additional information.

1.9.5.1.9 Core-Concrete Interaction - Ability to Cool Core Debris

NRC Position:

Containment integrity could be breached in the event of a severe accident in which the core melts through the reactor vessel, resulting in interaction between core debris and concrete, which can generate large quantities of hydrogen and other gases. It is the NRC position that sufficient reactor cavity floor space be provided to enhance debris spreading, and that a method for quenching debris in the reactor cavity be incorporated. The NRC staff has not formulated specific criteria for debris bed coolability and reviews each vendor's design to determine how they address the general criteria for debris spreading and quenching.

AP1000 Response:

The AP1000 design provides superior protection against core-concrete interaction by reliably depressurizing the reactor vessel and flooding the reactor cavity to cool the vessel and prevent debris from relocating from the vessel into the containment. Based on the DOE/ARSAP analysis of the thermal-hydraulics of in-vessel debris retention (see Section 19.39 and Appendix 19B as supported by Theofanous, T. G., et al., Reference 62) performed using the Risk Oriented Accident Analysis Methodology, the AP1000 has a large margin to reactor vessel failure in the depressurized, flooded cavity condition. This strategy eliminates the large uncertainties associated with ex-vessel debris relocation that could result in containment failure even while meeting the NRC criteria for debris coolability in the cavity.

In the event that cavity flooding fails, the floor area under the vessel provides debris spreading area to enhance the coolability of the debris. The AP1000 containment design drains the water from the reactor coolant system, core makeup tanks and accumulators to the reactor cavity to provide enough water to quench ex-vessel debris. The heat is ultimately removed from the containment via the passive containment cooling system, and the condensate is returned to the cavity to continue to provide cooling water to the debris bed.

1.9.5.1.10 High Pressure Core Melt Ejection

NRC Position:

Direct containment heating associated with the ejection of molten core debris, under high pressure, from the reactor vessel can result in a rapid addition of energy to the containment atmosphere. It is the NRC position that, pending completion of ongoing research, it is prudent to provide protection against this potential failure mode. This protection should include the following two aspects:

- Providing a rate of reactor coolant system depressurization to preclude molten core ejection and creep rupture of steam generator tubes
- Arranging the reactor cavity so that high-pressure core debris ejection resulting from reactor vessel failure does not impinge on the containment boundary

AP1000 Response:

The AP1000 design includes an automatic depressurization system that is redundant, diverse, independent of ac power sources, and automatically actuated. The automatic depressurization system can also be manually actuated. Any of the automatic depressurization system lines can sufficiently reduce the reactor coolant system pressure to help preclude direct containment heating. Subsection 5.4.6 and Section 6.3 provide additional information on the automatic depressurization system.

In addition, the reactor cavity region and lower containment of the AP1000 are designed to preclude transport of significant core debris to the upper containment in the unlikely event of a high pressure melt ejection scenario from the reactor vessel. This is a passive feature involving the geometric configuration of the reactor cavity lower containment. There is no direct pathway from the cavity to the upper compartment.

1.9.5.1.11 Containment Performance

NRC Position:

The NRC opinion is that because there are substantial uncertainties in core damage predictions, and because it is very important to maintain defense in depth, it is necessary that the containment boundary serve as a reliable barrier against fission product release for credible severe accident challenges. Hence, a containment performance criterion has been proposed by the NRC.

The objective of the containment performance criterion is to provide a leaktight barrier against radioactive releases for two distinct categories of severe accident challenges:

- Rapid energy release, hydrogen combustion, and initial release of stored reactor coolant system energy

- Slow energy release, including decay heat and noncondensible gas generation, due to core-concrete interaction

The NRC position is that the reactor containment boundary should serve as a reliable barrier against fission product release for credible severe accident challenges. A conditional containment failure probability of 0.1 should be used unless a deterministic containment performance goal can offer comparable protection.

An alternate deterministic criterion proposed in SECY-90-016 (Reference 29) states that "...The containment should maintain its role as a reliable leak tight barrier by ensuring that containment stresses do not exceed ASME service level C limits for a minimum period of 24 hours following the onset of core damage..."

This capability should, to the extent practical, be provided by the passive capability of the containment and any related passive design features. The NRC further believes that following this 24-hour period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

AP1000 Response:

The AP1000 design includes several features to minimize the potential for large fission product releases in the event of a severe accident. These features are aimed at both the prevention and the mitigation of severe accident phenomena that can threaten containment integrity. An adequate margin to containment performance is maintained.

The AP1000 containment is continuously cooled by natural air circulation outside the steel shell. During accident conditions, water drains on the outside of the containment vessel to increase heat transfer. The containment design best-estimate performance analysis alone shows that the maximum containment pressure reached maintains the containment shell stresses below the ASME Code Service Level C stress intensity limits, using a factor of safety of 1.5 for buckling of the top head.

Additionally, the probability of containment bypass scenarios is reduced by improved containment isolation, by designing to protect against interfacing system LOCAs, thereby reducing the associated core melt frequency, and by reducing the steam generator tube rupture core melt frequency.

The interfacing system LOCA core melt frequency is reduced by the use of several features, including effective leak testing of the normal residual heat removal system motor-operated isolation valves. A third valve is provided to the normal residual heat removal system suction line. It is a motor-operated valve located outside containment. This prevents inadvertently aligning the reactor coolant system to the normal residual heat removal system. The normal residual heat removal system design pressure is 900 psig. Therefore the ultimate rupture strength of the system prevents it from failing when exposed to the normal reactor coolant system operating pressure (2250 psia). See the position on intersystem LOCA for additional information on the normal residual heat removal system design against overpressurization.

Steam generator tube rupture core melt frequency is reduced by incorporating multiple levels of defense that are both redundant and diverse. The first level of defense relies on the use of nonsafety-related active systems and operator action. The second level of defense uses safety-related passive systems and equipment, such as the core makeup tanks and passive residual heat removal heat exchangers, without the safety-related automatic depressurization of the reactor coolant system. The third level of defense uses the redundant and diverse safety-related automatic depressurization system valves to depressurize the reactor coolant system and initiate low-pressure passive injection. Any of these levels of defense can prevent core damage during a steam generator tube rupture event.

Finally, containment isolation capabilities are substantially improved by reducing the number of penetrations and the number of open paths. Most of the open containment penetration lines use fail-closed valves for automatic isolation.

1.9.5.1.12 ABWR Containment Vent Design

This issue is specific to BWRs and PWRs with ice condenser containments. Therefore this issue does not apply to the AP1000 design.

1.9.5.1.13 Equipment Survivability

NRC Position:

Safety-related equipment used to mitigate design basis events is subject to a comprehensive set of criteria such as redundancy, diversity, environmental qualification, and quality assurance to provide reasonable assurance that they perform their intended functions, if needed. However, equipment used to mitigate the effects of severe accidents should not be treated in the same manner because of large differences in the likelihood of occurrence. There should be reasonable assurance that the equipment will operate in the severe accident environment for which they are intended and over the time span for which they are needed. However, equipment provided only for severe accident protection need not be subject to the 10 CFR 50.49, environmental qualification requirements, 10 CFR 50, Appendix B quality assurance requirements, and 10 CFR 50 Appendix A, redundancy and diversity requirements.

AP1000 Response:

The equipment used to mitigate severe accidents is identified in the AP1000 PRA evaluation report. Because of the nature of the passive safety features of the AP1000, there is very little equipment in this category. Equipment used to mitigate severe accidents is designed to survive the environmental conditions identified in the AP1000 PRA evaluation.

1.9.5.1.14 Operating Basis Earthquake (OBE)/Safe Shutdown Earthquake (SSE)

NRC Position:

Currently, 10 CFR 100 requires that the magnitude of the operating basis earthquake be at least one-half that of the safe shutdown earthquake. This forces the safety-related system design at some plants to be controlled by the operating basis earthquake, but the NRC agrees that the operating basis earthquake should not control the safety-related system design. Therefore, the NRC recommends eliminating the operating basis earthquake from the design of systems, structures, and components. Until final rulemaking is approved for 10 CFR 100, Appendix A, the elimination of the operating basis earthquake from the design of passive plants will require an exemption from current regulations, with acceptable supporting justification from the designer. The details of this process will be resolved with the NRC through the appropriate code-related activities or supplemental regulatory guidance.

AP1000 Response:

The operating basis earthquake is not used as a design basis for AP1000 safety-related structures, systems, and components. For safety-related equipment, the safe shutdown earthquake is used as the design basis. In specifying design criteria for this earthquake, consideration is given to lower magnitude earthquakes having a greater probability of occurrence, as well as to larger magnitude earthquakes having a lower probability.

Cyclic stresses due to earthquakes are included in the design of those components sensitive to fatigue. Analysis methods and allowable stresses provide margin for the design requirements for the safe shutdown earthquake. Sections 3.7 and 3.10 provide additional information.

1.9.5.1.15 In-Service Testing of Pumps and Valves

NRC Position:

Periodic testing according to ASME Code, Section XI is required to confirm operability of safety-related pumps and valves. The NRC believes that these testing requirements do not necessarily verify the capability of the components to perform their intended safety function. To address this concern, the NRC has issued Generic Letters 89-04 (Reference 38) and 89-10 (Reference 39), and has proposed rulemaking to extend in-service testing beyond code components and to demonstrate capability to perform safety functions. Reference 29 includes the following provisions to be applied to safety-related pumps and valves (not limited to only ASME Code Class 1, 2, or 3):

- Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves.
- Designs should incorporate provisions to test motor-operated valves under design basis differential pressure.
- Check valve testing should incorporate the use of advanced, nonintrusive techniques to address degradation and performance characteristics.
- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, nonintrusive techniques.

In June 1990, the NRC position was approved, additionally noting that due consideration should be given to the practicality of designing testing capability, particularly for large pumps and valves.

The NRC concluded that this was an issue for passive plant designs in SECY-94-084 (Reference 67), because the safety-related passive systems rely on the proper operation of equipment such as check valves and depressurization valves to mitigate the effects of transients.

AP1000 Response:

The AP1000 safety-related passive systems include the following design features:

- The AP1000 does not include any safety-related pumps.
- The motor-operated valve design is simplified by extending opening and closing times and by using simplified, conservative valve designs.
- Safety-related motor-operated valves are designed to be cycled with the plant at power.

- Features are included in the design to provide proper operational testing of the appropriate check valves, motor-operated valves, and air-operated valves, including flow and differential pressure testing during shutdown conditions.

Subsection 3.9.8.4 defines the responsibility for the in-service testing program for ASME Code Class 1, 2, and 3 valves.

Subsection 3.9.6 summarizes the requirements for the in-service testing program, including industry standards and NRC recommendations. The AP1000 system and valve designs generally allow implementation of the NRC recommendations in Generic Letters 89-04 and 89-10. Requirements for nonsafety-related pumps and valves that support the operation of systems that preclude unnecessary operation of the safety-related passive systems are outlined in Subsection 3.9.6.

The AP1000 in-service testing program provides for periodic testing of the safety-related passive system components. The safety-related passive system components and systems are designed to meet the intent of the ASME Code, Section XI, for in-service inspection.

The AP1000 is designed for the following basic types of in-service testing of safety-related components:

- Periodic functional testing of active components during power operation (such as cycling of specific valves)

- Periodic flow/differential pressure operability testing of active components

- Periodic leak testing of the containment isolation valves.

- Periodic system flow or heat transfer rate testing of passive safety-related injection or cooling features during plant shutdown

The passive system design includes specific features to support in-service test performance:

- Remotely operated valves can be exercised during plant operation.

- Level, pressure, flow, and valve position instrumentation is provided for monitoring passive system equipment during plant operation and testing.

- Permanently installed test lines and connections are provided for performance of the containment isolation valve leakage testing.

1.9.5.2 Other Evolutionary and Passive Design Issues

Other evolutionary and passive design issues were identified in Reference 30.

1.9.5.2.1 Industry Codes and Standards

NRC Position:

SECY-91-273 (Reference 40) discusses NRC concerns with the use of recently developed or modified design codes and industry standards that the ALWR vendors are using in applications, but that have not yet been reviewed by the NRC for acceptability. The NRC recommends using the newest codes and standards endorsed by the NRC in the review of passive design applications. Unapproved revisions to codes and standards will be reviewed on a case-by-case basis.

AP1000 Response:

When the AP1000 design is based on revisions of industry codes and standards later than those required by NRC regulation, such use is explicitly discussed in the appropriate section. Use of codes and standards later than those recommended in NRC guidance documents is also discussed in the appropriate section.

Appendix 1A discusses regulatory guide conformance. For those standards endorsed by regulatory guides and subsequently superseded by a more recent revision, when the later revision is used its use is discussed or indicated in Appendix 1A.

1.9.5.2.2 Electrical Distribution

NRC Position:

The Commission approved the recommendations in SECY-91-078 (Reference 41) for evolutionary plant designs to include the following:

1. An alternate power source for nonsafety-related loads unless design margins for loss of nonsafety-related loads are no more severe than turbine-trip-only events in current plants

2. At least one offsite circuit to each redundant safety division supplied directly from offsite power sources with no intervening nonsafety-related buses

The applicability of this issue to passive designs is discussed in SECY-94-084 (Reference 67).

AP1000 Response:

See the response to station blackout in Subsection 1.9.5.1.

1.9.5.2.3 Seismic Hazard Curves and Design Parameters

NRC Position:

To assess the seismic risk associated with an ALWR design, EPRI proposed the use of generic bounding seismic hazard curves for sites in the central and eastern United States. EPRI proposes that these curves be used in the seismic PRA. NRC regulations do not require performance of a seismic PRA to determine site acceptability.

The NRC has compared the proposed EPRI ALWR seismic hazard bounding curve for rock sites to hazard curves derived by Lawrence Livermore National Laboratories (LLNL) using historical earthquake methodology in NUREG/CR-4885 and to hazard curves generated by EPRI for the Seabrook site. The LLNL hazard curves are generally higher than the EPRI results for the same sites.

The proposed EPRI bounding curve is exceeded for accelerations below 0.1g and the NRC questions the adequacy of the proposed EPRI bounding curve at higher peak accelerations. The NRC concludes that the EPRI bounding hazards curve is nonconservative and also that its use in a seismic PRA assessment would underpredict the core damage frequency. Therefore, the EPRI curves are not sufficiently conservative for ALWR designer use.

The Combined License applicant must demonstrate that site-specific seismic parameters meet the certified design parameters, or a site-specific analysis will be required to confirm site acceptability.

AP1000 Response:

The AP1000 includes a seismic margin assessment performed in lieu of a seismic PRA. The seismic margin assessment follows the guidelines established in NUREG-1407 (Reference 42). This assessment demonstrates that the AP1000, located at a site having the most severe seismic inputs meeting the AP1000 site interface requirements, has a seismic risk comparable to that at existing nuclear power plants.

1.9.5.2.4 Leak-Before-Break

NRC Position:

GDC 4 provides the basis for the leak-before-break (LBB) analysis that has been approved for PWR primary piping, and the pressurizer surge, accumulator, and residual heat removal piping. In addition, it has been used for primary piping inside containment and for piping at least 6 inches nominal diameter and for both austenitic and carbon steel (clad with stainless) materials.

The NRC will evaluate the acceptability in ALWR designs, based on the justification provided by a deterministic fracture mechanics analysis submitted as part of the design. The NRC concluded that the analyses should be based on specific data, such as piping geometry, materials, and piping loads. However, the analyses may incorporate an initial set of bounding values and preliminary stress analysis results during the design certification phase. Subsequent verification of the preliminary analysis will be required.

The LBB approach has established certain limitations for excluding piping susceptible to failure from degradation mechanisms. In addition, the LBB introduced acknowledged inconsistency in the design basis, but the NRC published clarifications for the intended treatment of the containment, emergency core cooling systems, and environmental qualification in the LBB application.

The NRC position on LBB for the AP1000 is presented in SECY-95-172 (Reference 68).

AP1000 Response:

The AP1000 incorporates the leak-before-break approach for most high-energy lines inside containment that are 6 inches in diameter or larger. Detailed methodology and criteria are defined in Subsection 3.6.3 and are consistent with those accepted by the NRC on existing nuclear power plants.

1.9.5.2.5 Classification of Main Steam Line of Boiling Water Reactors (BWRs)

This issue is specific to BWRs and therefore does not apply to the AP1000 design.

1.9.5.2.6 Tornado Design Basis

NRC Position:

WASH-1300 (Reference 43) and Regulatory Guide 1.76 contain the current NRC regulatory position for design basis tornados. Based on a contractor review of Regulatory Guide 1.76, the NRC recommends a maximum tornado speed of 300 mph be used for design basis tornado for passive ALWR designs.

The tornado design basis requirements have been used in establishing structural requirements against effects not covered explicitly in review guidance such as Regulatory Guides or the SRP. The Combined License applicant will have to demonstrate that the design will also be sufficient to withstand other site hazards such as aviation crashes, nearby explosions, and explosion debris and missiles.

AP1000 Response:

The AP1000 is designed in accordance with the NRC recommendations for a maximum tornado wind speed of 300 mph, as described in Section 3.3. The AP1000 site interface defined in Chapter 2 provides information to evaluate other site hazards if appropriate.

1.9.5.2.7 Containment Bypass

NRC Position:

Reasonable efforts should be made to minimize the possibility of containment bypass leakage, and ALWR designs should allow for a certain amount of leakage in the containment design. The NRC is evaluating the need for containment spray for all ALWRs. The containment spray provides containment temperature and pressure suppression effects and scrubs the containment atmosphere of fission products, mitigating the effects on the fission product bypass distribution.

AP1000 Response:

Although the phenomenon described for this item is primarily applicable to BWRs, the AP1000 has a variety of design features that help to reduce the potential for containment bypass leakage.

The response to the containment performance issue in Subsection 1.9.5 provides additional information pertaining to various improvements that help to reduce containment bypass.

The safety-related passive containment cooling system design also contributes to the containment performance. The system includes multiple flow paths to provide cooling water for containment during severe accident conditions. The containment is also capable of successfully removing core decay heat with air-cooling alone.

The containment has a significantly reduced number of penetrations. The number of normally open containment penetrations is also reduced. The result is a low containment leak rate and a low probability of bypass.

The response to intersystem LOCA in Subsection 1.9.5.1 provides additional information pertaining to applicable AP1000 design features that reduce the potential for intersystem LOCA and the potential for containment bypass.

Improvements are made to the steam generator design, such as the use of improved tube materials and tube supports. These improvements reduce the potential for tube leakage, which contributes to a reduction in containment bypass. Subsection 5.4.2 provides additional information on the steam generator design.

During a steam generator tube rupture event, the safety-related passive core cooling system automatically mitigates the effects of the event, including automatic safety-related protection against steam generator overfill.

The safety-related passive core cooling system provides long-term pH control for the containment sump, which helps to reduce the levels of airborne radioactivity, thereby reducing the consequences of leakage from the containment. Section 6.3 includes additional information on the passive core cooling system.

The diverse actuation system includes containment isolation features to provide isolation for the most risk-significant containment penetrations. PRA Chapter 24 discusses the provisions for isolating risk significant containment penetrations.

The performance of the passive fission product removal process and minimal potential for containment bypass precludes the need for a safety-related containment spray system on AP1000.

1.9.5.2.8 Containment Leak Rate Testing

NRC Position:

SECY-91-348 (Reference 44) proposes changes to 10 CFR 50, Appendix J to allow an increased interval from 24 months to 30 months for Type C containment leakage rate tests, until rule change proceedings are completed.

AP1000 Response:

10 CFR 50 Appendix J has been revised since SECY-91-348 was issued. AP1000 type C testing and compliance with 10 CFR 50 Appendix J is discussed in Subsection 6.2.5.

1.9.5.2.9 Post-Accident Sampling System

NRC Position:

Regulatory Guide 1.97 and NUREG-0737 (Reference 45) provide guidance regarding the design of the post-accident sampling system. 10 CFR 50.34 required the capability to obtain and analyze samples from containment and the reactor coolant system that may contain TID-14844 source term radioactive materials, without exceeding specified radiation exposures. The analysis and quantification are required for certain specified radionuclides that are indicators of the degree of core damage, containment hydrogen, dissolved gases, chloride, and boron concentrations.

The NRC concluded that adequate capability for monitoring post-accident hydrogen is provided by the safety-grade containment hydrogen monitoring instrumentation.

The NRC requires sampling the reactor coolant system for dissolved hydrogen, chloride, and oxygen. The time for taking these samples can be extended to 24 hours after the accident.

The NRC requires sampling the reactor coolant system for boron and for activity measurements. The time for taking these samples can be extended to 8 hours after power operation for boron and 24 hours after power operation for activity measurements.

AP1000 Response:

The post-accident sampling system is a subsystem of the primary sampling system, described in Subsection 9.3.3.

The primary sampling system is designed to conform to the guidelines of the model Safety Evaluation Report on eliminating post-accident sampling system requirements from technical specifications for operating plants (Federal Register Volume 65, Number 211, October 31, 2000). The primary sampling system conforms with the most recent NRC position.

1.9.5.2.10 Level of Detail

NRC Position:

The Staff Requirements Memorandum for SECY-90-377 (Reference 47) provided guidance on the level of detail to be provided for a design certification application under 10 CFR 52. The guidance was that the application should include the information traditionally provided in a final safety analysis report, less the site-specific and as-procured information. This information should be supplemented by design inspections, tests, analysis, and acceptance criteria for those areas where the NRC is unable to make a final safety decision because of not having the site-specific information or the as-procured information, or because the technology is evolving so rapidly that it would be inappropriate to lock in the design.

AP1000 Response:

The AP1000 submittals are consistent with the requirements of 10 CFR 52 and the position in Reference 47.

1.9.5.2.11 Prototyping

NRC Position:

10 CFR 52.47 requires that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident conditions. Further, the interdependent effects among the safety features of the design must be found acceptable by analysis, appropriate test programs, experience, or a combination thereof. SECY-91-057 (Reference 48) informed the Commission of the steps the NRC was taking to identify the research needs for the AP600. SECY-91-074 (Reference 49) outlined the process the NRC would use to determine the need for a prototype or other demonstration facility for advanced reactor designs. SECY-91-273 (Reference 40) presented to the Commission the staff's recommendations for reviewing, monitoring and approving the Westinghouse test program to support the AP600 design certification application. SECY-92-030 (Reference 50) presented the Commission with the NRC opinion that there was a need for a full-height, full-pressure integral systems test to support the issuance of a final design approval.

AP1000 Response:

The Westinghouse testing program to assess the analytical methodologies used for the AP1000 safety analysis is described in Section 1.5 and is in conformance with the NRC position.

1.9.5.2.12 Inspections, Test, Analyses, and Acceptance Criteria (ITAAC)

NRC Position:

10 CFR 52 requires that the design certification application include the proposed tests, inspections, analyses, and the associated acceptance criteria. For certified standard designs, these tests, inspections, and analyses must apply to those portions of the facility covered by the design certification.

The Staff Requirements Memorandum for SECY-91-178 (Reference 51) provided guidance regarding development of ITAAC for final design approval and design certification applications.

AP1000 Response:

The AP1000 design certification application includes ITAACs.

1.9.5.2.13 Reliability Assurance Program

NRC Position:

SECY-89-013 (Reference 52) requires a reliability assurance program for design certification. The program would ensure that the design reliability of safety significant systems, structures, and components is maintained over the life of a plant.

The NRC is working on the development of a detailed guidance document consisting of two levels. The vendor submittal is the first level, consisting of a top-level program that identifies the scope, conceptual framework, and essential elements of an effective program. The Combined License applicant fully develops and implements the program based on the plant-specific design information.

AP1000 Response:

Section 16.2 includes a description of the reliability assurance program. The program description identifies the scope, conceptual framework, and essential elements of the program. The reliability assurance program confirms that the performance of the safety-related systems, structures, and components is consistent with the assumptions made for the design basis analysis.

In addition, the reliability assurance program monitors the long-term performance of important nonsafety-related structures, systems, and components that provide defense-in-depth against unnecessary actuation of the passive safety-related systems.

1.9.5.2.14 Site-Specific Probabilistic Risk Assessments (PRAs)

NRC Position:

10 CFR 52.47 requires all applicants for standard design certification to provide a PRA with enveloping analyses for seismic events and tornadoes. The Combined License applicant is responsible for the site-specific PRA information that addresses site-specific events such as river flooding, storm surge, tsunami, volcanism, and hurricanes.

AP1000 Response:

The AP1000 PRA submitted as a part of the design certification application is based on a site that bounds a large percentage of plant sites in the United States and is described in Chapter 2. APP-GW-GLR-101 (Reference 73) identifies the potential external events that may impact the AP1000 risk on a site-specific basis. This technical report considers a wide range of site-specific external events as long as a site can show that the external events listed in this report bound those applicable to the site. The report also discusses impact of site selection on PRA Level 3 requirements.

1.9.5.2.15 Severe Accident Mitigation Design Alternatives

NRC Position:

The National Environmental Policy Act (NEPA) requires that alternatives be investigated for actions that may significantly affect the quality of the human environment. The timing of the NEPA hearing is at the Early Site Permit or Combined License stage. One objective of the 10 CFR 52 design certification rulemaking is to preclude changes to a certified standard plant design. The U.S. Court of Appeals has required the NRC to include consideration of severe accident mitigation design alternatives (SAMDAs) as a part of their environmental impact review for operating license applications. If this same process is followed for a plant design that had been certified, it may be necessary to reopen issues that had been resolved in the design certification rulemaking. To avoid this situation, the NRC issued SECY-91-229 (Reference 53) which recommended that SAMDAs be specifically addressed during the design certification rulemaking.

AP1000 Response:

The severe accident mitigation design alternatives (SAMDA) evaluation for AP1000 is contained in Appendix 1B.

FSAR Position:

The severe accident mitigation design alternatives (SAMDA) evaluation for AP1000 contained in generic DCD Appendix 1B is not incorporated into this document, but is addressed in the COL application Environmental Report.

1.9.5.2.16 Generic Rulemaking Related to Design Certification

NRC Position:

SECY-91-262 (Reference 54) provides the NRC recommendations to proceed with design-specific rulemaking where appropriate for passive designs, as information becomes available from ongoing efforts on those issues, independent of the design review and certification processes. In SECY-93-087 the NRC staff concludes that the design of passive plants is not sufficiently developed to determine whether generic rulemaking should be initiated for passive plant designs.

Generic rulemaking activities for source terms during severe accidents are ongoing, and the results may be used during design certification of the passive plants, focusing on updating 10 CFR 100 siting criteria, and planning to incorporate the revised source criteria in 10 CFR 50.

AP1000 Response:

No response necessary. See Subsection 1.9.5.1.1 for a discussion of the use of a physically based source term.

1.9.5.3 Passive Design Issues

Issues related to the passive design were outlined in Reference 30.

1.9.5.3.1 Regulatory Treatment of Non-Safety Systems

NRC Position:

The NRC believes that its review of passive designs requires not only a review of the passive safety-related systems, but also a review of the functional capability and availability of the active nonsafety-related systems to provide significant defense-in-depth and accident and core damage prevention capability. The NRC issued a commission policy paper SECY-94-084 (Reference 67), on the regulatory treatment of non-safety systems (RTNSS), that outlines the process for resolving the RTNSS issue. This process includes a combination of probabilistic and deterministic criteria to identify risk-significant nonsafety-related systems.

AP1000 Response:

The AP1000 nonsafety-related active systems are designed to provide reliable support for normal plant operations and to provide defense-in-depth to minimize unnecessary challenges to the safety-related passive systems. These active systems are designed for more probable component and system failures. The systems include reliable, proven equipment and component designs. These active systems are capable of being powered by the nonsafety-related diesel-generators. The systems have nonsafety-related automatic actuation and controls that are separate from those of the safety-related systems.

These systems are designed to provide highly reliable performance. The design standards and operability provisions for these systems are discussed in Subsection 3.2.2.6. Availability controls were developed for nonsafety related structures, systems, and components found to the important via the RTNSS process. The availability controls for the AP1000 are documented in Section 16.3 and are the same as those for the AP600.

1.9.5.3.2 Definition of Passive Failure

NRC Position:

The NRC considered redefining failure of check valves in passive safety systems, where the valve fails to provide the mechanical movement to complete its intended safety function, to that of an active failure, as defined in Appendix A to 10 CFR 50. The NRC was concerned, since safety-related check valves in passive designs operate under different conditions (low flow and pressure without pump pressure to open valves) than current generation reactors and evolutionary designs. The check valves have increased safety significance to the operation of the passive safety-related systems, and operating experience has shown that they have a lower reliability than originally anticipated. The Staff position is described in SECY-94-084 (Reference 67).

AP1000 Response:

AP1000 is designed to tolerate the single failure of a check valve to change position to perform a safety-related function. Valve redundancy is provided for the core makeup tank discharge check valves (to close), the in-containment refueling water storage tank gravity injection check valves (to open), the containment recirculation gravity injection check valves (to open), and containment isolation line check valves (to close). The redundancy in the design for each of these safety-related flow paths is sufficient to accommodate the single failure of a check valve to reposition as required to perform its safeguards function.

Section 6.3 provides additional information on the failures assumed for the passive core cooling system including exceptions to the single failure criteria.

1.9.5.3.3 SBWR Stability

This issue is applicable to BWRs only.

1.9.5.3.4 Safe Shutdown Requirements

NRC Position:

GDC 34 requires that a residual heat removal system be provided to remove residual heat from the reactor core so that specified, acceptable fuel design limits are not exceeded. Regulatory Guide 1.139 and Branch Technical Position 5-1 implement this requirement and set forth conditions to cold shutdown (200°F for a PWR) using only safety-related systems within 36 hours.

The NRC evaluated the alternate means of addressing GDC 34 using passive safety-related systems to achieve a safe shutdown condition of 420°F. Additionally, the NRC reviewed the acceptability of using active, nonsafety-related systems to take a plant to cold shutdown conditions. The results of this review are presented in SECY-94-084 (Reference 67).

AP1000 Response:

The AP1000 includes safety-related passive systems and equipment that are designed to automatically establish and indefinitely maintain safe shutdown conditions for the plant following design basis events.

Sections 6.3 and 7.4 provide additional information pertaining to safe shutdown, using the safety-related passive systems.

1.9.5.3.5 Control Room Habitability

NRC Position:

10 CFR 50, Appendix A, GDC 19 requires adequate radiation protection to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of five rem whole body, or its equivalent, to any part of the body, for the duration of the accident. Section 6.4 of the Standard Review Plan defines this dose criterion in terms of specific whole-body and organ doses (5 rem to whole body, and 30 rem each to thyroid and skin). The NRC requires that the analyses of main control room habitability be based on the dose criterion defined in GDC 19 of Appendix A to 10 CFR 50 and Section 6.4 of the Standard Review Plan (5 rem to whole body, and 30 rem each to thyroid and skin). In addition, the analyses of control room habitability should be based on the duration of the accident according to GDC 19 of Appendix A to 10 CFR 50.

AP1000 Response:

The AP1000 design includes a passive, safety-related main control room habitability system to meet the requirements of GDC 19. Section 6.4 provides additional information.

As described in Subsection 15.6.5.3, the main control room operator doses following a design basis loss of coolant accident are within the dose criterion of GDC 19 (5 rem TEDE as applied to the AP1000 design).

1.9.5.3.6 Radionuclide Attenuation

NRC Position:

The NRC is concerned that use of the auxiliary building for holdup may require additional restrictions to be placed on the auxiliary building during normal operation. In addition, the NRC is continuing its evaluation of the need for a containment spray system for passive plant designs.

AP1000 Response:

The AP1000 design does not have a safety-related containment spray or take credit for auxiliary building holdup for mitigation of the design basis loss of coolant accident. The design includes a low-leakage-rate containment (0.10 percent per day) together with credit for aerosol removal by naturally occurring processes and pool scrubbing in containment. The low-leakage containment and natural aerosol removal are adequate to meet 10 CFR 50.34 dose limits, consistent with the physically-based source term.

1.9.5.3.7 Simplification of Off-Site Emergency Planning

NRC Position:

The NRC states that changes to emergency planning regulatory requirements may be appropriate, but that an NRC determination on this issue will require detailed design evaluation. Summaries of specific NRC conclusions are as follows:

- Unique characteristics of the designs should be considered in determining the extent of emergency planning, including the ability to prevent significant release of radioactive material or to provide delay times for all but the most unlikely events.

- A very low likelihood of all containment bypass sequences will be required before relaxing emergency planning requirements.

- Lack of information on source term and risk precludes further NRC evaluation of emergency preparedness for the passive plants at this time.

- Emergency planning requirements following the TMI-2 accident were not premised on specific assumptions regarding severe accident probability. So, as a policy matter, even very low calculated probabilities may not be a sufficient basis for changes to emergency planning requirements.

The industry and the NRC are working to determine a process, including developing technical criteria and methods, that would justify simplification of offsite emergency planning. The results of this process would be used as input to a generic rulemaking proposal to be initiated by nuclear industry organizations.

AP1000 Response:

The AP1000 PRA evaluation risk assessment includes calculations of the AP1000 response to severe accidents. This response includes the release of radionuclides following a severe accident. This analysis supports the technical basis for simplification of offsite emergency planning. The offsite emergency planning is discussed in Section 13.3.

1.9.5.4 Additional Licensing Issue

Post-72 Hour Support Actions

The AP1000 includes safety-related passive systems and equipment that are sufficient to automatically establish and maintain safe shutdown conditions for the plant following design basis events, assuming that the most limiting single failure occurs. The safety-related passive systems maintain safe shutdown conditions after an event -- without operator action, without onsite and offsite ac power sources.

The AP1000 includes nonsafety-related active systems and equipment designed to provide multiple levels of defense for a wide range of events. For the more probable events, these nonsafety-related systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related passive systems. These nonsafety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the standby nonsafety-related ac power sources is available.

Although event scenarios that result in an extended loss of the nonsafety-related systems or both offsite and onsite ac power sources for more than 72 hours are very unlikely, this potential is considered in the AP1000 design.

The actions described below are required following an extended loss of these nonsafety-related systems.

The safety functions required include the following:

- Core cooling, inventory, and reactivity control
- Containment cooling and ultimate heat sink
- Main control room habitability and post-accident monitoring
- Spent fuel pool cooling

The AP1000 design includes both onsite equipment and safety-related connections for use with transportable equipment and supplies to provide the following extended support actions:

- Provide electrical power to supply the post-accident and spent fuel pool monitoring instrumentation, using the ancillary diesel generators or a portable, engine-driven ac generator that both connect to electrical connections at the ancillary diesel generator electric panel. See Section 8.3 for additional information.

- Provide makeup water to the passive containment cooling water storage tank to maintain external containment cooling water flow, using one of the two PCS recirculation pumps powered by an ancillary diesel generator or a portable, engine-driven pump that connects to a safety-related makeup connection. See Subsection 6.2.2 for additional information.

- Ventilation and cooling of the main control room, the instrumentation and control rooms, and the dc equipment rooms is provided by open doors and ancillary fans or portable fans powered by an ancillary diesel generator or a portable, engine-driven ac generator.

- Provide makeup water to the spent fuel pool from the passive containment cooling water storage tank, passive containment cooling water ancillary water storage tank, and from the long term makeup connection. See Subsection 6.2.2.2.4 for a discussion of the operation of the passive containment cooling system and Subsection 9.1.3.4.3 and 9.1.3.5 for discussion of makeup to the spent fuel pool.

- Provide a vent path between the fuel handling area and outside environment to vent water vapor generated by elevated spent fuel pool water temperature. See Subsection 9.1.3.4.3.4 for additional information.

These actions are accomplished by the site support personnel, in coordination with the main control room operators. These actions are performed separate from, but in parallel with, other actions taken by the plant operators to directly mitigate the consequences of an event.

1.9.5.5 Operational Experience

Operational experience highlighted in NRC bulletins, generic letters, and information notices has been incorporated into the AP1000 design. Generic letters and bulletins are identified in WCAP-15800 (Reference 65). The applicability of each generic letter and bulletin to the AP1000 is assessed in WCAP-15800. If required, additional information for applicable issues is provided in the referenced sections of the DCD.

Table 1.9-204 lists the Bulletins and Generic Letters addressed by topical discussion in this FSAR. Table 1.9-204 also lists Bulletins and Generic Letters categorized as part of the first un-numbered COL Information Item identified at the end of Table 1.8-2 and listed in Table 1.8-202 as COL Information Item 1.9-2. Table 1.9-204 provides the appropriate cross-references for the discussion of the topics addressed by those Bulletins and Generic Letters. Bulletins or Generic Letters issued after those listed in the generic DCD are also included in Table 1.9-204. Issues identified as “procurement” or “maintenance” or “surveillance” in WCAP-15800 are addressed as part of the scope of the certified design and are not specifically identified in Table 1.9-204. Issues identified as “procedural” in WCAP-15800 are addressed by the procedures discussed in Section 13.5 and are not specifically identified in Table 1.9-204. Other items in WCAP-15800, including the Circulars and Information Notices, are considered to have been adequately addressed based on the guidance identified in Regulatory Guide 1.206 and the NRC Standard Review Plans.

1.9.6 References

1. NUREG-0696, "Functional Criteria for Emergency Response Facilities," 1981.

2. Report NP-2770-LD, "EPRI PWR Safety Valve Test Report," December 1982.

3. NUREG-0933, "A Prioritization of Generic Safety Issues," June 2001.

4. NUREG-0371, "Task Action Plans for Generic Activities (Category A)," U.S. Nuclear Regulatory Commission, November 1978.

5. NUREG-0927, Revision 1, "Evaluation of Water Hammer Occurrences in Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1984.

6. NRC letter to all PWR Licensees of Operating Reactors, Applicants for Operating Licensees and Holders of Construction Permits, and Ft. St. Vrain, "Staff Recommended Actions Stemming from NRC Integrated Program for the Resolution of Unresolved Safety Issues Regarding Steam Generator Tube Integrity," (Generic Letter 85-02) April 17, 1985.

7. NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, A-5 Regarding Steam Generator Tube Integrity," U.S. Nuclear Regulatory Commission, September 1988.

8. NUREG-0577, Revision 1, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," U.S. Nuclear Regulatory Commission, October 1983.

9. IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

10. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1980.

11. NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants," U.S. Nuclear Regulatory Commission, January 1979.

12. NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plant Stations," U.S. Nuclear Regulatory Commission, February 1981.

13. IEEE 344-1987, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

14. NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants."

15. NUREG-0660, "NRC Action Plan Developed as a result of the TMI-2 accident," May 1980.

16. NUREG-0985, "Nuclear Regulatory Commission Human Factors Program Plan Revision 2," April 1986.

17. IEEE 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," Institute of Electrical and Electronics Engineers.

18. NUREG-0471, "Generic Task Problem Descriptions (Category B, C, and D Tasks)" and NUREG-0933, "A Prioritization of Generic Safety Issues," June 1978.

