APPENDIX 1A

CONFORMANCE WITH REGULATORY GUIDES

Criteria Section	Referenced Criteria	AP1000 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 Guide 1.2 – Withdrawn			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 Steamline 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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.
D.5		N/A	See comment on Criteria Section D.2.
Reg. Guide 1.7, Rev. 2, 11/78 – Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident			
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	
Reg. Guide 1.8, Rev. 3, 5/00 – Qualification and Training of Personnel for Nuclear Power Plants			
General		N/A	Not applicable to AP1000 design certification. Section 13.2.1 defines the responsibility for the training program for plant personnel.
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.10 – Withdrawn			
Reg. Guide 1.11, Rev. 0, 3/71 – Instrument Lines Penetrating Primary Reactor Containment			
General		Conforms	The AP1000 has no instrument lines penetrating primary reactor containment.
C.1.a		Conforms	
C.1.b	10 CFR 100	Conforms	
C.1.c-e		Conforms	
C.2		Conforms	
E.1		Conforms	
E.2		N/A	This section applies only to plants for which a notice of hearing on application for construction permit was published between January 5, 1967, and December 30, 1969. Therefore, it is not applicable to the AP1000.
E.3		N/A	This section applies only to plants for which a notice of hearing on application for construction permit was published on or before December 30, 1966. Therefore, it is not applicable to the AP1000.
Reg. Guide 1.12, Rev. 2, 3/97 – Instrumentation for Earthquakes			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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	
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.
Reg. Guide 1.13, Rev. 1, 12/75 – Spent Fuel Storage Facility Design Basis			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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.
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

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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.
Reg. Guide 1.17 – Withdrawn			
Reg. Guide 1.18 – Withdrawn			
Reg. Guide 1.19 – Withdrawn			
Reg. Guide 1.20, Rev. 2, 5/76 – Comprehensive Vibration Assessment Program For Reactor Internals During Preoperational and Initial Startup Testing			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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.
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			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			The calibration of effluent monitoring systems is performed according to written plant procedures. Section 11.5.7 defines the responsibility for the radiation monitoring program. Section 13.5.1 defines the responsibility for the plant procedure preparation.
C.1		N/A	Not applicable to AP1000 design certification. Section 11.5.7 defines the responsibility for the radiation monitoring program.
C.2		Conforms	
C.3-14		N/A	Not applicable to AP1000 design certification. Section 11.5.7 defines the responsibility for the radiation monitoring program.
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	
Reg. Guide 1.23, Second Proposed Rev. 1, 4/86 – Onsite Meteorological Programs			
General		Conforms	The onsite meteorological measurement program is site-specific and will be defined as indicated in DCD 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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 – Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containment Components of Nuclear Power Plants			
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. 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

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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.

Criteria Section	Referenced Criteria	AP1000 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
C.4		Conforms	Technical Specifications exist for the passive containment cooling system and air inlet and air baffle.
Reg. Guide 1.28, Rev. 3, 8/85 – Quality Assurance Program Requirements (Design and Construction)			
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	
Reg. Guide 1.29, Rev. 3, 9/78 – Seismic Design Classification			
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 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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.30, Rev. 0, 8/72 – Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment			
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.
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			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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.
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.
Reg. Guide 1.33, Rev. 2, 2/78 – Quality Assurance Program Requirements (Operation)			
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.
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 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 – Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants			
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.
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			
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.
Reg. Guide 1.39, Rev. 2, 9/77 – Housekeeping Requirements for Water-Cooled Nuclear Power Plants			
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 Position	Clarification/Summary Description of Exceptions
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.44, Rev. 0, 5/73 – Control of the Use of Sensitized Stainless Steel			
C.1-2		Conforms	
C.3		Exception	Product forms with simple shapes are not corrosion tested provided they are water quenched following solution anneal. Cast metal and weld metal containing more than five percent delta ferrite also are not corrosion tested as they are not susceptible to sensitization.
C.4		Exception	Welding operations necessarily result in the weld heat affected zone (HAZ) being between the temperature of 800°F to 1500°F.
C.5		Conforms	
C.6		Exception	Extensive testing has been completed and documentation is available such that an intergranular corrosion test is not required for AP1000 welding procedures.
Reg. Guide 1.45, Rev. 0, 5/73 – Reactor Coolant Pressure Boundary Leakage Detection Systems			
C.1		Conforms	
C.2		Conforms	
C.3		Exception	The AP1000 reactor coolant pressure boundary leakage detection methods are selected and designed in accordance with the guidelines of this regulatory guide. No credit is taken for airborne particulate radiation measurement in quantifying the leak rate.
C.4		Conforms	
C.5		Conforms	
C.6		Exception	Airborne particulate radioactivity monitoring is not used to determine reactor coolant pressure boundary leakage.
C.7		Conforms	
C.8		Conforms	
C.9		Conforms	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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. 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

