3.2 Classification of Structures, Components, and Systems

ABWR Standard Plant structures, systems and components are categorized as nuclear safetyrelated or non-nuclear safety-related (see Table 3.2-1). The safety-related structures, systems and components, perform nuclear safety-related functions as defined here, and are classified in accordance with Subsection 3.2.3. In addition, specific design requirements are identified for the safety-related equipment commensurate with their safety classification (see Table 3.2-2 and 3.2-3).

A safety-related function is a direct or support function that is necessary to assure:

- (1) The integrity of the reactor coolant pressure boundary.
- (2) The capability to shut down the reactor and maintain it in a safe condition.
- (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guidelines exposures of 10CFR100.

3.2.1 Seismic Classification

ABWR Standard Plant safety-related structures, systems, and components, including their foundations and supports, that are required to perform nuclear safety-related functions during or after a safe shutdown earthquake (SSE) are designated as Seismic Category I.

All safety-related ABWR Standard Plant structures, components, and systems are classified as Seismic Category I, except those (e.g., pipe whip restraints), as noted on Table 3.2-1, which need not function during but shall remain functional after the event of an SSE. Also some nonsafety-related structures, systems, and components are classified as Seismic Category I as noted on Table 3.2-1.

The Seismic Category I structures, systems and components are designed to withstand, without loss of function, the appropriate seismic loads (as discussed in Section 3.7) in combination with other appropriate loads.

The seismic classifications indicated in Table 3.2-1 meet the requirements of Regulatory Guide 1.29 except as otherwise noted in the table.

3.2.2 Quality Group Classifications

Quality group classifications as defined in NRC Regulatory Guide 1.26 are shown in Table 3.2- 1 for all components under the heading, "Quality Group Classification". Although not within the scope of Regulatory Guide 1.26 definitions, component supports, core support structures and primary containment boundary that are within the scope of ASME Code Section III, are assigned per Tables 3.2-2 and 3.2-3, a quality group classification as identified in Table 3.2-1. Quality group classifications and design and fabrication requirements defined in Regulatory Guide 1.26 are indicated in Tables 3.2-1 and 3.2-3, respectively. Figure 6.2-38 depicts quality group classifications of the components in major systems.

3.2.3 Safety Classifications

Safety-related structures, systems, and components of the ABWR Standard Plant are classified for design requirements as Safety Class 1, Safety Class 2, or Safety Class 3 in accordance with their nuclear safety importance. These safety classifications are identified on Table 3.2-1 for principal structures, systems, and components. Components within a system are assigned different safety classes depending upon their differing safety importance; a system may thus have components in more than one safety class. Safety classification for supports within the scope of ASME Code Section III, depends upon that of the supported component.

The definitions of the safety classes in this section are based on Section 3.3 of ANS Standard 52.1, and examples of their broad application are given. Because of specific design considerations, these general definitions are subject to interpretation and exceptions. Table 3.2- 1 identifies component classifications on a component-by-component basis.

Minimum design requirements for various safety-related classes are delineated in Tables 3.2-2 and 3.2-3. Where possible, reference is made to accepted industry codes and standards which define design requirements commensurate with the safety-related function(s) to be performed. In cases where industry codes and standards have no specific design requirements, the sections that summarize the requirements to be implemented in the design are indicated.

3.2.3.1 Safety Class 1

Safety Class 1 (SC-1) applies to all components of the reactor coolant pressure boundary (as defined in 10CFR50.2), and their supports, whose failure could cause a loss of reactor coolant at a rate in excess of the normal makeup system, and which are within the scope of the ASME Code Section III.

Safety Class 1 components are identified in Table 3.2-1.

3.2.3.2 Safety Class 2

Safety Class 2 (SC-2) applies to pressure-retaining portions, and their supports, of primary containment and to other mechanical equipment, requirements for which are within the scope of the ASME Code Section III, that are not included in SC-1 and are designed and relied upon to accomplish the following nuclear safety-related functions:

- (1) Provide primary containment radioactive material holdup or isolation
- (2) Provide emergency heat removal for the primary containment atmosphere to an intermediate heat sink, or emergency removal of radioactive material from the primary containment atmosphere
- (3) Introduce emergency negative reactivity to make the reactor subcritical
- (4) Ensure emergency core cooling where the equipment provides coolant directly to the core (e.g., emergency core cooling systems)
- (5) Provide or maintain sufficient reactor coolant inventory for emergency core cooling (e.g., suppression pool)

Safety Class 2 includes the pressure-retaining portions of the following:

- (1) Those components of the control rod system which are necessary for emergency negative reactivity insertion
- (2) Emergency core cooling systems
- (3) Primary containment vessel
- (4) Post-accident containment heat removal systems
- (5) Pipes having a nominal pipe size of 25A or smaller that are part of the reactor coolant pressure boundary

Safety Class 2 structures, systems, and components are identified in Table 3.2-1.