19. NUREG-0484, Revision 1, "Methodology for Combining Dynamic Responses," May 1980.

20. IEEE 317-1983, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers.

21. ANSI/ANS-58.8-1984. "Time Response Design Criteria for Nuclear Safety Related Operator Actions."

22. NUREG/CR-2425, "Sediment and Radionuclide Transport in Rivers," December 1982.

23. NUREG-0693, "Analysis of Ultimate-Heat-Sink Cooling Ponds," November 1980.

24. NUREG-0733, "Analysis of Ultimate Heat-Sink Spray Ponds," August 1981.

25. NUREG-0858, "Comparison Between Field Data and Ultimate Heat Sink Cooling-Pond and Spray-Pond Models," September 1980.

26. ANSI 5.1, "Decay Heat Power in Light Water Reactors," American National Standards Institute, 1979.

27. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," September 1980.

28. ANSI 56.5-1979, "PWR and BWR Containment Spray System Design Criteria."

29. USNRC, SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues And Their Relationship to Current Regulatory Requirements," January 12, 1990.

30. NRC letter, Subject: Identification of Issues Concerning the Evolutionary and Passive Plant Designs, Dennis M. Crutchfield, USNRC Director, Division of Advanced Reactors and Special Projects, to E. E. Kintner, Chairman ALWR Steering Committee, February 27, 1992.

31. Staff Requirements Memorandum, Subject: SECY-90-016 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements, Samuel J. Chilk, USNRC Secretary, to James M. Taylor, USNRC Executive Director for Operations, June 26, 1990.

32. NUREG-1150, "Severe Accident Risk: An Assessment for Five U.S. Nuclear Power Plants," June 1989.

33. "Passive ALWR Source Term," D. E. Leaver, et al., DOE/ID-10321, February 1991.

34. NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," Revision 1, September 1984.

35. NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44," June 1988.

36. NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," June 1988.

37. Branch Technical Position CMEB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," July 1986.

38. Generic Letter 89-04, "Guidance on Developing Acceptable Inservice Testing Programs," April 3, 1989.

39. Generic Letter 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," June 28, 1989.

40. SECY-91-273, "Review of Vendors' Test Programs to Support the Design Certification of Passive Light Water Reactors," August 27, 1991.

41. SECY-91-078, "Chapter 11 of EPRI's Requirements Document and Additional Evolutionary Light Water Reactor Certification Issues," March 25, 1991.

42. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," June 1991.

43. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," October 1975.

44. SECY-91-348, preliminary untitled SECY related to containment leakrate testing, issued to the Commission for review, and not yet released by the NRC.

45. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

46. NUREG-4330, "Review of Light Water Reactor Regulatory Requirements, Volume 1, Identification of Regulatory Requirements That May Have Marginal Importance to Risk," April 1986.

47. SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," November 8, 1990.

48. SECY-91-057, "Early Review of AP600 and SBWR Research Needs," March 1, 1991.

49. SECY-91-074, "Prototype Decisions for Advanced Reactor Designs," March 19, 1991.

50. SECY-92-030, "Integral System Testing Requirements for Westinghouse's AP600 Plant," January 27, 1992.

51. SECY-91-178, "Inspections, Tests, Analyses, and Acceptance Criteria (ITAAC) for Design Certifications and Combined Licenses," June 12, 1991.

52. SECY-89-013, "Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs)," January 19, 1989.

53. SECY-91-229, "Severe Accident Mitigation Design Alternatives for Certified Standard Designs," July 31, 1991.

54. SECY-91-262, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," August 16, 1991.

55. Not used.

56. WCAP-15993, "Evaluation of the AP1000 Conformance to Inter-System Loss-of-Coolant Accident Acceptance Criteria," Revision 1, March 2003.

57. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," June 30, 1995.

58. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," L. Soffer, et al., February 1995.

59. SECY-94-302, "Source Term-Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," December 19, 1994.

60. SECY-94-194, "Proposed Revisions to 10 CFR Part 100 and 10 CFR Part 50, and New Appendix S to 10 CFR Part 50," July 27, 1994.

61. Not used.

62. Theofanous, T. G., et al., "In-Vessel Coolability and Retention of a Core Melt," DOE/ID-10460, July 1995.

63. WCAP-15799, "AP1000 Compliance with SRP Acceptance Criteria," Revision 1, August 2003.

64. NCRP Report No. 116, Limitation of Exposure to Ionizing Radiation, March 31, 1993.

65. WCAP-15800, "Operational Assessment for AP1000," Revision 3, July 2004.

66. SECY-98-161, "The Westinghouse AP1000 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems," July 1, 1998.

67. SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994.

68. SECY-95-172, "Key Technical Issues Pertaining to the Westinghouse AP1000 Standardized Passive Reactor Design," June 30, 1995.

69. WCAP-15992, "AP1000 Adverse Systems Interactions Evaluation Report," Revision 1, February 2003.

70. WCAP-15776, "Safety Criteria for the AP1000 Instrumentation and Control Systems," April 2002.

71. APP-GW-GLR-079, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Westinghouse Electric Company LLC.

72. APP-GW-GLR-040, "Plant Operations, Surveillance, and Maintenance Procedures," Westinghouse Electric Company LLC.

73. APP-GW-GLR-101, "AP1000 Probabilistic Risk Assessment External Events Evaluation to Support COL Applications," Westinghouse Electric Company LLC.

201. NUMARC 87-00, Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors, Revision 1, August 1991.

**Table 1.9-1 (Sheet 1 of 17)
Regulatory Guide/Section Cross-References**

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 2, 1970)	This regulatory guide is not applicable to AP1000.
1.2	Withdrawn	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-coolant Accident for Boiling Water Reactors (Rev. 2, June 1974)	This regulatory guide is not applicable to AP1000.
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors (Rev. 2, June 1974)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.4.
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Rev. 0, March 10, 1971)	This regulatory guide is not applicable to AP1000.
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Rev. 0, March 10, 1971)	8.1 8.3.1 8.3.2 16.1 Bases 16 (TS Bases 3.8.1)
1.7	Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident (Rev. 2, November 1978)	6.1.1 6.2.4 15.6.3 Appendix 15A
1.7	Control of Combustible Gas Concentrations in Containment (Rev. 3, March 2007)	DCD discussion only
1.8	Qualification and Training of Personnel for Nuclear Power Plants (Rev. 3, 1 May 2000)	This regulatory guide is not applicable to AP1000 design certification. 12.1 (NEI 07-08A) Appendix 12AA Appendix 12AA (NEI 07-03A) 13.1.1.4 13.1.3.1 13.2 (NEI 06-13A) 16 (TS 5.3.1) 17.5 (QAPD, IV)
1.9	Selection, Design, and Qualification of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (Proposed Rev. 3, November 1988)	This regulatory guide is not applicable to AP1000.
1.10	Withdrawn	
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 0, March 10, 1971)	3.6.2 6.2.3
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 1, March 2010)	DCD discussion only
1.12	Instrumentation for Earthquakes (Rev. 2, March 1997)	3.7.4 3.7.4.1

Table 1.9-1 (Sheet 2 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.13	Spent Fuel Storage Facility Design Basis (Proposed Rev. 2, December 1981)	9.1.2 9.1.3 9.1.4 16.1 Bases
1.13	Spent Fuel Storage Facility Design Basis (Rev. 2, March 2007)	16 (TS Bases 3.7.11) 16 (TS Bases 3.7.12)
1.14	Reactor Coolant Pump Flywheel Integrity (Rev. 1, August 1975)	5.4.1
1.15	Withdrawn	
1.16	Reporting of Operating Information - Appendix A Technical Specifications (Rev. 4, August 1975).	This regulatory guide is not applicable to AP1000 design certification.
1.17	Withdrawn	
1.18	Withdrawn	
1.19	Withdrawn	
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 2, May 1976)	3.9.2 14
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 3, March 2007)	DCD discussion only
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	11.5 2.3.3.4.1 11.5.1.2 11.5.4.1 11.5.4.2 12.3.4
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 17, 1972)	7.1 7.2 7.4
1.23	Onsite Meteorological Program (Second Proposed Rev. 1, April 1986)	2.3
1.23	Meteorological Monitoring Programs for Nuclear Power Plants (Rev. 1, March 2007)	2.3.2.2.3 2.3.3.3 2.3.3.3.1 2.3.3.3.2 2.3.3.3.4 2.3.3.3.5.1 2.3.3.3.5.2 2.3.3.5.1 2.3.3.5.2 2.3.3.5.3
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 23, 1972)	This regulatory guide is not applicable to AP1000.
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0, March 23, 1972)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.25.

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 3, February 1976)	3.2.2
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 4, March 2007)	5.2.4.1 17.5 (QAPD IV)
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	6.2.2
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 3, August 1985)	2.5 17 14.2.2.2 17.5 (QAPD, II, 17.1) 17.5 (QAPD, IV)
1.29	Seismic Design Classification (Rev. 3, September 1978)	3.2.1
1.29	Seismic Design Classification (Rev. 4, March 2007)	17.5 (QAPD IV)
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 11, 1972)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 3, April 1978)	4.5.1 4.5.2 5.2.3 5.3.2 6.1.1 6.1.1.2
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Rev. 2, February 1977)	8.1 8.2 8.3.1 8.3.2 16.1 Bases
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants and Rev. 3, March 2004	16 (TS Bases 3.8.1)
1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	3.11.2.1 3D.4.1.2 3D.6.4 16 (TS 5.4.1) 17.5 (QAPD, IV)
1.34	Control of Electroslag Weld Properties (Rev. 0, December 28, 1972)	4.5.2 5.2.3 5.3.2
1.35	Inservice Inspection of Ungrouted Tendons in Pre-stressed Concrete Containments (Rev. 3, July 1990)	This regulatory guide is not applicable to AP1000.
1.35.1	Determining Prestressing Forces for Inspection of Prestressed Concrete Containments (Rev. 0, July 1990)	This regulatory guide is not applicable to AP1000.
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 23, 1973)	5.2.3 6.1.1
1.37	Quality Assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	17

Table 1.9-1 (Sheet 4 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.37	Quality Assurance Requirements for cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 1, March 2007)	17.5 (QAPD, II, 13.2) 17.5 (QAPD IV)
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	17
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	17
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 16, 1973)	This regulatory guide is not applicable to AP1000.
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments (Rev. 0, March 16, 1973)	14
1.42	Withdrawn	
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	5.2.3 5.3.2
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	4.5.1 4.5.2 5.2.3 5.3.2 6.1.1 10.3 6.1.1.2
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)	5.2.5 16.1 Bases 16 (TS Bases 3.4.7) 16 (TS Bases 3.4.9)
1.46	Withdrawn	
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	6.3 7.2 7.3 7.4 7.5 8.3.2
1.48	Withdrawn	
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	16
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	5.2.3 5.3.2 6.1.1
1.51	Withdrawn	
1.52	Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 3, June 2001)	6.4 16 (TS 3.7.6)

Table 1.9-1 (Sheet 5 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)	7.1 7.2 7.4 15.2 15.3 15.4 15.5 15.6
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 2, November 2003)	DCD discussion only
1.54	Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants (Rev. 1, July 2000 and Rev. 2, October 2010)	6.1.2 1.9.4.2.3 6.1.2.1.6
1.55	Withdrawn	
1.56	Maintenance of Water Purity in Boiling Water Reactors (Rev. 1, July 1978)	This regulatory guide is not applicable to AP1000.
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 0, June 1973)	3.8.2 3.8.3
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 1, March 2007)	DCD discussion only
1.58	Withdrawn	
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	2.4 3.4 Not referenced; see Appendix 1A
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	2.5 3.7.1
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)	3.7.1 3.9.23.10 Appendix 3D
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 1, March 2007)	DCD discussion only
1.62	Manual Initiation of Protective Actions (Rev. 0, October 1973)	7.1 7.2
1.63	Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants (Task EE 405-4) (Rev. 3, February 1987)	8.3.1 Appendix 3D
1.64	Withdrawn	
1.65	Materials and Inspections for Reactor Vessel Closure Studs (Rev. 0, October 1973)	5.3.2
1.66	Withdrawn	
1.67	Withdrawn	
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 2, August 1978)	14 16.1 Bases
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants (Rev. 3, March 2007)	14.2.1 14.2.3 14.2.8 14.2.5.2 16 (TS Bases 3.1.8)

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants (Rev. 1, January 1977)	This regulatory guide is not applicable to AP1000.
1.68.2	Initial Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, July 1978)	14
1.68.3	Preoperational Testing of Instrument and Air Control Systems (Task RS 709-4) (Rev. 0, April 1982)	9.3.1 14
1.69	Concrete Radiation Shields for Nuclear Power Plants (Rev. 0, December 1973)	3.8.4 12.3
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Rev. 3, November 1978)	1.1 1.1.6.1
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 0, December 1973)	5.2.3.4.6
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 1, March 2007)	DCD discussion only
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	This regulatory guide is not applicable to AP1000.
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	3.11 Appendix 3D
1.74	Withdrawn	
1.75	Physical Independence of Electric Systems (Rev. 2, September 1978)	7.1 7.2 7.3 7.4 7.5 8.1 8.3.1 8.3.2 9.5.1
1.75	Criteria for Independence of Electrical Safety Systems (Rev 3, February 2005)	DCD discussion only
1.76	Design Basis Tornado for Nuclear Power Plants (Rev. 0, April 1974)	2.3 3.3
1.76	Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants (Rev. 1, March 2007)	2.3.1.3.2
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	The guidance of Reg. Guide 1.183, “Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors” will be followed instead of Reg. Guide 1.77. 16.1 Bases 16 (TS Bases 3.2.1) 16 (TS Bases 3.2.2) 16 (TS Bases 3.2.4) 16 (TS Bases 3.2.5)

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.78	Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 1, December 2001)	2.2 6.4 9.4.1 9.5.1 16.1 Bases 2.2.3.1.3 2.2.3.1.3.3 2.2.3.1.3.4 2.3.4.2.2 6.4.3 6.4.4.2 16 (TS Bases 3.7.6) Table 19.58-201
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	14
1.80	Withdrawn	
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plant (Rev. 1, January 1975)	This regulatory guide is not applicable to AP1000.
1.82	Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident (Task 203-4) (Rev. 3, November 2003)	6.3
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	5.4.2
1.84	Design and Fabrication Code Case Acceptability ASME Section III Division 1 (Rev. 32, June 2003)	4.5.1 4.5.2 5.2.1 5.2.3 10.3
1.84	Design, Fabrication, and Materials Code Case Acceptability, ASME Section III (Rev. 33, August 2005)	DCD discussion only
1.85	Withdrawn	
1.86	Termination of Operating Licenses for Nuclear Reactors (Rev. 0, June 1974)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Rev. 1, June 1975)	This regulatory guide is not applicable to AP1000.
1.88	Withdrawn	
1.89	Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants (Task EE 042-2) (Rev. 1, June 1984)	3.11 Appendix 3D
1.90	Inservice Inspection of Prestressed Concrete Containment Structures With Grouted Tendons (Rev. 1, August 1977)	This regulatory guide is not applicable to AP1000.
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites (Rev. 1, February 1978)	19.58 2.2 2.2.1 2.2.3.1.1 2.2.3.1.2 2.2.3.1.3 3.5.1.5

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 1, February 1976; Rev. 2, July 2006)	3.7 Appendix 3D
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis (Rev. 2, July 2006)	DCD discussion only
1.93	Availability of Electric Power Sources (Rev. 0, December 1974)	8.1 8.3 16.1 Bases 16 (TS Bases 3.8.1) 16 (TS Bases 3.8.5)
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.95	Withdrawn	
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Rev. 1, June 1976)	This regulatory guide is not applicable to AP1000.
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident (Rev. 3, May 1983)	7.5 18.8 16.1 Bases Appendix 3D Table 7.5-1 RN-14-046 Appendix 12AA 16 (TS Bases 3.3.17) RN-14-152
1.97	Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants (Rev. 4, June 2006)	Not referenced; see Appendix 1A
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (Rev. 0, March 1976)	This regulatory guide is not applicable to AP1000.
1.99	Radiation Embrittlement of Reactor Vessel Materials (Task ME 305-4) (Rev. 2, May 1988)	5.3.2 5.3.3 16.1 Bases
1.100	Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants (Task EE 108-5) (Rev. 2, June 1988)	3.10 Appendix 3D
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 3, August 1992)	This regulatory guide is not applicable to AP1000 design certification.
1.101	Emergency Response Planning and Preparedness for Nuclear Power Reactors (Rev. 3, August 1992)	9.5.1.8.2.2 Table 9.5-201 13.3 (Emergency Plan I.C.1)
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 4, July 2003)	Not referenced
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors (Rev. 5 June 2005)	Not referenced; see Appendix 1A
1.102	Flood Protection for Nuclear Power Plants (Rev. 1, September 1976)	3.4
1.103	Withdrawn	
1.104	Withdrawn	
1.105	Instrument Setpoints for Safety-Related Systems (Task 1C 010-5) (Rev. 3, December 1999)	7.1 16

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves (Rev. 1, March 1977)	8.1
1.107	Qualifications for Cement Grouting Tendons for Prestressing Tendons in Containment Structures (Rev. 1, February 1977)	This regulatory guide is not applicable to AP1000.
1.108	Withdrawn	
1.109	Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50 Appendix I (Rev. 1, October 1977)	11.3.3 11.2.3.5 11.3.3.4 12.4.1.9.3 Table 12.4-201
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Rev. 0, March 1976)	11.2 11.3 11.2.3.5.3 11.3.3.4.3
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors (Rev. 1, July 1977)	2.3 2.3.5.1 2.3.3.5.1.2 2.3.4.2 12.4.1.9.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors (Rev. 0-R, May 1977)	11.2.3 11.3.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors (Rev. 1, March 2007)	DCD discussion only
1.113	Estimating Aquatic Dispersion of Effluents From Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, April 1977)	This regulatory guide is not applicable to AP1000 design certification. 11.2.3.3
1.114	Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit (Rev. 2, May 1989)	This regulatory guide is not applicable to AP1000 design certification. 13.1.2.1.1.3.2.1.2 13.1.2.2
1.115	Protection Against Low-Trajectory Turbine Missiles (Rev 1, July 1977)	3.5 3.8.4 3.5.1.3
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.117	Tornado Design Classification (Rev. 1, April 1978)	3.5 9.1.2
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 3, April 1995)	7.1 8.1 8.3
1.119	Withdrawn	
1.120	Fire Protection Guidelines for Nuclear Power Plants (Rev. 1, November 1977)	9.5.1

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	5.4.2 16.1 Bases 16 (TS Bases 3.4.18)
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)	3.7 Appendix 3D
1.123	Withdrawn	
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Rev. 1, January 1978)	3.9.3, 9.1.2.1, 9.1.1.1
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Supports (Rev. 2, February 2007)	DCD discussion only
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	2.4
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	4.2
1.127	Inspection of Water-Control Structures Associated With Nuclear Power Plants (Rev. 1, March 1978)	This regulatory guide is not applicable to AP1000.
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, October 1978)	8.3.2
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	DCD discussion only
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, February 1978)	16.1 Bases
1.129	Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants (Rev. 2, February 2007)	Table 8.1-201 8.3.2.1.4 16 (TS Bases 3.8.1)
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 1, October 1978)	3.9.3
1.130	Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Supports (Rev. 2, March 2007)	DCD discussion only
1.131	Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	3.11 Appendix 3D
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 1, March 1979)	This regulatory guide is not applicable to AP1000 design certification.
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 2, October 2003)	2.5.4.2.3
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	4.4.6.4 Not referenced; see Appendix 1A
1.134	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses (Rev. 3, March 1998)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	2.4
1.136	Material for Concrete Containments (Rev. 2, June 1981)	This regulatory guide is not applicable to AP1000.
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, October 1979)	9.5.4

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.138	Laboratory Investigation of Soils for Engineering Analysis and Design of Nuclear Power Plants (Rev. 0, April 1978)	This regulatory guide is not applicable to AP1000 design certification.
1.138	Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants (Rev. 2, December 2003)	2.5.4.2.4
1.139	Guidance for Residual Heat Removal (Rev. 0, May 1978)	6.3 7.4
1.140	Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants (Rev. 2, June 2001)	9.4.1 9.4.4 9.4.5 9.4.7 9.4.9 16.1 Bases 9.4.1.4 9.4.7.4 16 (TS Bases 3.9.6)
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	6.2.4
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (Rev. 1, October 1981)	3.8.3 3.8.4 3.8.5
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 2, November 2001)	3.8.4 10.4.8 11.2 11.3 11.4 11.5 11.2.1.2.5.2 11.2.3.6 11.3.3.6 11.4.5 11.4.6.2
1.144	Withdrawn	
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 1, November 1982)	This regulatory guide is not applicable to AP1000 design certification. 2.3.4.1 2.3.4.2
1.146	Withdrawn	
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1 (Rev. 12, May 1999)	5.2.4.3 6.6.3
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1 (Rev. 15, October 2007)	5.2.4 6.6
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (Rev. 0, March 1981)	3.10 5.4.8
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations (Rev. 2, April 1996)	This regulatory guide is not applicable to AP1000 design certification.
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations (Rev. 3, October 2001)	13.2 (NEI 06-13A)

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.150	Ultrasonic Testing of Reactor Pressure Vessel Welds During Preservice and Inservice Examinations (Rev. 1, February 1983)	5.2.4 5.3.2 5.3.4
1.151	Instrument Sensing Lines (Task 1C 126-5) (Rev. 0, July 1983)	7.1 7.5 7.6 7.7
1.152	Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants (Task 1C 127-5) (Rev. 1, January 1996)	7.1 7.2 7.3 7.4 7.5 7.6
1.152	Criteria for Use of Computers in Safety Systems of Nuclear Power Plants (Rev. 2, January 2006)	Not referenced; see Appendix 1A
1.153	Criteria for Power, Instrumentation, and Control Portions of Safety Systems (Task 1C 609-5) (Rev. 1, June 1996)	7.1 7.2 7.3 7.4 7.5 7.6
1.154	Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors (Rev. 0, January 1987)	5.3
1.155	Station Blackout (Task SI 501-4) (Rev. 0, August 1988)	8.2 8.3.1
1.155	Station Blackout (Rev. 0, August 1998)	Table 8.1-201
1.156	Environmental Qualification of Connection Assemblies for Nuclear Power Plants (Task EE 404-4) (Rev. 0, November 1987)	3.10 3.11 Appendix 3D
1.157	Best-Estimate Calculations of Emergency Core Cooling System Performance (Task RS 701-4) (Rev. 0, May 1989)	6.3
1.158	Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants (Task EE 006-5) (Rev. 0, February 1989)	3.10 3.11 Appendix 3D
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 0, August 1990)	This regulatory guide is not applicable to AP1000 design certification.
1.159	Assuring the Availability of Funds for Decommissioning Nuclear Reactors (Rev. 1, October 2003)	Not referenced; see Appendix 1A
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Rev. 2, March 1997)	3.8.6.5 17.5.6 The COL applicant is responsible for assessing conformance to Regulatory Guide 1.160 of monitoring the effectiveness of maintenance. 3.8.3.7 3.8.4.7 3.8.5.7 17.6 (NEI 07-02A)

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.161	Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb (Rev. 0, June 1995)	This regulatory guide is not applicable to AP1000 design certification.
1.162	Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels (Rev. 0, February 1996)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.163	Performance Based Containment Leak-Test Program (Rev. 0, September 1995)	6.2 16.1 Bases 6.2.5.1 6.2.5.2.2 16 (TS 5.5.8)
1.165	Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion (Rev. 0, March 1997)	2.5.2.2 2.5.2.2.2.4 2.5.2.4 2.5.2.4.1 2.5.2.4.6
1.166	Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions (Rev. 0, March 1997)	This regulatory guide is not applicable to AP1000 design certification. 3.7.4.4
1.167	Restart of a Nuclear Power Plant Shut Down by a Seismic Event (Rev. 0, March 1997)	This regulatory guide is not applicable to AP1000 design certification. 3.7.4.4
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.168	Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 1, February 2004)	DCD discussion only
1.169	Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.170	Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.171	Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.172	Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.173	Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants (Rev. 0, September 1997)	7
1.174	An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis (Rev. 0, July 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.174	An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis (Rev. 1, November 2002)	Not referenced; see Appendix 1A

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Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.175	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing (Rev. 0, July 1998)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.176	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance (Rev. 0, August 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.177	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications (Rev. 0, August 1998)	16.1 Bases 16 (TS Bases 3.5.1) 16 (TS Bases 3.7.10)
1.178	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping (Rev. 0, September 1998)	This regulatory guide is not applicable to AP1000 design certification.
1.178	An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping (Rev. 1, September 2003)	Not referenced; see Appendix 1A
1.179	Standard Format and Content of License Termination Plans for Nuclear Power Reactors (Rev. 0, January 1999)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.180	Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems (Rev. 1, October 2003)	Appendix 3D
1.181	Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e) (Rev. 0, September 1999)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.182	Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants (Rev. 0, May 2000)	16.1 Bases 16 (TS Bases SR 3.0.3) 17.6 (NEI 07-02A)
1.183	Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors (Rev. 0, July 2000)	2.3 Appendix 3D 4.2 6.5.1 15.4 15.6.3 15.7 16.1 Bases 16 (TS Bases 3.7.5) 16 (TS Bases 3.9.4) 16 (TS Bases 3.9.5) RN-14-152
1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.184	Decommissioning of Nuclear Power Reactors (Rev. 0, July 2000)	Not referenced; see Appendix 1A

Table 1.9-1 (Sheet 15 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.185	Standard Format and Content for Post-shutdown Decommissioning Activities Report (Rev. 0, August 2000)	This regulatory guide is not applicable to AP1000 design certification.
1.185	Standard Format and Content for Post- shutdown Decommissioning Activities Report (Rev. 0, July 2000)	Not referenced; see Appendix 1A
1.186	Guidance and Examples of Identifying 10 CFR 50.2 Design Bases (Rev. 0, December 2000)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.187	Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments (Rev. 0, November 2000)	This regulatory guide is not applicable to AP1000 design certification. Not referenced; see Appendix 1A
1.188	Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses (Rev. 1, September 2005)	Not referenced; see Appendix 1A
1.189	Fire Protection for Operating Nuclear Power Plants (Rev. 0, April 2001)	This regulatory guide is not applicable to AP1000 design certification.
1.189	Fire Protection for Nuclear Power Plants (Rev. 1, March 2007)	9.5.1.8.1.1 9.5.1.8.2.2 13.1.1.1.3.2.1.4
1.190	Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence (Rev. 0, March 2001)	5.3.2.6.2.2
1.191	Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown (Rev. 0, May 2001)	Not referenced; see Appendix 1A
1.192	Operation and Maintenance Code Case Acceptability, ASME OM Code (Rev. 0, June 2003)	3.9.6.3
1.193	ASME Code Cases Not Approved for Use (Rev. 1, August 2005)	Not referenced; see Appendix 1A
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	2.3.4.2.1.2
1.195	Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors (Rev. 0, May 2003)	Not referenced; see Appendix 1A
1.196	Control Room Habitability at Light-Water Nuclear Power Reactors (Rev. 1, January 2007)	6.4.3
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	9.4.1 6.4.5
1.198	Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites (Rev. 0, November 2003)	2.5.4.8
1.199	Anchoring Components and Structural Supports in Concrete (Rev. 0, November 2003)	3.8.3.5 3.8.4.5.1 3.8.5.5
1.200	An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities (Rev. 1, January 2007)	19.59.10.6
1.201	Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance (Rev. 1, May 2006)	Not referenced; see Appendix 1A

Table 1.9-1 (Sheet 16 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
1.202	Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors (Rev. 0, February 2005)	Not referenced; see Appendix 1A
1.203	Transient and Accident Analysis Methods (Rev. 0, December 2005)	Not referenced; see Appendix 1A
1.204	Guidelines for Lightning Protection of Nuclear Power Plants (Rev. 0, November 2005)	Table 8.1-201
1.205	Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants (Rev. 0, May 2006)	Not referenced; see Appendix 1A
1.206	Combined License Applications for Nuclear Power Plants (LWR Edition) (Rev. 0, June 2007)	See Appendix 1A 1.1.6.1 1.9.5.5 2.5.0 2.5.1 2.5.2.1.2 2.5.2.4.6 2.5.4 14.2.1 14.3.2.3.1 14.3.2.3.2 Table 8.1-201 Appendix 12AA (NEI 07-03A)
1.207	Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors (Rev. 0, March 2007)	Not referenced; see Appendix 1A
1.208	A Performance-Based Approach to Define the Site-Specific Earthquake Gound Motion (Rev. 0, March 2007)	2.5.1 2.5.2 2.5.3 2.5.4
1.209	Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants (Rev. 0, March 2007)	Not referenced; see Appendix 1A
Division 4 Regulatory Guides		
4.7	General Site Suitability Criteria for Nuclear Power Stations (Rev. 2, April 1998)	Not referenced; see Appendix 1A
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) — Effluent Streams and the Environment (Rev. 1, February 1979)	11.5.1.2 11.5.3 11.5.4 11.5.6.5
4.15	Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) — Effluent Streams and the Environment (Rev. 2, July 2007)	11.5.3
Division 5 Regulatory Guides		Note b
Division 8 Regulatory Guides		
8.2	Guide for Administrative Practices in Radiation Monitoring (Rev. 0, February 1973)	12.1 (NEI 07-08A) 12.3.4
8.4	Direct-Reading and Indirect-Reading Pocket Dosimeters (Rev. 0, February 1973)	Appendix 12AA (NEI 07-03A)
8.5	Criticality and Other Interior Evacuation Signals (Rev. 1, March 1981)	Appendix 12AA (NEI 07-03A)
8.6	Standard Test Procedure for Geiger-Muller Counters (Rev. 0, May 1973)	Appendix 12AA (NEI 07-03A)

Table 1.9-1 (Sheet 17 of 17)
Regulatory Guide/Section Cross-References

Division 1 Regulatory Guide		Chapter, Section or Subsection
8.7	Instructions for Recording and Reporting Occupational Radiation Dose Data (Rev. 2, November 2005)	12.1 (NEI 07-08A)
8.8	Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable (Rev. 3, June 1978)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA Appendix 12AA (NEI 07-03A) 13.1.1.3.2.2.3 13.1.2.1.1
8.9	Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (Rev. 1, July 1993)	12.1 (NEI 07-08A)
8.10	Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable (Rev. 1-R, May 1977)	12.1 (NEI 07-08A) 12.3.4 Appendix 12AA Appendix 12AA (NEI 07-03A) 13.1.1.3.2.2.3 13.1.2.1.1
8.13	Instruction Concerning Prenatal Radiation Exposure (Rev. 3, June 1999)	12.1 (NEI 07-08A)
8.15	Acceptable Programs for Respiratory Protection (Rev. 1, October 1999)	12.1 (NEI 07-08A)
8.27	Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants (Rev. 0, March 1981)	12.1 (NEI 07-08A)
8.28	Audible-Alarm Dosimeters (Rev. 0, August 1981)	12.1 (NEI 07-08A)
8.29	Instruction Concerning Risks from Occupational Radiation Exposure (Rev. 1, February 1996)	12.1 (NEI 07-08A)
8.34	Monitoring Criteria and Methods To Calculate Occupational Radiation Doses (Rev. 0, July 1992)	12.1 (NEI 07-08A)
8.35	Planned Special Exposures (Rev. 0, June 1992)	12.1 (NEI 07-08A)
8.36	Radiation Dose to the Embryo/Fetus (Rev. 0, July 1992)	12.1 (NEI 07-08A)
8.38	Control of Access to High and Very High Radiation Areas of Nuclear Plants (Rev. 1, May 2006)	12.1 (NEI 07-08A)

a. NEI templates are incorporated by reference. See Table 1.6-201.
b. Division 5 of the regulatory guides applies to materials and plant protection. As appropriate, the Division 5 regulatory guide topics are addressed in the DCD and plant-specific security plans (i.e., Physical Security Plan, Training and Qualification Plan, Safeguards Contingency Plan, and Cyber Security Plan).

Table 1.9-2 (Sheet 1 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
TMI Action Plan Items			
I.A.1.1	Shift Technical Advisor	f	Resolved per NUREG-0933
I.A.1.2	Shift Supervisor Administrative Duties	f	Resolved per NUREG-0933
I.A.1.3	Shift Manning	f	Resolved per NUREG-0933
I.A.1.4	Long-Term Upgrading	f	See Subsection 13.1.1 Resolved per NUREG-0933
I.A.2.1(1)	Qualifications - Experience	f	Resolved per NUREG-0933
I.A.2.1(2) I.A.2.1(2)	Training Immediate Upgrading of RO & SRO Training and Qualifications, Training	f f	 Resolved per NUREG-0933
I.A.2.1(3)	Facility Certification of Competence and Fitness of Applicants for Operator and Senior Operator Licenses	f	Resolved per NUREG-0933
I.A.2.2	Training and Qualifications of Operations Personnel	c	
I.A.2.3	Administration of Training Programs	f	Resolved per NUREG-0933
I.A.2.4	NRR Participation in Inspector Training	d	Not applicable to new plants
I.A.2.5	Plant Drills	c	
I.A.2.6(1)	Revise Regulatory Guide 1.8	f	Resolved per NUREG-0933
I.A.2.6(2)	Staff Review of NRR 80-117	c	
I.A.2.6(3)	Revise 10 CFR 55	e	
I.A.2.6(4)	Operator Workshops	c	
I.A.2.6(5)	Develop Inspection Procedures for Training Programs	c	
I.A.2.6(6)	Nuclear Power Fundamentals	a	
I.A.2.7	Accreditation of Training Institutions	c	
I.A.3.1	Revise Scope of Criteria for Licensing Examinations	f	Resolved per NUREG-0933
I.A.3.2	Operator Licensing Program Changes	c	
I.A.3.3	Requirements for Operator Fitness	c	
I.A.3.4	Licensing of Additional Operations Personnel	c	
I.A.3.5	Establish Statement of Understanding with INPO and DOE	d	Not applicable to new plants

Table 1.9-2 (Sheet 2 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.A.4.1(1)	Short-Term Study of Training Simulators	c	
I.A.4.1(2)	Interim Changes in Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(1)	Research on Training Simulators	f	Resolved per NUREG-0933
I.A.4.2(2)	Upgrade Training Simulator Standards	f	Resolved per NUREG-0933
I.A.4.2(3)	Regulatory Guide on Training Simulators	f	See Subsection 1.9.3, item (2)(i) Resolved per NUREG-0933
I.A.4.2(4)	Review Simulators for Conformance to Criteria	f	Resolved per NUREG-0933
I.A.4.3	Feasibility Study of Procurement of NRC Training Simulator	d	Not applicable to new plants
I.A.4.4	Feasibility Study of NRC Engineering Computer	d	Not applicable to new plants
I.B.1.1(1)	Prepare Draft Criteria	c	
I.B.1.1(2)	Prepare Commission Paper	c	
I.B.1.1(3)	Issue Requirements for the Upgrading of Management and Technical Resources	c	
I.B.1.1(4)	Review Responses to Determine Acceptability	c	
I.B.1.1(5)	Review Implementation of the Upgrading Activities	c	
I.B.1.1(6)	Prepare Revisions to Regulatory Guides 1.33 and 1.8	e	
I.B.1.1(7)	Issue Regulatory Guides 1.33 and 1.8	e	
I.B.1.2(1)	Prepare Draft Criteria	c	
I.B.1.2(2)	Review Near-Term Operating License Facilities	c	
I.B.1.2(3)	Include Findings in the SER for Each Near-Term Operating License Facility	c	
I.B.1.3(1)	Require Licensees to Place Plant in Safest Shutdown Cooling Following a Loss of Safety Function Due to Personnel Error	d	Not applicable to new plants
I.B.1.3(2)	Use Existing Enforcement Options to Accomplish Safest Shutdown Cooling	d	Not applicable to new plants
I.B.1.3(3)	Use Non-Fiscal Approaches to Accomplish Safest Shutdown Cooling	d	Not applicable to new plants
I.B.2.1(1)	Verify the Adequacy of Management and Procedural Controls and Staff Discipline	d	Not applicable to new plants

Table 1.9-2 (Sheet 3 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.B.2.1(2)	Verify that Systems Required to Be Operable Are Properly Aligned	d	Not applicable to new plants
I.B.2.1(3)	Follow-up on Completed Maintenance Work Orders to Ensure Proper Testing and Return to Service	d	Not applicable to new plants
I.B.2.1(4)	Observe Surveillance Tests to Determine Whether Test Instruments Are Properly Calibrated	d	Not applicable to new plants
I.B.2.1(5)	Verify that Licensees Are Complying with Technical Specifications	d	Not applicable to new plants
I.B.2.1(6)	Observe Routine Maintenance	d	Not applicable to new plants
I.B.2.1(7)	Inspect Terminal Boards, Panels, and Instrument Racks for Unauthorized Jumpers and Bypasses	d	Not applicable to new plants
I.B.2.2	Resident Inspector at Operating Reactors	d	Not applicable to new plants
I.B.2.3	Regional Evaluations	d	Not applicable to new plants
I.B.2.4	Overview of Licensee Performance	d	Not applicable to new plants
I.C.1(1)	Small Break LOCAs	f	Resolved per NUREG-0933
I.C.1(2)	Inadequate Core Cooling	f	Resolved per NUREG-0933
I.C.1(3)	Transients and Accidents	f	Resolved per NUREG-0933
I.C.1(4)	Confirmatory Analyses of Selected Transients	c	
I.C.2	Shift and Relief Turnover Procedures	f	Resolved per NUREG-0933
I.C.3	Shift Supervisor Responsibilities	f	Resolved per NUREG-0933
I.C.4	Control Room Access	f	Resolved per NUREG-0933
I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	g	See Subsection 1.9.3, item (3)(i)
I.C.6	Procedures for Verification of Correct Performance of Operating Activities	f	Resolved per NUREG-0933
I.C.7	NSSS Vendor Review of Procedures	f	Resolved per NUREG-0933
I.C.8	Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants	f	Resolved per NUREG-0933

Table 1.9-2 (Sheet 4 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.C.9	Long-Term Program Plan for Upgrading of Procedures	c	See subsections 13.5.1 and 1.9.3, item (2)(ii)
I.D.1	Control Room Design Reviews	g	See subsection 1.9.3, item (2)(iii)
I.D.2	Plant Safety Parameter Display Console	g	See subsection 1.9.3, item (2)(iv)
I.D.3	Safety System Status Monitoring	c	See subsection 1.9.3, item (2)(v)
I.D.4	Control Room Design Standard	c	
I.D.5(1)	Operator-Process Communication	c	
I.D.5(2)	Plant Status and Post-Accident Monitoring	g	See subsection 1.9.4, item I.D.5(2)
I.D.5(3)	On-Line Reactor Surveillance System	c	See subsection 1.9.4, item I.D.5(3)
I.D.5(4)	Process Monitoring Instrumentation	c	
I.D.5(5)	Disturbance Analysis Systems	d	Not applicable to new plants
I.D.6	Technology Transfer Conference	d	Not applicable to new plants
I.E.1	Office for Analysis and Evaluation of Operational Data	d	Not applicable to new plants
I.E.2	Program Office Operational Data Evaluation	d	Not applicable to new plants
I.E.3	Operational Safety Data Analysis	d	Not applicable to new plants
I.E.4	Coordination of Licensee, Industry, and Regulatory Programs	d	Not applicable to new plants
I.E.5	Nuclear Plant Reliability Data Systems	d	Not applicable to new plants
I.E.6	Reporting Requirements	d	Not applicable to new plants
I.E.7	Foreign Sources	d	Not applicable to new plants
I.E.8	Human Error Rate Analysis	d	Not applicable to new plants
I.F.1	Expand QA List	c, j	See subsections 1.9.4.2.1, item I.F.1 and 1.9.3, item (3)(ii)
I.F.2(1)	Assure the Independence of the Organization Performing the Checking Function	a	See subsection 17.5
I.F.2(2)	Include QA Personnel in Review and Approval of Plant Procedures	g	See subsection 1.9.3, item (3)(iii)
I.F.2(3)	Include QA Personnel in All Design, Construction, Installation, Testing, and Operation Activities	g	See subsection 1.9.3, item (3)(iii)