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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 pressurized water heat transfer 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. 2, 3/78 – 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			
General		N/A	There are no ESF atmosphere cleanup systems for the AP1000. The AP1000 does not require engineered safety feature atmosphere cleanup systems to meet limits on doses offsite or onsite.
Reg. Guide 1.53, Rev. 0, 6/73 – Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.54, Rev. 1, 7/00 – Service Level I, II and III Protective Coatings Applied to Nuclear Power Plants			
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 DCD subsection 6.1.2.1.6. Subsection 6.1.3 defines the responsibility for the coating program.
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 – Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components			
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			
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.
Reg. Guide 1.60, Rev. 1, 12/73 – Design Response Spectra for Seismic Design of Nuclear Power Plants			
C.1		Conforms	
C.2		Conforms	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.61, Rev. 0, 10/73 – Damping Values for Seismic Design of Nuclear Power Plants			
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	
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

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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			
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.
Reg. Guide 1.66 – Withdrawn			
Reg. Guide 1.67 – Withdrawn			
Reg. Guide 1.68, Rev. 2, 8/78 – Initial Test Program for Water-Cooled Nuclear Power Plants			
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.)

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
	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 and containment annulus vacuum breaker • 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
	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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
	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.

Criteria Section	Referenced Criteria	AP1000 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			
General		Conforms	
Reg. Guide 1.71, Rev. 0, 12/73 – Welder Qualification for Areas of Limited Accessibility			
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. For field application, the type of qualification is considered on a case-by-case basis due to the great variety of circumstances encountered.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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 – Physical Independence of Electric Systems			
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. 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.76, Rev. 0, 4/74 – Design Basis Tornado for Nuclear Power Plants			
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			
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			
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 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. DCD Section 13.3 defines the responsibility for addressing emergency planning.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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			
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. 2, 5/96 – Water Sources for Long Term Recirculation Cooling Following a Loss-of-Coolant Accident			
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.
Reg. Guide 1.83, Rev. 1, 7/75 – Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes			
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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	
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. 31, 5/99 – Design and Fabrication Code Case Acceptability ASME Section III Division 1			
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			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
C.2-5		Conforms	
Reg. Guide 1.86, Rev. 0, 6/74 – Termination of Operating Licenses for Nuclear Reactors			
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.
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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			
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.
Reg. Guide 1.92, Rev. 1, 2/76 – Combining Modal Responses and Spatial Components in Seismic Response Analysis			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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. 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			
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.
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.97, Rev. 3, 5/83 – Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident			
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. 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	
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.101, Rev. 3, 8/92 – Emergency Planning and Preparedness for Nuclear Power Reactors			
General	NUREG-0654, FEMA-REP-1 NUMARC/NESP-007	Conforms	DCD Section 13.3 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.
Reg. Guide 1.102, Rev. 1, 9/76 – Flood Protection for Nuclear Power Plants			
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