3.2.3.3 Safety Class 3

Safety Class 3, (SC-3) applies to those structures, systems, and components, not included in SC-1 or -2, that are designed and relied upon to accomplish the following nuclear safety-related functions:

- (1) Provide for functions defined in SC-1 or -2 by means of equipment, or portions thereof, that is not within the scope of the ASME Code Section III.
- (2) Provide secondary containment radioactive material holdup, isolation, or heat removal.
- (3) Except for primary containment boundary extension functions, ensure hydrogen concentration control of the primary containment atmosphere to acceptable limits.
- (4) Remove radioactive material from the atmosphere of confined spaces outside primary containment (e.g., control room or secondary containment) containing SC-1, -2, or -3 equipment.
- (5) Maintain geometry within the reactor to ensure core reactivity control or core cooling capability.
- (6) Structurally bear the load or protect SC-1, -2, or -3 equipment in accordance with the requirements.
- (7) Provide radiation shielding for the control room or offsite personnel.
- (8) Provide inventory of cooling water and shielding for stored spent fuel.
- (9) Ensure nuclear safety-related functions provided by SC-1, -2, or -3 equipment (e.g., provide heat removal for SC-1, -2, or -3 heat exchangers, provide lubrication of SC-2 or -3 pumps, provide fuel oil to the emergency diesel engine).
- (10) Provide actuation or motive power for SC-1, -2, or -3 equipment.
- (11) Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety-related functions required of SC-1, -2, or -3 equipment.
- (12) Supply or process signals or supply power required for SC-1, -2, or -3 equipment to perform their required nuclear safety-related functions.
- (13) Provide a manual or automatic interlock function to ensure or maintain proper performance of nuclear safety-related functions required of SC-1, -2, or -3 equipment.
- (14) Provide acceptable environments for SC-1, -2, or -3 equipment and operating personnel.
- (15) Monitor plant variables that are identified requiring Category 1 electrical instrumentation in Table 1 of Regulatory Guide 1.97.

Safety Class 3 includes the following:

- (1) Reactor trip and isolation system
- (2) Electrical and instrumentation auxiliaries necessary for operation of the safetyrelated systems and components
- (3) Systems or components which restrict the rate of insertion of positive reactivity
- (4) Secondary containment
- (5) Service water systems required for the purpose of:
	- (a) Removal of heat from SC-1, SC-2 or SC-3 equipment
	- (b) Emergency core cooling
	- (c) Post-accident heat removal from the suppression pool
- (d) Providing cooling water needs for the functioning of emergency systems
- (6) Initiating systems required to accomplish emergency core cooling, containment isolation and other safety-related functions
- (7) Spent fuel pool
- (8) Fuel supply for the onsite emergency electrical system
- (9) Emergency equipment area cooling
- (10) Compressed gas or hydraulic systems required to provide control or operation of safety-related systems

Safety Class 3 structures, systems and components of the ABWR design are identified in Table $3.2 - 1.$

3.2.4 Correlation of Safety Classes with Industry Codes

The design of plant safety-related equipment is commensurate with the safety importance of the equipment. Hence, the various safety classes have a gradation of design requirements. The correlation of safety classes with other design requirements is summarized in Tables [3.2-2](#page-59-0) and [3.2-3.](#page-60-0)

3.2.5 Non-Safety-Related Structures, Systems, and Components

3.2.5.1 Definition of Non-Nuclear Safety (NNS) Category

Structures, systems, and components that are not SC-1, -2, or -3, are non-nuclear safety-related (NNS) and are identified with "N" in the Safety Class column of Table [3.2-1.](#page-8-0)