Table 1.9-2 (Sheet 5 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
I.F.2(4)	Establish Criteria for Determining QA Requirements for Specific Classes of Equipment	a	See subsection 17.5
I.F.2(5)	Establish Qualification Requirements for QA and QC Personnel	a	See subsection 17.5
I.F.2(6)	Increase the Size of Licensees' QA Staff	f	Resolved per NUREG-0933
I.F.2(7)	Clarify that the QA Program Is a Condition of the Construction Permit and Operating License	a	See subsection 17.5
I.F.2(8)	Compare NRC QA Requirements with Those of Other Agencies	a	See subsection 17.5
I.F.2(9)	Clarify Organizational Reporting Levels for the QA Organization	f	Resolved per NUREG-0933
I.F.2(10)	Clarify Requirements for Maintenance of "As-Built" Documentation	a	See subsection 17.5
I.F.2(11)	Define Role of QA in Design and Analysis Activities	a	See subsection 17.5
I.G.1	Training Requirements	f, j	See subsection 1.9.4.2.1, item I.G.1 Resolved per NUREG-0933
I.G.2	Scope of Test Program	f, j	See subsection 1.9.4.2.1, item I.G.2 Resolved per NUREG-0933
II.A.1	Siting Policy Reformulation	c	
II.A.2	Site Evaluation of Existing Facilities	e	
II.B.1	Reactor Coolant System Vents	g	See subsection 1.9.3, item (2)(vi)
II.B.2	Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation	g	See subsection 1.9.3, item (2)(vii)
II.B.3	Post-Accident Sampling	g	See subsection 1.9.3, item (2)(viii)
II.B.4	Training for Mitigating Core Damage	f	Resolved per NUREG-0933
II.B.5(1)	Behavior of Severely Damaged Fuel	d	Not applicable to new plants
II.B.5(2)	Behavior of Core Melt	d	Not applicable to new plants
II.B.5(3)	Effect of Hydrogen Burning and Explosions on Containment Structures	d	Not applicable to new plants
II.B.6	Risk Reduction for Operating Reactors at Sites with High Population Densities	f	Resolved per NUREG-0933
II.B.7	Analysis of Hydrogen Control	e	
II.B.8	Rulemaking Proceedings on Degraded Core Accidents	g	See subsection 1.9.3, items (1)(i), (1)(xii), (2)(ix), (3)(iv), and (3)(v)
II.C.1	Interim Reliability Evaluation Program	c	

Table 1.9-2 (Sheet 6 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.C.2	Continuation of Interim Reliability Evaluation Program	c	
II.C.3	Systems Interaction	e	
II.C.4	Reliability Engineering	c	
II.D.1	Testing Requirements	g	See subsection 1.9.3, item (2)(x)
II.D.2	Research on Relief and Safety Valve Test Requirements	a	
II.D.3	Relief and Safety Valve Position Indication	g	See subsection 1.9.3, item (2)(xi)
II.E.1.1	Auxiliary Feedwater System Evaluation	g	See subsection 1.9.3, item (1)(ii)
II.E.1.2	Auxiliary Feedwater System Automatic Initiation and Flow Indication	g	See subsection 1.9.3, items (1)(ii) and (2)(xii)
II.E.1.3	Update Standard Review Plan and Develop Regulatory Guide	d, j	See subsection 1.9.4.2.1, item II.E.1.3 Resolved per NUREG-0933
II.E.2.1	Reliance on ECCS	e	
II.E.2.2	Research on Small Break LOCAs and Anomalous Transients	c	
II.E.2.3	Uncertainties in Performance Predictions	a	
II.E.3.1	Reliability of Power Supplies for Natural Circulation	g	See subsection 1.9.3, item (2)(xiii)
II.E.3.2	Systems Reliability	e	
II.E.3.3	Coordinated Study of Shutdown Heat Removal Requirements	e	
II.E.3.4	Alternate Concepts Research	c	
II.E.3.5	Regulatory Guide	e	
II.E.4.1	Dedicated Penetrations	g	See subsection 1.9.3, item (3)(vi)
II.E.4.2	Isolation Dependability	g	See subsection 1.9.3, item (2)(xiv)
II.E.4.3	Integrity Check	c	
II.E.4.4	Purging	g	See subsection 1.9.3, item (2)(xv)
II.E.5.1	Design Evaluation	b	
II.E.5.2	B&W Reactor Transient Response Task Force	b	
II.E.6.1	Test Adequacy Study	d, j	See subsection 1.9.4.2.1, item II.E.6.1 Resolved per NUREG-0933
II.F.1	Additional Accident Monitoring Instrumentation	g	See subsection 1.9.3, item (2)(xvii)
II.F.2	Identification of and Recovery from Conditions Leading to Inadequate Core Cooling	g	See subsection 1.9.3, item (2)(xviii)

Table 1.9-2 (Sheet 7 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.F.3	Instruments for Monitoring Accident Conditions	g	See subsection 1.9.3, item (2)(xix)
II.F.4	Study of Control and Protective Action Design Requirements	a	
II.F.5	Classification of Instrumentation, Control, and Electrical Equipment	d	Not applicable to new plants
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	g	See subsection 1.9.3, item (2)(xx)
II.H.1	Maintain Safety of TMI-2 and Minimize Environmental Impact	c	
II.H.2	Obtain Technical Data on the Conditions Inside the TMI-2 Containment Structure	b	
II.H.3	Evaluate and Feed Back Information Obtained from TMI	e	
II.H.4	Determine Impact of TMI on Socioeconomic and Real Property Values	d	Not applicable to new plants
II.J.1.1	Establish a Priority System for Conducting Vendor Inspections	d	Not applicable to new plants
II.J.1.2	Modify Existing Vendor Inspection Program	d	Not applicable to new plants
II.J.1.3	Increase Regulatory Control Over Present Non-Licensees	d	Not applicable to new plants
II.J.1.4	Assign Resident Inspectors to Reactor Vendors and Architect-Engineers	d	Not applicable to new plants
II.J.2.1	Reorient Construction Inspection Program	d	Not applicable to new plants
II.J.2.2	Increase Emphasis on Independent Measurement in Construction Inspection Program	d	Not applicable to new plants
II.J.2.3	Assign Resident Inspectors to All Construction Sites	d	Not applicable to new plants
II.J.3.1	Organization and Staffing to Oversee Design and Construction	f	See subsection 1.9.3, item (3)(vii) Not applicable to new plants
II.J.3.2	Issue Regulatory Guide	e	
II.J.4.1	Revise Deficiency Reporting Requirements	f	Resolved per NUREG-0933
II.K.1(1)	Review TMI-2 PNs and Detailed Chronology of the TMI-2 Accident	f	Resolved per NUREG-0933
II.K.1(2)	Review Transients Similar to TMI-2 That Have Occurred at Other Facilities and NRC Evaluation of Davis-Besse Event	b	
II.K.1(3)	Review Operating Procedures for Recognizing, Preventing, and Mitigating Void Formation in Transients and Accidents	f	Resolved per NUREG-0933
II.K.1(4)	Review Operating Procedures and Training Instructions	f	Resolved per NUREG-0933
II.K.1(5)	Safety-Related Valve Position Description	f	Resolved per NUREG-0933

Table 1.9-2 (Sheet 8 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(6)	Review Containment Isolation Initiation Design and Procedures	f	Resolved per NUREG-0933
II.K.1(7)	Implement Positive Position Controls on Valves That Could Compromise or Defeat AFW Flow	b	
II.K.1(8)	Implement Procedures That Assure Two Independent 100% AFW Flow Paths	b	
II.K.1(9)	Review Procedures to Assure That Radioactive Liquids and Gases Are Not Transferred out of Containment Inadvertently	f	Resolved per NUREG-0933
II.K.1(10)	Review and Modify Procedures for Removing Safety-Related Systems from Service	f, j	See subsection 1.9.4.2.1, item II.K.1(10) Resolved per NUREG-0933
II.K.1(11)	Make All Operating and Maintenance Personnel Aware of the Seriousness and Consequences of the Erroneous Actions Leading up to, and in Early Phases of, the TMI-2 Accident	f	Resolved per NUREG-0933
II.K.1(12)	One Hour Notification Requirement and Continuous Communications Channels	f	Resolved per NUREG-0933
II.K.1(13)	Propose Technical Specification Changes Reflecting Implementation of All Bulletin Items	f, j	See subsection 1.9.4.2.1, item II.K.1(13) Resolved per NUREG-0933
II.K.1(14)	Review Operating Modes and Procedures to Deal with Significant Amounts of Hydrogen	f	Resolved per NUREG-0933
II.K.1(15)	For Facilities with Non-Automatic AFW Initiation, Provide Dedicated Operator in Continuous Communication with CR to Operate AFW	f	Resolved per NUREG-0933
II.K.1(16)	Implement Procedures That Identify PZR PORV "Open" Indications and That Direct Operator to Close Manually at "Reset" Setpoint	f	Resolved per NUREG-0933
II.K.1(17)	Trip PZR Level Bistable so That PZR Low Pressure Will Initiate Safety Injection	f, j	See subsection 1.9.4.2.1, item II.K.1(17) Resolved per NUREG-0933
II.K.1(18)	Develop Procedures and Train Operators on Methods of Establishing and Maintaining Natural Circulation	b	
II.K.1(19)	Describe Design and Procedure Modifications to Reduce Likelihood of Automatic PZR PORV Actuation in Transients	b	
II.K.1(20)	Provide Procedures and Training to Operators for Prompt Manual Reactor Trip for LOFW, TT, MSIV Closure, LOOP, LOSG Level, and LO PZR Level	b	
II.K.1(21)	Provide Automatic Safety-Grade Anticipatory Reactor Trip for LOFW, TT, or Significant Decrease in SG Level	b	

Table 1.9-2 (Sheet 9 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.1(22)	Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When FW System Not Operable	g	See subsection 1.9.3, item (2)(xxi)
II.K.1(23)	Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation Safety Systems	b	
II.K.1(24)	Perform LOCA Analyses for a Range of Small-Break Sizes and a Range of Time Lapses Between Reactor Trip and RCP Trip	e, j	See subsection 1.9.4.2.1, item II.K.1(24)
II.K.1(25)	Develop Operator Action Guidelines	e	
II.K.1(26)	Revise Emergency Procedures and Train ROs and SROs	f	Resolved per NUREG-0933
II.K.1(27)	Provide Analyses and Develop Guidelines and Procedures for Inadequate Core Cooling Conditions	e	
II.K.1(28)	Provide Design That Will Assure Automatic RCP Trip for All Circumstances Where Required	e	
II.K.2(1)	Upgrade Timeliness and Reliability of AFW System	b	
II.K.2(2)	Procedures and Training to Initiate and Control AFW Independent of Integrated Control System	b	
II.K.2(3)	Hard-Wired Control-Grade Anticipatory Reactor Trips	b	
II.K.2(4)	Small-Break LOCA Analysis, Procedures and Operator Training	b	
II.K.2(5)	Complete TMI-2 Simulator Training for All Operators	b	
II.K.2(6)	Reevaluate Analysis of Dual-Level Setpoint Control	b	
II.K.2(7)	Reevaluate Transient of September 24, 1977	b	
II.K.2(8)	Continued Upgrading of AFW System	e	
II.K.2(9)	Analysis and Upgrading of Integrated Control System	e	
II.K.2(10)	Hard-Wired Safety-Grade Anticipatory Reactor Trips	b	See subsection 1.9.3, item (2)(xxiii)
II.K.2(11)	Operator Training and Drilling	b	
II.K.2(12)	Transient Analysis and Procedures for Management of Small Breaks	e	
II.K.2(13)	Thermal-Mechanical Report on Effect of HPI on Vessel Integrity for Small-Break LOCA With No AFW	b	
II.K.2(14)	Demonstrate That Predicted Lift Frequency of PORVs and SVs Is Acceptable	b	
II.K.2(15)	Analysis of Effects of Slug Flow on Once-Through Steam Generator Tubes After Primary System Voiding	b	
II.K.2(16)	Impact of RCP Seal Damage Following Small-Break LOCA With Loss of Offsite Power	g	See subsection 1.9.3, item (1)(iii)
II.K.2(17)	Analysis of Potential Voiding in RCS During Anticipated Transients	b	
II.K.2(18)	Analysis of Loss of Feedwater and Other Anticipated Transients	e	
II.K.2(19)	Benchmark Analysis of Sequential AFW Flow to Once-Through Steam Generator	b	

Table 1.9-2 (Sheet 10 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.2(20)	Analysis of Steam Response to Small-Break LOCA	b	
II.K.2(21)	LOFT L3-1 Predictions	b	
II.K.3(1)	Install Automatic PORV Isolation System and Perform Operational Test	g	See subsection 1.9.3, item (1)(iv)
II.K.3(2)	Report on Overall Safety Effect of PORV Isolation System	g	See subsection 1.9.3, item (1)(iv)
II.K.3(3)	Report Safety and Relief Valve Failures Promptly and Challenges Annually	f	Resolved per NUREG-0933
II.K.3(4)	Review and Upgrade Reliability and Redundancy of Non-Safety Equipment for Small-Break LOCA Mitigation	e	
II.K.3(5)	Automatic Trip of Reactor Coolant Pumps	f, j	See subsection 1.9.4.2.1, item II.K.3(5) Resolved per NUREG-0933
II.K.3(6)	Instrumentation to Verify Natural Circulation	e	
II.K.3(7)	Evaluation of PORV Opening Probability During Overpressure Transient	b	
II.K.3(8)	Further Staff Consideration of Need for Diverse Decay Heat Removal Method Independent of SGs	e	
II.K.3(9)	Proportional Integral Derivative Controller Modification	g	See subsection 1.9.4, item II.K.3(9)
II.K.3(10)	Anticipatory Trip Modification Proposed by Some Licensees to Confine Range of Use to High Power Levels	f	Resolved per NUREG-0933
II.K.3(11)	Control Use of PORV Supplied by Control Components, Inc. Until Further Review Complete	f	Resolved per NUREG-0933
II.K.3(12)	Confirm Existence of Anticipatory Trip Upon Turbine Trip	f	Resolved per NUREG-0933
II.K.3(13)	Separation of HPCI and RCIC System Initiation Levels	b	
II.K.3(14)	Isolation of Isolation Condensers on High Radiation	b	
II.K.3(15)	Modify Break Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems	b	
II.K.3(16)	Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification	b	
II.K.3(17)	Report on Outage of ECC Systems - Licensee Report and Technical Specification Changes	b	
II.K.3(18)	Modification of ADS Logic - Feasibility Study and Modification for Increased Diversity for Some Event Sequences	g	See subsection 1.9.3, item (1)(vii)
II.K.3(19)	Interlock on Recirculation Pump Loops	b	
II.K.3(20)	Loss of Service Water for Big Rock Point	b	
II.K.3(21)	Restart of Core Spray and LPCI Systems on Low Level - Design and Modification	b	
II.K.3(22)	Automatic Switchover of RCIC System Suction - Verify Procedures and Modify Design	b	

Table 1.9-2 (Sheet 11 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(23)	Central Water Level Recording	e	
II.K.3(24)	Confirm Adequacy of Space Cooling for HPCI and RCIC Systems	b	
II.K.3(25)	Effect of Loss of AC Power on Pump Seals	g	See subsection 1.9.3, item (1)(iii)
II.K.3(26)	Study Effect on RHR Reliability of Its Use for Fuel Pool Cooling	e	
II.K.3(27)	Provide Common Reference Level for Vessel Level Instrumentation	b	
II.K.3(28)	Study and Verify Qualification of Accumulators on ADS Valves	g	See subsection 1.9.3, item (1)(x)
II.K.3(29)	Study to Demonstrate Performance of Isolation Condensers with Non-Condensibles	b	
II.K.3(30)	Revised Small-Break LOCA Methods to Show Compliance with 10 CFR 50, Appendix K	f	Resolved per NUREG-0933
II.K.3(31)	Plant-Specific Calculations to Show Compliance with 10 CFR 50.46	f	Resolved per NUREG-0933
II.K.3(32)	Provide Experimental Verification of Two-Phase Natural Circulation Models	e	
II.K.3(33)	Evaluate Elimination of PORV Function	e	
II.K.3(34)	Relap-4 Model Development	e	
II.K.3(35)	Evaluation of Effects of Core Flood Tank Injection on Small-Break LOCAs	e	
II.K.3(36)	Additional Staff Audit Calculations of B&W Small-Break LOCA Analyses	e	
II.K.3(37)	Analysis of B&W Response to Isolated Small-Break LOCA	e	
II.K.3(38)	Analysis of Plant Response to a Small-Break LOCA in the Pressurizer Spray Line	e	
II.K.3(39)	Evaluation of Effects of Water Slugs in Piping Caused by HPI and CFT Flows	e	
II.K.3(40)	Evaluation of RCP Seal Damage and Leakage During a Small-Break LOCA	e	
II.K.3(41)	Submit Predictions for LOFT Test L3-6 with RCPs Running	e	
II.K.3(42)	Submit Requested Information on the Effects of Non Condensible Gases	e	
II.K.3(43)	Evaluation of Mechanical Effects of Slug Flow on Steam Generator Tubes	e	
II.K.3(44)	Evaluation of Anticipated Transients with Single Failure to Verify No Significant Fuel Failure	b	
II.K.3(45)	Evaluate Depressurization with Other Than Full ADS	b	
II.K.3(46)	Response to List of Concerns from ACRS Consultant	b	
II.K.3(47)	Test Program for Small-Break LOCA Model Verification Pretest Prediction, Test Program, and Model Verification	e	
II.K.3(48)	Assess Change in Safety Reliability as a Result of Implementing B&OTF Recommendations	e	

Table 1.9-2 (Sheet 12 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
II.K.3(49)	Review of Procedures (NRC)	e	
II.K.3(50)	Review of Procedures (NSSS Vendors)	e	
II.K.3(51)	Symptom-Based Emergency Procedures	e	
II.K.3(52)	Operator Awareness of Revised Emergency Procedures	e	
II.K.3(53)	Two Operators in Control Room	e	
II.K.3(54)	Simulator Upgrade for Small-Break LOCAs	e	
II.K.3(55)	Operator Monitoring of Control Board	e	
II.K.3(56)	Simulator Training Requirements	e	
II.K.3(57)	Identify Water Sources Prior to Manual Activation of ADS	b	
III.A.1.1(1)	Implement Action Plan Requirements for Promptly Improving Licensee Emergency Preparedness	f	Resolved per NUREG-0933
III.A.1.1(2)	Perform an Integrated Assessment of the Implementation	f	Not applicable to new plants
III.A.1.2	Upgrade Licensee Emergency Support Facilities	g	See subsection 1.9.3, item (2)(xxv)
III.A.1.3(1)	Maintain Supplies of Thyroid-Blocking Agent - Workers	c	
III.A.1.3(2)	Maintain Supplies of Thyroid-Blocking Agent - Public	c	
III.A.2.1(1)	Publish Proposed Amendments to the Rules	d	Resolved per NUREG-0933
III.A.2.1(2)	Conduct Public Regional Meetings	d	Not applicable to new plants
III.A.2.1(3)	Prepare Final Commission Paper Recommending Adoption of Rules	d	Not applicable to new plants
III.A.2.1(4)	Revise Inspection Program to Cover Upgraded Requirements	d	Resolved per NUREG-0933
III.A.2.2	Development of Guidance and Criteria	d	Resolved per NUREG-0933
III.A.3.1(1)	Define NRC Role in Emergency Situations	c	
III.A.3.1(2)	Revise and Upgrade Plans and Procedures for the NRC Emergency Operations Center	c	
III.A.3.1(3)	Revise Manual Chapter 0502, Other Agency Procedures, and NUREG-0610	c	
III.A.3.1(4)	Prepare Commission Paper	c	
III.A.3.1(5)	Revise Implementing Procedures and Instructions for Regional Offices	c	
III.A.3.2	Improve Operations Centers	c	
III.A.3.3	Communications	d	See subsection 9.5.2.5.2 Resolved per NUREG-0933
III.A.3.4	Nuclear Data Link	c	
III.A.3.5	Training, Drills, and Tests	c	
III.A.3.6(1)	Interaction of NRC and Other Agencies - International	c	

Table 1.9-2 (Sheet 13 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.A.3.6(2)	Federal	c	
III.A.3.6(3)	State and Local	c	
III.B.1	Transfer of Responsibilities to FEMA	c	
III.B.2(1)	The Licensing Process	c	
III.B.2(2)	Federal Guidance	c	
III.C.1(1)	Review Publicly Available Documents	d	Not applicable to new plants
III.C.1(2)	Recommend Publication of Additional Information	d	Not applicable to new plants
III.C.1(3)	Program of Seminars for News Media Personnel	d	Not applicable to new plants
III.C.2(1)	Develop Policy and Procedures for Dealing With Briefing Requests	d	Not applicable to new plants
III.C.2(2)	Provide Training for Members of the Technical Staff	d	Not applicable to new plants
III.D.1.1(1)	Review Information Submitted by Licensees Pertaining to Reducing Leakage from Operating Systems	g	See subsection 1.9.3, item (2)(xxvi)
III.D.1.1(2)	Review Information on Provisions for Leak Detection	a	
III.D.1.1(3)	Develop Proposed System Acceptance Criteria	a	
III.D.1.2	Radioactive Gas Management	a	
III.D.1.3(1)	Decide Whether Licensees Should Perform Studies and Make Modifications	a	
III.D.1.3(2)	Review and Revise SRP	a	
III.D.1.3(3)	Require Licensees to Upgrade Filtration Systems	a	
III.D.1.3(4)	Sponsor Studies to Evaluate Charcoal Adsorber	c	
III.D.1.4	Radwaste System Design Features to Aid in Accident Recovery and Decontamination	a	
III.D.2.1(1)	Evaluate the Feasibility and Perform a Value-Impact Analysis of Modifying Effluent-Monitoring Design Criteria	a	
III.D.2.1(2)	Study the Feasibility of Requiring the Development of Effective Means for Monitoring and Sampling Noble Gases and Radioiodine Released to the Atmosphere	a	
III.D.2.1(3)	Revise Regulatory Guides	a	
III.D.2.2(1)	Perform Study of Radioiodine, Carbon-14, and Tritium Behavior	c	
III.D.2.2(2)	Evaluate Data Collected at Quad Cities	e	
III.D.2.2(3)	Determine the Distribution of the Chemical Species of Radioiodine in Air-Water-Steam Mixtures	e	
III.D.2.2(4)	Revise SRP and Regulatory Guides	e	
III.D.2.3(1)	Develop Procedures to Discriminate Between Sites/Plants	c	
III.D.2.3(2)	Discriminate Between Sites and Plants That Require Consideration of Liquid Pathway Interdiction Techniques	c	
III.D.2.3(3)	Establish Feasible Method of Pathway Interdiction	c	

Table 1.9-2 (Sheet 14 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
III.D.2.3(4)	Prepare a Summary Assessment	c	
III.D.2.4(1)	Study Feasibility of Environmental Monitors	c	
III.D.2.4(2)	Place 50 TLDs Around Each Site	d	Not applicable to new plants
III.D.2.5	Offsite Dose Calculation Manual	c	
III.D.2.6	Independent Radiological Measurements	d	Not applicable to new plants
III.D.3.1	Radiation Protection Plans	c	
III.D.3.2(1)	Amend 10 CFR 20	d	Not applicable to new plants
III.D.3.2(2)	Issue a Regulatory Guide	d	Not applicable to new plants
III.D.3.2(3)	Develop Standard Performance Criteria	d	Not applicable to new plants
III.D.3.2(4)	Develop Method for Testing and Certifying Air-Purifying Respirators	d	Not applicable to new plants
III.D.3.3	In-plant Radiation Monitoring	g COL Item 12.3-2	See subsection 1.9.3, item (2)(xxvii) 12.3.4, Appendix 12AA
III.D.3.4	Control Room Habitability	g	See subsection 1.9.3, item (2)(xxviii)
III.D.3.5(1)	Develop Format for Data To Be Collected by Utilities Regarding Total Radiation Exposure to Workers	d	Not applicable to new plants
III.D.3.5(2)	Investigate Methods of Obtaining Employee Health Data by Nonlegislative Means	d	Not applicable to new plants
III.D.3.5(3)	Revise 10 CFR 20	d	Not applicable to new plants
IV.A.1	Seek Legislative Authority	d	Not applicable to new plants
IV.A.2	Revise Enforcement Policy	d	Not applicable to new plants
IV.B.1	Revise Practices for Issuance of Instructions and Information to Licensees	d	Not applicable to new plants
IV.C.1	Extend Lessons Learned from TMI to Other NRC Programs	c	
IV.D.1	NRC Staff Training	d	Not applicable to new plants
IV.E.1	Expand Research on Quantification of Safety Decision-Making	d	Not applicable to new plants
IV.E.2	Plan for Early Resolution of Safety Issues	d	Not applicable to new plants
IV.E.3	Plan for Resolving Issues at the CP Stage	d	Not applicable to new plants

Table 1.9-2 (Sheet 15 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
IV. E.4	Resolve Generic Issues by Rulemaking	d	Not applicable to new plants
IV.E.5	Assess Currently Operating Reactors	c	
IV.F.1	Increased OIE Scrutiny of the Power-Ascension Test Program	c	
IV.F.2	Evaluate the Impacts of Financial Disincentives to the Safety of Nuclear Power Plants	c	
IV.G.1	Develop a Public Agenda for Rulemaking	d	Not applicable to new plants
IV.G.2	Periodic and Systematic Reevaluation of Existing Rules	d	Not applicable to new plants
IV.G.3	Improve Rulemaking Procedures	d	Not applicable to new plants
IV.G.4	Study Alternatives for Improved Rulemaking Process	d	Not applicable to new plants
IV.H.1	NRC Participation in the Radiation Policy Council	d	Not applicable to new plants
V.A.1	Develop NRC Policy Statement on Safety	d	Not applicable to new plants
V.B.1	Study and Recommend, as Appropriate, Elimination of Nonsafety Responsibilities	d	Not applicable to new plants
V.C.1	Strengthen the Role of Advisory Committee on Reactor Safeguards	d	Not applicable to new plants
V.C.2	Study Need for Additional Advisory Committees	d	Not applicable to new plants
V.C.3	Study the Need to Establish an Independent Nuclear Safety Board	d	Not applicable to new plants
V.D.1	Improve Public and Intervenor Participation in the Hearing Process	d	Not applicable to new plants
V.D.2	Study Construction-During-Adjudication Rules	d	Not applicable to new plants
V.D.3	Reexamine Commission Role in Adjudication	d	Not applicable to new plants
V.D.4	Study the Reform of the Licensing Process	d	Not applicable to new plants
V.E.1	Study the Need for TMI-Related Legislation	d	Not applicable to new plants
V.F.1	Study NRC Top Management Structure and Process	d	Not applicable to new plants
V.F.2	Reexamine Organization and Functions of the NRC Offices	d	Not applicable to new plants
V.F.3	Revise Delegations of Authority to Staff	d	Not applicable to new plants
V.F.4	Clarify and Strengthen the Respective Roles of Chairman, Commission, and Executive Director for Operations	d	Not applicable to new plants

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Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
V.F.5	Authority to Delegate Emergency Response Functions to a Single Commissioner	d	Not applicable to new plants
V.G.1	Achieve Single Location, Long-Term	d	Not applicable to new plants
V.G.2	Achieve Single Location, Interim	d	Not applicable to new plants
Task Action Plan Items			
A-1	Water Hammer (former USI)	g	See subsection 1.9.4, item A-1
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems (former USI)	g	See subsection 1.9.4, item A-2
A-3	Westinghouse Steam Generator Tube Integrity (former USI)	g COL Item 5.4-1	See subsection 1.9.4, item A-3 5.4.2.5
A-4	CE Steam Generator Tube Integrity (former USI)	b	
A-5	B&W Steam Generator Tube Integrity (former USI)	b	
A-6	Mark I Short-Term Program (former USI)	b	
A-7	Mark I Long-Term Program (former USI)	b	
A-8	Mark II Containment Pool Dynamic Loads Long-Term Program (former USI)	b	
A-9	ATWS (former USI)	g	See subsection 1.9.4, item A-9
A-10	BWR Feedwater Nozzle Cracking (former USI)	b	
A-11	Reactor Vessel Materials Toughness (former USI)	g	See subsection 1.9.4, item A-11
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (former USI)	g	See subsection 1.9.4, item A-12
A-13	Snubber Operability Assurance	g	See subsection 1.9.4, item A-13
A-14	Flaw Detection	a	
A-15	Primary Coolant System Decontamination and Steam Generator Chemical Cleaning	c	
A-16	Steam Effects on BWR Core Spray Distribution	b	
A-17	Systems Interactions in Nuclear Power Plants (former USI)	c, j	See subsection 1.9.4.2.2, item A-17
A-18	Pipe Rupture Design Criteria	a	
A-19	Digital Computer Protection System	d	Not applicable to new plants
A-20	Impacts of the Coal Fuel Cycle	d	Not applicable to new plants
A-21	Main Steam Line Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification	a	
A-22	PWR Main Steam Line Break - Core, Reactor Vessel and Containment Building Response	a	

**Table 1.9-2 (Sheet 17 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues**

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-23	Containment Leak Testing	d COL Item 6.2-1	6.2.5.1
A-24	Qualification of Class e Safety-Related Equipment (former USI)	g	See subsection 1.9.4, item A-24
A-25	Non-Safety Loads on Class e Power Sources	g	See subsection 1.9.4, item A-25
A-26	Reactor Vessel Pressure Transient Protection (former USI)	g	See subsection 1.9.4, item A-26
A-27	Reload Applications	d	Not applicable to new plants
A-28	Increase in Spent Fuel Pool Storage Capacity	g	See subsection 1.9.4, item A-28
A-29	Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage	c, j	See subsection 1.9.4.2.2, item A-29
A-30	Adequacy of Safety-Related DC Power Supplies	e	
A-31	RHR Shutdown Requirements (former USI)	g	See subsection 1.9.4, item A-31
A-32	Missile Effects	e	
A-33	NEPA Review of Accident Risks	i	
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	e	
A-35	Adequacy of Offsite Power Systems	g	See subsection 1.9.4, item A-35
A-36	Control of Heavy Loads Near Spent Fuel (former USI)	g	See subsection 1.9.4, item A-36
A-37	Turbine Missiles	a	
A-38	Tornado Missiles	a	
A-39	Determination of Safety Relief Valve Pool Dynamic Loads and Temperature Limits (former USI)	b	See subsection 1.9.4, item A-39
A-40	Seismic Design Criteria - Short Term Program (former USI)	g	See subsection 1.9.4, item A-40
A-41	Long Term Seismic Program	c	
A-42	Pipe Cracks in Boiling Water Reactors (former USI)	b	
A-43	Containment Emergency Sump Performance (former USI)	g	See subsection 1.9.4, item A-43
A-44	Station Blackout (former USI)	g	See subsection 1.9.4, item A-44
A-45	Shutdown Decay Heat Removal Requirements (former USI)	c	
A-46	Seismic Qualification of Equipment in Operating Plants (former USI)	g	See subsection 1.9.4, item A-46
A-47	Safety Implications of Control Systems (former USI)	g	See subsection 1.9.4, item A-47

Table 1.9-2 (Sheet 18 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	g	See subsection 1.9.4, item A-48
A-49	Pressurized Thermal Shock (former USI)	g	See subsection 1.9.4, item A-49
B-1	Environmental Technical Specifications	d	Not applicable to new plants
B-2	Forecasting Electricity Demand	d	Not applicable to new plants
B-3	Event Categorization	a	
B-4	ECCS Reliability	e	
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	c, j	See subsection 1.9.4.2.2, item B-5
B-6	Loads, Load Combinations, Stress Limits	e	
B-7	Secondary Accident Consequence Modeling	a	
B-8	Locking Out of ECCS Power Operated Valves	a	
B-9	Electrical Cable Penetrations of Containment	c	
B-10	Behavior of BWR Mark III Containments	b	
B-11	Subcompartment Standard Problems	d	Not applicable to new plants
B-12	Containment Cooling Requirements (Non-LOCA)	c	
B-13	Marviken Test Data Evaluation	d	Not applicable to new plants
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	e	
B-15	CONTEMPT Computer Code Maintenance	a	
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	e	
B-17	Criteria for Safety-Related Operator Actions	c	See subsection 1.9.4, item B-17
B-18	Vortex Suppression Requirements for Containment Sumps	e	
B-19	Thermal-Hydraulic Stability	c	
B-20	Standard Problem Analysis	d	Not applicable to new plants
B-21	Core Physics	a	
B-22	LWR Fuel	a	See subsection 1.9.4, item B-22
B-23	LMFBR Fuel	a	
B-24	Seismic Qualification of Electrical and Mechanical Components	e	
B-25	Piping Benchmark Problems	d	Not applicable to new plants
B-26	Structural Integrity of Containment Penetrations	c	
B-27	Implementation and Use of Subsection NF	d	Not applicable to new plants

Table 1.9-2 (Sheet 19 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-28	Radionuclide/Sediment Transport Program	d	Not applicable to new plants
B-29	Effectiveness of Ultimate Heat Sinks	d	See subsection 1.9.4, item B-29 Not applicable to new plants
B-30	Design Basis Floods and Probability	d	Not applicable to new plants
B-31	Dam Failure Model	a	
B-32	Ice Effects on Safety-Related Water Supplies	e	See subsection 1.9.4, item B-32
B-33	Dose Assessment Methodology	d	Not applicable to new plants
B-34	Occupational Radiation Exposure Reduction	e	
B-35	Confirmation of Appendix I Models for Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light Water Cooled Power Reactors	d	Not applicable to new plants
B-36	Develop Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	g	See subsection 1.9.4, item B-36
B-37	Chemical Discharges to Receiving Waters	d	Not applicable to new plants
B-38	Reconnaissance Level Investigations	a	
B-39	Transmission Lines	a	
B-40	Effects of Power Plant Entrainment on Plankton	a	
B-41	Impacts on Fisheries	a	
B-42	Socioeconomic Environmental Impacts	d	Not applicable to new plants
B-43	Value of Aerial Photographs for Site Evaluation	d	Not applicable to new plants
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	d	Not applicable to new plants
B-45	Need for Power - Energy Conservation	e	
B-46	Cost of Alternatives in Environmental Design	a	
B-47	Inservice Inspection of Supports - Classes 1, 2, 3, and MC Components	a	
B-48	BWR CRD Mechanical Failure (Collet Housing)	b	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	d	Not applicable to new plants
B-50	Post-Operating Basis Earthquake Inspections	a	
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	e	
B-52	Fuel Assembly Seismic and LOCA Responses	e	

Table 1.9-2 (Sheet 20 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
B-53	Load Break Switch	g	See subsection 1.9.4, item B-53
B-54	Ice Condenser Containments	c	
B-55	Improved Reliability of Target Rock Safety Relief Valves	b	
B-56	Diesel Reliability	g	See subsection 1.9.4, item B-56
B-57	Station Blackout	e	
B-58	Passive Mechanical Failures	c	
B-59	(N-1) Loop Operation in BWRs and PWRs	d	Not applicable to new plants
B-60	Loose Parts Monitoring System	c	
B-61	Allowable ECCS Equipment Outage Periods	g	See subsection 1.9.4, item B-61
B-62	Reexamination of Technical Bases for Establishing SLs, LSSSs, and Reactor Protection System Trip Functions	a	
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	g	See subsection 1.9.4, item B-63
B-64	Decommissioning of Reactors	f	Resolved per NUREG-0933
B-65	Iodine Spiking	a	
B-66	Control Room Infiltration Measurements	g	See subsection 1.9.4, item B-66
B-67	Effluent and Process Monitoring Instrumentation	e	
B-68	Pump Overspeed During LOCA	a	
B-69	ECCS Leakage Ex-Containment	e	
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps	c	
B-71	Incident Response	e	
B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles	d	Not applicable to new plants
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	e	
C-1	Assurance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electrical Equipment	g	See subsection 1.9.4, item C-1
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	c	
C-3	Insulation Usage Within Containment	e	
C-4	Statistical Methods for ECCS Analysis	d	See subsection 1.9.4, item C-4 Not applicable to new plants

Table 1.9-2 (Sheet 21 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
C-5	Decay Heat Update	d	See subsection 1.9.4, item C-5 Not applicable to new plants
C-6	LOCA Heat Sources	d	See subsection 1.9.4, item C-6 Not applicable to new plants
C-7	PWR System Piping	c	
C-8	Main Steam Line Leakage Control Systems	b	
C-9	RHR Heat Exchanger Tube Failures	a	
C-10	Effective Operation of Containment Sprays in a LOCA	g	See subsection 1.9.4, item C-10
C-11	Assessment of Failure and Reliability of Pumps and Valves	c	
C-12	Primary System Vibration Assessment	c	
C-13	Non-Random Failures	e	
C-14	Storm Surge Model for Coastal Sites	a	
C-15	NUREG Report for Liquids Tank Failure Analysis	a	
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	a	
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	g	See subsection 1.9.4, item C-17
D-1	Advisability of a Seismic Scram	a	
D-2	Emergency Core Cooling System Capability for Future Plants	a	
D-3	Control Rod Drop Accident	c	
New Generic Issues			
1.	Failures in Air-Monitoring, Air-Cleaning, and Ventilating Systems	a	
2.	Failure of Protective Devices on Essential Equipment	a	
3.	Set Point Drift in Instrumentation	c	
4.	End-of-Life and Maintenance Criteria	c	
5.	Design Check and Audit of Balance-of-Plant Equipment	e	
6.	Separation of Control Rod from Its Drive and BWR High Rod Worth Events	c	
7.	Failures Due to Flow-Induced Vibrations	a	
8.	Inadvertent Actuation of Safety Injection in PWRs	e	
9.	Reevaluation of Reactor Coolant Pump Trip Criteria	e	
10.	Surveillance and Maintenance of TIP Isolation Valves and Squib Charges	a	
11.	Turbine Disc Cracking	e	
12.	BWR Jet Pump Integrity	b	
13.	Small Break LOCA from Extended Overheating of Pressurizer Heaters	a	