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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. 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			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.110, Rev. 0, 3/76 – Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors			
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.
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			
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. O-R, 5/77 – Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors			
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	
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			
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.
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			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.115, Rev. 1, 1/77 – Protection Against Low-Trajectory Turbine Missiles			
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).
Reg. Guide 1.116, Rev. O-R, 5/77 – Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems			
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.
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 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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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	
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 – Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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." 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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.
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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.
Reg. Guide 1.125, Rev. 1, 10/78 – Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.128, Rev. 1, 10/78 – Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants			
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 – Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants			
General	IEEE Std. 450-1975	N/A	Not applicable to AP1000 design certification. Section 8.3 defines the responsibility for battery testing.
Reg. Guide 1.130, Rev. 1, 10/78 – Service Limits and Loading Combinations for Class 1 Plate-And-Shell-Type Component Supports			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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	
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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.
C.1-14		Conforms	
Reg. Guide 1.132, Rev. 1, 3/79 – Site Investigations for Foundations of Nuclear Power Plants			
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.
Reg. Guide 1.133, Rev. 1, 5/81 – Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors			
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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.
Reg. Guide 1.134, Rev. 3, 3/98 – Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses			
General		N/A	Not applicable to AP1000 plant design certification. DCD Section 13.5 defines the responsibility for administrative procedures.
Reg. Guide 1.135, Rev. 0, 9/77 – Normal Water Level and Discharge at Nuclear Power Plants			
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 – Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.139, Rev. 0, 5/78 – Guidance for Residual Heat Removal			
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 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-1-1997 Article SA-4500	Exception	Exhaust ductwork upstream of the containment air filtration system exhaust filters that has a negative operating pressure are 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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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	
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 are 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			Containment isolation for AP1000 fluid systems conforms to Reference 42 with the following exceptions and/or clarifications. 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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	
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			
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 Position	Clarification/Summary Description of Exceptions
C.1.1.1	Regulatory Guide 1.143, Table 1	Conforms	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. 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.
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, waste 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 tank rooms 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. Two liquid radwaste system monitor tanks are three levels up at elevation 100′-0″. 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. The radwaste building that contains three liquid radwaste monitor tanks is designed to the Uniform Building Code. The basemat and curbed structure will

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			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 Position	Clarification/Summary Description of Exceptions
C.2.1	Regulatory Guide 1.143, Table 1	Conforms	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. 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.
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 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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		
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.

Reg. Guide 1.144 – Withdrawn

Reg. Guide 1.145, Rev. 1, 11/82 – Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

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

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
			to bound the majority of them. Chapter 2 provides the interface criteria.
Reg. Guide 1.146 – Withdrawn			
Reg. Guide 1.147, Rev. 12, 5/99 – Inservice Inspection Code Case Acceptability ASME Section XI Division 1			
General	ASME Code, Section XI	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 – Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.150, Rev. 1, 2/83 – Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations			
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 difference between the pressure boundary integrity of an instrument sensing line in accordance with the appropriate parts of ASME Code, Section III, or ANSI B31.1, as applicable, and the availability of the protection function of safety-related instruments. 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.
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.152, (Task 1C 127-5), Rev. 1, 1/96 – Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants			
General	ANSI/IEEE-ANS-7-4.3.2-1993	Conforms	
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			
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.
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			
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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 – Assuring the Availability of Funds for Decommissioning Nuclear Reactors			
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.
Reg. Guide 1.160, Rev. 2, 3/97 – Monitoring the Effectiveness of Maintenance at Nuclear Power Plants			
General		N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for a Plant Maintenance Program.
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.
Reg. Guide 1.162, Rev. 0, 2/96 – Format and Content of Report for Thermal Annealing of Reactor Pressure Vessels			
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.
Reg. Guide 1.163, Rev. 0, 9/95 – Performance Based Containment Leak-Test Program			
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	