Some NNS structures, systems and components have one or more selected but limited, requirements that are specified to ensure acceptable performance of specific NNS functions. The selected requirements are established on a case-by-case basis commensurate with the specific NNS function performed (see Table [3.2-2\)](#page-59-0). The functions performed by this subset of NNS structures, systems, and components are:

- (1) Process, extract, encase, or store radioactive waste.
- (2) Ensure required cooling for the stored fuel (e.g., spent fuel pool cooling system).
- (3) Provide cleanup of radioactive material from the reactor coolant system or the fuel storage cooling system.
- (4) Monitor radioactive effluents to ensure that release rates or total releases are within limits established for normal operations and transient events.
- (5) Resist failure that could prevent any SC-1, -2, or -3 equipment from performing its nuclear safety-related function (see Table 3.2-2).
- (6) Structurally bear the load or protect NNS equipment providing any of the functions listed in this Subsection 3.2.5.1.
- (7) Provide permanent shielding for protection of SC-1, -2, or -3 equipment or of onsite personnel.
- (8) Provide operational, maintenance or post-accident recovery functions involving radioactive materials without undue risk to the health and safety of the public.
- (9) Following a control room evacuation, provide an acceptable environment for equipment required to achieve or maintain a safe shutdown condition.
- (10) Handle spent fuel, the failure of which could result in fuel damage such that significant quantities of radioactive material could be released from the fuel.
- (11) Ensure reactivity control of stored fuel.
- (12) Protect safety-related equipment necessary to attain or maintain safe shutdown following a fire.
- (13) Monitor variables to:
	- (a) Verify that plant operating conditions are within technical specification limits (e.g., emergency core cooling water storage tank level, safety-related cooling water temperature).
	- (b) Indicate the status of protection system bypasses that are not automatically removed as a part of the protection system operation.
	- (c) Indicate status of safety-related equipment.
	- (d) Aid in determining the cause or consequences of events for post-accident investigation.

3.2.5.2 Design Requirements for NNS Structures, Systems and Components

The design requirements for NNS equipment are specified by the designer with appropriate consideration of the intended service of the equipment and expected plant and environmental conditions under which it will operate.

Where appropriate, the Seismic Category I, ASME Code Section III, or IEEE Class 1E requirements are specified for NNS equipment in Table [3.2-1.](#page-8-0) Generally, design requirements are based on applicable industry codes and standards. Where these are not available, accepted industry or engineering practice is followed.

3.2.5.3 Main Steam Line Leakage Path

The ABWR main steam leakage path utilizes the large volume and surface area in the main steam piping, bypass line, and condenser to hold up and plate out the release of fission products following postulated core damage. In this manner, the main steam piping, bypass line, and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after an SSE.

The main steamlines and all branch lines 65A nominal pipe size in diameter and larger, up to and including the first valve (including lines and valve supports) are designed by the use of an appropriate dynamic seismic system analysis to withstand the safe shutdown earthquake (SSE) design loads in combination with other appropriate loads, within the limits specified. The mathematical model for the dynamic seismic analyses of the main steamlines and branch line piping includes the turbine stop valves and piping to the turbine casing and the turbine bypass valves and piping to the condenser. The dynamic input loads for design of the main steamlines in the reactor building and the control building are derived from a time history model analysis or an equivalent method as described in Section 3.7.

Dynamic input loads for the design of the main steamlines in the turbine building are derived as follows: For locations on the basemat, the ARS shall be based upon Regulatory Guide 1.60 Response spectra normalized to 0.6g (i.e., 2 times ARS of the site envelope). For locations at the operating deck level (either operating deck or turbine deck), the ARS used shall be the same as used at the reactor building end of the main steam tunnel. Seismic Anchor motions shall be similarly calculated.

Figure [3.2-1](#page-61-0) depicts the classification requirements for the main steamline leakage path as described below.