Table 1.9-2 (Sheet 22 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
14.	PWR Pipe Cracks	c, j	See subsection 1.9.4.2.3, item 14
15.	Radiation Effects on Reactor Vessel Supports	c	See subsection 1.9.4, item 15
16.	BWR Main Steam Isolation Valve Leakage Control Systems	e	
17.	Loss of Offsite Power Subsequent to LOCA	a	
18.	Steam Line Break with Consequential Small LOCA	e	
19.	Safety Implications of Nonsafety Instrument and Control Power Supply Bus	e	
20.	Effects of Electromagnetic Pulse on Nuclear Power Plants	c	
21.	Vibration Qualification of Equipment	a	
22.	Inadvertent Boron Dilution Events	c, j	See subsection 1.9.4.2.3, item 22
23.	Reactor Coolant Pump Seal Failures	c	See subsection 1.9.4, item 23
24.	Automatic Emergency Core Cooling System Switch to Recirculation	a, j	See subsection 1.9.4.2.3, item 24
25.	Automatic Air Header Dump on BWR Scram System	b	
26.	Diesel Generator Loading Problems Related to SIS Reset on Loss of Offsite Power	e	
27.	Manual vs. Automated Actions	e	
28.	Pressurized Thermal Shock	e	
29.	Bolting Degradation or Failure in Nuclear Power Plants	c	See subsection 1.9.4, item 29
30.	Potential Generator Missiles - Generator Rotor Retaining Rings	a	
31.	Natural Circulation Cooldown	e	
32.	Flow Blockage in Essential Equipment Caused by Corbicula	e	
33.	Correcting Atmospheric Dump Valve Opening Upon Loss of Integrated Control System Power	e	
34.	RCS Leak	a	
35.	Degradation of Internal Appurtenances in LWRs	a	
36.	Loss of Service Water	c	
37.	Steam Generator Overfill and Combined Primary and Secondary Blowdown	e	
38.	Potential Recirculation System Failure as a Consequence of Injection of Containment Paint Flakes or Other Fine Debris	a	
39.	Potential for Unacceptable Interaction Between the CRD System and Non-Essential Control Air System	e	
40.	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	b	
41.	BWR Scram Discharge Volume Systems	b	
42.	Combination Primary/Secondary System LOCA	e	

Table 1.9-2 (Sheet 23 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
43.	Reliability of Air Systems	f, j	See subsection 1.9.4.2.3, item 43 Resolved per NUREG-0933
44.	Failure of Saltwater Cooling System	e	
45.	Inoperability of Instrumentation Due to Extreme Cold Weather	g	See subsection 1.9.4, item 45
46.	Loss of 125 Volt DC Bus	e	
47.	Loss of Off-Site Power	c	
48.	LCO for Class e Vital Instrument Buses in Operating Reactors	e	
49.	Interlocks and LCOs for Redundant Class e Tie Breakers	e	
50.	Reactor Vessel Level Instrumentation in BWRs	c	
51.	Proposed Requirements for Improving the Reliability of Open Cycle Service Water Systems	g	See subsection 1.9.4, item 51
52.	SSW Flow Blockage by Blue Mussels	e	
53.	Consequences of a Postulated Flow Blockage Incident in a BWR	a	
54.	Valve Operator-Related Events Occurring During 1978, 1979, and 1980	e	
55.	Failure of Class e Safety-Related Switchgear Circuit Breakers to Close on Demand	a	
56.	Abnormal Transient Operating Guidelines as Applied to a Steam Generator Overfill Event	e	
57.	Effects of Fire Protection System Actuation	c	See subsection 1.9.4, item 57
58.	Inadvertent Containment Flooding	a	
59.	Technical Specification Requirements for Plant Shutdown when Equipment for Safe Shutdown is Degraded or Inoperable	d	Not applicable to new plants
60.	Lamellar Tearing of Reactor Systems Structural Supports	e	
61.	SRV Line Break Inside the BWR Wetwell Airspace of Mark I and II Containments	c	
62.	Reactor Systems Bolting Applications	e	
63.	Use of Equipment Not Classified as Essential to Safety in BWR Transient Analysis	a	
64.	Identification of Protection System Instrument Sensing Lines	c	
65.	Probability of Core-Melt Due to Component Cooling Water System Failures	e	
66.	Steam Generator Requirements	c	
67.2.1	Integrity of Steam Generator Tube Sleeves	d	Not applicable to new plants
67.3.1	Steam Generator Overfill	e	
67.3.2	Pressurized Thermal Shock	e	

Table 1.9-2 (Sheet 24 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
67.3.3	Improved Accident Monitoring	e, j	See subsection 1.9.4.2.3, item 67.3.3
67.3.4	Reactor Vessel Inventory Measurements	e	
67.4.1	RCP Trip	e	
67.4.2	Control Room Design Review	e	
67.4.3	Emergency Operating Procedures	e	
67.5.1	Reassessment of SGTR Design Basis	d	
67.5.1	Reassessment of Radiological Consequences	d	Not applicable to new plants
67.5.2	Reevaluation of SGTR Design Basis	d	Not applicable to new plants
67.5.3	Secondary System Isolation	a	
67.6.0	Organizational Responses	e	
67.7.0	Improved Eddy Current Tests	e	
67.8.0	Denting Criteria	e	
67.9.0	Reactor Coolant System Pressure Control	e	
67.10.0	Supplement Tube Inspections	d	Not applicable to new plants
68.	Postulated Loss of Auxiliary Feedwater System Resulting from Turbine-Driven Auxiliary Feedwater Pumps Steam Supply Line Rupture	e	
69.	Make-up Nozzle Cracking in B&W Plants	c	
70.	PORV and Block Valve Reliability	g	See subsection 1.9.3, item (1)(iv)
71.	Failure of Resin Demineralizer Systems and Their Effects on Nuclear Power Plant Safety	a	
72.	Control Rod Drive Guide Tube Support Pin Failures	a	
73.	Detached Thermal Sleeves	a, j	See subsection 1.9.4.2.3, item 73
74.	Reactor Coolant Activity Limits for Operating Reactors	a	
75.	Generic Implications of ATWS Events at the Salem Nuclear Plant	g, j	See subsection 1.9.4.2.3, item 75
76.	Instrumentation and Control Power Interactions	a	
77.	Flooding of Safety Equipment Compartments by Back-flow Through Floor Drains	e	
78.	Monitoring of Fatigue Transient Limits for Reactor Coolant System	c	
79.	Unanalyzed Reactor Vessel Thermal Stress During Natural Circulation Cooldown	c	See subsection 1.9.4.2.3, item 79
80.	Pipe Break Effects on Control Rod Drive Hydraulic Lines in the Drywells of BWR Mark I and II Containments	a	
81.	Impact of Locked Doors and Barriers on Plant and Personnel Safety	a	

**Table 1.9-2 (Sheet 25 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues**

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
82.	Beyond Design Basis Accidents in Spent Fuel Pools	c, j	See subsection 1.9.4.2.3, item 82
83.	Control Room Habitability	c	See subsection 1.9.4.2.3, item 83
84.	CE PORVs	c	
85.	Reliability of Vacuum Breakers Connected to Steam Discharge Lines Inside BWR Containments	a	
86.	Long Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping	b	
87.	Failure of HPCI Steam Line Without Isolation	g	See subsection 1.9.4, item 87
88.	Earthquakes and Emergency Planning	c	
89.	Stiff Pipe Clamps	h (Medium)	
90.	Technical Specifications for Anticipatory Trips	a	
91.	Main Crankshaft Failures in Transamerica DeLaval Emergency Diesel Generators	c	
92.	Fuel Crumbling During LOCA	a	
93.	Steam Binding of Auxiliary Feedwater Pumps	g	See subsection 1.9.4, item 93
94.	Additional Low Temperature Overpressure Protection for Light Water Reactors	g	See subsection 1.9.4, item 94
95.	Loss of Effective Volume for Containment Recirculation Spray	c	
96.	RHR Suction Valve Testing	e	
97.	PWR Reactor Cavity Uncontrolled Exposures	e	
98.	CRD Accumulator Check Valve Leakage	a	
99.	RCS/RHR Suction Line Valve Interlock on PWRs	f	Resolved per NUREG-0933
100.	OTSG Level	b	
101.	BWR Water Level Redundancy	c	
102.	Human Error in Events Involving Wrong Unit or Wrong Train	c	
103.	Design for Probable Maximum Precipitation	g	See subsection 1.9.4, item 103
104.	Reduction of Boron Dilution Requirements	a	
105.	Interfacing Systems LOCA at BWRs	c	See subsection 1.9.4, item 105
106.	Piping and Use of Highly Combustible Gases in Vital Areas	c	See subsection 1.9.4, item 106
107.	Main Transformer Failures	a	
108.	BWR Suppression Pool Temperature Limits	a	
109.	Reactor Vessel Closure Failure	a	
110.	Equipment Protective Devices on Engineered Safety Features	a	

Table 1.9-2 (Sheet 26 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
111.	Stress Corrosion Cracking of Pressure Boundary Ferritic Steels in Selected Environments	d	Not applicable to new plants
112.	Westinghouse RPS Surveillance Frequencies and Out-of-Service Times	d	Not applicable to new plants
113.	Dynamic Qualification Testing of Large Bore Hydraulic Snubbers	c	See subsection 1.9.4, item 113
114.	Seismic-Induced Relay Chatter	e	
115.	Enhancement of the Reliability of Westinghouse Solid State Protection System	c	
116.	Accident Management	a	
117.	Allowable Time for Diverse Simultaneous Equipment Outages	a	
118.	Tendon Anchorage Failure	f	Resolved per NUREG-0933.
119.1	Piping Rupture Requirements and Decoupling of Seismic and LOCA Loads	d	Not applicable to new plants
119.2	Piping Damping Values	a	
119.3	Decoupling the OBE from the SSE	d	Not applicable to new plants
119.4	BWR Piping Materials	d	Not applicable to new plants
119.5	Leak Detection Requirements	d	Not applicable to new plants
120.	On-Line Testability of Protection Systems	c, j	See subsection 1.9.4.2.3, item 120
121.	Hydrogen Control for Large, Dry PWR Containments	c	See subsection 1.9.4, item 121
122.1.a	Failure of Isolation Valves in Closed Position	e	
122.1.b	Recovery of Auxiliary Feedwater	e	
122.1.c	Interruption of Auxiliary Feedwater Flow	e	
122.2	Initiating Feed-and-Bleed	c	
122.3	Physical Security System Constraints	a	
123.	Deficiencies in the Regulations Governing DBA and Single-Failure Criteria Suggested by the Davis-Besse Event of June 9, 1985	a	
124.	Auxiliary Feedwater System Reliability	g	See subsection 1.9.4, item 124
125.I.1	Availability of the STA	a	
125.I.2.a	Need for a Test Program to Establish Reliability of the PORV	e	
125.I.2.b	Need for PORV Surveillance Tests to Confirm Operational Readiness	e	
125.I.2.c	Need for Additional Protection Against PORV Failure	a	
125.I.2.d	Capability of the PORV to Support Feed-and-Bleed	e	
125.I.3	SPDS Availability	c	

Table 1.9-2 (Sheet 27 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
125.I.4	Plant-Specific Simulator	a	
125.I.5	Safety Systems Tested in All Conditions Required by Design Basis Analysis	a	
125.I.6	Valve Torque Limit and Bypass Switch Settings	a	
125.I.7.a	Recover Failed Equipment	a	
125.I.7.b	Realistic Hands-On Training	a	
125.I.8	Procedures and Staffing for Reporting to NRC Emergency Response Center	a	
125.II.1.a	Two-Train AFW unavailability	a	
125.II.1.b	Review Existing AFW Systems for Single Failure	e	
125.II.1.c	NUREG-0737 Reliability Improvements	a	
125.II.1.d	AFW/Steam and Feedwater Rupture Control System/ICS Interactions in B&W Plants	a	
125.II.2	Adequacy of Existing Maintenance Requirements for Safety-Related Systems	a	
125.II.3	Review Steam/Feedline Break Mitigation Systems for Single Failure	a	
125.II.4	Thermal Stress of OTSG Components	a	
125.II.5	Thermal-Hydraulic Effects of Loss and Restoration of Feedwater on Primary System Components	a	
125.II.6	Reexamine PRA-Based Estimates of the Likelihood of a Severe Core Damage Accident Based on Loss of All Feedwater	a	
125.II.7	Reevaluate Provisions to Automatically Isolate Feedwater from Steam Generator During a Line Break	c	
125.II.8	Reassess Criteria for Feed-and-Bleed Initiation	a	
125.II.9	Enhanced Feed-and-Bleed Capability	a	
125.II.10	Hierarchy of Impromptu Operator Actions	a	
125.II.11	Recovery of Main Feedwater as Alternative to AFW	a	
125.II.12	Adequacy of Training Regarding PORV Operation	a	
125.II.13	Operator Job Aids	a	
125.II.14	Remote Operation of Equipment Which Must Now Be Operated Locally	a	
126.	Reliability of PWR Main Steam Safety Valves	d	
127.	Testing and Maintenance of Manual Valves in Safety-Related Systems	a	
128.	Electrical Power Reliability	h (High)	See subsection 1.9.4, item 128 Resolved per NUREG-0933.
129.	Valve Interlocks to Prevent Vessel Drainage During Shutdown Cooling	a	

Table 1.9-2 (Sheet 28 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
130.	Essential Service Water Pump Failures at Multiplant Sites	f	See subsection 1.9.4, item 130
131.	Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System in Westinghouse Plants	e	
132.	RHR Pumps Inside Containment	a	
133.	Update Policy Statement on Nuclear Plant Staff Working Hours	d	Not applicable to new plants
134.	Rule on Degree and Experience Requirements	c	
135.	Steam Generator and Steam Line Overfill	c	See subsection 1.9.4, item 135
136.	Storage and Use of Large Quantities of Cryogenic Combustibles On Site	d	Not applicable to new plants
137.	Refueling Cavity Seal Failure	a	
138.	Deinerting Upon Discovery of RCS Leakage	a	
139.	Thinning of Carbon Steel Piping in LWRs	d	Not applicable to new plants
140.	Fission Product Removal Systems	a	
141.	LBLOCA With Consequential SGTR	a	
142.	Leakage Through Electrical Isolators in Instrumentation Circuits	c	See subsection 1.9.4, item 142
143.	Availability of Chilled Water Systems	c, j	See subsection 1.9.4.2.3, item 143
144.	Scram Without a Turbine/Generator Trip	a	
145.	Actions to Reduce Common Cause Failures	c	
146.	Support Flexibility of Equipment and Components	d	Not applicable to new plants
147.	Fire-Induced Alternate Shutdown Control Room Panel Interactions	d	Not applicable to new plants
148.	Smoke Control and Manual Fire-Fighting Effectiveness	d	Not applicable to new plants
149.	Adequacy of Fire Barriers	a	
150.	Overpressurization of Containment Penetrations	a	
151.	Reliability of Recirculation Pump Trip During an ATWS	c	
152.	Design Basis for Valves That Might Be Subjected to Significant Blowdown Loads	a	
153.	Loss of Essential Service Water in LWRs	c, j	See subsection 1.9.4.2.3, item 153
154.	Adequacy of Emergency and Essential Lighting	a	

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Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
155.1	More Realistic Source Term Assumptions	g	
155.2	Establish Licensing Requirements For Non-Operating Facilities	d	Not applicable to new plants
155.3	Improve Design Requirements For Nuclear Facilities	a	
155.4	Improve Criticality Calculations	a	
155.5	More Realistic Severe Reactor Accident Scenario	a	
155.6	Improve Decontamination Regulations	a	
155.7	Improve Decommissioning Regulations	a	
156	Systematic Evaluation Program	f	Not applicable to new plants
156.6.1	Pipe Break Effects on Systems and Components	High	The AP1000 is a new plant that takes the effects of a pipe break into account and therefore issue 156.6.1 is not applicable.
157	Containment Performance	c	
158	Performance Of Safety-Related Power-Operated Valves Under Design Basis Conditions	c	
159	Qualification Of Safety-Related Pumps While Running On Minimum Flow	a	
160	Spurious Actuations Of Instrumentation Upon Restoration Of Power	a	
161	Use Of Non-Safety-Related Power Supplies In Safety-Related Circuits	a	
162	Inadequate Technical Specifications For Shared Systems At Multiplant Sites When One Unit Is Shut Down	a	
163	Multiple Steam Generator Tube Leakage	h (Medium) h (High)	See Subsection 1.9.4.2.3, item 163
164	Neutron Fluence In Reactor Vessel	a	
165	Spring-Actuated Safety And Relief Valve Reliability	c	
166	Adequacy Of Fatigue Life Of Metal Components	c	
167	Hydrogen Storage Facility Separation	a	
168	Environmental Qualification Of Electrical Equipment	f	See subsection 1.9.4.2.3, item 168 Not applicable to new plants
169	BWR MSIV Common Mode Failure Due To Loss Of Accumulator Pressure	a	
170	Fuel Damage Criteria For High Burnup Fuel	c	

Table 1.9-2 (Sheet 30 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
171	ESF Failure From LOOP Subsequent To A LOCA	c	
172	Multiple System Responses Program	e	
173.A	Spent Fuel Storage Pool Operating Facilities	c	
173.B	Spent Fuel Storage Pool Permanently Shutdown Facilities	c	
174	Fastener Gaging Practices	c	
175	Nuclear Power Plant Shift Staffing	c	
176	Loss Of Fill-Oil In Rosemount Transmitters	c	
177	Vehicle Intrusion At TMI	g	
178	Effect Of Hurricane Andrew On Turkey Point	d	Not applicable to new plants
179	Core Performance	c	
180	Notice Of Enforcement Discretion	d	Not applicable to new plants
181	Fire Protection	d	Not applicable to new plants
182	General Electric Extended Power Uprate	b	
183	Cycle-Specific Parameter Limits In Technical Specifications	d	Not applicable to new plants
184	Endangered Species	d	Not applicable to new plants
185	Control of Recriticality following Small-Break LOCA in PWRs	h (High)	See subsection 1.9.4.2.3, item 185 Not applicable to new plants
186	Potential Risk and Consequences of Heavy Load Drops	a	
186	Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants	Continue	1.9.4.2.3 9.1.5.3
187	The Potential impact of Postulated Cesium Concentration on Equipment Qualification in the Containment Sump in Nuclear Power Plants.	a	
188	Steam Generator Tube Leaks/Ruptures Concurrent with Containment Bypass	a	
189	Susceptibility of Ice Condenser Containments to Early Failure from Hydrogen Concentration during a Severe Accident	a	
189	Susceptibility of Ice Condenser and Mark III Containments to Early Failure from Hydrogen Combustion During a Severe Accident Description	Continue	Not applicable to the AP1000.
190	Fatigue Evaluation Of Metal Components For 60-Year Plant Life	c	
191	Assessment Of Debris Accumulation On PWR Sump Performance	h (High)	See subsections 6.3.2.2.7 and 1.9.4.2.3, item 191

Table 1.9-2 (Sheet 31 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
199	Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States	Issue to be Prioritized by NRC in the Future	2.5
Human Factors Issues			
HF1.1	Shift Staffing	f	13.1.2.3 18.6
HF1.2	Engineering Expertise on Shift	c	
HF1.3	Guidance on Limits and Conditions of Shift Work	c	
HF2.1	Evaluate Industry Training	d	Not applicable to new plants
HF2.2	Evaluate INPO Accreditation	d	Not applicable to new plants
HF2.3	Revise SRP Section 13.2	d	Not applicable to new plants
HF3.1	Develop Job Knowledge Catalog	d	Not applicable to new plants
HF3.2	Develop License Examination Handbook	d	Not applicable to new plants
HF3.3	Develop Criteria for Nuclear Power Plant Simulators	e	
HF3.4	Examination Requirements	e	
HF3.5	Develop Computerized Exam System	d	Not applicable to new plants
HF4.1	Inspection Procedure for Upgraded Emergency Operating Procedures	c, i	
HF4.2	Procedures Generation Package Effectiveness Evaluation	d	Not applicable to new plants
HF4.3	Criteria for Safety-Related Operator Actions	e	
HF4.4	Guidelines for Upgrading Other Procedures	c, j	See subsection 1.9.4.2.4, item HF4.4
HF4.5	Application of Automation and Artificial Intelligence	e	
HF5.1	Local Control Stations	c	See subsection 1.9.4, item HF5.1
HF5.2	Review Criteria for Human Factors Aspects of Advanced Controls and Instrumentation	g	See subsection 1.9.4, item HF5.2
HF5.3	Evaluation of Operational Aid Systems	e	
HF5.4	Computers and Computer Displays	e	
HF6.1	Develop Regulatory Position on Management and Organization	e	

Table 1.9-2 (Sheet 32 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
HF6.2	Regulatory Position on Management and Organization at Operating Reactors	e	
HF7.1	Human Error Data Acquisition	d	Not applicable to new plants
HF7.2	Human Error Data Storage and Retrieval	d	Not applicable to new plants
HF7.3	Reliability Evaluation Specialist Aids	d	Not applicable to new plants
HF7.4	Safety Event Analysis Results Applications	d	Not applicable to new plants
HF8	Maintenance and Surveillance Program	c	
Chernobyl Issues			
CH1.1A	Symptom-Based EOPs	d	Not applicable to new plants
CH1.1B	Procedure Violations	d	Not applicable to new plants
CH1.2A	Test, Change, and Experiment Review Guidelines	d	Not applicable to new plants
CH1.2B	NRC Testing Requirements	d	Not applicable to new plants
CH1.3A	Revise Regulatory Guide 1.47	d	Not applicable to new plants
CH1.4A	Engineered Safety Feature Availability	d	Not applicable to new plants
CH1.4B	Technical Specification Bases	d	Not applicable to new plants
CH1.4C	Low Power and Shutdown	d	Not applicable to new plants
CH1.5	Operating Staff Attitudes Toward Safety	d	Not applicable to new plants
CH1.6A	Assessment of NRC Requirements on Management	d	Not applicable to new plants
CH1.7A	Accident Management	d	Not applicable to new plants
CH2.1A	Reactivity Transients	d	Not applicable to new plants
CH2.2	Accidents at Low Power and at Zero Power	e	
CH2.3A	Control Room Habitability	e	
CH2.3B	Contamination Outside Control Room	d	Not applicable to new plants
CH2.3C	Smoke Control	d	Not applicable to new plants
CH2.3D	Shared Shutdown Systems	d	Not applicable to new plants

Table 1.9-2 (Sheet 33 of 33)
Listing of Unresolved Safety Issues and Generic Safety Issues

Action Plan Item/Issue No.	Title	Applicable Screening Criteria	Notes
CH2.4A	Firefighting With Radiation Present	d	Not applicable to new plants
CH3.1A	Containment Performance	d	Not applicable to new plants
CH3.2A	Filtered Venting	d	Not applicable to new plants
CH4.1	Size of the Emergency Planning Zones	a	
CH4.2	Medical Services	a	
CH4.3A	Ingestion Pathway Protective Measures	d	Not applicable to new plants
CH4.4A	Decontamination	d	Not applicable to new plants
CH4.4B	Relocation	d	Not applicable to new plants
CH5.1A	Mechanical Dispersal in Fission Product Release	d	Not applicable to new plants
CH5.1B	Stripping in Fission Product Release	d	Not applicable to new plants
CH5.2A	Steam Explosions	d	Not applicable to new plants
CH5.3	Combustible Gas	a	
CH6.1A	The Fort St. Vrain Reactor and the Modular HTGR	a	
CH6.1B	Structural Graphite Experiments	d	Not applicable to new plants
CH6.2	Assessment	d	Not applicable to new plants

Notes:

a. Issue has been prioritized as **Low, Drop** or has not been prioritized.
b. Issue is not an AP1000 design issue. Issue is applicable to GE, B&W, or CE designs only.
c. Issue resolved with no new requirements.
d. Issue is not a design issue (Environmental, Licensing, or Regulatory Impact Issue; or covered in an existing NRC program).
e. Issue superseded by one or more issues.
f. Issue is not an AP1000 design certification issue. Issue is applicable to current operating plants or is programmatic in nature.
g. Issue is resolved by establishment of new regulatory requirements and/or guidance.
h. Issue is unresolved pending generic resolution (for example, prioritized as **High, Medium**, or possible resolution identified).
i. The AP600 DSER (Draft NUREG-01512) identified this item as not being required to be addressed by 10 CFR 52.47.
j. The AP600 DSER (Draft NUREG-01512) identified this item as required to be discussed.

Table 1.9-201
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Table 1.9-202 (Sheet 1 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
1	Introduction and Interfaces, Initial Issuance, 03/2007	N/A	No specific acceptance criteria associated with these general requirements.
2.0	Site Characteristics and Site Parameters, Initial Issuance, 03/2007	N/A	No specific acceptance criteria are identified.
2.1.1	Site Location and Description	Acceptable	
2.1.2	Exclusion Area Authority and Control	Acceptable	
2.1.3	Population Distribution	Acceptable	
2.2.1 – 2.2.2	Identification of Potential Hazards in Site Vicinity	Acceptable	
2.2.3	Evaluation of Potential Accidents	Acceptable	
2.3.1	Regional Climatology	Acceptable	
2.3.2	Local Meteorology	Acceptable	
2.3.3	Onsite Meteorological Measurements Programs	Acceptable	
2.3.4	Short-Term Atmospheric Dispersion Estimates for Accident Releases	Acceptable	
2.3.5	Long-Term Atmospheric Dispersion Estimates for Routine Releases	Acceptable	
2.4.1	Hydrologic Description	Acceptable	
2.4.2	Floods, Rev. 4, 03/2007	Acceptable	
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers, Rev. 4, 03/2007	Acceptable	
2.4.4	Potential Dam Failures	Acceptable	
2.4.5	Probable Maximum Surge and Seiche Flooding	Acceptable	
2.4.6	Probable Maximum Tsunami Hazards	Acceptable	
2.4.7	Ice Effects	Acceptable	
2.4.8	Cooling Water Canals and Reservoirs	Acceptable	
2.4.9	Channel Diversions	Acceptable	
2.4.10	Flooding Protection Requirements	Acceptable	
2.4.11	Low Water Considerations	Acceptable	
2.4.12	Groundwater	Acceptable	
2.4.13	Accidental Releases of Radioactive Liquid Effluents in Ground and Surface Waters	Acceptable	

Table 1.9-202 (Sheet 2 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
2.4.14	Technical Specifications and Emergency Operation Requirements	Acceptable	
2.5.1	Basic Geologic and Seismic Information, Rev.4, 03/2007	Acceptable	
2.5.2	Vibratory Ground Motion, Rev. 4, 03/2007	Exception	Exception is taken to the guidance in R.G. 1.206 (C.I.2.5.2.4). Hazard curves were run at 15th and 85th fractiles instead of 16th and 84th fractiles as they are very close approximations (+/- 1 sigma).
2.5.3	Surface Faulting, Rev. 4, 03/2007	Acceptable	
2.5.4	Stability of Subsurface Materials and Foundations	Acceptable	
2.5.5	Stability of Slopes	Acceptable	
3.2.1	Seismic Classification, Rev. 2, 03/2007		See Notes d and e.
3.2.2	System Quality Group Classification, Rev. 2, 03/2007		See Notes d and e.
3.3.1	Wind Loadings	Acceptable	See Notes d, e, and f.
3.3.2	Tornado Loadings	Acceptable	See Notes d, e, and f.
3.4.1	Internal Flood Protection for Onsite Equipment Failures	Acceptable	See Notes d, e, and f.
3.4.2	Analysis Procedures		See Notes d and e.
3.5.1.1	Internally Generated Missiles (Outside Containment)		See Notes d and e.
3.5.1.2	Internally Generated Missiles (Inside Containment)		See Notes d and e.
3.5.1.3	Turbine Missiles	Acceptable	See Notes d, e, and f.
3.5.1.4	Missiles Generated by Tornadoes and Extreme Winds		See Notes d and e.
3.5.1.5	Site Proximity Missiles (Except Aircraft), Rev.4, 03/2007	Acceptable	See Notes d, e, and f.
3.5.1.6	Aircraft Hazards	Acceptable	See Notes d, e, and f. Aircraft hazard event probability is consistent with SRP 2.2.3, Rev. 3, Technical Rationale 2.
3.5.2	Structures, Systems, and Components to be Protected from Externally-Generated Missiles		See Notes d and e.
3.5.3	Barrier Design Procedures		See Notes d and e.
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment		See Notes d and e.
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
3.6.3	Leak-Before-Break Evaluation Procedures, Rev. 1, 03/2007	Acceptable	See Notes d, e, and f.
3.7.1	Seismic Design Parameters		See Notes d and e.

Table 1.9-202 (Sheet 3 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
3.7.2	Seismic System Analysis	Acceptable	See Notes d, e, and f.
3.7.3	Seismic Subsystem Analysis		See Notes d and e.
3.7.4	Seismic Instrumentation, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
3.8.1	Concrete Containment, Rev. 2, 03/2007		See Notes d and e.
3.8.2	Steel Containment, Rev. 2, 03/2007		See Notes d and e.
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments, Rev. 2, 03/2007		See Notes d and e.
3.8.4	Other Seismic Category I Structures, Rev. 2, 03/2007		See Notes d and e.
3.8.5	Foundations, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
3.9.1	Special Topics for Mechanical Components		See Notes d and e.
3.9.2	Dynamic Testing and Analysis of Systems, Structures, and Components		See Notes d and e.
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
3.9.4	Control Rod Drive Systems		See Notes d and e.
3.9.5	Reactor Pressure Vessel Internals		See Notes d and e.
3.9.6	Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints	Acceptable	See Notes d, e, and f.
3.9.7	Risk-Informed Inservice Testing, Rev. 0, 08/1998	N/A	
3.9.8	Risk-Informed Inservice Inspection of Piping, Rev.0, 09/2003	N/A	
3.10	Seismic and Dynamic Qualification of Mechanical and Electrical Equipment		See Notes d and e.
3.11	Environmental Qualification of Mechanical and Electrical Equipment	Acceptable	See Notes d, e, and f.
3.12	ASME Code Class 1, 2, and 3 Piping Systems, Piping Components and their Associated Supports, Initial Issuance, 03/2007		See Note g.
3.13	Threaded Fasteners - ASME Code Class 1, 2, and 3, Initial Issuance, 03/2007		See Note g.
4.2	Fuel System Design		See Notes d and e.
4.3	Nuclear Design		See Notes d and e.
4.4	Thermal and Hydraulic Design, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
4.5.1	Control Rod Drive Structural Materials		See Notes d and e.
4.5.2	Reactor Internal and Core Support Structure Materials		See Notes d and e.

Table 1.9-202 (Sheet 4 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
4.6	Functional Design of Control Rod Drive System, Rev. 2, 03/2007		See Notes d and e.
5.2.1.1	Compliance with the Codes and Standards Rule, 10 CFR 50.55a	Acceptable	See Notes d, e, and f.
5.2.1.2	Applicable Code Cases		See Notes d and e.
5.2.2	Overpressure Protection		See Notes d and e.
5.2.3	Reactor Coolant Pressure Boundary Materials	Acceptable	See Notes d, e, and f.
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection, Rev. 2, 03/2007		See Notes d and e.
5.3.1	Reactor Vessel Materials, Rev. 2, 03/2007		See Notes d and e.
5.3.2	Pressure-Temperature Limits, Upper-Shelf Energy, and Pressurized Thermal Shock, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
5.3.3	Reactor Vessel Integrity, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
5.4	Reactor Coolant System Component and Subsystem Design, Rev. 2, 03/2007	N/A	No specific acceptance criteria associated with these general requirements.
5.4.1.1	Pump Flywheel Integrity (PWR), Rev. 2, 03/2007		See Notes d and e.
5.4.2.1	Steam Generator Materials		See Notes d and e.
5.4.2.2	Steam Generator Program, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
5.4.6	Reactor Core Isolation Cooling System (BWR), Rev. 4, 03/2007	N/A	
5.4.7	Residual Heat Removal (RHR) System, Rev. 4, 03/2007		See Notes d and e.
5.4.8	Reactor Water Cleanup System (BWR)	N/A	
5.4.11	Pressurizer Relief Tank		See Notes d and e.
5.4.12	Reactor Coolant System High Point Vents, Rev. 1, 03/2007		See Notes d and e.
5.4.13	Isolation Condenser System (BWR), Initial Issuance, 03/2007	N/A	
6.1.1	Engineered Safety Features Materials, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
6.1.2	Protective Coating Systems (Paints) – Organic Materials	Acceptable	See Notes d, e, and f.
6.2.1	Containment Functional Design		See Notes d and e.
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments		See Notes d and e.
6.2.1.1.B	Ice Condenser Containments, Rev. 2, 07/1981	N/A	
6.2.1.1.C	Pressure-Suppression Type BWR Containments, Rev. 7, 03/2007	N/A	

Table 1.9-202 (Sheet 5 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
6.2.1.2	Subcompartment Analysis		See Notes d and e.
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)		See Notes d and e.
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures, Rev. 2, 03/2007		See Notes d and e.
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies		See Notes d and e.
6.2.2	Containment Heat Removal Systems, Rev. 5, 03/2007		See Notes d and e.
6.2.3	Secondary Containment Functional Design		See Notes d and e.
6.2.4	Containment Isolation System		See Notes d and e.
6.2.5	Combustible Gas Control in Containment	Acceptable	See Notes d, e, and f.
6.2.6	Containment Leakage Testing	Acceptable	See Notes d, e, and f.
6.2.7	Fracture Prevention of Containment Pressure Boundary, Rev. 1, 03/2007		See Notes d and e.
6.3	Emergency Core Cooling System	Acceptable	See Notes d, e, and f.
6.4	Control Room Habitability System	Acceptable	See Notes d, e, and f.
6.5.1	ESF Atmosphere Cleanup Systems		See Notes d and e.
6.5.2	Containment Spray as a Fission Product Cleanup System, Rev. 4, 03/2007		See Notes d and e.
6.5.3	Fission Product Control Systems and Structures		See Notes d and e.
6.5.4	Ice Condenser as a Fission Product Cleanup System, Rev. 3, 12/1988	N/A	
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System, Rev. 1, 03/2007	N/A	
6.6	Inservice Inspection and Testing of Class 2 and 3 Components, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
6.7	Main Steam Isolation Valve Leakage Control System (BWR), Rev. 2, 07/1981	N/A	
7	Instrumentation and Controls –Overview of Review Process, Rev. 5, 03/2007		See Notes d and e.
Appendix 7.0-A	Review Process for Digital Instrumentation and Control Systems, Rev. 5, 03/2007		See Notes d and e.
7.1	Instrumentation and Controls –Introduction, Rev. 5, 03/2007		See Notes d and e.

Table 1.9-202 (Sheet 6 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
7.1-T Table 7-1	Regulatory Requirements, Acceptance Criteria, and Guidelines for Instrumentation and Control Systems Important to Safety, Rev. 5, 03/2007		See Notes d and e.
Appendix 7.1-A	Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety, Rev. 5, 03/2007		See Notes d and e.
Appendix 7.1-B	Guidance for Evaluation of Conformance to IEEE Std 279, Rev. 5, 03/2007		See Notes d and e.
Appendix 7.1-C	Guidance for Evaluation of Conformance to IEEE Std 603, Rev. 5, 03/2007		See Notes d and e.
Appendix 7.1-D	Guidance for Evaluation of the Application of IEEE Std 7-4.3.2 Initial Issuance 03/2007		See Notes d and e.
7.2	Reactor Trip System, Rev. 5, 03/2007		See Notes d and e.
7.3	Engineered Safety Features Systems, Rev. 5, 03/2007		See Notes d and e.
7.4	Safe Shutdown Systems, Rev. 5, 03/2007		See Notes d and e.
7.5	Information Systems Important to Safety, Rev. 5, 03/2007		See Notes d and e.
7.6	Interlock Systems Important to Safety, Rev. 5, 03/2007		See Notes d and e.
7.7	Control Systems, Rev. 5, 03/2007		See Notes d and e.
7.8	Diverse Instrumentation and Control Systems, Rev. 5, 03/2007		See Notes d and e.
7.9	Data Communication Systems, Rev. 5, 03/2007		See Notes d and e.
8.1	Electric Power – Introduction	N/A	No specific acceptance criteria associated with these general requirements.
8.2	Offsite Power System, Rev. 4, 03/2007	Acceptable	See Notes d, e, and f
8.3.1	A-C Power Systems (Onsite)	Acceptable	See Notes d, e, and f.
8.3.2	D-C Power Systems (Onsite)	Acceptable	See Notes d, e, and f.
8.4	Station Blackout, Initial Issuance, 03/2007		See Note g.
9.1.1	Criticality Safety of Fresh and Spent Fuel Storage and Handling		See Notes d and e.
9.1.2	New and Spent Fuel Storage, Rev. 4, 03/2007		See Notes d and e.
9.1.3	Spent Fuel Pool Cooling and Cleanup System, Rev. 2, 03/2007		See Notes d and e.
9.1.4	Light Load Handling System (Related to Refueling)	Acceptable	See Notes d, e, and f.
9.1.5	Overhead Heavy Load Handling Systems, Rev. 1, 03/2007	Acceptable	See Notes d, e, and f.
9.2.1	Station Service Water System, Rev. 5, 03/2007	Acceptable	See Notes d, e, and f.
9.2.2	Reactor Auxiliary Cooling Water Systems, Rev. 4, 03/2007		See Notes d and e.
9.2.4	Potable and Sanitary Water Systems		See Notes d and e.

Table 1.9-202 (Sheet 7 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
9.2.5	Ultimate Heat Sink	Acceptable	See Notes d, e, and f.
9.2.6	Condensate Storage Facilities	Acceptable	See Notes d, e, and f.
9.3.1	Compressed Air System, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
9.3.2	Process and Post-accident Sampling Systems		See Notes d and e.
9.3.3	Equipment and Floor Drainage System		See Notes d and e.
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)		See Notes d and e.
9.3.5	Standby Liquid Control System (BWR)	N/A	
9.4.1	Control Room Area Ventilation System	Acceptable	See Notes d, e, and f.
9.4.2	Spent Fuel Pool Area Ventilation System		See Notes d and e.
9.4.3	Auxiliary and Radwaste Area Ventilation System		See Notes d and e.
9.4.4	Turbine Area Ventilation System		See Notes d and e.
9.4.5	Engineered Safety Feature Ventilation System		See Notes d and e.
9.5.1	Fire Protection Program, Rev. 5, 03/2007	Acceptable	See Notes d, e, and f.
9.5.2	Communications Systems	Acceptable	See Notes d, e, and f.
9.5.3	Lighting Systems		See Notes d and e.
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	Acceptable	See Notes d, e, and f.
9.5.5	Emergency Diesel Engine Cooling Water System		See Notes d and e.
9.5.6	Emergency Diesel Engine Starting System		See Notes d and e.
9.5.7	Emergency Diesel Engine Lubrication System		See Notes d and e.
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System		See Notes d and e.
10.2	Turbine Generator	Acceptable	See Notes d, e, and f.
10.2.3	Turbine Rotor Integrity, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
10.3	Main Steam Supply System, Rev. 4, 03/2007	Acceptable	See Notes d, e, and f.
10.3.6	Steam and Feedwater System Materials	Acceptable	See Notes d, e, and f.
10.4.1	Main Condensers		See Notes d and e.
10.4.2	Main Condenser Evacuation System	Acceptable	See Notes d, e, and f.
10.4.3	Turbine Gland Sealing System		See Notes d and e.
10.4.4	Turbine Bypass System		See Notes d and e.
10.4.5	Circulating Water System	Acceptable	See Notes d, e, and f.
10.4.6	Condensate Cleanup System		See Notes d and e.
10.4.7	Condensate and Feedwater System, Rev. 4, 03/2007	Acceptable	See Notes d, e, and f.
10.4.8	Steam Generator Blowdown System (PWR)		See Notes d and e.