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.165, Rev. 0, 3/97 – Identification and Characterization of Seismic Sources and Determination Safe Shutdown Earthquake Ground Motion			
General		N/A	Subsection 2.5.2.1 defines the responsibility to address site-specific information related to the vibratory ground motion aspects.
Reg. Guide 1.166, Rev. 0, 3/97 – Pre-Earthquake Planning and Immediate Nuclear Power Plant Operator Postearthquake Actions			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
Reg. Guide 1.167, Rev. 0, 3/97 – Restart of a Nuclear Power Plant Shut Down by a Seismic Event			
General		N/A	Not applicable to AP1000 design certification. Subsection 13.5.1 defines the responsibility for the plant procedure preparation.
Reg. Guide 1.168, Rev. 0, 9/97 – Verification, Validation, Reviews, and Audits 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 1.169, 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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
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 – An Approach for using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis			
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.
Reg. Guide 1.175, Rev. 0, 7/98 – An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing			
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.
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			
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.
Reg. Guide 1.178, Rev. 0, 9/98 – An Approach for Plant-Specific Risk-informed Decisionmaking Inservice Inspection of Piping			
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.

Criteria Section	Referenced Criteria	AP1000 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			
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.
Reg. Guide 1.180, Rev. 1, 10/03 – Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference in Safety-Related Instrumentation and Control Systems			
General		Conforms	See Appendix 3D for a discussion of the EMI/RFI qualification.
Reg. Guide 1.181, Rev. 0, 9/99 – Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)			
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.
Reg. Guide 1.182, Rev. 0, 5/00 – Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants			
General		N/A	Not applicable to AP1000 design certification. Section 17.5 defines the responsibility for a Plant Maintenance Program.
Reg. Guide 1.183, Rev. 0, 7/00 – Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors			
General		Conforms	
Reg. Guide 1.184, Rev. 0, 8/00 – Decommissioning of Nuclear Power Reactors			
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.
Reg. Guide 1.185, Rev. 0, 8/00 – Standard Format and Content for Post-shutdown Decommissioning Activities Report			
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 Position	Clarification/Summary Description of Exceptions
Reg. Guide 1.186, Rev. 0, 12/00 – Guidance and Examples of Identifying 10 CFR 50.2 Design Bases			
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.
Reg. Guide 1.187, Rev. 0, 11/00 – Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments			
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.
Reg. Guide 1.189, Rev. 0, 4/01 – Fire Protection for Operating Nuclear Power Plants			
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.
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. 197, Rev. 0, 5/03 – Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
DIVISION 4 – Environmental and Siting			
Reg. Guide 4.7 Rev. 2, 4/98 – General Site Suitability Criteria for Nuclear Power Stations			
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.
DIVISION 5 – Materials and Plant Protection			
Reg. Guide 5.9 Rev. 2, 12/83 – Specifications for Ge (Li) Spectroscopy Systems for Material Protection Measurements Part 1: Data Acquisition Systems			
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.
Reg. Guide 5.12, Rev. 0, 11/73 – General Use of Locks in the Protection and Controls of Facilities and Special Nuclear Materials			
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	
Reg. Guide 5.65, Rev. 0, 9/86 – Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls			
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.

Criteria Section	Referenced Criteria	AP1000 Position	Clarification/Summary Description of Exceptions
DIVISION 8 – Occupational Health			
Reg. Guide 8.2, Rev. 0, 2/73 – Guide for Administrative Practices in Radiation Monitoring			
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.
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			
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 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	
Reg. Guide 8.10, Rev. 1-R, 5/77 – Operating Philosophy For Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable			
General		N/A	Not applicable to AP1000 design certification. Subsection 12.1.3 defines the responsibility for operational considerations of ALARA.
Reg. Guide 8.12 – Withdrawn			
Reg. Guide 8.13, Rev. 3, 6/99 – Instruction Concerning Prenatal Radiation Exposure			
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.
Reg. Guide 8.14 – Withdrawn			
Reg. Guide 8.15, Rev. 1, 10/99 – Acceptable Programs for Respiratory Protection			
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.
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 8.38, Rev. 0, 6/93 – Control of Access to High and Very High Radiation Areas of Nuclear Plants			
General		Conforms	

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.