- (1) Main steam piping from the reactor pressure vessel up to and including the outboard isolation valve is classified as QG A (SC-1) and Seismic Category I.
- (2) Main steam piping beyond the outboard isolation valve up to the seismic interface restraint and connecting branch lines up to the first normally closed valve is classified as QG B (SC-2) and Seismic Category I.
- *(3)* [*The main steamline from the seismic interface restraint up to but not including the turbine stop valve (including branch lines to the first normally closed valve) is classified as QG B and inspected in accordance with applicable portions of the American Society of Mechanical Engineers (ASME) Section XI. This portion of the steamline is classified as non-Seismic Category I and analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B, 10CFR Part 50*

are applicable to ensure that the quality of the piping material is commensurate with its importance to safety during normal operational, transient, and accident conditions.] ***

The seismic interface restraint provides a structural barrier between the Seismic Category I portion of the main steamline in the reactor building and the non-Seismic Category I portions of the main steamline in the turbine building. The seismic interface restraint is located inside the Seismic Category I building. The classification of the main steamline in the turbine building as non-Seismic Category I is consistent with the classification of the turbine building.

At the interface between Seismic and non-Seismic Category I main steam piping system, the Seismic Category I dynamic analyses will be extended to either the first anchor point in the non-seismic system or to a sufficient distance in the non-seismic system so as to not degrade the validity of the Seismic Category I analysis.

- *(4)* [*To ensure the integrity of the remainder of main steamline leakage path, the following requirements are met:*
	- *(a) The main steam piping between the turbine stop valve and the turbine inlet, the turbine bypass line from the bypass valve to the condenser, and the main steam drain line from the first valve to the condenser are not required to be classified as safety-related nor as Seismic Category I, but are analyzed using a dynamic seismic analysis to demonstrate their structural integrity under SSE loading conditions.*
	- *(b) The condenser anchorage is seismically analyzed to demonstrate that it is capable of sustaining the SSE loading conditions without failure.*] *†*

[*A plant-specific walkdown of non-seismically designed systems, structures, and components overhead, adjacent to, and attached to the main steamline leakage path (i.e., the main steam piping, the bypass line, and the main condenser) shall be conducted to confirm by inspection that the as-built main steam piping, bypass lines to the condenser, and the main condenser are not compromised by non-seismically designed systems, structures and components.*] ***

3.2.6 Quality Assurance

Structures, systems, and components that perform nuclear safety-related functions conform to the quality assurance requirement of 10CFR50 Appendix B as shown in Table [3.2-1](#page-8-0) under the heading, "Quality Assurance Requirements," and in Table [3.2-2](#page-59-0). Some NNS structures, systems, and components meet the same requirements as noted on Table [3.2-1.](#page-8-0) The Quality Assurance Program is described in Chapter 17.

^{*} See Subsection 3.9.1.7.

[†] See Subsection 3.9.1.7.

Table 3.2-1 Classification Summary

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Table 3.2-1 Notes and Footnotes

- * The RHR/ECCS low pressure flooder spargers are part of the reactor pressure vessel system, see Item B1.5.
- † The ECCS high pressure core flooder spargers are part of the Reactor Pressure Vessel System, see Item B1.5.
- ‡ Pool suction piping, suction piping from condensate storage tank, test line to pool, pump discharge piping and return line to pool.
- f These sample lines are totally within containment and the fission product monitor provides no isolation function.
- ** Includes Reactor Building, Control Building, and Service Building thermal and radiological environmental control functions within the ABWR Standard Plant.
- †† Controls environment in Main and Local control rooms, diesel-generator rooms, battery rooms, ECCS-RCIC, pump rooms within the ABWR Standard Plant.
- ‡‡ Controls environment in rooms or areas containing non-safety-related equipment within the ABWR Standard Plant.
	- a. A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, signal processors, and mechanical modules include turbines, strainers, and orifices. Safety-related motor control centers, power centers, metal clad switchgear, and remote mulitplexing units in the Reactor Building are located outside the Secondary Containment in the emergency electric equipment rooms. The specific location of many of the electrical modules in the Reactor Building are given on Table 9A.6-2.
	- b. $1, 2, 3, N$ = Nuclear safety-related function designation defined in Subsections 3.2.3 and [3.2.5.](#page-4-0)
	- $c. \quad C = \quad \text{Primary Containment}$ H = Service Building

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- g. 1. Lines 25A and smaller which are part of the reactor coolant pressure boundary and are ASME Code Section III, Class 2 and Seismic Category I.
	- 2. All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate and monitor safety systems shall be Safety

^{*} Pump House structures are out of the ABWR Standard Plant scope.

Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

- 3. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate and monitor safety systems shall be Code Group D from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.
- 4. All other instrument lines:
	- i. Through the root valve the lines shall be of the same classification as the system to which they are attached.
	- ii. Beyond the root valve, if used to actuate a safety system, the lines shall be of the same classification as the system to which they are attached.
	- iii. Beyond the root valve, if not used to actuate a safety system, the lines may be Code Group D.
- 5. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system may be Code Group D.
- 6. All safety-related instrument sensing lines shall be in conformance with the criteria of Regulatory Guides 1.11 and 1.151.
- h. Safety/Relief valve discharge line (SRVDL) piping to the quencher shall be Quality Group C and Seismic Category I. In addition, all welds in the SRVDL piping in the wetwell above the surface of the suppression pool shall be non-destructively examined to the requirements of ASME Boiler and Pressure Vessel Code, Section III, Class 2.

SRVDL piping from the safety/relief valve to the quenchers in the suppression pool consists of two parts: the first part is located in the drywell and is attached at one end to the safety/relief valve and attached at its other end to the diaphragm floor penetration. This first part of the SRVDL is analyzed with the main steam piping as a complete system. The second part of the SRVDL is in the wetwell and extends from the penetration to the quenchers in the suppression pool. Because of the penetration on this part of the line, it is physically decoupled from the main steam piping and the first part of the SRVDL piping and is, therefore, analyzed as a separate piping system.

i. Electrical devices include components such as switches, controllers, solenoids, fuses, junction boxes, and transducers which are discrete components of a larger

subassembly/module. Nuclear safety-related devices are Seismic Category I. Failsafe devices are non-Seismic Category I.

- j. The control rod driver insert lines from the drive flange up to and including the first valve on the hydraulic control unit are Safety Class 2, and non-safety-related beyond the first valve.
- k. The hydraulic control unit (HCU) is a factory-assembled engineered module of valves, tubing, piping, and stored water which controls two control rod drives by the application of pressures and flows to accomplish rapid insertion for reactor scram.

Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Groups A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connection to conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components, and instruments).

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but of the remaining parts and details. For example: (1) all welds are LP inspected; (2) all socket welds are inspected for gap between pipe and socket bottom; (3) all welding is performed by qualified welders; and (4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer standards and proven design techniques which are not explicitly defined within the codes for Quality Groups A, B, or C. This is supplemented by the QC technique described.

- l. The turbine stop valve is designed to withstand the SSE and maintain its integrity.
- m. The RCIC turbine and pump are designed and fabricated to ASME Code Section III.
- n. All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards is used as an alternate to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those defined in Paragraph 136.4, Nonboiler External Piping, ANSI B31.1.
- o. The following qualifications are met with respect to the certification requirements:
	- 1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valve to turbine

casing utilizes quality control procedures commensurate with the importance of the prevention of faults.

2. A certification obtained from the manufacturer of these valves and steam loads demonstrates that the quality control program as defined has been accomplished.

The following requirements shall be met in addition to the Quality Group D requirements:

- 1. All longitudinal and circumferential butt weld joints shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrate examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1.
- 2. All fillet and socket welds shall be examined by either magnetic particle or liquid penetrant methods. All structural attachment welds to pressure retaining materials shall be examined by either magnetic particle or liquid penetrate methods. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1
- 3. All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.
- p. A quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction.
- q. Detailed seismic design criteria for the offgas system are provided in Subsection 11.3.4.8.
- r. See Subsection 3.2.5.3.
- s. The recirculation motor cooling system (RMCS) is classified Quality Group B and Safety Class 2 which is consistent with the requirements of 10CFR50.55a. The RMCS, which is part of the reactor coolant pressure boundary (RCPB) meets 10CFR50.55a (c)(2). Postulated failure of the RMCS piping cannot cause a loss of reactor coolant in excess of normal makeup (CRD return or RCIC flow), and the RMCS is not an engineered safety feature. Thus, in the event of a postulated failure of the RMCS piping during normal operation, the reactor can be shutdown and cooled down in an orderly manner, and reactor coolant makeup can be provided by a normal make up system (e.g., CRD return or RCIC system). Thus, per

 $10CFR50.55a(c)(2)$, the RMCS need not be classified Quality Group A or Safety Class 1, however, for plant availability, the system is designed, fabricated and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class 1 criteria as specified in Subsection 3.9.3.1.4 and Figure 5.4-4.