Table 1.9-202 (Sheet 8 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
10.4.9	Auxiliary Feedwater System (PWR)		See Notes d and e.
11.1	Source Terms		See Notes d and e.
11.2	Liquid Waste Management System	Acceptable	See Notes d, e, and f.
11.3	Gaseous Waste Management System	Acceptable	See Notes d, e, and f.
11.4	Solid Waste Management System	Acceptable	See Notes d, e, and f.
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems, Rev. 4, 03/2007	Acceptable	See Notes d, e, and f.
12.1	Assuring that Occupational Radiation Exposures Are As Low As Is Reasonably Achievable	Exception	See Notes d, e, and f. An exception is taken to following the guidance of RG 1.206 to address RG 8.20, 8.25, and RG 8.26. NUREG-1736, Final Report (published 2001) lists RG 8.20 and RG 8.26 as "outdated" and recommends the methods of RG 8.9 R1. RG 8.25 states it is not applicable to nuclear facilities licensed under 10 CFR Part 50, and, by extension, to 10 CFR Part 52. An exception is taken to RG 8.8 C.3.b. RG 1.16 C.1.b (3) data is no longer reported. Reporting per C.1.b (2) is also no longer required.
12.2	Radiation Sources	Exception	See Notes d, e, and f. A general description of miscellaneous sealed sources related to radiography is provided in FSAR text. Other requested details are maintained on-site for NRC review and audit upon their procurement.
12.3 - 12.4	Radiation Protection Design Features	Acceptable	See Notes d, e, and f.
12.5	Operational Radiation Protection Program	Acceptable	See Notes d, e, and f.

Table 1.9-202 (Sheet 9 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
13.1.1	Management and Technical Support Organization, Rev. 5, 03/2007	Exception	See Notes d, e, and f. Design and construction responsibilities are not defined in numbers. The experience requirements of corporate staff are set by corporate policy and not provided here in detail, however the experience level of the corporate staff, as discussed Subsections 13.1.1, 13.1.1.1, and Appendix 13AA, in the area of nuclear plant development, construction, and management establishes that the applicant has the necessary capability and staff to ensure that design and construction of the facility will be performed in an acceptable manner. Resumes and/or other documentation of qualification and experience of initial appointees to appropriate management and supervisory positions are available for NRC after position vacancies are filled.
13.1.2 - 13.1.3	Operating Organization, Rev. 6, 03/2007	Exception	See Notes d, e, and f. The SRP requires resumes of personnel holding plant managerial and supervisory positions to be included in the FSAR. Current industry practice is to have the resumes available for review by the regulator when requested but not be kept in the FSAR. Additionally, at time of COLA, most positions are unfilled.

Table 1.9-202 (Sheet 10 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
13.2.1	Reactor Operator Requalification Program; Reactor Operator Training	Exception	See Notes d, e, and f. SRP requires meeting the guidance of NUREG-0711. NEI 06-13A, Template for an Industry Training Program Description, which is incorporated by reference in Section 13.2, does not address meeting the guidance of NUREG-0711. NEI 06-13A, is approved by NRC to meet the regulatory requirements for the FSAR description of the Training Program. SRP requires meeting the guidance of Regulatory Guide 1.149, "Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations" RG 1.149 is not addressed in NEI 06-13A. Level of detail is consistent with NEI 06-13A.
13.2.2	Non-Licensed Plant Staff Training	Exception	See Notes d, e, and f. Level of detail is consistent with NEI 06-13A.
13.3	Emergency Planning	Acceptable	See Notes d, e, and f.
13.4	Operational Programs	Acceptable	See Notes d, e, and f.
13.5.1.1	Administrative Procedures – General, Initial Issuance, 03/2007	Exception	The procedure development schedule is addressed in the COL application (not in the SAR as requested by this SRP).
13.5.2.1	Operating and Emergency Operating Procedures, Rev. 2, 03/2007	Exception	See Notes d, e, and f. Procedures are generally identified in this section by topic, type, or classification in lieu of the specific title and represent general areas of procedural coverage.
13.6	Physical Security	Acceptable	See Security Plan developed in accordance with NEI 03-12.
13.6.1	Physical Security - Combined License Review Responsibilities, Initial Issuance, 03/2007	Acceptable	See Security Plan developed in accordance with NEI 03-12
13.6.2	Physical Security - Design Certification, Initial Issuance, 03/2007		See notes d and e.
13.6.3	Physical Security - Early Site Permit, Initial Issuance, 03/2007	N/A	

Table 1.9-202 (Sheet 11 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
14.2	Initial Plant Test Program - Design Certification and New License Applicants	Exception	See Notes d, e, and f. The level of detail is consistent with DCD section content addressing nonsafety-related systems.
14.2.1	Generic Guidelines for Extended Power Uprate Testing Programs, Initial Issuance, 08/2006	N/A	No power uprate is sought.
14.3	Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007	Acceptable	
14.3.1	[Reserved]		
14.3.2	Structural and Systems Engineering - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.3	Piping Systems and Components - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.4	Reactor Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.5	Instrumentation and Controls - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.6	Electrical Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.7	Plant Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007	Acceptable	See Notes d, e, and f.
14.3.8	Radiation Protection - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.9	Human Factors Engineering - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.10	Emergency Planning - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007	Acceptable	See Notes d, e, and f.
14.3.11	Containment Systems - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007		See Notes d and e.
14.3.12	Physical Security Hardware - Inspections, Tests, Analyses, and Acceptance Criteria, Initial Issuance, 03/2007	Acceptable	See Notes d, e, and f.
15	Introduction - Transient and Accident Analysis		See Notes d and e.
15.0.1	Radiological Consequence Analyses Using Alternative Source Terms, Rev. 0, 07/2000		See Notes d and e.

Table 1.9-202 (Sheet 12 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
15.0.2	Review of Transient and Accident Analysis Method, Rev. 0, 12/2005		See Notes d and e.
15.0.3	Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors, Initial Issuance, 03/2007		See Notes d and e.
15.1.1 - 15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve, Rev. 2, 03/2007		See Notes d and e.
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)		See Notes d and e.
15.2.1 - 15.2.5	Loss of External Load; Turbine Trip; Loss of Condenser Vacuum; Closure of Main Steam Isolation Valve (BWR); and Steam Pressure Regulator Failure (Closed), Rev. 2, 03/2007		See Notes d and e.
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries, Rev. 2, 03/2007		See Notes d and e.
15.2.7	Loss of Normal Feedwater Flow		See Notes d and e.
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR), Rev. 2, 03/2007		See Notes d and e.
15.3.1 - 15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions, Rev. 2, 03/2007		See Notes d and e.
15.3.3 - 15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break		See Notes d and e.
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition		See Notes d and e.
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power		See Notes d and e.
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)		See Notes d and e.
15.4.4 - 15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate, Rev. 2, 03/2007		See Notes d and e.
15.4.6	Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR), Rev. 2, 03/2007		See Notes d and e.

Table 1.9-202 (Sheet 13 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position, Rev. 2, 03/2007		See Notes d and e.
15.4.8	Spectrum of Rod Ejection Accidents (PWR)		See Notes d and e.
15.4.8.A	Radiological Consequences of a Control Rod Ejection Accident (PWR), Rev. 1, 07/1981		See Notes d and e.
15.4.9	Spectrum of Rod Drop Accidents (BWR)	N/A	
15.5.1 - 15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory, Rev. 2, 03/2007		See Notes d and e.
15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve, Rev. 2, 03/2007		See Notes d and e.
15.6.5	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary		See Notes d and e.
15.8	Anticipated Transients Without Scram, Rev. 2, 03/2007		See Notes d and e.
15.9	Boiling Water Reactor Stability, Initial Issuance, 03/2007	N/A	
16	Technical Specifications	Acceptable	See Notes d, e, and f.
16.1	Risk-informed Decision Making: Technical Specifications, Rev. 1, 03/2007	N/A	This SRP applies to the Technical Specifications change process.
17.1	Quality Assurance During the Design and Construction Phases, Rev. 2, 07/1981	Acceptable	See Notes d, e, and f. This section covers the requirements of SRP Section 17.1 through reference to quality assurance plan which is maintained separately as described in Sections 17.1 and 17.5.
17.2	Quality Assurance During the Operations Phase, Rev. 2, 07/1981		See Notes d and e.
17.3	Quality Assurance Program Description, Rev. 0, 08/1990		See Notes d and e.
17.4	Reliability Assurance Program (RAP), Initial Issuance, 03/2007		See Notes d and e.
17.5	Quality Assurance Program Description - Design Certification, Early Site Permit and New License Applicants, Initial Issuance, 03/2007	Acceptable	See Notes d, e, and f. This section covers the requirements of SRP Section 17.5 through reference to Quality Assurance Program Description which is maintained separately and developed in accordance with NEI 06-14A.
17.6	Maintenance Rule, Initial Issuance, 03/2007	Acceptable	Content developed in accordance with NEI 07-02A

Table 1.9-202 (Sheet 14 of 14)[a]
Conformance with SRP Acceptance Criteria

Criteria Section [b]	Reference Criteria	FSAR Position [c]	Comments/Summary of Exceptions
18.0	Human Factors Engineering, Rev. 2, 03/2007	Acceptable	See Notes d, e, and f.
19.0	Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors, Rev. 2, 06/2007	Acceptable	See Notes d, e, and f.
19.1	Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Rev. 2, 06/2007	Acceptable	See Notes d, e, and f.
19.2	Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance, Initial Issuance, 06/2007		See Note g.

a) This table is provided as a one-time aid to facilitate NRC review. This table becomes historical information and need not be updated
b) If no revision or date is specified, it is Rev. 3, 03/2007.
c) Consult the AP1000 Design Control Document (DCD) Appendix 1A to determine extent of conformance with Regulatory Guides (except Regulatory Guide 1.206).
d) Conformance with a previous revision of this SRP is documented in AP1000 Design Control Document (Section 1.9.2 and WCAP-15799).
e) Conformance with the design aspects of this SRP is as stated in the AP1000 DCD.
f) Conformance with the plant or site-specific aspects of this SRP is as stated under “FSAR Position.”
g) This SRP is not applicable to the AP1000 certified design.

Table 1.9-203
Deleted

Table 1.9-204 (Sheet 1 of 6)
Generic Communications Assessment

Number	Title	Comment
BULLETIN		
80-06	Engineered Safety Feature (ESF) Reset Controls (3/80)	See Note a.
80-10	Contamination of Nonradioactive System and Resulting Potential for Unmonitored, Uncontrolled Release of Radioactivity to Environment (5/80)	Appendix 12AA
80-15	Possible Loss of Emergency Notification System (ENS) with Loss of Offsite Power (6/80)	9.5.2.5.1 Emergency Plan
88-11	Pressurizer Surge Line Thermal Stratification	3.9.3.1.2
02-01	Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity	5.2.4 See Note a.
02-02	Reactor Pressure Vessel Head and Vessel Head Penetration Nozzle Inspection Programs	5.2.4 See Note a.
03-01	Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized-Water Reactors	6.3 See Note a.
03-02	Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity	5.2.4.3 See Note a.
03-03	Potentially Defective 1-inch Valves for Uranium Hexafluoride Cylinders	N/A
03-04	Rebaselining of Data in the Nuclear Materials Management and Safeguards System	N/A One time report.

Table 1.9-204 (Sheet 2 of 6)
Generic Communications Assessment

Number	Title	Comment
04-01	Inspection of Alloy 82/182/600 Materials Used in the Fabrication of Pressurizer Penetrations and Steam Space Piping Connections at Pressurized-Water Reactors	See Note a.
05-01	Material Control and Accounting at Reactors and Wet Spent Fuel Storage Facilities	13.5.2.2.9
05-02	Emergency Preparedness and Response Actions for Security-Based Events	13.3
GENERIC LETTERS		
80-22	Transmittal of NUREG-0654 “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans” (3/80)	13.3
80-26	Qualifications of Reactor Operators (3/80)	13.2 18.10
80-51	On-Site Storage Of Low-Level Waste (6/90)	11.4.6
80-55	Possible Loss of Hotline With Loss Of Off-Site Power	See Bulletin 80-15
80-77	Refueling Water Level (8/80)	16.1 See Note a.
80-094	Emergency Plan (11/80)	13.3
80-099	Technical Specification Revisions for Snubber Surveillance (11/80)	Snubbers no longer in generic Tech Specs See Note a.
80-108	Emergency Planning (12/80)	13.3
81-02	Analysis, Conclusions and Recommendations Concerning Operator Licensing (1/81)	13.2

Table 1.9-204 (Sheet 3 of 6)
Generic Communications Assessment

Number	Title	Comment
81-10	Post-TMI Requirements for the Emergency Operations Facility (2/81)	13.3
81-38	Storage of Low-Level Radioactive Waste at Power Reactor Sites (11/81)	11.4.6
81-40	Qualifications of Reactor Operators (12/81)	13.1 13.2
82-02	Commission Policy on Overtime (2/82)	16.1
82-04	Use of INPO See-in Program (3/82)	13.1 13.5
82-12	Nuclear Power Plant Staff Working Hours (6/82)	13.1.2.3 13.5.1
82-13	Reactor Operator and Senior Reactor Operator Examinations (6/82)	For information only.
82-18	Reactor Operator and Senior Reactor Operator Requalification Examinations (10/82)	13.2
83-06	Certificates and Revised Format For Reactor Operator and Senior Reactor Operator Licenses (1/83)	13.2
83-11	Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions (2/83)	13.1 See Note a.
83-12	Issuance of NRC FORM 398 - Personal Qualifications Statement - Licensee (2/83)	13.2
83-17	Integrity of the Requalification Examinations for Renewal of Reactor Operator and Senior Reactor Operator Licenses (4/83)	13.1
83-22	Safety Evaluation of "Emergency Response Guidelines" (6/83)	18.9

Table 1.9-204 (Sheet 4 of 6)
Generic Communications Assessment

Number	Title	Comment
83-40	Operator Licensing Examination (12/83)	13.2
84-10	Administration of Operating Tests Prior to Initial Criticality (10 CFR 55.25) (4/84)	13.2
84-14	Replacement and Requalification Training Program (5/84)	13.2
84-17	Annual Meeting to Discuss Recent Developments Regarding Operator Training, Qualifications, and Examinations (7/84)	Administrative
84-20	Scheduling Guidance for Licensee Submittals of Reloads That Involve Unreviewed Safety Questions (8/84)	13.5
85-04	Operating Licensing Examinations (1/85)	Administrative
85-05	Inadvertent Boron Dilution Events (1/85)	13.5
85-14	Commercial Storage At Power Reactor Sites Of Low Level Radioactive Waste Not Generated By The Utility (8/85)	Administrative
85-18	Operator Licensing Examinations (9/85)	Administrative
85-19	Reporting Requirements On Primary Coolant Iodine Spikes (9/85)	16.1
86-14	Operator Licensing Examinations (8/86)	Administrative
87-14	Operator Licensing Examinations (8/87)	Administrative
88-05	Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants (3/88)	5.2.4 See Note a.
88-14	Instrument Air Supply System Problems Affecting Safety-Related Equipment (8/88)	9.3.7
88-18	Plant Record Storage on Optical Disk (10/88)	17

Table 1.9-204 (Sheet 5 of 6)
Generic Communications Assessment

Number	Title	Comment
89-07	Power Reactors Safeguards Contingency Planning for Surface Vehicle Bombs (4/89)	13.6
89-07 S1	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	13.6
89-08	Erosion/Corrosion-Induced Pipe Wall Thinning	10.1.3.1
89-12	Operator Licensing Examination (7/89)	13.2
89-15	Emergency Response Data System (8/89)	9.5.2 13.3
89-17	Planned Administrative Changes to the NRC Operator Licensing Written Examination Process (9/89)	N/A
91-14	Emergency Telecommunications (9/91)	9.5.2 13.3
91-16	Licensed Operators and Other Nuclear Facility Personnel Fitness for Duty (10/91)	13.7
92-01	Reactor Vessel Structural Integrity (1/92)	5.3.2.6.3
93-01	Emergency Response Data System Test Program	13.3
93-03	Verification of Plant Records	17
96-02	Reconsideration of Nuclear Power Plant Security Requirements Associated with an Internal Threat (2/96)	13.6
03-01	Control Room Habitability	6.4 See Note a.
04-01	Requirements for Steam Generator Tube Inspections	5.4.2.5 16.1 See Note a.

Table 1.9-204 (Sheet 6 of 6)
Generic Communications Assessment

Number	Title	Comment
04-02	Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors	6.3.8.1 See Note a.
06-01	Steam Generator Tube Integrity and Associated Technical Specifications	5.4.2.5 16.1 See Note a.
06-02	Grid Reliability and the Impact on Plant Risk and the Operability of Offsite Power	8.2.1.1 8.2.2 See Note a.
06-03	Potentially Nonconforming Hemyc and MT Fire Barrier Configurations	9.5.1.8 See Note a.
07-01	Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients.	17.6 See Note a.

(a) The design aspects of this topic are as stated in the AP1000 DCD.

1.10 Nuclear Power Plants to be Operated on Multi-Unit Sites

The certification for the AP1000 is for a single unit. Dual siting of AP1000 is achievable, provided that the centerlines of the units are sufficiently separated. The primary consideration in setting this separation distance is the space needed to support plant construction via the use of a heavy-lift crane.

Security controls during construction and operation are addressed in the Physical Security Plan.

Management and administrative controls are established to identify potential hazards to structures, systems, and components (SSCs) of an operating unit as a result of construction activities at a unit under construction. Controls within this section are not required unless there is an operating unit on the site, i.e., a unit with fuel loaded into the reactor vessel. Advance notification, scheduling and planning allow site management to implement interim controls to reduce the potential for impact to SSCs.

This section presents an assessment of the potential impacts of construction of one unit on SSCs important to safety for an operating unit, in accordance with 10 CFR 52.79(a)(31). This assessment includes:

- Identification of potential construction activity hazards
- Identification of SSCs important to safety and limiting conditions for operation (LCOs) for the operating unit
- Identification of potentially impacted SSCs and LCOs
- Identification of applicable managerial and administrative controls

1.10.1 Potential Construction Activity Hazards

The power blocks for Units 2 and 3 meet the minimum required separation of 800 feet between plant centerlines.

Construction activities may include site exploration, grading, clearing, and installation of drainage and erosion-control measures; boring, drilling, dredging, pile driving and excavating; transportation, storage and warehousing of equipment; and construction, erection, and fabrication of new facilities.

Construction activities and their representative hazards to an operating unit are shown in Table 1.10-201.

1.10.2 Potentially Impacted SSCs and Limiting Conditions for Operation

The construction activities described above were reviewed for possible impact to operating unit SSCs important to safety.

- VCSNS Unit 1 SSCs important to safety are described in Chapter 3 of the Unit 1 Updated Final Safety Analysis Report (UFSAR).
- Limiting conditions for operation of VCSNS Unit 1 are located in Appendix A of the Unit 1 Operating License (Technical Specifications).
- Units 2 and 3 SSCs important to safety are described in Chapter 3.

- As indicated in Chapter 16, the LCOs for Units 2 and 3 are located in Part 4 of the COL Application.

The initial assessment consisted of a review of individual SSCs and LCOs to determine whether an item is applicable, or may be eliminated due to either examination or being internal and specific to an operating unit. The assessment identified the SSCs that could reasonably be expected to be impacted by construction activities unless administrative and managerial controls are established. The results of the assessment are presented in Table 1.10-202.

Periodic assessment during construction is addressed in Appendix 13AA, Subsection 13AA.1.1.1.1.8.

1.10.3 Managerial and Administrative Controls

To eliminate or mitigate construction hazards that could potentially impact operating unit SSCs important to safety, specific managerial and administrative controls have been identified as shown in Table 1.10-203.

Although not all of the managerial and administrative construction controls are necessary to protect the operating unit, the identified controls are applied to any operating unit as a conservative measure. This conservative approach provides reasonable assurance of protecting the identified SSCs from potential construction hazards and preventing the associated LCOs specified in the operating unit Technical Specifications from being exceeded as a result of construction activities, as discussed below.

The majority of the operating unit SSCs important to safety are contained and protected within safety-related structures. The managerial controls protect these internal SSCs from postulated construction hazards by maintaining the integrity and design basis of the safety-related structures and foundations. Heavy load drop controls, crane boom failure standoff requirements, ground vibration controls and construction generated missile(s) control are examples of managerial controls that provide this protection.

Other managerial controls support maintaining offsite power, control of hazardous materials and gases, and protection of cooling water supplies and safety system instrumentation. These managerial controls prevent or mitigate external construction impacts that could affect SSCs important to safety. These controls also prevent or mitigate unnecessary challenges to safety systems caused by plant construction hazards, such as disruption of offsite transmission lines or impact to plant cooling water supplies.

The above discussed controls to eliminate or mitigate construction hazards that could potentially impact operating unit SSCs important to safety are in place when there is an operating nuclear unit on the site. Additional controls may be established during construction as addressed in Appendix 13AA, Subsection 13AA.1.1.1.1.8.

Table 1.10-201 (Sheet 1 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Site Exploration, Grading, Clearing, Installation of Drainage and Erosion Control Measures	• Overhead Power Lines • Transmission Towers • Underground Conduits, Piping, Tunnels, etc. • Site Access and Egress • Drainage Facilities and Structures • Onsite Transportation Routes • Slope Stability • Soil Erosion and Local Flooding • Construction-Generated Dust and Equipment Exhausts • Encroachment on Plant Control Boundaries • Encroachment on Structures and Facilities
Boring, Drilling, Pile Driving, Dredging, Demolition, Excavation	• Underground Conduits, Piping, Tunnels, etc. • Foundation Integrity • Structural Integrity • Slope Stability • Erosion and Turbidity Control • Groundwater and Groundwater Monitoring Facilities • Dewatering Structures, Systems and Components • Nearby Structures, Systems and Components • Vibratory Ground Motion

Table 1.10-201 (Sheet 2 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Equipment Movement, Material Delivery, Vehicle Traffic	• Overhead Power Lines • Transmission Towers • Underground Conduits, Piping, Tunnels • Crane Load Drops • Crane or Crane Boom Failures • Vehicle Accidents • Rail Car Derailments
Equipment and Material Laydown, Storage, Warehousing	• Releases of Flammable, Hazardous or Toxic Materials • Wind-Generated, Construction-Related Debris and Missiles
General Construction, Erection, Fabrication	• Physical Integrity of Structures, Systems and Components • Adjacent or Nearby Structures, Systems and Components • Instrumentation and Control Systems and Components • Electrical Systems and Components·Cooling Water Systems and Components • Waste Heat Environmental Controls and Parameters • Radioactive Waste Release Points and Parameters • Abandonment of Structures, Systems or Components • Relocation of Structures, Systems or Components • Removal of Structures, Systems or Components

Table 1.10-201 (Sheet 3 of 3)
Potential Hazards from Construction Activities

CONSTRUCTION ACTIVITY HAZARD	POTENTIAL IMPACT
Connection, Integration, Testing	• Instrumentation and Control Systems and Components • Electrical and Power Systems and Components • Cooling Water Systems and Components

Table 1.10-202 (Sheet 1 of 2)
Hazards During Construction Activities

CONSTRUCTION HAZARD	IMPACTED SSCs
Impact on Overhead Power Lines	• Offsite Power System
Impact on Transmission Towers	• Offsite Power Systems
Impact on Utilities, Underground Conduits, Piping, Tunnels, Tanks	• Fire Protection System • Service Water System[1]
Impact of Construction-Generated Dust and Equipment Exhausts	• Control Room Emergency HVAC Systems[1] • Diesel Generators
Impact of Vibratory Ground Motion	• Offsite Power System • Onsite Power Systems • Instrumentation and Seismic Monitors
Impact of Crane or Crane Boom Failures	• Safety-Related Structures
Impact of Releases of Flammable, Hazardous or Toxic Materials	• Control Room Emergency HVAC Systems[1]
Impact of Wind-Generated, Construction-Related Debris and Missiles	• Safety-Related Structures • Control Room Emergency HVAC Systems[1]
Impact on Electrical Systems and Components	• Offsite Power System • Onsite Power Systems
Impact on Cooling Water Systems and Components	• Service Water System[1] • Ultimate Heat Sink[1]
Impact on Radioactive Waste Release Points and Parameters	• Gaseous and Liquid Radioactive Waste Management Systems
Impact of Relocation of Structures, Systems or Components	• Fire Protection System • Service Water System[1]
Impact of Site Groundwater Depression and Dewatering	• Safety-Related Structures and Foundations
Impact of Equipment Delivery and Heavy Equipment Delivery	• Safety-Related Structures and Foundations

Table 1.10-202 (Sheet 2 of 2)
Hazards During Construction Activities

CONSTRUCTION HAZARD	IMPACTED SSCs
Impact of Local Flooding	• Safety-related structures, systems, and components (SSCs)

1 Not applicable to AP1000 operating units.

Table 1.10-203 (Sheet 1 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact on Transmission Power Lines and Offsite Power Lines	• Safe standoff clearance distances are established for transmission power lines, including verification of standoff distance for modules, the reactor vessel and other equipment to be transported beneath energized electric lines to meet minimum standoff clearance requirements. • Physical warning or caution barriers and signage are erected along transport routes.
Impact on Transmission Towers	• Establish controls or physical barriers to avoid equipment collisions with electric transmission support towers.
Impact on Utilities, Underground Conduits, Piping, Tunnels, Tanks	• Grading, excavation, and pile driving require location and identification of equipment or underground structures that must be relocated, removed, or left in place and protected prior to the work activity.
Impact of Construction-Generated Dust and Equipment Exhausts	• Fugitive dust and dust generation is controlled. Potentially affected system air intakes and filters are periodically monitored.
Impact of Vibratory Ground Motion	• Construction administrative procedures, methods, and controls are implemented to prevent exceeding ground vibration and instrumentation limit settings.
Impact of Crane or Crane Boom Failures	• Construction standoff distance controls prevent heavy load impacts from crane boom failures and crane load drops. Drop analyses may be substituted if minimum standoff distances are not practical.

Table 1.10-203 (Sheet 2 of 3)
Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact of Releases of Flammable, Hazardous or Toxic Materials and Missile Generation	• Environmental, safety and health controls limit transport, storage, quantities, type and use of flammable, hazardous, toxic materials and compressed gasses. Construction safety and storage controls maintain potential missile generation events from compressed gasses within the operating unit design basis.
Impact of Wind-Generated, Construction-Related Debris and Missiles	• Administrative controls address equipment, material storage and transport during high winds or high wind warnings. • Plant procedures are followed during severe weather conditions which may call for power reduction or shut down.
Impact on Electrical Systems and Components	• Affected operating unit electrical systems and components within the construction area are identified and isolated or relocated or otherwise protected.
Impact on Cooling Water Systems and Components	• Transport of heavy load equipment over buried cooling water piping is prohibited without evaluation.
Impact on Radioactive Waste Release Points and Parameters	• Engineering evaluation and managerial controls are implemented, as necessary, to prevent radioactive releases beyond the established limits due to construction activity.
Impact of Relocation of Structures, Systems or Components	• Administrative controls identify SSCs that require relocation. Temporary or permanent design changes are implemented if necessary.
Impact of Equipment Delivery and Heavy Equipment Delivery	• Rail transport speed limits and maximum rail loading weights onsite are established. • General equipment and heavy equipment movement controls and limitations are established.

Table 1.10-203 (Sheet 3 of 3) Managerial and Administrative Construction Controls

CONSTRUCTION HAZARDS TO SSCs	MANAGERIAL CONTROL
Impact of Local Flooding	• Site grading and drainage provisions consider potential flooding impacts from local intense precipitation
Impact of Site Groundwater Dewatering	• Administrative controls address groundwater level monitoring

Appendix 1A Conformance With Regulatory Guides

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
DIVISION 1 – Power Reactors			
Reg. Guide 1.1, Rev. 0, 11/70 – Net Positive Suction Head For Emergency Core Cooling and Containment Heat Removal System Pumps			
General		N/A	The AP1000 passive safety systems make maximum use of natural phenomena (gravity, natural circulation, and gas driven injection) and fail-safe position valves, and thus require no active pumps, diesel-generators, or fans. The AP1000 normal residual heat removal system is designed to take suction from the cask loading pit, the in-containment refueling water storage tank, and from containment, however it is not a safety-related system, and does not control or mitigate the consequences of an accident in the licensing basis accident analyses.
Reg. Guide 1.2 – Withdrawn			
Reg. Guide 1.3, Rev. 2, 6/74 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.4, Rev. 2, 6/74 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors			
General		Exception	The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors" will be followed instead of Reg. Guide 1.4.
Reg. Guide 1.5, Rev. 0, 3/71 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.6, Rev. 0, 3/71 – Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems			
General		Exception	The AP1000 main ac power system is a nonsafety-related system. This regulatory guide is applicable only to the Class 1E dc and UPS system.
D.1		Conforms	Guidance applies only to the Class 1E dc and UPS system, since the AP1000 ac power system is a nonsafety-related system.
D.2		N/A	The main ac power system is a nonsafety-related system. Therefore, this regulatory position is not applicable. However, the AP1000 design includes connections to a preferred (offsite) power source and two nonsafety-related onsite standby diesel generators.
D.3		Conforms	
D.4		N/A	See comment on Criteria Section D.2.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
D.5		N/A	See comment on Criteria Section D.2.

Reg. Guide 1.7, Rev. 2, 11/78 and Rev. 3, 03/07 – Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident

Conformance of the design aspects with Revision 2 of the Regulatory Guide is as stated below in the DCD.

C.1		Conforms	Mixing of the containment atmosphere is accomplished through natural passive processes (natural circulation), not with an active system.
C.2		Conforms	
C.3	Regulatory Guide 1.29 Regulatory Guide 1.26	Conforms	The hydrogen recombiners are passive autocatalytic recombiners and nonsafety-related. They do not require and are not supplied with power.
C.4		Conforms	
C.5		Conforms	
C.6		Conforms	

Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

C.2		Conforms	
C.4		Conforms	

Reg. Guide 1.8, Rev. 3, 5/00 – Qualification and Training of Personnel for Nuclear Power Plants

Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. Subsection 13.2.1 defines the responsibility for the training program for plant personnel.

Conformance for programmatic and/or operational aspects is documented below.

C.1		Conforms	
C.2	Section 4 of ANSI/ ANS-3.1-1993	Exception	Not able to meet Regulatory Guide 1.8, Rev. 3 qualification requirements for licensed personnel prior to operations.

Reg. Guide 1.9, Rev. 2, 12/79 – Selection, Design, and Qualification of Diesel Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants

General		N/A	Guidelines apply to Class 1E diesel-generators. They are not applicable to the AP1000.
C.1-14		N/A	Guidelines apply to Class 1E diesel-generators. They are not applicable to the AP1000.

Reg. Guide 1.10 – Withdrawn

Reg. Guide 1.11, Rev.1, 3/10 – Instrument Lines Penetrating Primary Reactor Containment

General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		Conforms	
C.2.a-c		Conforms	
C.2.d	10 CFR 100	Conforms	
C.3.a-c		N/A	Design features described are not applicable to AP1000 Design.
C.4		Conforms	Regulatory position describes the AP1000 implementation.
C.5		Conforms	
C.6		Conforms	
C.7.a-b		N/A	There are no instrument lines penetrating containment that are not associated with protection or safety system instrumentation.

Reg. Guide 1.12, Rev. 2, 3/97 – Instrumentation for Earthquakes

Conformance of the design aspects is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		Exception	Two elevations (excluding the foundation) on a structure internal to the containment are specified in the draft regulatory guide. A second sensor internal to the containment is not provided because access to a sensor at a lower elevation is inconsistent with maintaining occupational radiation exposures as low as reasonably achievable (ALARA) and the containment seismic analyses show such a location to be unnecessary. The response of the containment internal structures is well represented by the response obtained at elevation 138′-0″. Two independent Category I structure foundations where the response is different from that of the containment structure are also specified. Since all seismic Category I structures are part of the nuclear island, which has a common basemat, no additional foundation sensors are required.
C.2		Conforms	Should the seismic response at multiple units at the same site be evaluated as not essentially the same, multiple seismic monitoring systems will be installed at the units. If the seismic response is essentially the same at the other units, the system will be installed at only one unit; however annunciation will be provided in the main control room of each unit.
C.3		Conforms	
C.4		Conforms	The system power panel provides timing signals to components of the entire system. The triaxial acceleration sensor input signals exceeding a preset value are used as the actuation signal for system recording and analysis.
C.5		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.6		Conforms	The triaxial acceleration sensor input signals exceeding a preset value are used as the actuation signal for system recording and analysis.
C.7		Conforms	See Criteria Section C.2.
C.8		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for development of procedures.
Conformance for programmatic and/or operational aspects is documented below.			
C.3		Conforms	
C.8		Conforms	
Reg. Guide 1.13, Rev. 1, 12/75 and Rev. 2, 03/07 – Spent Fuel Storage Facility Design Basis			
Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated below in the DCD.			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4	Regulatory Guide 1.25	Exception	The ventilation system is not designed to mitigate the consequences of a fuel handling accident.
C.5		Conforms	
C.6		Conforms	
C.7		Conforms	
C.8		Exception	Normal makeup supply (demineralized water) is not seismic Category I. Long-term post-accident supply piping is seismic Category I.
Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.7		Conforms	
Reg. Guide 1.14, Rev. 1, 8/75 – Reactor Coolant Pump Flywheel Integrity			
1.a	ASTM A.20	Exception	The flywheel is made of a bi-metallic design. Heavy alloy segments are fitted to a stainless steel hub and held in place by a retaining ring. Therefore, the specific guidelines in this section are not directly applicable to the AP1000.
1.b		Exception	Fracture toughness and tensile properties are checked for components that are required for structural integrity of the bi-metallic flywheel.
1.c		N/A	This guideline is not applicable to the flywheel assembly. Therefore, the guideline is not applicable to the AP1000 reactor coolant pump.
1.d		Conforms	The components of the flywheel that are relied upon for structural integrity require no welding.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
2.a-b		Conforms	
2.c-e	ASME Code, Section III	Exception	The limits and methods of ASME Code, Section III, Paragraph F-1331.1(b), (replacement for Paragraph F-1323.1) are not directly applicable to the flywheel assembly. The calculated stress levels in the flywheel are evaluated against the ASME Code, Section III, Subsection NG stress limits used as guidelines and the recommended stress limits in Positions 4.a and 4.c of the Standard Review Plan 5.4.1.1.
2.f		Exception	The calculated stress levels in the flywheel satisfy the ASME Code, Section III, Subsection NG stress limits used as guidelines and the recommended stress limits in Position 4.a of the Standard Review Plan 5.4.1.1.
2.g		Conforms	
3		Conforms	
4.a	ASME Code, Section III, NB-2545 or NB-2546,NB-2540, NB-2530	Exception	The inspections and guidelines referenced in the regulatory guide were developed for steel flywheels in shaft seal pumps. The paragraphs of Subsection NB referenced in the regulatory guide apply only to forged and plate steel components. The bi-metallic flywheel design will be manufactured using multiple processes and materials. In accordance with the regulatory guide, each structural component of the bi-metallic flywheel will be inspected prior to final assembly according to its fabrication and the procedures outlined in Section III, NB-2500 of the ASME Code. Inspection of the flywheel assembly inside the sealed enclosure following a spin test is not practical.
4.b	ASME Code, Section XI	Exception	Inservice inspection of the flywheel assembly is not required to support safe operation of the reactor coolant pump. Planned, routine inspections of the flywheel assembly require considerable occupational radiation exposure and are not recommended. Inservice inspection of the flywheel assemblies requires extensive disassembly. Postulated missiles from the failure of the flywheel are contained within the stator shell and the pressure boundary is not breached. Vibration of the shaft due to a small flywheel fracture or leak in the enclosure does not result in stresses in the pressure boundary of sufficient magnitude to result in a break in the primary pressure boundary.
Reg. Guide 1.15 – Withdrawn			
Reg. Guide 1.16, Rev. 4, 8/75 – Reporting of Operating Information – Appendix A Technical Specifications			
General		N/A	Not applicable to AP1000 design certification. Reporting requirements associated with the technical specifications are identified in Tech Spec. Section 5.6. DCD subsection 1.1.1 defines the responsibility for finalizing the technical specification.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions

Reg. Guide 1.17 – Withdrawn

Reg. Guide 1.18 – Withdrawn

Reg. Guide 1.19 – Withdrawn

Reg. Guide 1.20, Rev. 2, 5/76 and Rev. 3, 3/07 – Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational and Initial Startup Testing

Conformance with Revision 2 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

General		Conforms	The AP1000 internals are similar to those for a three-loop XL Westinghouse 17 x 17 robust fuel assembly core internals, a core shroud and the new incore instrumentation system. The upper internals are not significantly changed from standard designs. An internals vibration measurement program is conducted during hot functional testing. The results are evaluated based on pre-established allowable levels. During hot functional testing the AP1000 internals are subjected to operational system flow conditions that are similar to those imposed on previous 3XL three-loop designs. The duration of the hot functional flow testing is the same as that for the previous design. Pre- and post-test inspections are conducted to confirm that the AP1000 internals experience no excessive motion or wear.
C.1		Conforms	Although the AP1000 internals do not represent a first of a kind or unique design on the basis of the arrangement, design, size, or operating conditions, for the purposes of the reactor internals preoperational test program, the first operational AP1000 reactor vessel internals are classified as a prototype. Subsequent plants will be classified as Non-Prototype Category I based on the designation of the first AP1000 as a Valid Prototype. See Subsections 3.9.2.3 and 3.9.2.4 for additional information on the vibration assessment of the reactor vessel internals.
C.2		Conforms	A comprehensive vibration assessment program will be developed for the first AP1000 reactor vessel internals. With regard to transients, data are acquired only during the hot functional test. Additionally, data are calculated over the ranges of hot functional test temperatures and during startup, shutdown, and steady-state operation of various combinations of reactor coolant pumps. Subsection 3.9.8 addresses information provided about the vibration assessment program.
C.3		Conforms	Subsequent to completion of the vibration assessment program for the first AP1000 reactor vessel internals, the vibration analysis program will address the criteria for Non-prototype Category I internals.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.21, Rev. 1, 6/74 – Measuring Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	The design guidance of this regulatory guide for the selection of locations and type of effluent measurements to cover major or potentially significant pathways of release of radioactive materials during normal reactor operation, including anticipated operational occurrences, are incorporated in the plant design and in the requirements of the radiological effluent technical specifications. The calibration of effluent monitoring systems is performed according to written plant procedures. Subsection 11.5.8 defines the responsibility for the radiation monitoring program. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
C.1		N/A	Not applicable to AP1000 design certification. Subsection 11.5.8 defines the responsibility for the radiation monitoring program.
C.2		Conforms	
C.3-14		N/A	Not applicable to AP1000 design certification. Subsection 11.5.8 defines the responsibility for the radiation monitoring program.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.3-C.5		Conforms	
C.6		Conforms	
C.7-C.14		Conforms	
Reg. Guide 1.22, Rev. 0, 2/72 – Periodic Testing of Protection System Actuation Functions			
General		Conforms	Safety actuation circuitry is provided with a capability for testing with the reactor at power. The protection system, including the engineered safety features test cabinet design, conforms to this regulatory guide. The protection functions are tested at power to the greatest extent practical. Only the device function and/or system level function is not universally tested. The logic associated with the devices has the capability for testing at power, at the subsystem and/or component level.
D.1		Conforms	The AP1000 protection system is designed to permit periodic testing.
D.2-4		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.23, Second Proposed Rev. 1, 4/86 – Onsite Meteorological Programs			
Reg. Guide 1.23, Rev. 1, 3/07 Meteorological Monitoring Programs for Nuclear Power Plants			
Conformance of the DCD scope of second proposed Rev. 1 aspects is as stated below in the DCD.			
General		Conforms	The onsite meteorological measurement program is site-specific and will be defined as indicated in subsection 2.3.6. The number and location of meteorological instrument towers are determined by actual site parameters. Subsection 2.3.6 defines the responsibility for the onsite meteorological program. The data display and processing system has the capability to record the data from the meteorological instruments and display the information in the main control room.
Conformance with Rev. 1 of this Regulatory Guide for site-specific aspects is documented below.			
General		Conforms	Plant operations phase program will conform.
Reg. Guide 1.24, Rev. 0, 3/72 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure			
General		N/A	This regulatory guide applies to the evaluation of a waste gas storage tank failure. The AP1000 design does not include waste gas storage tanks. Therefore, it is not applicable to the AP1000.
Reg. Guide 1.25, Rev. 0, 3/72 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors			
General		Exception	The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" will be followed instead of Reg. Guide 1.25.
Reg. Guide 1.26, Rev. 3, 2/76 and Rev. 4, 3/07 – Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containment Components of Nuclear Power Plants			
Conformance with Revision 3 of the Regulatory Guide for DCD scope of work is as stated below in the DCD.			
C.1		Exception	A portion of the chemical and volume control system that is defined as reactor coolant pressure boundary uses an alternate classification in conformance with the requirements of 10 CFR 50.55a(a)(3). The alternate classification is discussed in Section 5.2.
C.1.a		Exception	For the AP1000 plant design, Quality Group B is reserved for the containment boundary including any extensions such as containment isolation valves and associated piping. Quality Group C is essentially equivalent quality except that it has less stringent ISI. For equipment such as passive safety system accumulators, minor leakage is not a problem for the following reasons: a. It is located inside containment so activity releases are contained.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			b. Minor leakage does not affect its functional performance, especially considering the limited duration of post-accident operation. c. There is continuous water level and gas pressure monitoring of the passive safety system accumulators that detects leaks. This approach results in the change of quality group (from Quality Group B to Quality Group C) for various components such as the IRWST. Portions of systems that provide emergency core cooling system functions and are constructed using ASME Code, Section III, Class 3 criteria have the additional requirement that radiography will be conducted on a random sample of welds during construction, see subsection 3.2.2.5.
C.1.b		Exception	The AP1000 residual heat removal system is a nonsafety-related system, but it is classified as Quality Group C. The passive core cooling system provides the safety-related function that the residual heat removal system provides in current plants with active safety-related systems.
C.1.c		N/A	Applies to boiling water reactors only.
C.1.d		Conforms	Portions of the feedwater and steam systems are Quality Group B, up to the isolation valves.
C.1.e		Conforms	
C.2.a		Conforms	The component cooling water and the service water systems are Quality Group D since they perform no safety-related functions.
C.2.b		Conforms	Component cooling water is not required for safe shutdown of the AP1000. The reactor cooling pumps do not have seals and do not require seal water supply.
C.2.c		Conforms	
C.2.d		N/A	Regulatory Guide 1.143 supersedes this guideline.
C.2.e		N/A	Regulatory Guide 1.143 supersedes this guideline.
C.3		Exception	Systems that are normally radioactive are classified as Quality Group D. AP1000 also classifies as Quality Group D, nonsafety-related systems and components which have functions that have been identified as important as part of the implementation of the regulatory treatment of nonsafety-related systems or as defense-in-depth systems. Some structures, systems, and components that have the potential to be contaminated with radioactive fluids but normally do not contain radioactive fluids are not classified as Quality Group D.
Conformance with Revision 4 of this Regulatory Guide for remaining scope is documented below.			
General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.27, Rev. 2, 1/76 – Ultimate Heat Sink for Nuclear Power Plants			
C.1		Conforms	The passive containment cooling system water storage tank is sized to provide water cooling to the containment vessel and provide heat removal to meet the requirements of General Design Criterion 38 to reduce and maintain the containment temperature and pressure following a postulated loss-of-coolant accident for 3 days following passive containment cooling system actuation. This water delivery is done in conjunction with the flow of air over the containment shell to provide the containment cooling. After 3 days of water delivery from the PCCWST, the PCS cooling water supply is continued through either: • simple operator action via installed safety-related piping and connection utilizing offsite or available onsite supplies of water and an offsite pump to resupply water to the tank; or, • simple operator action utilizing onsite seismically analyzed pumps, piping and 4 days of water inventory within the passive containment cooling ancillary water storage tank to resupply the PCCWST. Supplemental supplies would then be available from either onsite storage facilities or an offsite source. Since the passive containment cooling system can function with replenished water supplies from either onsite or offsite, the system meets the guideline of providing an ultimate heat sink for more than 30 days.
C.2		Conforms	The AP1000 design conforms to this regulatory position, provided that the definition of a single failure of a man-made structure does not include the safety-related, seismically-designed containment structure assembly. The AP1000 uses the atmosphere as the ultimate heat sink. A baffle located between the containment shell and the shield building sustains the natural circulation that provides for air flow over the containment shell to carry heat away. The baffle is composed of a large number of panels and will continue to function if damaged by an external missile.
C.3		Conforms	The seismically-designed passive containment cooling system water storage, integral to the containment structure meets this regulatory position.
C.4		Conforms	Technical Specifications exist for the passive containment cooling system and air inlet and air baffle.
Reg. Guide 1.28, Rev. 0, 6/72 and Rev. 3, 8/85 – Quality Assurance Program Requirements (Design and Construction)			
Conformance for DCD scope of work is as stated below in the DCD.			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	ANSI/ASME N45.2-1977 ANSI/ASME NQA-1-1983 through NQA-1a-1983 Addenda	Conforms	The Westinghouse quality assurance program is described in Chapter 17. Refer to "Westinghouse Electric Company Quality Management System" (QMS) referenced therein for Westinghouse positions on regulatory guides within the scope of the quality assurance program. In some cases current industry consensus standards have replaced the standards specifically referenced by certain regulatory guides. In particular, the N45.2 series standards have been replaced by ASME NQA-1. Therefore, the "Quality Management System" may reference ASME NQA-1 rather than the N45.2 series standards when describing the Westinghouse position. QMS complies with ASME NQA-1-1994.
2.	Criteria 17 10 CFR 50 Appendix B	Conforms	

Design, procurement, and construction activities associated with VCSNS Units 2 and 3 that may occur before the COL is issued will be conducted in accordance with the existing SCE&G Unit 1 QA Program requirements (Regulatory Guide 1.28, Rev. 0). Design and construction activities that occur following COL issuance will be conducted in accordance with the QAPD submitted as Part 13 of the application and is the QAPD that was evaluated and discussed in Table 1.9-202 for conformance to SRP 17.1.

Conformance for remaining scope is documented below.

General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1-1994 in lieu of the identified outdated standards.

Reg. Guide 1.29, Rev. 3, 9/78 and Rev. 4 3/07 – Seismic Design Classification

Conformance with Revision 3 of the Regulatory Guide for DCD scope of work is as stated below in the DCD.

C.1.a		Conforms	
C.1.b		Conforms	
C.1.c		Conforms	
C.1.d		Exception	The AP1000 normal residual heat removal system is nonsafety-related. The safety-related function of decay heat removal is provided by the safety-related passive residual heat removal heat exchanger of the passive core cooling system that is seismic Category I. The spent fuel pool cooling system does not have active components that are required for the safety-related decay heat removal function. This function is provided passively through a large heat sink of water in the pool. The spent fuel pool is sized to keep the fuel covered for at least 72 hours without active cooling or makeup following a loss of ac power sources. The 72-hour sizing calculation accounts for the maximum loss of water due to the rupture of non-seismic piping. Seismic Category I components within the spent fuel pool cooling system include the containment penetration, the connections for makeup, and the spent fuel pool.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1.e		N/A	Applies to boiling water reactors only.
C.1.f		Conforms	
C.1.g		Exception	The AP1000 does not have a safety-related auxiliary feedwater system. The safety-related decay heat removal function is provided by the passive residual heat removal heat exchanger. The safety-related functions of the essential service water system are provided by the passive residual heat removal heat exchangers and the passive containment cooling system. The component cooling system is a nonsafety-related system, since it performs no safety-related functions.
C.1.h		Conforms	
C.1.i		N/A	The diesel-generators are nonsafety-related. Therefore, this section is not applicable to the AP1000.
C.1.j		Conforms	
C.1.k		Conforms	
C.1.l		Conforms	
C.1.m		Conforms	
C.1.n		Exception	Structures or equipment whose failure results in incapacitating injury to the occupants of the main control room are classified as seismic Category II and covered under Position 2 of this regulatory guide.
C.1.o		Conforms	
C.1.p		Conforms	
C.1.q		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4		Conforms	

Conformance with Revision 4 of this Regulatory Guide for remaining scope is documented below.

C.4		Conforms	

Reg. Guide 1.30, Rev. 0, 8/72 – Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

Conformance for DCD scope of work is as stated below in the DCD.

General	ANSI/ASME N45.2.4-1972	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the Quality Assurance program.

Conformance for remaining scope is documented below.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.31, Rev. 3, 4/78 – Control of Ferrite Content in Stainless Steel Weld Metal			
General		Conforms	
C.1-5		Conforms	
Reg. Guide 1.32, Rev. 2, 2/77 – Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants			
Reg. Guide 1.32, Rev. 3, 03/04 – Criteria for Power Systems for Nuclear Power Plants			
Conformance of the design aspects with Revision 2 of the Regulatory Guide is as stated below in the DCD.			
1.	IEEE Std. 308-1974	Exception	Regulatory Guide 1.32 endorses IEEE Std. 308-1974 (Reference 5) which has been superseded by IEEE Std. 308-1991(Reference 6). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.32. The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000.
1.a	Regulatory Guide 1.93	N/A	The AP1000 has no safety-related ac power system. Therefore, the guidelines specified in this criterion section recommending the availability of offsite power "within a few seconds" are not applicable.
1.b	IEEE Std. 308-1974, Section 5.3.4	Exception	See comment on Criterion Section 1.
1.c	IEEE Std. 450-1975	Exception	Battery performance discharge test is per IEEE-Std 450-1995 as described in Bases for Technical Specification, Surveillance Requirement 3.8.7.6.
1.d	Regulatory Guide 1.6 Regulatory Guide 1.75	Exception	The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000.
1.e	Regulatory Guide 1.75	Exception	The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000.
1.f	Regulatory Guide 1.9	N/A	Guidelines apply to Class 1E diesel generators. Therefore, they are not applicable to the AP1000.
2.a	IEEE Std. 308-1974, Section 8.2, 8.3.1; Regulatory Guide 1.81	N/A	The AP1000 is a single-unit plant. This criterion is not applicable to the AP1000. When two or more AP1000s are located adjacent, electrical systems are not shared.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
2.b.	Regulatory Guide 1.93	Exception	The guidelines are applicable to the Class 1E dc and UPS system only. There are no safety-related ac power systems in the AP1000. See comments on Regulatory Guide 1.93.
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.33, Rev. 2, 2/78 – Quality Assurance Program Requirements (Operation)			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N18.7-1976 ANS-3.2	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the Quality Assurance program. Regulatory Guide 1.33 is used in a specific manner for determining documentation adequacy in regard to the ongoing qualification method based on the assumption the utility programs are in conformance with Regulatory Guide 1.33.
Conformance for remaining scope is documented below.			
C.1		Conforms	
C.2		Clarification	See separate conformance statement for each identified Regulatory Guide.
C.3 – C.5		Conforms	
Reg. Guide 1.34, Rev. 0, 12/72 – Control of Electroslag Weld Properties			
General	ASME Code, Sections III and IX	Conforms	The AP1000 prohibits the use of electroslag welding on reactor coolant pressure boundary components. AP1000 safety-related components that use electroslag welding conform to the provisions of the ASME Code and this regulatory guide.
C.1-5		Conforms	
Reg. Guide 1.35, Rev. 3, 7/90 – Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structure. Therefore, this regulatory guide is not applicable to the AP1000.
Reg. Guide 1.35.1, Rev. 0, 7/90 – Determining Prestressing Forces for Inspection of Prestressed Concrete Containments			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structure. Therefore, this regulatory guide is not applicable to the AP1000.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.36, Rev. 0, 2/73 – Nonmetallic Thermal Insulation for Austenitic Stainless Steel			
General		Conforms	
C.1		Conforms	
C.2.a	ASTM C692-71 RDT M12-1T	Conforms	
C.2.b		Conforms	
C.3-4		Conforms	
Reg. Guide 1.37, Rev. 0, 3/73 and Rev. 1, 3/07 – Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.			
General	ANSI N45.2.1-1973	Exception	The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance regulatory guides have been replaced by ASME NQA-1. ANSI N45.2.1, which is referenced in Regulatory Guide 1.37, has been incorporated into NQA-1 Subpart 2.1. The technical requirements specified in ANSI N45.2.1 and NQA-1 Subpart 2.1 are compatible. Therefore, compliance with NQA-1 Subpart 2.1 satisfies Regulatory Guide 1.37. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.38, Rev. 2, 5/77 – Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.2-1972	Exception	The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance regulatory guides have been replaced by ASME NQA-1. Refer to the Regulatory Guide 1.28 position. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.39, Rev. 2, 9/77 – Housekeeping Requirements for Water-Cooled Nuclear Power Plants			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.3-1973	Exception	The ANSI N45.2 series of standards that are referenced by the current revisions of the Quality Assurance regulatory guides have been replaced by ASME NQA-1. Refer to the Regulatory Guide 1.28 position. Section 17.5 defines the responsibility for the Quality Assurance program.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.40, Rev. 0, 3/73 – Qualification Tests of Continuous-Duty, Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants			
General	IEEE Std. 334-1971	N/A	The AP1000 does not have continuous-duty safety-related motors installed inside the containment. Therefore, the regulatory guide is not applicable to the AP1000.
Reg. Guide 1.41, Rev. 0, 3/73 – Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments			
General		Exception	The guidelines are followed for Class 1E dc power systems during the preoperational testing of AP1000 redundant onsite electric power systems to verify proper load group assignments, except as follows. Complete preoperational testing of the startup, sequence loading, and functional performance of the load groups is performed where practical. In those cases where it is not practical to perform complete functional performance testing, an evaluation is used to supplement the testing.
Reg. Guide 1.42 – Withdrawn			
Reg. Guide 1.43, Rev. 0, 5/73 – Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components			
General		Conforms	The guidelines of this regulatory guide are followed during the welding process used for cladding ferritic steel components of the AP1000 with austenitic stainless steel. No qualifications are provided for by this regulatory guide for ASME SA-533 material and equivalent chemistry for forging grade ASME SA-508, Class 3, material. The reactor vessel, steam generator channel heads, accumulators, and core makeup tanks design specification restricts the low alloy steel forging material to ASME SA-508, Class 3, which is made to a fine grain practice only. Cladding of ASME SA-508, Class 2 is not applicable to the AP1000 design. The fabricator monitors and records the weld parameters to verify agreement with the parameters established by the procedure qualification as stated in Regulatory Position C.3.
C.1-3		N/A	The AP1000 material, specifically ASME SA-533 and SA-508 Class 3 made to a fine grain practice, is not subjected to the controls in this regulatory guide.
Reg. Guide 1.44, Rev. 0, 5/73 – Control of the Use of Sensitized Stainless Steel			
C.1 – C.6		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.45, Rev. 0, 5/73 – Reactor Coolant Pressure Boundary Leakage Detection Systems			
Conformance of the design aspects is as stated below in the DCD.			
C.1		Conforms	
C.2		Conforms	
C.3		Exception	The AP1000 uses two methods for leak detection, including sump level and flow monitoring, and airborne particulate radioactivity monitoring.
C.4		Conforms	
C.5		Conforms	
C.6		Conforms	
C.7		Conforms	
C.8		Conforms	
C.9		Conforms	
Conformance with programmatic and/or operational aspects is documented below.			
C.7		Conforms	
Reg. Guide 1.46 – Withdrawn			
Reg. Guide 1.47, Rev. 0, 5/73 – Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems			
General	IEEE Std. 279-1971, Section 4.13; and 10 CFR 50 App. B, Criterion XIV	Conforms	
C.1-4		Conforms	
Reg. Guide 1.48 – Withdrawn			
Reg. Guide 1.49, Rev. 1, 12/73 – Power Levels of Nuclear Power Plants			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
Reg. Guide 1.50, Rev. 0, 5/73 – Control of Preheat Temperature for Welding of Low-Alloy Steel			
General	ASME Code, Sections III and IX	N/A	The guidelines of this regulatory guide are followed during the initial fabrication of low-alloy steel components of the AP1000.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			This regulatory guide is considered as applicable to ASME Code, Section III, Class 1 components. The AP1000 practice for Class 1 components is in agreement with the guidance of this regulatory guide except for Regulatory Positions C.1(b) and 2. For AP1000 Class 2 and 3 components, the guidelines provided by this regulatory guide are not applied, however all requirements of the ASME Boiler and Pressure Vessel Code are imposed.
C.1(b)		Conforms	The welding procedures are qualified within the preheat temperature ranges required by ASME Code, Section IX. Experience has shown excellent quality of welds using the ASME qualification procedures.
C.2		Exception	The AP1000 position is that the guidance specified in this regulatory guide is both unnecessary and impractical. Code acceptable low-alloy steel welds have been and are being made under present procedures. It is not necessary to maintain the preheat temperature until a post-weld heat treatment has been performed in accordance with the guidance provided by this regulatory guide, in the case of large components. In some cases of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or final post-weld heat treatment has been followed. In other cases, an extended preheat practice has been utilized in accordance with the reactor vessel design specification. In this practice, the weld temperature is maintained at 400°F to 750°F for 4 hours after welding. The weld temperature may then be lowered to ambient without performing an intermediate or final post-weld heat treatment at 1100°F. The welds have shown high integrity. Westinghouse practices are documented in WCAP-8577 (Reference 9) which has been accepted by the Nuclear Regulatory Commission.

Reg. Guide 1.51 – Withdrawn

Reg. Guide 1.52, Rev. 3, 6/01 – Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants

Reg. Guide 1.52, Rev. 3, 6/01 – Design, Inspection and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants

Conformance with the design and operational aspects is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	The AP1000 main control room emergency habitability system (VES) includes a passive filtration system that is contained entirely within the main control room envelope. The passive filtration portion of the AP1000 VES contains no active equipment. This regulatory guide is followed where appropriate for the AP1000 MCR passive filtration system.
C.1		Conforms	The performance, design, construction, acceptance testing, and QA of the AP1000 MCR passive filtration system will be in accordance with ASME AG-1-1997 where appropriate. Also, it will be designed to ASME N509-1989 and tested to ASME N510-1989.
C.2		Conforms	The AP1000 MCR passive filtration is designed to operate under the environmental conditions specified by the guidelines in this regulatory position.
C.3.1 – 3.2		Exception	Redundant filtration trains are not needed in the AP1000 passive filtration system. The passive filtration system contains no active components or moving parts. Adequate margin is provided in the design to prevent the likelihood of a passive failure.
C.3.3		Conforms	The maximum pressure surge is experienced during the highly unlikely failure of the upstream pressure regulator valve in the VES. The passive filtration system is designed to operate under this pressure surge.
C.3.4		Conforms	All components in the MCR passive filtration system are designed to be seismic Category I.
C.3.5		Conforms	The mechanical design of the MCR passive filtration system is such that it can withstand the buildup of postulated radioactivity levels and maintain functionality.
C.3.6		Conforms	The AP1000 passive filtration system volumetric air-flow rate is below 30,000 CFM.
C.3.7		Conforms	The flow instrumentation is not required to be safety related. The instrumentation does not perform a safety function. The instrumentation monitoring flow from the VES emergency air storage tanks indicates whether there is sufficient flow coming from the compressed air tanks to induce the passive filtration.
C.3.8		Exception	The AP1000 passive filtration system does not rely upon, use, or need a power supply and/or electrical distribution.
C.3.9		Conforms	The AP1000 passive filtration system is designed to operate for 72 hours following a DBA.
C.3.10		Conforms	
C.3.11		Exception	There are no outdoor air intakes for the AP1000 passive filtration system. The system uses breathable compressed air that is stored in compressed air tanks during the post-72-hour operation time.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.12		Exception	The AP1000 passive filtration system is located completely within the CRE. Leakages as explained in this regulatory position are not applicable to this system.
C.4.1 – 4.2		Exception	There are no moisture separators and/or heaters in the AP1000 passive filtration line.
C.4.3 – 4.7		Conforms	
C.4.8		Exception	There are no water drains in the AP1000 passive filtration line.
C.4.9		Exception	The credited adsorber efficiencies are 90% for elemental iodine and 30% for organic iodine. These efficiencies assume no humidity control.
C.4.10		Conforms	Type II adsorbers are used in this application
C.4.11		Conforms	The AP1000 passive filtration line uses impregnated activated carbon as the absorbent. The absorber is designed for a minimum average atmosphere residence time of 0.25 seconds per 2 inches of absorbent bed.
C.4.12		Conforms	
C.4.13		Conforms	
C.4.14		Exception	The passive filtration line requires no fans.
C.5.1		N/A	Only one bank of filters is used.
C5.2		Exception	This system is not used for normal HVAC, and the filters should not build up unusual levels of particulate once installed.
C.6.1		Conforms	
C.6.2 – 6.6		Conforms	
C.7		Conforms	
Table 1		Exception	The Technical Specification methyl iodide penetration acceptance limit for the AP1000 activated carbon adsorber is 5%, which correlates to 90% removal efficiency of both organic and elemental iodine. The calculated design basis for the AP1000 passive filtration adsorbers assumes a 30% organic iodine removal efficiency and a 90% elemental iodine efficiency. A 1% bypass leakage is accounted for by testing to increased organic iodine removal efficiency.

Reg. Guide 1.53, Rev. 0, 6/73 – Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

Reg. Guide 1.53, Rev. 2, 11/03 – Application of the Single-Failure Criterion to Safety Systems

Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.This guidance is completely within the scope of the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	IEEE Std. 379-1972	Exception	Regulatory Guide 1.53 endorses IEEE Std. 379-72 (Reference 10), which has been superseded by IEEE Std. 379-2000 (Reference 11). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.53. IEEE Std. 379-2000 is endorsed by DG-1118 (Proposed Revision of Regulatory Guide 1.53). The guidelines are applicable to safety-related dc power systems. There are no safety-related ac power sources in the AP1000.
Reg. Guide 1.54, Rev. 1, 7/00 and Rev. 2 10/10 – Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants			RN-13-021
Conformance with Revision 1 of this regulatory guide is as stated below.			
General	ASTM D 3843-00, ASTM D 3911-95, ASTM D 5144-00	Exception	Some coatings inside containment are nonsafety-related and satisfy appropriate ASTM Standards. See subsection 6.1.2 for additional information. Application is controlled by procedures using qualified personnel to provide a high quality product. The paint materials for coatings inside the containment are subject to 10 CFR Part 50 Appendix B Quality Assurance requirements. The quality assurance features of the AP1000 coatings systems are outlined in subsection 6.1.2.1.6. Subsection 6.1.3 defines the responsibility for the coating program.
General		Exception	Revision 2 of the regulatory guide is applied to use for protective coatings on the containment vessel shell and attachments to the containment vessel shell. RN-13-021
Conformance with Revision 2 of this regulatory guide is as stated below.			
General	ASTM D 3843-00, ASTM D 3911-08, ASTM D 5144-08	Exception	Some coatings inside containment are nonsafety-related and satisfy appropriate ASTM Standards. See subsection 6.1.2 for additional information. Application is controlled by procedures using qualified personnel to provide a high quality product. The paint materials for coatings inside the containment are subject to 1 0 CFR Part 50 Appendix B Quality Assurance requirements. The quality assurance features of the AP1 000 coatings systems are outlined in DCD subsection 6.1.2.1.6. Subsection 6.1.3 defines the responsibility for the coating program. RN-13-021
General		Exception	Revision 2 of this regulatory guide is applied to use only for protective coatings on the containment vessel shell and attachments to the containment vessel shell. Other protective coatings use Revision 1 of the regulatory guide. RN-13-021

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	ASTM D 3911-08	Exception	Regulatory Guide 1.54 provides additional acceptance criteria for Service Level I coatings when tested per ASTM D 3911-08. Of the additional acceptance criteria a) Peeling and delamination shall not be permitted and b) Cracking is not considered a failure unless it is accompanied by delamination or loss of adhesion. These are not applicable to a Service Level I epoxy top coat applied over inorganic zinc. This is because delamination and/or detachment of epoxy coatings does not impact plant safety and does not adversely affect underlying inorganic zinc coating or the ability of the inorganic zinc coating to perform its safety functions. RN-14-129

Conformance with programmatic and/or operational aspects of Revision 1 is documented below. Revision 2 is not utilized for the programmatic and/or operational aspects. RN-13-021

General		Conforms	

Reg. Guide 1.55 – Withdrawn

Reg. Guide 1.56, Rev. 1, 7/78 – Maintenance of Water Purity in Boiling Water Reactors

General		N/A	Applies to boiling water reactors only.

Reg. Guide 1.57, Rev. 0, 6/73 and Rev. 1, 03/07 – Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

General	ASME Code, Section III	Exception	The regulatory guide was issued in 1973. It refers to the ASME Code through the Summer 1973 Addenda. The acceptance criteria have been defined in greater detail in SRP 3.8.2. The AP1000 complies with the SRP acceptance criteria with the exception that the operating basis earthquake is excluded.

Reg. Guide 1.58 – Withdrawn

Reg. Guide 1.59, Rev. 2, 8/77 – Design Basis Floods for Nuclear Power Plants

Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

C.1-4	Regulatory Guide 1.29	N/A	The maximum water level due to the probable maximum flood is established as a site interface in Chapter 2 and is used in the design of the AP1000. Subsection 2.4.1.2 defines the responsibility for addressing site-specific information on historical flooding and potential flooding factors.

Conformance with programmatic and/or operational aspects is documented below.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Exception	Regulatory Guide 1.59, Appendix A indicates use of ANSI N170-1976 “Standards for Determining Design Basis Flooding at Power Reactor Sites.” In place of this standard, ANSI/ANS 2.8-1992 “Determining Design Basis Flooding at Power Reactor Sites” was used. ANSI/ANS 2.8-1992 was withdrawn on July 26, 2002. However, a replacement standard has not been issued. NUREG-0800 2.4.3 Revision 4, March 2007and 2.4.4 Revision 3, March 2007 include ANSI/ANS 2.8-1992 as a reference. ANSI/ANS 2.8-1992 is also specifically identified in the review procedures subsection of NUREG-0800 2.4.4.

Reg. Guide 1.60, Rev. 1, 12/73 – Design Response Spectra for Seismic Design of Nuclear Power Plants

Conformance of the design aspects is as stated below in the DCD.

C.1		Conforms	
C.2		Conforms	

Reg. Guide 1.61, Rev. 0, 10/73 and Rev. 1, 3/07 – Damping Values for Seismic Design of Nuclear Power Plants

Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

General		Conforms	Damping values used in the AP1000 safe shutdown earthquake analyses are shown in Table 3.7.1-1. These values are based on Regulatory Guide 1.61, on the recommendations of ASCE 4-86 (Reference 12), and on values used and accepted on past projects (Reference 13). The values are conservative relative to realistic damping values reported in the literature (Reference 14) A site interface is established to verify that the site is within the range considered in the design.

Reg. Guide 1.62, Rev. 0, 10/73 – Manual Initiation of Protective Actions

C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4		Conforms	
C.5		Conforms	
C.6	IEEE Std. 279-1971, Section 4.16	Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.63, Rev. 3, 2/87 – Electric Penetration Assemblies in Containment Structures for Nuclear Power Plants			
General	IEEE Std. 317-1983 IEEE Std. 741-1986, Section 5.4	Exception	Regulatory Guide 1.63 endorses IEEE Std. 741-1986 (Reference 15), which has been superseded by IEEE Std. 741-1997 (Reference 16). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.63. Electric penetration assemblies are in conformance with IEEE Std. 317-1983 (Reference 17), Reference 16, and this regulatory guide with the clarification discussed below. The majority of low voltage control circuits are self-limiting in that circuit resistance limits the fault current to a level that does not damage the penetration. Where, on a case-by-case basis, a circuit is found not to be self-limiting, primary and backup breaker or fuse coordination or the addition of a subfeed over current protection as in the case of motor control centers, provide for safe operation. The energy levels in the instrument systems are such that damage cannot occur to the containment penetration.
Reg. Guide 1.64 – Withdrawn			
Reg. Guide 1.65, Rev. 0, 10/73 – Materials and Inspections for Reactor Vessel Closure Studs			
Conformance of the design aspects is as stated below in the DCD.			
C.1.a	ASME Code, Section III, Subsection NB	Conforms	The reactor vessel closure stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. The material meets the criteria of Appendix G to 10 CFR 50. The reactor vessel design specification requires the maximum tensile strength of 170,000 psi for the closure stud material.
C.1.b		Conforms	
C.2	ASME Code, Section III, NB-2580	Conforms	The guidelines of this regulatory guide are followed during the fabrication of the stud bolts and nuts.
C.3		Conforms	The guidance of this regulatory guide is followed during the venting and filling of the AP1000 pressure vessel.
C.4	ASME Code, Section XI; ASME Code, Section III, NB-2545 or NB-2546	Conforms	The guidelines of this regulatory guide are followed during the inservice examination of the AP1000 pressure vessel stud bolting.
Conformance with programmatic and/or operational aspects is documented below.			
C.3		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.4		Exception	ASME XI ISI criteria for reactor vessel closure stud examinations are applied in lieu of the ASME III NB 2545 and NB 2546 surface examinations. The volumetric examinations currently required by ASME XI provide improved (since 1973) detection of bolting degradation.

Reg. Guide 1.66 – Withdrawn

Reg. Guide 1.67 – Withdrawn

Reg. Guide 1.68, Rev. 2, 8/78 and Rev. 3, 3/07 – Initial Test Program for Water-Cooled Nuclear Power Plants

Conformance with Revision 2 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1	App. A.1.a	Conforms	Applies to AP1000 reactor coolant system components. (Pressurizer power-operated relief valves and reactor vessel internal vent valves are not design features of the AP1000. Jet pumps are applicable to boiling water reactors only.)
	App. A.1.b	Conforms	Applies to the AP1000 reactivity control system, except the systems for boiling water reactors such as rod worth minimizers. Standby liquid control system is not a design feature of the AP1000.
	App. A.1.c	Conforms	
	App. A.1.d	Conforms	The functions of these systems are replaced by the passive residual heat removal heat exchanger of the passive core cooling system. Reactor core isolation cooling system is not a design feature of the AP1000.
	App. A.1.e	Conforms	
	App. A.1.f	Conforms	
	App. A.1.g	Conforms	
	App. A.1.h	Conforms	The characteristics of the AP1000 passive safety systems allow the support systems such as the cooling water systems, the heating, ventilating, and air conditioning and the ac power sources to be nonsafety-related and simplified. The capability of these systems is established by testing. Cold water interlocks are not a design feature of the AP1000.
	App. A.1.i	Conforms	The AP1000 has no secondary containment. Therefore, this guideline applies only to primary containment. The following systems or functions are not design features of the AP1000 and are therefore not tested: • Containment supplementary leak collection • Standby gas treatment • Secondary containment system • Containment annulus and cleanup • Bypass leakage tests on pressure suppression • Ice condenser systems • Containment penetration cooling

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
	App. A.1.j	Conforms	Recirculation flow control, traversing incore probes, automatic dispatching control systems and hotwell level control are not design features of the AP1000.
	App. A.1.k	Conforms	
	App. A.1.l	Conforms	Condenser off gas systems are not a design feature of the AP1000.
	App. A.1.m	Conforms	
	App. A.1.n	Conforms	Seal water, boron recovery, shield cooling, refueling water storage tank heating, and equipment for establishing and maintaining subatmospheric pressures are not design features of the AP1000.
	App. A.1.o	Conforms	
	App. A.2	Conforms	As applicable for pressurized water reactor.
	App. A.3	Conforms	As applicable for pressurized water reactor.
	App. A.4	Conforms	As applicable for pressurized water reactor. Compliance with A.4.t is met for the AP1000 with the provisions to perform the pre-operational tests of the passive RHR heat exchanger, as well as the low power tests described in DCD test abstracts 14.2.10.3.6, “Natural Circulation (First Plant Only)” and 14.2.10.3.7, “Passive Residual Heat Removal Heat Exchanger (First Plant Only).” Natural circulation testing of the reactor coolant system will be performed using the steam generators and the PRHR for the first plant only, in conformance with the AP1000 position on TMI item I.G.1 as outlined in Subsection 1.9.4.2.1.
	App. A.5	Conforms	
C.2 through C9		N/A	Section 14.2 describes the AP1000 plant initial test program. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.
General	Appendix B	N/A	Section 14.2 describes the AP1000 plant initial test program. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.
General	Appendix C	N/A	Section 14.2 describes the AP1000 plant initial test program. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.

Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

C2-C.9 Appendix B Appendix C		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.68.1, Rev. 1, 1/77 – Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.68.2, Rev. 1, 7/78 – Initial Test Program to Demonstrate Remote Shutdown Capability for Water Cooled Nuclear Power Plants			
General		Conforms	
Reg. Guide 1.68.3, (Task RS 709-4), 4/82 – Preoperational Testing of Instrument and Control Air Systems			
General	Regulatory Guide 1.68	Conforms	
Reg. Guide 1.69, Rev. 0, 12/73 – Concrete Radiation Shields for Nuclear Power Plants			
General	ANSI N101.6-1972	Exception	Regulatory Guide 1.69 endorses ANSI N101.6-1972 (Reference 18), which has been superseded by ANSI/ANS 6.4 1997 (Reference 19) and ACI 349-R01 (Reference 44). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.69.
Reg. Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition), Rev. 3, 11/78			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	
Conformance with programmatic and/or operational aspects is documented below.			
General		Exception	The format and content of the FSAR follow Regulatory Guide 1.206 and the AP1000 Design Control Document as required by Appendix D of 10 CFR Part 52.
Reg. Guide 1.71, Rev. 0, 12/73 and Rev. 1, 3/07 – Welder Qualification for Areas of Limited Accessibility			
Conformance of the design aspects with Revision 0 of the Regulatory Guide is as stated below in the DCD.			
General		Exception	Current practice does not require qualification or requalification of welders for areas of limited accessibility as described by this regulatory guide. The performance of required nondestructive evaluations helps to confirm weld quality. Limited accessibility qualification or requalification in excess of ASME Code, Section III or IX requirements is considered an unduly restrictive requirement for component fabrication, where the welders' physical position relative to the welds is controlled and does not present significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding is closely supervised.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			For field application, the type of qualification is considered on a case-by-case basis due to the great variety of circumstances encountered.

Conformance with Revision 1 of the Regulatory Guide during the operational phase (i.e., after the construction phase is completed per the DCD) is documented below.

General		Conforms	

Reg. Guide 1.72, Rev. 2, 11/78 – Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin

General	ASME Code CCN-155-1 (1792-1)	N/A	The AP1000 does not have safety-related spray pond piping components. Therefore, this regulatory guide is not applicable to the AP1000.

Reg. Guide 1.73, Rev. 0, 1/74 – Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

General	IEEE Std. 382-1972	Exception	Qualification of valve appurtenances, such as motor operators, solenoid valves, and limit switches, is in accordance with this regulatory guide. For safety-related motor-operated valves located inside containment, environmental qualification is performed in accordance with IEEE Standards 382-1996 (Reference 21) and 323-1974 (Reference 22).
C.1-6		Conforms	

Reg. Guide 1.74 – Withdrawn

Reg. Guide 1.75, Rev. 2, 9/78 and Rev. 3, 2/05 – Physical Independence of Electric Systems

Reg. Guide 1.75, Rev. 3, 2/05 - Criteria for Independence of Electrical Safety Systems

Conformance with Revision 2 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

General	IEEE Std. 384-1974	Exception	Regulatory Guide 1.75 endorses IEEE Std. 384-74 (Reference 23) which has been superseded by a later revision, IEEE Std. 384-81 (Reference 24). It is the later version that is used for the referenced purposes. This version has not yet been endorsed by a regulatory guide. The differences between the two revisions are not expected to contribute to conflicting design configurations because the jurisdiction of Regulatory Guide 1.75 with regard to the onsite ac power sources is limited. Specifically, since the AP1000 does not use safety-related ac power sources, the guidelines of Regulatory Guide 1.75 are applicable on a very limited basis to provide guidance on the Class 1E/non-Class 1E electrical separation and isolation for the following ac components that employ safety-related and nonsafety-related circuits: a) Class 1E dc battery chargers b) Reactor coolant pump switchgear c) Class 1E dc and UPS system regulating transformers.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			See subsection 8.3.2.4.2 for exceptions related to spacial separation between separation groups. Two fuses in series may be used as an isolation device for Class 1E and non-Class 1E isolation.

Reg. Guide 1.76, Rev. 0, 4/74 and Rev. 1, 3/07 – Design Basis Tornado for Nuclear Power Plants

Reg. Guide 1.76, Rev. 1, 3/07 – Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants

Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guide is completely within the scope of the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		Exception	The design basis tornado for the AP1000 is defined by the following parameters: Maximum wind speed: 300 mph Maximum rotational speed: 240 mph Translational speed: 60 mph (maximum) 5 mph (minimum) Radius to maximum wind from center of tornado: 150 feet Atmospheric pressure drop: 2.0 psi Rate of pressure drop: 1.2 psi/sec. Chapter 2 provides design basis tornado interface parameters.
C.2		Conforms	

Reg. Guide 1.77, Rev. 0, 5/74 – Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Exception	The guidance of Reg. Guide 1.183, "Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors" will be followed instead of Reg. Guide 1.77.