- t. A quality assurance program for the Fire Protection System meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800), is applied.
- u. Special seismic qualification and quality assurance requirements are applied.
- v. See Regulatory Guide 1.143, Paragraph C.5 for the offgas vault seismic requirements.
- w. The condensate storage tank will be designed, fabricated, and tested to meet the intent of API Standard API 650. In addition, the specification for this tank will require: (1) 100% surface examination of the side wall to bottom joint and (2) 100% volumetric examination of the side wall weld joints.
- x. The cranes and fuel servicing equipment are designed to hold up their loads and to maintain their positions over the units under conditions of SSE.
- y. All off-engine components are constructed to the extent possible to the ASME Code, Section III, Class 3.
- z. Components associated with safety-related function (e.g., isolation) are safetyrelated.
- aa. Structures which support or house safety-related mechanical or electrical components are safety-related.
- bb. All quality assurance requirements shall be applied to ensure that the design, construction and testing requirements are met.
- cc. A quality assurance program, which meets or exceeds the guidance of Generic Letter 85-06, is applied to all non-safety-related ATWS equipment.
- dd. Deleted.
- ee. Figure [3.2-1](#page-61-0) depicts the classification requirements for the feedwater system. At the interface between Seismic and non-Seismic Category I feedwater piping system, the Seismic Category I dynamic analyses will be extended to either the first anchor point

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in the non-seismic system or to sufficient distance in the non-seismic system so as not to degrade the validity of the Seismic Category I analysis.

- ff. Deleted
- gg. The Head Holding Pedestal is non-safety related and Seismic Category I. All other reactor vessel servicing equipment is non-seismic Category I.
- hh. Light fixtures and bulbs are not seismically qualified but fixtures which receive Class 1E power are seismically supported (see Subsections 9.5.3.2.2.1 and 9.5.3.2.3.1).

Table 3.2-2 Minimum Design Requirements for an Assigned Safety Designation

* For structural design requirements that are not covered here and in Table [3.2-3,](#page-60-0) see Section 3.8.

- † Safety designations are defined in Subsections [3.2.3](#page-1-0) and [3.2.5](#page-4-0).
- ‡ Table [3.2-3](#page-60-0) shows applicable codes and standards for components and structures in accordance with their quality group identified in Table [3.2-1](#page-8-0).

Non-nuclear safety (NNS) related structures, systems and equipment that are not assigned a Quality Group in Table 3.2-1 are designed to requirements of applicable industry codes and standards (see Subsection [3.2.5.2](#page-5-0)).

Some NNS structures, systems, and components are optionally designed to Quality Group C or D requirements of Table 3.2-3, per Quality Group designation on Table [3.2-1](#page-8-0).

 f Seismic Category I structures, systems and components meet design and analysis requirements of Subsection 3.7.

Some NNS structures, systems and components are optionally designed to Seismic Category I design criteria as noted on Table [3.2-1](#page-8-0). Some safety-related components (e.g., Pipe whip restraints) have no safety-related function in the event of an SSE, and are not Seismic Category I.

Safety-related electrical equipment and instrumentation are designated SC-3 and are designed to meet IEEE Class 1E (as well as Seismic Category I) design requirements.

Some NNS electrical equipment and instrumentation are optionally designed to IEEE Class 1E requirements as noted on Table [3.2-1.](#page-8-0)

†† Safety-related structures, systems and components meet the quality assurance requirements of 10CFR50, Appendix B, as described in Chapter 17.

Some NNS structures, systems, and components meet the QA requirements as noted on Table [3.2-1](#page-8-0).

Table 3.2-3 Quality Group Designations—Codes and Industry Standards

* Applicable Standards or Subsections of the ASME Code Section III.

† For pumps classified in Group D, ASME Code Section VIII, Division 1, shall be used as a guide in calculating the wall thickness for pressure-retaining parts ad in sizing the cover bolting.

‡ Tanks shall be designed to meet the intent of API, AWWA, and/or ANSI B96.1 Standards as applicable.

 f See Subsection 3.9.1.7. The change restriction is limited only to the applicability to design of piping and</sup> piping supports. See Table 1.8-21 for restriction to change of ASME Code Edition for design of piping and supports only.

3.2-62

Classification of Structures, Components, and Systems