Reg. Guide 1.78, Rev. 1, 12/01 – Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

Conformance with the design aspects is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		N/A	This criterion is site-specific. It is not applicable to AP1000 design certification. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards.
C.2		N/A	This criterion is site-specific. Therefore, this is not applicable to AP1000 design certification. Subsection 2.1.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Subsection 6.4.7 defines the responsibility for addressing site-specific information amount and location of possible sources of toxic chemicals in or near the plant relative to control room habitability.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.1		N/A	This criterion is site-specific. It is not applicable to AP1000 design certification. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Subsection 6.4.7 defines the responsibility for addressing site-specific information amount and location of possible sources of toxic chemicals in or near the plant relative to control room habitability.
C.3.2		Conforms	
C.3.3		Exception	For AP1000 design certification, the atmospheric dispersion factors are not calculated (since there are no specific site data), but are selected so as to bound the majority of existing sites. Section 2.3 provides additional information.
C.3.4		Conforms	
C.4.1		N/A	This criterion is site-specific. It is not applicable to AP1000 design certification. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Subsection 6.4.7 defines the responsibility for addressing site-specific information amount and location of possible sources of toxic chemicals in or near the plant relative to control room habitability.
C.4.2		Conforms	
C.4.3		Conforms	
C.5		N/A	Not applicable to AP1000 design certification. Subsection 2.1.1 defines the responsibility for addressing site-specific information on identification of site location and description, exclusion area authority and control, and population distribution. Subsection 2.2.1 defines the responsibility for addressing site-specific information on identification of site-specific potential hazards. Section 13.3 defines the responsibility for addressing emergency planning.
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.79, Rev. 1, 9/75 – Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors			
General		Conforms	Preoperational testing is performed to test the functioning of the accumulators, core makeup tanks, passive residual heat removal heat exchanger, and automatic depressurization system, in a manner consistent with this regulatory guide. However, the AP1000 does not have high-head or low-head active safety-injection pumps. Therefore, many of the specific guidelines of this regulatory guide do not apply.
Reg. Guide 1.80 – Withdrawn			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.81, Rev. 1, 1/75 – Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plant			
General		N/A	The AP1000 is a single unit plant. When two or more AP1000s are located adjacent, electrical systems are not shared.
Reg. Guide 1.82, Rev. 3, 11/03 – Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident			
Conformance with the design aspects is as stated below in the DCD.			
General		Conforms	The AP1000 does not have high-head or low-head safety-injection pumps that need to take suction from the containment. The AP1000 does have a gravity-driven recirculation path that employs a containment recirculation arrangement. This containment recirculation can also be used to feed the normal residual heat removal pumps if they are available. The containment recirculation design conforms with the guidelines of this regulatory guide.
Conformance with programmatic and/or operational aspects is documented below.			
C.1.1.2		Conforms	
C.1.1.5		Conforms	
Reg. Guide 1.83, Rev. 1, 7/75 – Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes			
Conformance of the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 58324, 11/12/2009).			
General		Conforms	A program for in-service inspection of AP1000 steam generator tubing is established and performed in accordance with the guidelines of this regulatory guide. The baseline inspection will be performed in accordance with Regulatory Position C.3.a. Should the Combined License applicant request a baseline examination at the manufacturing facility, it will be performed in accordance with Regulatory Position C.3.a.
C.1		Conforms	
C.2		Exception	The specification of equipment in Regulatory Position C.2.c does not represent state-of-the-art equipment for gathering and storing eddy current information. When an eddy current inspection of an AP1000 steam generator is done in the manufacturing facility, more capable equipment than that specified in the regulatory guide is used. The steam generator design is compatible with robotic eddy current inspection equipment.
C.3		Exception	As noted in the comment on Criteria Section C.2, any eddy current inspection done in the manufacturing facility uses equipment of more current technology than that specified in Criteria Section C.2.
C.4-7		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.8		Exception	The only corrective action recognized by the regulatory guide is plugging of the tube to remove it from service. Sleeving of tubes is in many cases an acceptable repair method. The AP1000 steam generator design provides increased access to tubes to implement the sleeving repair method or other repair methods which may be developed.

Reg. Guide 1.84, Rev. 32, 6/03 and Rev. 33, 8/05 – Design and Fabrication Code Case Acceptability ASME Section III Division 1

Reg. Guide 1.84, Rev. 33, 8/05 – Design, Fabrication, and Materials Code Case Acceptability, ASME Section III

Conformance with Revision 32 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	The ASME Code Cases required for design certification are listed in Table 5.2-3. These cases are included in Regulatory Guide 1.84 or have been accepted by the US Nuclear Regulatory Commission staff as part of the review of AP1000.
C.1	ASME Code, Section III, Code Cases	Conforms	As applicable for pressurized water reactor.
C.2-5		Conforms	

Reg. Guide 1.85, Rev. 31, 5/99 – Materials Code Case Acceptability - ASME Code, Section III, Division 1

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	Refer to the discussion on Regulatory Guide 1.84. Subsequent to Revision 31 Reg. Guide 1.85 was combined with Reg. Guide 1.84. The guidance and conditions included in the previous revisions of Reg. Guide 1.85 remains valid.
C.1	ASME Code, Section III, Code Cases	Conforms	As applicable for pressurized water reactor.
C.2-5		Conforms	

Reg. Guide 1.86, Rev. 0, 6/74 – Termination of Operating Licenses for Nuclear Reactors

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.87, Rev. 1, 6/75 – Guidance for Construction of Class 1 Components in Elevated Temperature Reactors			
General		N/A	The AP1000 is not an elevated temperature reactor. See Section 1.2 for a general description of the plant and plant parameters. This regulatory guide is not applicable to the AP1000.
Reg. Guide 1.88 – Withdrawn			
Reg. Guide 1.89, (Task EE 042-2), Rev. 1, 6/84 – Environmental Qualification of Certain Electric Equipment Important to Safety For Nuclear Power Plants			
General	IEEE Std. 323-1974	Conforms	Conformance of AP1000 Class 1E equipment with 10 CFR 50.49, Reference 26 and this regulatory guide is demonstrated by an appropriate combination of the following: type testing, operating experience, qualification by analysis and ongoing qualification.
C.1	App. A	Conforms	As applicable for pressurized water reactor.
	App. B	Conforms	As applicable for pressurized water reactor.
	Regulatory Guide 1.97	Conforms	As applicable for pressurized water reactor.
C.2		Conforms	
C.2.a	App. C	Conforms	As applicable for pressurized water reactor.
C.2.b		Conforms	
C.2.c	App. D	Conforms	
C.2.c.1		Exception	Source term definition is discussed in the exceptions to Regulatory Guide 1.4, Positions C.1.a and C.1.b.
C.2.c.2		Exception	The fission product inventories in the fuel are discussed in the exception to Regulatory Guide 1.25, Position C.1.d.
C.2.c.3-8		Conforms	
C.2.d		Conforms	
C.3-6		Conforms	
C.7	App. E	Conforms	
Reg. Guide 1.90, Rev. 1, 8/77 – Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structures. Therefore, this regulatory guide is not applicable to the AP1000.
Reg. Guide 1.91, Rev. 1, 2/78 – Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites			
Conformance of the design aspects is as stated below in the DCD.			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	Onsite explosive materials conform to these guidelines. Offsite explosive materials are site-specific. See subsection 2.2.1 for information for identification of site-specific potential hazards. See subsection 19.58 for site-specific hazards evaluation.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.92, Rev. 1, 2/76 and Rev. 2, 07/06 – Combining Modal Responses and Spatial Components in Seismic Response Analysis

Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

C.1		Conforms	
C.2		Conforms	
C.3		Conforms	

Reg. Guide 1.93, Rev. 0, 12/74 – Availability of Electric Power Sources

C.1-2		N/A	The ac power sources are nonsafety-related. Therefore, these guidelines do not apply to the AP1000.
C.3		N/A	The function of the nonsafety-related diesel-generators for the AP1000 is to provide ac power for equipment and lighting during loss of offsite power but is not required for safe shutdown. Therefore, these guidelines do not apply to the AP1000.
C.4		N/A	See discussion on Criteria Section C.3.
C.5		Exception	AP1000 does not follow the guidance of C.5 for a 2-hour completion time for the limiting conditions of operation associated with the loss of one dc power subsystem. If one of the Class 1E dc electrical power subsystems is inoperable, the remaining Class 1E dc electrical power subsystems have the capacity to support a safe shutdown and to mitigate all design basis accidents, based on conservative analysis. Because of the passive system design and the use of fail-safe components, the remaining Class 1E dc electrical power subsystems have the capacity to support a safe shutdown and to mitigate most design basis accidents following a subsequent worst-case single failure. Also, with passive/fail-safe design, the risk associated with the loss of one Class 1E dc subsystem is similar to the loss of one ac supply for a conventional unit.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			AP1000 uses a 6-hour completion time for the limiting conditions of operation associated with the loss of one dc power subsystem to be consistent with the guidance in C.1 for a conventional plant with the loss of one ac source. The 6-hour completion time is reasonable based on engineering judgement balancing the risks of operation without one dc subsystem against the risks of a forced shutdown. Additionally, the completion time reflects a reasonable time to assess plant status; attempt to repair or replace, thus avoiding an unnecessary shutdown, and if necessary, prepare and effect an orderly and safe shutdown.
Reg. Guide 1.94, Rev. 1, 4/76 – Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.5-1974	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the quality assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.95 – Withdrawn			
Reg. Guide 1.96, Rev. 1, 6/76 – Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.97, Rev. 3, 5/83 and Rev. 4, 6/06 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident			
Reg. Guide 1.97, Rev. 4, 6/06 – Criteria For Accident Monitoring Instrumentation For Nuclear Power Plants			
Conformance with Revision 3 of the Regulatory Guide is as stated below in the DCD.			
General	ANS-4.5-1980	Conforms	The variables to be monitored are selected according to usage and need in the plant Emergency Response Guidelines. They are assigned design and qualification Category 1, 2, or 3 and classified as Type A, B, C, D, or E. Due to AP1000 specific design features, the selection of some plant-specific variables and their classifications and categories are different from those of this regulatory guide. For example, the use of the passive residual heat removal system as the safety grade heat sink allows steam generator wide range level to be category 2, not category 1 as specified in Regulatory Guide 1.97. The AP1000 has no Type A variables. See Section 7.5 for additional information.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			Since Category 3 instrumentation is not part of a safety-related system, it is not qualified to provide information when exposed to a post-accident adverse environment. Category 3 instrumentation is subject to servicing, testing, and calibration programs that are specified to maintain their capability. However, these programs are not in accordance with Regulatory Guide 1.118, which applies to safety-related systems.
C.1-2		Conforms	
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Exception	Portable equipment outside the DCD scope conforms to Revision 3 of this Regulatory Guide for consistency with DCD scope since Revision 4 indicates that partial implementation is not advised.
Reg. Guide 1.98, Rev. 0, 3/76 – Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor			
General		N/A	Applies to boiling water reactors only.
Reg. Guide 1.99, (Task ME 305-4), Rev. 2, 5/88 – Radiation Embrittlement of Reactor Vessel Materials			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
Reg. Guide 1.100, (Task EE 108-5), Rev. 2, 6/88 – Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants			
General	IEEE Std. 344-1987	Conforms	
Reg. Guide 1.101, Rev. 3, 8/92 and Rev. 5, 6/05 – Emergency Planning and Preparedness for Nuclear Power Reactors			
Reg. Guide 1.101, Rev. 5, 6/05 – Emergency Response Planning and Preparedness for Nuclear Power Reactors			
Conformance of the design aspects of Rev. 3 is as stated below in the DCD.			
General	NUREG-0654, FEMA-REP-1 NUMARC/NESP-007	Conforms	DCD Section13.3 defines the responsibility for addressing emergency planning. RG 1.101 (Revision 2) references NUREG-0654/ FEMA-REP-1 and item II.H, "Emergency Facilities and Equipment" of NUREG-0654/FEMA-REP-1, is applicable to the technical support center (TSC), operations support center (OSC), and the emergency operations facility (EOF) in the AP1000 design. Subsection 18.2.6 defines the responsibility for designing the EOF in accordance with the AP1000 human factors engineering program, including specification of its location. The AP1000 design conforms with the design criteria of item II.H that pertain to the TSC and the OSC.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with Rev. 5 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Exception	Rev. 5 is not applicable for this site. Rev. 3 and 4 are essentially the same except for endorsement of NEI 99-01 which is not directly applicable to the AP1000 passive design. The EP conforms to Rev. 3 and 4 with the exception that the EALs are written with necessary modifications to address the passive plant design.
General		Conforms	
Reg. Guide 1.102, Rev. 1, 9/76 – Flood Protection for Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
C.1		Conforms	
C.2	Regulatory Guide 1.59, C.2	Conforms	
C.3		Conforms	
Reg. Guide 1.103 – Withdrawn			
Reg. Guide 1.104 – Withdrawn			
Reg. Guide 1.105, Rev. 3, 12/99 – Instrument Setpoints for Safety-Related Systems			
General	ISA-S67.04-1994	Conforms	The technical specifications setpoints provide the margin from the nominal setpoint to the safety-analysis limit to account for drift when measured at the rack during periodic testing. The allowances between the technical specification limit and the safety limit include the following items: a) the inaccuracy of the instrument; b) process measurement accuracy; c) uncertainties in the calibration; and d) environmental effects on equipment accuracy caused by postulated or limiting postulated events (only for those systems required to mitigate consequences of an accident). The setpoints are chosen such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis. The instrumentation range is based on the span necessary for the associated function. Narrow range instruments are used where necessary. Instruments are selected based on expected environmental and accident conditions. The need for qualification testing is evaluated and justified on a channel-by-channel basis. Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices are not supplied.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			A plant-specific setpoint analysis must be performed to provide technical specification setpoints prior to plant startup. AP1000 conforms to the documentation requirements of the 1994 criteria.
Reg. Guide 1.106, Rev. 1, 3/77 – Thermal Overload Protection for Electric Motors on Motor-Operated Valves			
C.1	IEEE 279-1971, Sections 4.1, 4.2, 4.3, 4.4, 4.5, 4.10, and 4.13	Exception	Regulatory Guide 1.106 endorses IEEE Std. 279-1971 Reference 27, which has been replaced by IEEE STD 603-1991, (Reference 51). The AP1000 uses IEEE Std. 603-1991. This standard is endorsed by Regulatory Guide 1.153. The only safety-related electric motor-operated valves are dc.
C.2		Conforms	
Reg. Guide 1.107, Rev. 1, 2/77 – Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures			
General		N/A	The AP1000 does not have a concrete containment and does not use a prestressing tendon in the containment structure. Therefore, these guidelines are not applicable to the AP1000.
Reg. Guide 1.108 – Withdrawn			
Reg. Guide 1.109, Rev. 1, 10/77 – Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	This is applicable to the evaluation of specific sites. AP1000 design certification application evaluates how the AP1000 design is expected to compare with existing plants. This comparison is made based on the calculation of anticipated annual releases for the AP1000.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.110, Rev. 0, 3/76 – Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General	10 CFR 50, App. I, Section II.D	Exception	The disposal of effluents for the AP1000 is within the limits of Appendix I of 10 CFR 50, and the radwaste treatment systems have sufficient capacity to control effluents. The AP1000 approach to the design of radwaste systems is the result of a nuclear industry-sponsored program to optimize the radwaste systems design. A site-specific cost-benefit analysis is not required for sites that meet the site interface criteria.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance with Revision 0 this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.111, Rev. 1, 7/77 – Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. This is applicable to the evaluation of specific sites. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to design certification.
Reg. Guide 1.112, Rev. 0-R, 5/77 and Rev. 1, 3/07 – Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors			
Conformance of the design aspects with Revision 0-R of the Regulatory Guide is as stated below in the DCD.			
C.1	10 CFR 20.1(c); 10 CFR 50.34a; 10 CFR 50.36a; 10 CFR 50, App. I	Exception	The reference to 10 CFR 20.1(c) is no longer valid in the current version of 10 CFR Part 20.
C.2	NUREG-0016; NUREG-0017	Exception	Revision 1 of NUREG-0017 is used.
C.3		Conforms	
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General	ANSI 18.1-1999	Conforms	
Reg. Guide 1.113, Rev. 1, 4/77 – Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I			
Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. This is applicable to the evaluation of specific sites. Interface data are provided. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to design certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.114, Rev. 2, 5/89 – Guidance to Operators at the Controls and to Senior Operators in the Control Room of a Nuclear Power Unit			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 13.2 defines the responsibility for training and Section 13.5 defines the responsibility for procedures.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.115, Rev. 1, 1/77 – Protection Against Low-Trajectory Turbine Missiles			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	The SRP 3.5.1.3 issued in 1981 and Regulatory Guide 1.115, issued in 1977, provide criteria for protection against the effects of potential turbine missiles. Reference 28 issued in 1984 states that "the Nuclear Regulatory Commission staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability (P_2xP_3) to the missile generation probability (P_1) and, in the process, has attempted to integrate the various aspects of the issue into a single coherent evaluation." The AP1000 turbine is arranged in a radial orientation. The two low pressure turbines incorporate fully integral rotors. The turbine conforms with the criteria given in Reference 28 and WCAP-16650 (Reference 52).
Conformance with Rev. 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.116, Rev. O-R, 5/77 – Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems			
Conformance for DCD scope of work is as stated below in the DCD.			
General	ANSI N45.2.8-1975	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the Quality Assurance program.
Conformance for remaining scope is documented below.			
General		Exception	Quality assurance requirements utilize the more recently NRC endorsed NQA-1 in lieu of the identified outdated standards.
Reg. Guide 1.117, Rev. 1, 4/78 – Tornado Design Classification			
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
APPENDIX			
General		Conforms	For the AP1000, the leaktight integrity of the primary containment is maintained.
Reg. Guide 1.118, Rev. 3, 4/95 – Periodic Testing of Electric Power and Protection Systems			
General	IEEE Std. 338-1987	Conforms	Guidelines apply to safety-related dc power systems. Since the AP1000 has no safety-related ac power sources, the guidelines do not apply to the AP1000 ac power sources.
Reg. Guide 1.119 – Withdrawn			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.120, Rev. 1, 11/77 – Fire Protection Guidelines for Nuclear Power Plants			
General		Exception	The AP1000 design conforms with the Branch Technical Position CMEB 9.5.1 (Reference 32), which is attached to Section 9.5.1 of the Standard Review Plan, NUREG-0800 (Reference 33), as described in Section 9.5.1. Therefore, these guidelines are not applicable to the AP1000.
Reg. Guide 1.121, Rev. 0, 8/76 – Bases for Plugging Degraded Pressurizer Water Reactor Steam Generator Tubes			
General		Conforms	The only corrective action recognized by this regulatory guide is plugging of the tube to remove it from service. Sleeving of tubes is in many cases an acceptable repair method. The AP1000 steam generator design provides increased access to tubes to implement the sleeving repair method or other repair methods which may be developed.
C.1		Exception	Westinghouse interprets the term "unacceptable defects" to apply to those imperfections resulting from service induced mechanical or corrosion degradation of the tube walls which have penetrated to a depth or a length or a combination of both in excess of the plugging limit.
C.2.a.(1)		Exception	Westinghouse interprets this criterion to exclude the local region of the crack tip for Inconel tubing.
C.2.a.(2)		Exception	Tube minimum wall requirements are calculated in accordance with the following criteria. For normal plant operation, allowable membrane stress, Pm, is limited to a margin of three against exceeding the ultimate tensile strength of the material. As this regulatory guide constitutes an operating criterion, the allowable stress limits are based on expected lower bound material properties rather than ASME Code minimum values. Expected strength properties are obtained from statistical analysis of tensile test data of actual production tubing.
C.2.a(3)		Conforms	
C.2.a(4)		Exception	Refer to the discussion on Criteria Section C.2.a(2).
C.2.a.(5)-(6)		Conforms	
C.2..b.		Exception	In cases where sufficient inspection data exist to establish degradation allowance, the rate used is an average time-rate determined from the mean of the test data. Where requirements for minimum wall are markedly different for various areas of the tube bundle, such as the U-bend area versus straight length in Westinghouse designs, separate plugging limits are established to address the varying requirements in a manner which does not require unnecessary plugging of tubes.
C.3.a - c		Conforms	
C.3.d.(1)-(3)		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.e.(1)-(5)		Conforms	
C.3.e.(6)		Exception	Computer code names and references are supplied rather than actual codes.
C.3.e.(7)-(10)		Conforms	
C.3.f.(1)-(3)		Conforms	

Reg. Guide 1.122, Rev. 1, 2/78 – Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

C.1-3		Conforms	

Reg. Guide 1.123 – Withdrawn

Reg. Guide 1.124, Rev. 1, 1/78 and Rev. 2, 02/07 – Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports

Reg. Guide 1.124, Rev. 2, 02/07 – Service Limits and Loading Combinations for Class 1 Linear-Type Supports

Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD.

General	ASME Code, Section III Subsection NF	Conforms	Many of the items addressed in this regulatory guide have since been incorporated into later ASME Code, Section III Editions and Addenda. The design conforms to this regulatory guide with the following interpretations to maintain consistency with the ASME Code: 1. References to ASME Code, Section III, Subsection NF and Appendix XVII paragraphs are interpreted to be references to the corresponding paragraph in Subsection NF of the ASME Code. 2. References to ASME Code Case 1644 are interpreted to be references to the accepted versions of ASME Code Cases N-249 and N-71.
C.1		Conforms	
C.2	Code Case 1644	Conforms	Values of Su at these elevated temperatures are determined by test rather than via the method 2 as given by this regulatory position.
C.3	ASME Code, Section III, Appendix XVII	Conforms	
C.4	ASME Code, Section III, Appendix XVII-2110(b)	Exception	Paragraph B.1(b) of this regulatory guide states that "Allowable service limits for bolted connections are derived from tensile and shear stress limits and their nonlinear interaction. They also change with the size of the bolt. For this reason, the increases permitted by ASME Code, Section III, Subsections NF-3231.1, XVII-2110(a), and F-1370(a) are not directly applicable to allowable shear stresses and allowable stresses for bolts and bolted connections." This regulatory position also states that "This increase of level A or B service limits does not apply to limits for bolted connections and shear stresses."

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			As stated above, the increase in bolt allowable stress under emergency and faulted conditions is not permitted because the interaction between the allowable tension and shear stress in bolts is nonlinear, and the allowable tension and shear stress vary with the bolt size. The ASME Code, NF-3225, allows small increases in allowable stresses for Level B, Level C (previously termed "emergency"), Level D (previously termed "faulted"), and test conditions. The ASME Code rules are adequate since they satisfy the two objectives raised in the above quoted paragraph and will be used without further restrictions or justifications. This position is based on the following. 1. The interaction curve between the shear and tension stress in bolts is more closely represented by an ellipse and not a line. 2. The ASME Code specifies stress limits for bolts and represents this tension/shear relationship as a non-linear interaction equation (ellipse). This interaction equation has a built-in safety factor that ranges between two and three (depending on whether the bolt load is predominately tension or shear) based on the actual strength of the bolt as determined by test. See Reference 34. 3. This regulatory position states that "Any increases of limits for shear stresses above 1.5 times the ASME Code, Level A service limits should be justified." Concerning allowable shear stresses, the AP1000 uses the ASME Code, Subsection NF requirements. The ASME Code shear stress limits (NF-3300 and Tables NF-3523.2 and NF-3623.2-1) generally meet the guidance provided by this regulatory position that shear stresses be maintained within 1.5 times Level A service limits. This limit may be exceeded slightly in some limited cases such as Level D limits for SA-36 material, in which case the NF shear stress limit of .42 Su is 13 percent greater than this regulatory guide limit of 1.5 x .4 Fy. Su and Fy are the material tensile and yield strengths, respectively.
C.5.a	ASME Code, Section III,	Exception	The AP1000 evaluates supports to current Level B stress limits for the upset load combination. Effects of constraint of free-end displacements are included in the upset loading condition while no further increase in allowable stresses over and above the Level B limits is permitted. The operating basis earthquake has been eliminated from the AP1000 design basis.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.5.b-c	ASME Code, Section III, Subsection NF-3262.3, Appendix XVII-4200, Appendix XVII-4110(a)	Conforms	The operating basis earthquake has been eliminated from the AP1000 design basis.
C.6	ASME Code, Section III, Appendix XVII-2000, 2110(a) Subsection NF 2362.3, Appendix XVII-4200, 4110(a), II-1400	Conforms	
C.7.a	ASME Code, Section III, Appendix XVII-2000, and F-1370(a)	Conforms	
C.7.b		Exception	The AP1000 uses the provisions of the ASME Code, Section III, Appendix F to determine faulted condition allowable loads for supports designed by the load rating method. The method described in this regulatory position is conservative and inconsistent with the remainder of the faulted stress limits.
C.7.c	ASME Code, Section III Appendix XVII-4200, and F-1370(b)	Conforms	
C.7.d	ASME Code, Section III, II-1400, and F-1370(b)	Conforms	
C.8		Exception	The reduction of allowable stresses to no greater than Level B limits for support structures in those systems with normal safety-related functions occurring during emergency or faulted plant conditions is overly conservative for components which are not required to mechanically function (inactive components). For service Level C and D loading conditions, Level C limits are used for the support of active components.

Conformance with Revision 2 for programmatic and/or operational aspects of the Regulatory Guide is as stated below. This guidance is completely within the scope of the DCD.

Reg. Guide 1.125, Rev. 1, 10/78 – Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	
C.1-6		Conforms	

Reg. Guide 1.126, Rev. 1, 3/78 – An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

C.1-2		Exception	Westinghouse uses the densification model described in the Nuclear Regulatory Commission-approved topical reports WCAP-10851-A and WCAP-13589-A. Westinghouse conforms to the methodology of this regulatory guide when implementation of the methodology is required.
C.3-4		Conforms	
C.5		Conforms	

Reg. Guide 1.127, Rev. 1, 3/78 – Inspection of Water-Control Structures Associated With Nuclear Power Plants

General		N/A	The AP1000 does not have water-control structures. Therefore, this guideline is not applicable to the AP1000. Subsection 2.5.6 defines the responsibility for embankments and dams.

Reg. Guide 1.128, Rev. 1, 10/78 and Rev. 2, 2/07– Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

Reg. Guide 1.128, Rev. 2, 2/07 – Installation Design and Installation of Vented Lead-Acid Storage Batteries for Nuclear Power Plants

Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

General	IEEE Std. 484-1975	Exception	Regulatory Guide 1.128 endorses IEEE Std. 484-75 (Reference 36) which has been superseded by IEEE Std. 484-1996 (Reference 37). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 1.128.

Reg. Guide 1.129, Rev. 1, 2/78 and Rev. 2, 2/07 – Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

Reg. Guide 1.129, Rev. 2, 2/07 – Maintenance, Testing, and Replacement of Vented Lead-Acid Storage Batteries for Nuclear Power Plants

Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD.

General	IEEE Std. 450-1975	N/A	Not applicable to AP1000 design certification. Section 8.3 defines the responsibility for battery testing.

Conformance with Revision 2 of the Regulatory Guide for programmatic and/or operational aspects is as stated below.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	IEEE Std. 450-2002	Exception	Approved Generic Technical Specifications are based on IEEE Std 450-1995.

Reg. Guide 1.130, Rev. 1, 10/78 – Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports

Reg. Guide 1.130, Rev. 2, 3/07 - Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Supports

Conformance with Revision 1 of the Regulatory Guide is as stated below in the DCD.This guidance is completely within the scope of the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General	ASME Code, Section III Subsection NF	Exception	Many of the items addressed in this regulatory guide have since been incorporated into later ASME Code, Section III, Editions and Addenda. The plant design conforms to this regulatory guide with the following interpretations to maintain consistency with the ASME Code: 1. Regulatory guide references to ASME Code, Section III, Subsection NF and Appendix XVII paragraphs are interpreted to be references to the corresponding paragraph in the ASME Code, Subsection NF. 2. Regulatory guide references to ASME Code Case 1644 are interpreted to be references to the latest acceptable versions of the ASME Code Case N-249 and N-71. Paragraph B.1 of this regulatory guide states that "Allowable stress limits for bolted connections are derived on a different basis that varies with the size of the bolt. For this reason, the increases permitted by NF-3222.3 and F-1323.1(a) of ASME Code, Section III are not directly applicable to bolts and bolted connections." The ASME Code rules are adequate for bolt design without further restriction or justification. The maximum stress increase factor allowed is 25 percent for the Service Level D condition, and the stress allowables do not vary with bolt size. The AP1000 takes exception to the guideline stated in Paragraph B.5 of this regulatory guide, that systems whose safety-related function occurs during emergency or faulted plant conditions should meet Level A limits. The reduction of allowable stresses to no greater than Level A limits for support structures in those systems with normal safety-related functions occurring during emergency or faulted plant conditions is overly conservative for components which are not required to mechanically function (inactive components). For service, Level C and D loading conditions, Level C limits are used for the support of active components.
C.1		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2	Code Case 1644	Conforms	
C.3		Exception	Design margins of two for flat plates and three for shells are unnecessarily restrictive for normal, upset, and emergency conditions, as well as inconsistent with ASME Code requirements. For these loading conditions, the AP1000 limits the allowable buckling strength to 1/2 of the critical buckling strength.
C.4	ASME Code, Section III, NF-3221.1, NF-3221.2, NF-3222, NF-3262.2, II-1400	Exception	This regulatory position recommends that design stress limits be used in conjunction with a loading combination that includes operating basis earthquake. The ASME Code rules (in which Level B stress limits are typically used for the upset load combination) provide a conservative design basis. The AP1000 uses the latest rules (as of 4/2001) without further restriction or justification. The operating basis earthquake has been eliminated from the AP1000 design basis. Refer also to the discussion on Criteria Section C.3.
C.5.a	ASME Code, Section III, NF-3224	Exception	Refer to the discussion on Criteria Section C.3.
C.5.b-c	ASME Code, Section III, NF-3262.2, II-1400	Conforms	
C.6.a	ASME Code, Section III, F-1323.1(a), F-1370(c)	Conforms	
C.6.b	ASME Code, Section III, NF-3262.1	Exception	The limit based on the test load given in this regulatory position is overly conservative and is inconsistent with ASME Code requirements. The AP1000 uses the provisions of the ASME Code, Section III, Appendix F to determine faulted condition allowable loads for supports designed by the load rating method.
C.6.c		Conforms	
C.6.d	ASME Code Section III, F-1324, F-1370(c)	Conforms	
C.7		Conforms	
Reg. Guide 1.131, Rev. 0, 8/77 – Qualification Tests of Electric Cables, Field Splices and Connections for Light-Water-Cooled Nuclear Power Plants			
General	IEEE Std. 383-1974	Conforms	The insulating and jacketing material for electrical cables are selected to meet the fire and flame test requirements of IEEE Standard 1202 or IEEE Standard 383 excluding the option to use the alternate flame source, oil or burlap.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1-14		Conforms	

Reg. Guide 1.132, Rev. 1, 3/79 and Rev. 2, 10/03 – Site Investigations for Foundations of Nuclear Power Plants

Conformance of the design aspects for Revision 1 of this Regulatory Guide is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. Section 2.5 defines the responsibility for site investigations and the site specific information related to basic geological, seismological, and geotechnical engineering of the site.

Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

4.3		Exception	Only borings used for down-hole geophysical logging were surveyed for deviations (Section 4.3.1.2). Only color photographs of rock cores were taken-no soil sample photographs (Section 4.3.2). One or more borings for each major structure was not continuously sampled (Section 4.3.2.2).

Reg. Guide 1.133, Rev. 1, 5/81 – Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors

Conformance of the design aspects is as stated below in the DCD.

General		Conforms	A digital metal impact monitoring system (DMIMS) monitors the reactor coolant system for the presence of loose metallic parts. The system actuates audible and visual alarms if a signal exceeds the preset alarm level. The digital metal impact monitoring system is not a Class 1E system. It serves as a diagnostic aid to detect loose parts in the reactor coolant system before damage occurs. Database calibration is made prior to plant startup and the capability for periodic online channel checks and channel functional tests are incorporated in the digital metal impact monitoring system design.
C.1.a-i		Conforms	
C.2		Conforms	
C.3.a		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for development of procedures.
C.3.b		Conforms	
C.4-5		Conforms	
C.6		N/A	Not applicable to AP1000 design certification. Reporting Requirements associated with the technical specification are identified in Tech. Spec. Section 5.6. DCD subsection 1.1.1 defines the responsibility to finalize the technical specification.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2b		Conforms	Procedures are addressed in Section 13.5
C.3a		Conforms	Procedures are addressed in Section 13.5
C.4g		Conforms	Procedures are addressed in Section 13.5
C.4h		Conforms	Procedures are addressed in Section 13.5
C.4i		Conforms	ALARA is addressed in Chapter 12 and Section 13.5
C.4j		Conforms	Training is addressed in Section 13.2
C.6		Exception	Regulatory Guide 1.16 has been withdrawn. Event reporting is performed in accordance with 10 CFR 50.72 and 50.73 utilizing the guidance of NUREG-1022

Reg. Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses

Reg. Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Licensed Personnel at Nuclear Power Plants

Conformance of the design aspects is as stated below in the DCD.

General		N/A	Not applicable to AP1000 plant design certification. Section 13.5 defines the responsibility for administrative procedures.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.135, Rev. 0, 9/77 – Normal Water Level and Discharge at Nuclear Power Plants

Conformance of the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (74 FR 39349, 08/06/2009).

General		Conforms	The normal ground and surface water levels and surface water discharges for the AP1000 are determined using the postulated site parameters. Chapter 2 provides additional information.

Reg. Guide 1.136, Rev. 2, 6/81 – Materials, Construction, and Testing of Concrete Containments

General		N/A	The AP1000 does not have a concrete containment. Therefore, this guideline is not applicable to the AP1000.

Reg. Guide 1.137, Rev. 1, 10/79 – Fuel-Oil Systems for Standby Diesel Generators

General		N/A	The AP1000 diesel-generators and the associated fuel-oil systems are nonsafety-related. Therefore, this guideline is not applicable to the AP1000.

Reg. Guide 1.138, Rev. 0, 4/78 and Rev. 2, 12/03 – Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants

Reg. Guide 1.138, Rev. 2, 12/03 – Laboratory Investigations of Soils and rocks for Engineering Analysis and Design of Nuclear Power Plants

Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Not applicable to AP1000 design certification. Subsection 2.5.4.6.2 defines the responsibility to establish the properties of the foundation soils including laboratory investigations of underlying materials.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.139, Rev. 0, 5/78 – Guidance for Residual Heat Removal			
Conformance with the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (73 FR 32750, 06/10/2008).			
C.1.a		Exception	The AP1000 employs a full pressure/temperature passive residual heat removal heat exchanger that is automatically actuated. The heat exchanger does not rely on ac or dc power. Fail-safe valves are used to manually isolate the heat exchanger. When these valves are open, the reactor coolant pumps (if available) or natural circulation produces flow through the heat exchangers. The heat exchanger is safety-related, seismically designed, and can tolerate single active failure. Continued operation of the heat exchanger brings the reactor coolant system pressure and temperature down to the point where the stress in the reactor coolant system pressure boundary is low. This temperature is about 400°F which allows an reactor coolant system pressure of 1/10 of design (250 psia).
C.1.b		Conforms	
C.1.c		Exception	See the comment on Criteria Section C.1.a. The passive residual heat removal heat exchanger does not rely on pumps, ac power sources, air systems, or water cooling systems.
C.2		Conforms	
C.3		Conforms	
C.4		N/A	The passive residual heat removal heat exchanger does not have pumps. Therefore, this guideline is not applicable to the AP1000.
C.5	IEEE Std. 338; Regulatory Guide 1.22; Regulatory Guide 1.68	Conforms	IEEE Std. 338-1987 (Reference 31) is current standard.
C.6		N/A	The passive residual heat removal heat exchanger provides this function. As a result, the auxiliary feedwater system has been replaced by a nonsafety-related startup feedwater system. Therefore, this guideline is not applicable to the AP1000.
C.7	Regulatory Guide 1.33	N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for the quality assurance program.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.140, Rev. 2, 06/01 – Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup System in Light-Water-Cooled Nuclear Power Plants			
C.1		Conforms	Regulatory Guide 1.140 endorses ASME Standard N509-1989 (Reference 39), ASME Standard N510-1989 (Reference 40), and ASME AG-1-1997 (Reference 38). The AP1000 uses the latest version of the industry standards (as of 3/2002).
C.2.1-2.4		Conforms	
C.3.1-3.2		Conforms	
C.3.3	ERDA 76-21, Section 5.6; ASME N509-1989 Section 4.9	Conforms	
C.3.4	Regulatory Guide 8.8	Conforms	
C.3.5		Conforms	
C.3.6	ASME AG-1-1997Article SA-4500	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure is designed to meet at least SMACNA design standards.
	ASME AG-1-1997,Section TA	Conforms	
C.4.1	ASME AG-1-1997,Section FB	Conforms	
C.4.2	ASME AG-1-1997, Section CA	Conforms	
C.4.3	ASME AG-1-1997, Section FC, and Section TA	Conforms	
C.4.4	ASME AG-1-1997, Section FG	Conforms	
C.4.5	ERDA 76-21, Section 4.4; ASME AG-1a-2000, Section HA	Conforms	
C.4.6	ASME N509-1989, Section 5.6; ASME AG-1a-2000, Section HA	Conforms	
C.4.7	ASME AG-1-1997, Section CA	Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.4.8	ASME AG-1-1997, Section FD or FE	Conforms	
C.4.9	ASME AG-1-1997, Section FD and FE or, Section FF	Conforms	
C.4.10	ASME AG-1-1997 Section SA	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure is designed to meet at least SMACNA design standards.
C.4.11		Conforms	
C.4.12	ASME AG-1-1997 Section DA	Conforms	
C.4.13	ASME AG-1-1997, Section BA and SA	Conforms	
C.5.1	ERDA 76-21, Section 2.3.8; ASME AG-1a-2000, Section HA	Conforms	
C.5.2		Conforms	
C.6	ASME N510-1989	Conforms	
C.7	ANSI N509-1989	Conforms	
Reg. Guide 1.141, Rev. 0, 4/78 – Containment Isolation Provisions for Fluid Systems			
General	ANSI N271-1976	Exception	Regulatory Guide 1.141 endorses ANSI N271-1976 (Reference 41) that has been superseded by ANS 56.2-1984 (Reference 42). The AP1000 uses the latest version of industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.141. Containment isolation for AP1000 fluid systems conforms to Reference 42 with the following exceptions and/or clarifications.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			ANS 56.2-1984, Section 3.6.3 states that "remote manual closure of isolation valves on engineered safeguards features or engineered safeguards features-related systems is acceptable when provisions are made to detect possible failure of the fluid lines inside and outside containment." The AP1000 engineered safeguards features are designed to avoid the need for transport of post-accident fluids outside of containment and thus avoid the concern associated with remote manual isolation of engineered safety feature lines. Non-engineered safety feature lines capable of providing engineered safety feature functions are provided with the capability for remote manual isolation. The nonsafety-related normal residual heat removal system has provisions to isolate on high containment radiation. Radiation monitors are provided inside containment to assess continuation of the functions.
C.1		Conforms	
C.2		Conforms	
C.3		Conforms	
C.4	ANSI N271-1976, Section 4.4.8, Section 3.5 or 3.6.7	Conforms	
C.5	Regulatory Guide 1.7 & 1.89	Conforms	
C.6	ANSI N271-1976, Section 3.5 or 3.7	Conforms	
Reg. Guide 1.142, Rev. 2, 11/01 – Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)			
General	ACI 349-97	Exception	Regulatory Guide 1.142 endorses ACI 349-97 (Reference 43) that has been superseded by ACI 349-01 (Reference 44). The AP1000 uses the latest version of industry standards as of October 2001). This version is not endorsed by a regulatory guide but its use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.142. In the following evaluation of conformance, the design is shown as conforming since the requirements of ACI-2001 are similar to those of ACI349-1997.
C.1		N/A	The compartments within the containment are not designed to be leaktight since they must communicate with one another to preclude subcompartment pressurization. Therefore, this guideline is not applicable to the AP1000.
C.2	ANS 6.4-1997	Conforms	
C.3	ANSI/ACI 349-97	Conforms	
C.4		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.5		Conforms	
C.6	ACI 349-97, Section 9.2.1	Conforms	
C.7		Conforms	
C.8		Conforms	
C.9		N/A	The AP1000 does not include a pressure-suppression containment. Therefore, this guideline is not applicable to the AP1000.
C.10	ACI 349-97, App. C	Conforms	
C.11		Conforms	
C.12	ACI 349-97, App. A	Conforms	
C.13		Conforms	
C.14		N/A	The AP1000 containment vessel is steel.
C.15		Conforms	The provisions in Section 11.6 of ACI 349-01 are the same as those in ACI 318-99 (Reference 46).

Reg. Guide 1.143, Rev. 2, 11/01 - Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light Water Cooled Nuclear Power Plants

Conformance of the design aspects is as stated below in the DCD.

General			The AP1000 Radwaste Building provides space to store dry active waste and space for mobile waste processing systems and equipment and three liquid waste monitor tanks which contain liquid effluents which have been processed and are acceptable for release to the environment (within the requirements of classification RW-IIC of Regulatory Guide 1.143). It does not contain installed systems and components used to process, store, or handle gaseous or liquid waste.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions	
C.1.1.1	Regulatory Guide 1.143, Table 1	Exception	Components in the liquid radwaste systems are designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the exception of the use of the 2013 Edition of ASME, Section VIII, Div. 1, instead of the 1998 Edition (with 1999 Addenda) that is referenced in Appendix A of Regulatory Guide 1.143. This version is not endorsed by a regulatory guide, but it's use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.143. Equipment classifications and design codes are listed in Table 3.2-3. Pressure vessels are designed and built according to ASME, Section VIII, Div. 1. Atmospheric tanks are per API 650 or ASME, Section III and heat exchangers to ASME Section VIII, Div. 1 and TEMA (for shell and tube). Piping and valves are per ANSI B31.1 except the containment penetrations and isolation valves are per ASME, Section III, Class 2. Pumps are according to manufacturer's standards.	RN-15-005
C.1.1.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code, Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ANSI B31.1. Pump materials are provided according to manufacturer's standards.	
C.1.1.3		Conforms	The auxiliary building that contains the liquid radwaste system with the exception of three monitor tanks is designed to seismic Category I criteria. The seismic Category I structure will retain the maximum liquid inventory of the system. The lowest level of the auxiliary building, elevation 66′-6″, contains the liquid radwaste system effluent holdup tanks, a monitor tank, and chemical waste tank within a common flood zone. This flood zone has watertight floors and walls. The enclosed volume within this flood zone is sufficient to contain the contents of the system. The effluent holdup tank rooms, the monitor tank room, and chemical waste tank room each have one or two floor drains that lead to the sump. Tank overflows or spills will be collected in the auxiliary building sump. The sump is automatically pumped to a waste holdup tank. The waste holdup tank rooms each have two normally isolated floor drains that are connected to a sump pump. The waste holdup tank overflows or spills will be manually pumped to a waste holdup tank. Two liquid radwaste system monitor tanks are above at elevations 92'-6" and 107'-2". Overflows or spills from these monitor tanks drain by gravity down through the drain system to a waste holdup tank. The seismic Category I criteria exceed the operating basis earthquake required by regulatory position C.6 of Regulatory Guide 1.143.	RN-14-075 RN-14-075 RN-14-075 RN-13-013

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
			The radwaste building that contains three liquid radwaste monitor tanks is designed to the Uniform Building Code. The basemat and curbed structure will retain the maximum liquid inventory of any of the three monitor tanks. Tank overflows or spills will be collected by the radioactive waste drain system and routed to the auxiliary building sump. The sump is automatically pumped to a waste holdup tank. The Uniform Building Code design of the radwaste building meets the requirements of regulatory position C.6 of Regulatory Guide 1.143 for structures classified as RW-IIC.
C.1.2.1		Conforms	Atmospheric tanks in the liquid radwaste system have level sensors, transmitters, and alarms. Local alarm is not provided because the tanks are located in shielded areas that are not normally occupied by people.
C.1.2.2		Conforms	Tank overflows, drains and sample lines that may contain radioactive water are routed to the liquid radwaste system for processing.
C.1.2.3		Conforms	Please refer to the discussion of conformance to C.1.1.3, which addresses the provisions in the buildings that contain radioactive waste to contain any spills.
C.1.2.4		Conforms	Please refer to the discussion of conformance to C.1.1.3, which addresses the provisions in the building that contain radioactive waste to contain any spills. The measures to prevent contamination of clean areas via ductwork due to leakage are as follows: the annex building general area HVAC system normally maintains the personnel areas at a slightly positive pressure with respect to adjacent areas, including the auxiliary building. Interfaces with the adjacent buildings are limited to doorways, airlocks, and ductwork. Ductwork connecting the annex building and adjacent areas consists entirely of supply air ductwork handling outside air for the fuel handling area, health physics area, containment purge supply, and main control room. The main control room supplemental air filtration unit is in the HVAC equipment room; however, this unit has no radioactive material during normal plant operation.
C.1.2.5		Conforms	This guideline does not apply because the liquid radwaste treatment system has no outdoor tanks. No other outside tanks store radioactive fluids.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.2.1	Regulatory Guide 1.143, Table 1	Exception	Components in the gaseous radwaste systems are designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143 with the exception of the use of the 2013 Edition of ASME, Section VIII, Div. 1, instead of the 1998 Edition (with 1999 Addenda) that is referenced in Appendix A of Regulatory Guide 1.143. This version is not endorsed by a regulatory guide, but it's use should not result in deviations from the design philosophy otherwise stated in Regulatory Guide 1.143. Heat exchangers are designed and built according to ASME, Section VIII, Div. 1 and TEMA (for shell and tube). Piping and valves are per ANSI B31.1. Pumps are according to manufacturer's standards. RN-15-005
C.2.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ANSI B31.1. Pump materials are provided according to manufacturer's standards.
C.2.3		Conforms	The guard bed and the delay beds, including supports, in the gaseous radwaste system are designed for seismic loads according to the requirements of Regulatory Guide 1.143. These are the only AP1000 components used to store or delay the release of gaseous radioactive waste. The beds are located in the seismic Category I auxiliary building at elevation 66′-6″. Seismic loads for this equipment will be established using one-half of the safe shutdown earthquake (SSE) floor response spectra. The loads due to this seismic response spectra are equivalent or greater than those due to an operating basis earthquake (OBE). Other equipment and supports will be designed in accordance with the codes indicated in Table 3.2-3.
C.3		Conforms	The regulatory guidance applies to the AP1000 solid waste processing system except for components and subsystems used to solidify or concentrate liquid waste. The AP1000 solid waste processing system does not have these components/subsystems. These functions are provided by contractors who process these wastes using mobile systems.
C.3.1	Regulatory Guide 1.143, Table 1, Reg. Pos. 3.2 and 3.3	Conforms	The solid radwaste system is designed and tested to the requirements set forth in the codes and standards listed in Table 1 of Regulatory Guide 1.143. The spent resin tanks are designed and tested in accordance with ASME Code, Section VIII, Div. 1. Piping and valves are designed and tested according to ANSI B31.1. The pumps are designed to manufacturers' standards and tested in accordance with the Hydraulic Institute standards.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3.2	ASME Code, Section II	Conforms	Materials, except elastomers for gaskets, seals, seats, diaphragms, and packing, are provided in accordance with the ASME Code, Section II when the ASME Code is the design and fabrication standard. Piping and valves materials are per ASTM specifications consistent with ANSI B31.1. Pump materials are provided according to manufacturer's standards.
C.3.3		Conforms	The seismic Category I auxiliary building will retain the maximum liquid and spent resin inventory of the spent resin tanks. The seismic Category I criteria exceed the operating basis earthquake required by regulatory position C.6 of Regulatory Guide 1.143.
C.3.4		Conforms	The equipment and components used to collect, process, and store solid radwaste are nonseismic as permitted by this paragraph.
C.4.1	Regulatory Guide 8.8	Conforms	Design Control Document section 12, "Radiation Protection," discusses the measures taken to maintain the radiation exposure to personnel as low as reasonably achievable.
C.4.2		Conforms	The quality assurance program for design, fabrication, procurement, and installation of radwaste systems is in accordance with the overall quality assurance program described in Chapter 17, which meets the requirements of Regulatory Guide 1.143, position C.7.
C.4.3	ASME Code, Section IX	Conforms	Pressure-containing components in the radwaste systems are of welded construction to the maximum practical extent. Flanged joints and quick connect fittings are used only where maintenance or operational requirements indicate that they are preferable. Screwed connections are not used except for some instrumentation and vents and drains where welded construction is not suitable. Process lines are 1 in. or larger. Butt welds are used in process lines, which contain radioactive fluids. Nonconsumable backing rings are not used in process piping welds. Process pipe welding is performed as required by ANSI B31.1. Component welding is performed as required by the applicable construction code.
C.4.4		Conforms	Hydrostatic testing is performed as required by the applicable construction codes.
C.4.5		Conforms	In-service testing of the containment penetrations and isolation valves is performed as described in Design Control Document subsection 3.9.6. Other tests, on nonsafety equipment, are performed on an item-by-item basis as judged necessary to confirm proper operation of the systems.
C.5	10 CFR Part 20		

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.5.1		Conforms	Systems containing enough activity to be possibly classified as RW-IIa are located in the Auxiliary Building. The Auxiliary Building is seismic Category I.
C.5.2		Conforms	
C.5.3	10 CFR Part 71 Appendix A	Conforms	AP1000 systems and components that store or process radioactive waste are located in the Auxiliary Building.
C.5.4	10 CFR Part 71 Appendix A	Conforms	AP1000 systems and components that store or process radioactive waste are located in the Auxiliary Building and radwaste building.
C6.1.1	Table 2	Conforms	
C6.1.2	Table 3	Conforms	
C6.1.3	Table 4	Conforms	
C6.1.4	Table 1 & 4	Conforms	
C6.2.1	UBC 1997, ASCE 7-95	Conforms	The Radwaste Building is designed to UBC-1997 and ASCE 7-98.
C6.2.2		Conforms	Shield structures, if used, will comply with Regulatory Guide 1.143, position C.6.2.
C.7	ANSI/ANS55.6-1993	Conforms	The quality assurance program for design, fabrication, procurement, and installation of radwaste systems is in accordance with the overall quality assurance program described in Chapter 17, which meets the requirements of Regulatory Guide 1.143, position C.6.

Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.144 – Withdrawn

Reg. Guide 1.145, Rev. 1, 11/82 (Revised 2/83 to correct page 1.145-7) – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

Conformance of the design aspects of the Regulatory Guide is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. The atmospheric dispersion factors for use in determining potential accident consequences are selected to be representative of existing nuclear power plant sites and to bound the majority of them. Chapter 2 provides the interface criteria.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.146 – Withdrawn

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.147, Rev. 12, 5/99 and Rev. 14, 8/05 – Inservice Inspection Code Case Acceptability ASME Section XI Division 1			
Conformance with Revision 12 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated in the DCD.			
General	ASME Code, Section XI	Conforms	
Conformance with Revision 14 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.148, (Task SC 704-5), Rev. 0, 3/81 – Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants			
General	ANSI N27.8.1-1975	Conforms	
C.1.a		Conforms	
C.1.b	ASME Code, Section III, NCA-3250	Conforms	
C.1.c(1)	ASME Code, Section III, NCA-3252(a)(b)	Conforms	
C.1.c(2)		Conforms	
C.1.c(3)	ASME Code, Section III, NCA-3256	Conforms	
C.1.d		Conforms	
C.1.e	Regulatory Guide 1.84, Regulatory Guide 1.85	Conforms	
C.2.a-d		Conforms	
Reg. Guide 1.149, Rev. 2, 4/96 and Rev. 3, 10/01 – Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations			
Reg. Guide 1.149, Rev. 3, 10/01 – Nuclear Power Plant Simulation Facilities for Use in Operator Training and License Examinations			
Conformance with Revision 2 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.2.1 defines the responsibility to develop and implement training programs for plant personnel. These training programs will address the scope of licensing examinations.
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		Conforms	During cold licensing, training is conducted using a simulator with limited scope in accordance with Appendix D of ANSI/ANS-3.5-1998. Operator Licensing examinations are conducted on a simulator meeting the applicable requirements of ANSI/ANS-3.5-1998.
Reg. Guide 1.150, Rev. 1, 2/83 – Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations			
Conformance with the design aspects is as stated below in the DCD. The programmatic and/or operational aspects are not applicable since this guidance was withdrawn by NRC (73 FR 7766, 02/11/2008).			
General		Conforms	The reactor vessel design includes features that permit conformance to the pre-service and in-service inspection of this regulatory guide. Guidelines for such features as positioning of welds, vessel contour, and weld surface preparation are included.
Reg. Guide 1.151, (Task 1C 126-5), Rev. 0, 7/83 – Instrument Sensing Lines			
General	ISA-S67.02	Conforms	This regulatory guide addresses the use of ISA-S67.02, “Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants,” 1980, to provide a basis for the design and installation of instrument sensing lines. It establishes the applicable ASME code requirements and boundaries for the design and installation of instrument sensing line in accordance with the appropriate parts of ASME Code, Section III. Where there is a conflict between the ISA standard and the ASME Code, Section III, the ASME Code requirements are followed. Because this instrument configuration (permanently sealed, fluid-filled tubing) is excluded from the scope of Section III, the tubing is designed and fabricated to ASME B31.1. Industry standard ISA-S67.02 reiterates and clarifies the practice of controlling documents such as interface requirements and regulations. The AP1000 uses the Piping and Instrumentation Diagram as the approved document to designate the safety classification system boundaries.
C.1		Conforms	
C.2	ASME Code, Class 2 SC I	Conforms	Safety-related instrumentation has safety class pressure boundaries, including the sensing line, valves, and instrumentation sensors. The pressure boundary is the same safety class as the equipment to which it is connected. The AP1000 credits design features such as flow restrictors and diaphragms as class separation. For that portion of a sensing line from the ASME Code, Class 1 piping or vessel out to the class separation device, ISA-S67.02 includes the ASME Code, Class 1 requirement. For that portion of the sensing line from the class separation device to the sensor is designated as ASME Code, Class 2 requirement. The AP1000 has no sensing lines penetrating the containment barrier.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.3	ASME Code, Class 3 SC I	Exception	The guidelines apply to the AP1000 sensing lines, except the sensing lines that are connected to some ASME Code, Class 3 components that do not have a seismic design requirement. Sensing lines from these components are not ASME Code, seismic Category I.
C.4-6		Conforms	

Reg. Guide 1.152, (Task 1C 127-5), Rev. 1, 1/96 and Rev. 2, 1/06 – Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants

Reg. Guide 1.152, Rev. 2, 1/06 – Criteria for Use of Computers in Safety Systems of Nuclear Power Plants

Conformance of the design aspects with Revision 1 of the Regulatory Guide is as stated below in the DCD.

General	ANSI/ IEEE-ANS-7-4.3.2 -1993	Conforms	

Conformance with Revision 2 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Exception	The Cyber Security Program is based on March 2009 revisions of the 10 CFR 73.54 regulations in lieu of Revision 2 of this Regulatory Guide.

Reg. Guide 1.153, Rev. 1, 6/96 – Criteria for Power, Instrumentation, and Control Portions of Safety Systems

General	IEEE Std. 603-1991 including January 30, 1995 Correction sheet	Conforms	

Reg. Guide 1.154, Rev. 0, 1/87 – Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors

Conformance of the design aspects with the Regulatory Guide is as stated below in the DCD.

General		N/A	See Section 5.3 for additional information on pressurized thermal shock. Subsection 5.3.6 defines the responsibility to document reactor vessel materials and material evaluation.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.155, (Task SI 501-4), Rev. 0, 8/88 – Station Blackout

General	10 CFR 50.63	N/A	There are no safety-related ac power sources. Therefore, this regulatory guide is not applicable to the AP1000.

Reg. Guide 1.156, (Task EE 404-4), Rev. 0, 11/87 – Environmental Qualification of Connection Assemblies for Nuclear Power Plants

General	IEEE Std. 572-1985	Conforms	

Reg. Guide 1.157, (Task RS 701-4), Rev. 0, 5/89 – Best-Estimate Calculations of Emergency Core Cooling System Performance

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
C.1		Conforms	
C.2		Conforms	
C.3.1	10 CFR 50, App. A	Conforms	
C.3.2-12		Conforms	
C.3.13-14		N/A	Applies to boiling water reactors only.
C.3.15-16		Conforms	
C.4.1	10 CFR 50.46(a)(1)(i)	Conforms	
C.4.2-4		Conforms	
C.4.5		Conforms	
Reg. Guide 1.158, (Task EE 006-5), Rev. 0, 2/89 – Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants			
General	IEEE Std. 535-1986	Conforms	
Reg. Guide 1.159, Rev. 0, 8/90 and Rev. 1, 10/03 – Assuring the Availability of Funds for Decommissioning Nuclear Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.160, Rev. 2, 3/97 – Monitoring the Effectiveness of Maintenance at Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for a Plant Maintenance Program. The COL applicant is responsible for assessing conformance to Regulatory Guide 1.160 of monitoring the effectiveness of maintenance.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.161, Rev. 0, 6/95 – Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy Less Than 50 Ft-Lb			
General		N/A	The design and material specification for the reactor vessel do not permit a Charpy value less than 50 ft.-lb. Subsection 5.3.6.4 defines the responsibility for reactor vessel materials properties verification.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.162, Rev. 0, 2/96 – Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.163, Rev. 0, 9/95 – Performance Based Containment Leak-Test Program			
Conformance of the design aspects is as stated below in the DCD.			
1	NEI94-01 ANSI/ ANS 56.8-1994	Conforms	
2	NEI Section 11.3.2	Conforms	
3	NEI 94-01 Section 9.2.1 NEI 94-01 Section 10.2.3.3	Conforms	
Conformance with Revision 0 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.165, Rev. 0, 3/97 – Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Subsection 2.5.2.1 defines the responsibility to address site-specific information related to the vibratory ground motion aspects.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		N/A	Seismic analysis performed in accordance with Regulatory Guide 1.208.
Reg. Guide 1.166, Rev. 0, 3/97 – Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.167, Rev. 0, 3/97 – Restart of a Nuclear Power Plant Shut Down by a Seismic Event			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
Conformance with Rev. 0 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.168, Rev. 0, 9/97 and Rev. 1, 2/04 – Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
Conformance of the design aspects with Rev. 0 of the Regulatory Guide is as stated below in the DCD.			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Conformance with Rev. 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.169, Rev. 0, 9/97 – Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.170, Rev. 0, 9/97 – Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.171, Rev. 0, 9/97 – Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.172, Rev. 0, 9/97 – Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.173, Rev. 0, 9/97 – Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants			
General		Conforms	See Chapter 7 for a discussion of the instrumentation and control software program.
Reg. Guide 1.174, Rev. 0, 7/98 and Rev. 1, 11/02 – An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis			
Conformance of the design aspects for Revision 0 of this Regulatory Guide is as stated below in the DCD.			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

This Regulatory Guide is outside the scope of the FSAR.

Reg. Guide 1.175, Rev. 0, 7/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing

Reg. Guide 1.175, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing

Conformance of the design aspects is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Inservice testing of ASME Section III components is discussed in subsection 3.9.6.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

Risk-informed inservice testing is not being utilized for this plant.

Reg. Guide 1.176, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance

General		N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Quality assurance is discussed in Chapter 17.

Reg. Guide 1.177, Rev. 0, 8/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications

Conformance of the design aspects is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. The standard AP1000 Technical Specification is provided in DCD Chapter 16.

Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.178, Rev. 0, 9/98 and Rev. 1, 9/03 – An Approach for Plant-Specific Risk-informed Decisionmaking Inservice Inspection of Piping

Conformance of the design aspects is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. The AP1000 is a standardized design. Inservice inspection is discussed in DCD Subsection 5.2.4 and Section 6.6.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

Risk-informed inservice inspection is not being utilized for this plant.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.179, Rev. 0, 9/99 – Standard Format and Content of License Termination Plans for Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. The AP 1000 is a standardized design. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.180, Rev. 1, 10/03 – Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	See Appendix 3D for a discussion of the EMI/RFI qualification.
Conformance with Rev. 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	Exclusion zones are established through administrative controls to prohibit the activation of portable EMI/RFI emitters (e.g., welders and transceivers) in areas where safety-related I&C systems are installed.
Reg. Guide 1.181, Rev. 0, 9/99 – Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.182, Rev. 0, 5/00 – Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for a Plant Maintenance Program.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.183, Rev. 0, 7/00 – Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.184, Rev. 0, 7/00 – Decommissioning of Nuclear Power Reactors			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.185, Rev. 0, 7/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.186, Rev. 0, 12/00 – Guidance and Examples of Identifying 10 CFR 50.2 Design Bases			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.187, Rev. 0, 11/00 – Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.188, Rev. 1, 9/05 – Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 1.189, Rev. 0, 4/01 and Rev. 1, 3/07 – Fire Protection for Operating Nuclear Power Plants			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions

Reg. Guide 1.189, Rev. 1, 3/07 – Fire Protection for Nuclear Power Plants

Conformance with Revision 0 of the Regulatory Guide is documented below in the DCD. Conformance of the design aspects is as stated in the DCD.

General		N/A	Subsection 9.5.1 describes the AP1000 Fire Protection System. Subsection 9.5.1.8 defines the responsibility for completing a fire protection program.

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.190, Rev. 0, 4/01 – Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence

General		N/A	Subsection 5.3.2.6 describes the calculational and dosimetry methods for determining pressure vessel neutron fluence for the AP1000 subsection 5.3.6.4 defines the responsibility for reactor vessel materials properties verification.

Reg. Guide 1.191, Rev. 0, 5/01 – Fire Protection Program for Nuclear Power Plants During Decommissioning and Permanent Shutdown

Conformance with the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

		N/A	This Regulatory Guide is outside the scope of the FSAR.

Reg. Guide 1.192, Rev. 0, 6/03 – Operation and Maintenance Code Case Acceptability, ASME OM Code

Conformance with the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.193, Rev. 1, 8/05– ASME Code Cases Not Approved for Use

Conformance with Revision 0 of the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.194, Rev. 0, 6/03 – Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

Conformance with Revision 0 of the Regulatory Guide is documented in the DCD. Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions

Reg. Guide 1.195, Rev. 0, 5/03 – Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors

This Regulatory Guide is not applicable to the AP1000 certified design.

Reg. Guide 1.196, Rev. 1, 1/07 – Control Room Habitability at Light-Water Nuclear Power Reactors

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below. This Regulatory Guide is not applicable to the AP1000 certified design.

General		Conforms	

Reg. Guide 1.197, Rev. 0, 5/03 – Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors

Conformance with the design aspects is as stated in the DCD.

General		Conforms	The design of the AP1000 control room and associated HVAC systems facilitates testing to demonstrate control room envelope integrity.
C.1.1-2.6		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for procedures. Section 14.4 describes the responsibilities required to perform the AP1000 plant initial test program.

Conformance with programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.198, Rev. 0, 11/03 – Procedures and Criteria for Assessing Seismic Soil Liquefaction at Nuclear Power Plant Sites

Conformance with programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.199, Rev. 0, 11/03 – Anchoring Components and Structural Supports in Concrete

Conformance with Revision 0 of the Regulatory Guide is as stated below in the DCD. This guidance is completely within the scope of the DCD.

C.1 – C.7		Conforms	

Reg. Guide 1.200, Rev. 1, 1/07 – An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities

Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 1.201, Rev. 1, 5/06 – Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance

This Regulatory Guide is not applicable to the AP1000 certified design.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.202, Rev. 0, 2/05 – Standard Format and Content of Decommissioning Cost Estimates for Nuclear Power Reactors			
This Regulatory Guide is outside the scope of the FSAR.			
Reg. Guide 1.203, Rev. 0, 12/05 – Transient and Accident Analysis Methods			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Reg. Guide 1.204, Rev. 0, 11/05 – Guidelines for Lightning Protection of Nuclear Power Plants			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 1.205, Rev. 0, 5/06 – Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Reg. Guide 1.206, Rev. 0, 6/07 – Combined License Applications for Nuclear Power Plants (LWR Edition)			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General	Format	Conforms	
General	Content	Exception	Exceptions to content are identified in Table 1.9-202.
Reg. Guide 1.207, Rev. 0, 3/07 – Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors			
This Regulatory Guide is not applicable to the AP1000 certified design.			
Reg. Guide 1.208, Rev. 0, 3/07 – A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion			
Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Appendix C, Section C.3		Exception 3.4	Exception is taken to requirement that 0.05 and 0.95 fractile hazard curves be provided. These were not run. Hazard curves were run at 0.15 and 0.85th percentile instead of 0.16 and 84th as they are very close approximations (+/- 1 sigma).
Reg. Guide 1.209, Rev. 0, 3/07 – Guidelines for Environmental Qualification of Safety-Related Computer-Based Instrumentation and Control Systems in Nuclear Power Plants			
This Regulatory Guide is not applicable to the AP1000 certified design.			
DIVISION 4 – Environmental and Siting			
Reg. Guide 4.7 Rev. 2, 4/98 – General Site Suitability Criteria for Nuclear Power Stations			
Conformance with the design aspects is as stated below in the DCD.			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		N/A	Chapter 2 defines the site-related parameters for which the AP1000 plant is designed. These interface parameters envelop most potential sites in the United States. The guidelines in this regulatory guide are site-specific. Chapter 2 defines the responsibility for determining general site suitability.

Conformance with programmatic and/or operational aspects is documented below.

General		Conforms	

Reg. Guide 4.15 Rev. 2, 7/07 – Quality Assurance for Radiological Monitoring Programs (Inception through Normal Operations to License Termination) – Effluent Streams and the Environment

		Exception	The Guidance of Rev. 1, February 1979 will be followed as per the justification provided in FSAR Subsection 11.5.3.

DIVISION 5 – Materials and Plant Protection

The plant-specific physical security plans include no substantive deviations from the NRC-endorsed template in NEI 03-12, Rev. 6. Therefore, the degree of conformance with Division 5 regulatory guides for the Physical Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan is consistent with the degree of conformance of NEI 03-12, Rev. 6.

Reg. Guide 5.9 Rev. 2, 12/83 – Specifications for Ge (Li) Spectroscopy Systems for Material Protection Measurements Part 1: Data Acquisition Systems

Reg. Guide 5.9, Rev. 2, 12/83 – Guidelines for Germanium Spectroscopy Systems for Measurement of Special Nuclear Material

Conformance of the design aspects is as stated below in the DCD.

General		N/A	Not applicable to AP1000 design certification. Laboratory Equipment is not included in the AP1000 design. Subsection 1.9.1.5 defines the responsibility for Regulatory Guides not applicable to Design Certification.

Conformance with programmatic and/or operational aspects is documented below.

		N/A	This Regulatory Guide is outside the scope of the FSAR.

Reg. Guide 5.12, Rev. 0, 11/73 – General Use of Locks in the Protection and Controls of Facilities and Special Nuclear Materials

Conformance of the design aspects is as stated below in the DCD.

C.1	UL-768	Conforms	
C.2	FF-P-110F	Conforms	
C.3	UL-437	Conforms	
C.4	FF-P-001480 (GSA FSS)	Conforms	
C.5-8		Conforms	

Conformance with programmatic and/or operational aspects is documented below.

		N/A	This Regulatory Guide is outside the scope of the FSAR.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 5.65, Rev. 0, 9/86 – Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls			
Conformance of the design aspects is as stated below in the DCD.			
General		Conforms	The AP1000 provides for physical protection of the vital area. Identification of the protected and vital areas and an outline of the physical protection system are presented in the AP1000 Security Design Report.
Conformance with programmatic and/or operational aspects is documented below.			
		N/A	This Regulatory Guide is outside the scope of the FSAR.
Reg. Guide 5.71, Rev. 0, 1/10 – Cyber Security Programs for Nuclear Facilities			
Conformance with regulatory positions C.1 through C.5 of Regulatory Guide 5.71, Rev. 0, is as stated in the Cyber Security Plan (CSP), with exceptions to the guidance as noted in Attachment A of the CSP.			
DIVISION 8 – Occupational Health			
Reg. Guide 8.2, Rev. 0, 2/73 – Guide for Administrative Practices in Radiation Monitoring			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Section 13.5 defines the responsibility for administrative procedures. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
Conformance with programmatic and/or operational aspects is documented below.			
General	10 CFR Part 20; ANSI 13.2-1969	Exception	The reference to 10 CFR 20.401 is no longer valid in the current version of 10 CFR Part 20. ANSI N13.2-1969 was reaffirmed in 1988.
Reg. Guide 8.4, Rev. 0, 2/73 - Direct-Reading and Indirect-Reading Pocket Dosimeters			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General	10 CFR Part 20 ANSI N13.5-1972	Exception	The reference to 10 CFR 20.202 (a) and 20.401 is no longer valid in the current version of 10 CFR Part 20. ANSI N13.5-1972 was reaffirmed in 1989. The two performance criteria specified in Regulatory Guide 8.4 (accuracy and leakage) for these devices are met using acceptance standards in ANSI N322-1997 “American National Standard Inspection, Test, Construction, and Performance Requirements for Direct Reading Electrostatic/Electroscope Type Dosimeters”.
Reg. Guide 8.5, Rev. 1, 3/81 - Criticality and Other Interior Evacuation Signals			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 8.6, Rev. 0, 5/73 - Standard Test Procedure for Geiger-Muller Counters			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Exception	Instrument calibration program is based upon criteria in ANSI N323A-1997 (with 2004 Correction Sheet) “Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments.” The ANSI 42.3-1969 Standard is no longer recognized as sufficient for calibration of modern instruments.
Reg. Guide 8.7, Rev. 2, 11/05 - Instructions for Recording and Reporting Occupational Radiation Dose Data			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.8, Rev. 3, 6/78 – Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable			
Conformance of the design aspects is as stated below in the DCD.			
1		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
1.a-c	Regulatory Guide 1.8	N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
1.d		Conforms	
2	ANSI N237-1976	Exception	Regulatory Guide 8.8 endorses ANSI-N237-1976 (Reference 49), which has been superseded by ANSI 18.1-1999 (Reference 50). The AP1000 uses the latest version of the industry standards (as of 4/2001). This version is not endorsed by a regulatory guide but its use should not result in deviation from the design philosophy otherwise stated in Regulatory Guide 8.8.
2.a	10 CFR 20-203	Conforms	
2.b-g		Conforms	
2.h	ANS N197 ANS 55.1 ANS N19	Conforms	ANS-55.1-1992-R2000 is Current Version
2.i		Conforms	
3		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
4.a		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
4.b-d		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
4.e		Conforms	
Conformance with Revision 3 of this Regulatory Guide for programmatic and/or operational aspects is documented below.			
C.1		Conforms	
C.3.a		Conforms	
C.3.b		Exception	Regulatory Guide 1.16 C.1.b.(3) data is no longer reported. Reporting per C.1.b(2) is also no longer required.
C.3.c		Conforms	
C.4.b-C.4.d	ANSI Z-88.2, Regulatory Guide 8.15, NUREG-0041	Conforms	Conformance is with the latest revision of NUREG-0041.
Reg. Guide 8.9, Rev. 1, 7/93 – Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program			
Conformance with the design aspects is as stated in the DCD. Conformance with programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.10, Rev. 1-R, 5/77 – Operating Philosophy For Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable			
Conformance of the design aspects is as stated below in the DCD.			
General		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.12 – Withdrawn			
Reg. Guide 8.13, Rev. 3, 6/99 – Instruction Concerning Prenatal Radiation Exposure			
Conformance of the design aspects is as stated below in the DCD.			
General	10 CFR 19.12	N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA. Section 13.5 defines the responsibility for administrative procedures.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.14 – Withdrawn			

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
Reg. Guide 8.15, Rev. 1, 10/99 – Acceptable Programs for Respiratory Protection			
Conformance of the design aspects is as stated below in the DCD.			
General	10 CFR 20.103	N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA. See Section 12.3 for information on radiation protection design features. See Section 12.5 for information on health physics facilities. Section 13.5 defines the responsibility for administrative procedures.
Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.19, Rev. 1, 6/79 – Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates			
General		Conforms	
Reg. Guide 8.27, Rev. 0, 3/81 – Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.28, Rev. 0, 8/81 – Audible-Alarm Dosimeters			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General	ANSI N13.27-1981	Conforms	
Reg. Guide 8.29, Rev. 1, 2/96 – Instruction Concerning Risks from Occupational Radiation Exposure			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.34, Rev. 0, 7/92 – Monitoring Criteria and Methods To Calculate Occupational Radiation Doses			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	
Reg. Guide 8.35, Rev. 0, 6/92 – Planned Special Exposures			
Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.			
General		Conforms	

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions

Reg. Guide 8.36, Rev. 0, 7/92 – Radiation Dose to the Embryo/Fetus

Conformance of the design aspects is as stated in the DCD. Conformance with this Regulatory Guide for programmatic and/or operational aspects is documented below.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	

Reg. Guide 8.38, Rev. 0, 6/93 and Rev. 1, 5/06 – Control of Access to High and Very High Radiation Areas in Nuclear Plants

Conformance of the design aspects for Revision 0 of this Regulatory Guide is as stated below in the DCD.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	

Conformance with Revision 1 of this Regulatory Guide for programmatic and/or operational aspects is documented below.

Criteria Section	Referenced Criteria	AP1000/ FSAR Position	Clarification/Summary Description of Exceptions
General		Conforms	

Note a. Above stated general alternatives regarding the use of previous revisions of the Regulatory Guide for design aspects as stated in the DCD is provided to preserve the finality of the certified design. Further, each stated conformance with the programmatic and/or operational aspects is only to the extent that a design change or departure from the approved DCD is not required to implement those programmatic and/or operational aspects. As the operational and programmatic aspects become more fully defined (for example, during the preparation, approval, or initial implementation of plant procedures), there exists a potential that a conflict could be identified between the design as certified in the DCD and the programmatic and/or operational aspects of the guidance. In such cases, the design certification (rule) becomes the controlling factor, and the design conformance to the Regulatory Guide is per the revision stated in the DCD.

Note b. A "Criteria Section" entry of "General" indicates a scope for the conformance statement of "all regulatory guide positions related to programmatic and/or operational aspects." Thus, an associated conformance statement of "Conforms" indicates that the applicant "complies with all regulatory guide positions related to programmatic and/or operational aspects."

1A.1 References

1. Sofer, et al., "Accident Source Terms for Light Water Nuclear Power Plants." NUREG-1465, February 1995.

2. Not used.

3. Not used.

4. Not used.

5. IEEE Std. 308-1974, IEEE Standard Criteria for Class 1E Power Systems for Nuclear Power Generating Stations, 1974.

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

7. Not used.

8. Not used.

9. WCAP-8577, "The Application of Pre-Heat Temperature After Welding of Pressure Vessel Steels," September 1975.

10. IEEE 379-72, IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems, 1972.

11. IEEE 379-2000, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems, 2000.

12. ASCE 4-86, Seismic Analysis of Safety-Related Nuclear Structures, September 1986.

13. RESAR/SP90, Westinghouse Advanced Pressurized Water Reactor, Paragraph 3.7.1.3, Damping Values, January 1989.

14. Design & Evaluation Guideline for Department of Energy Facilities Subjected to Natural Phenomena Hazards, Lawrence Livermore National Laboratory, UCRL-15910, R. P. Kennedy et al., May 1989.

15. IEEE 741-1986, IEEE Standard Criteria for Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations, 1986.

16. IEEE Std. 741-1997, IEEE Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations, 1997.

17. IEEE 317-1983, IEEE Standard for Electric Penetrations Assemblies in Containment Structures for Nuclear Power Generating Stations, 1983.

18. ANSI N101.6-1972, Atomic Industry Facility Design, Construction, and Operation Criteria, December 22, 1972.

19. ANSI/ANS 6.4 1997, Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants.

20. Not used.

21. IEEE 382-1996, IEEE Standard for Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants, 1996.

22. IEEE 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, 1974.

23. IEEE 384-74, IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits, 1974.

24. IEEE 384-81, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits, 1981.

25. ANS 5.4-1982, American National Standard Method for Calculating the Fractional Release of Volatile Fission Products From Oxide Fuel, 1982.

26. Title 10 Code of Federal Regulations Part 50, 50.49, Environmental qualification of electric equipment important to safety for nuclear power plants.

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

28. NRC Safety Evaluation Report, letter from B. D. Liaw to J. A. Martin, December 27, 1984.

29. Not used.

30. Not used.

31. IEEE Std. 338-1987, IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Stations Safety Systems, 1987.

32. BTP CMEB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants."

33. SRP (NUREG-0800) Section 9.5.1 "Fire Protection Program," Revision 3, July 1986.

34. Guide to Design Criteria for Bolted and Riveted Joints, Fisher and Struik, copyright 1974, John Wiley and Sons.

35. Not used.

36. IEEE 484-75, IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations, 1975.

37. IEEE 484-1996, IEEE Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations, 1996.

38. ASME AG-1-1997, "Code on Nuclear Air and Gas Treatment," 1997.

39. ASME Code N509-1989, "Nuclear Power Plant Air Cleaning Units and Components," 1989.

40. ASME Code N510-1989, "Testing of Nuclear Air Cleaning Systems," 1989.

41. ANSI N271-1976, "Containment Isolation Provisions for Fluid Systems," 1976.

42. ANS 56.2-1984, Containment Isolation Provisions for Fluid Systems, 1984.

43. ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures," 1997.

44. ACI 349-01, "Code Requirements for Nuclear Safety Related Concrete Structures," 2001.

45. Not used.

46. ACI 318-99, "Building Code Requirements for Reinforced Concrete," 1999.

47. Not used.

48. Not used.

49. ANSI N237-1976, Source Term Specification, 1976.

50. ANSI/ANS 18.1-1999, Radioactive Source Term for Normal Operation of Light Water Reactors, 1999.

51. IEEE Std. 603-1991, IEEE Standard Criteria for Power Generating Stations, 1991.

52. WCAP-16650-P (Proprietary) and WCAP-16650-NP (Non-Proprietary), "Analysis of the Probability of Generation of Missiles for Fully Integral Low Pressure Turbines," Revision 0, February 2007.

Appendix 1B Not Used

DCD Appendix 1B is not incorporated into this document. Rather, the severe accident mitigation design alternatives are addressed in the Environmental Report. As indicated in 10 CFR Part 52, Appendix D, Section III.B, "...the evaluation of severe accident mitigation design alternatives in appendix 1B of the generic DCD are not part of this appendix."

Appendix 1AA Not Used

FSAR Appendix 1AA is merged with DCD Appendix 1A and therefore Appendix 1AA is not used.